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INTERNATIONAL ATOMIC ENERGY AGENCY

EXPERTS MISSION TO REVIEW THE PSAR OF THE RRR FOR ARPANSA

FINAL REPORT, 10 July 2001

REPORT TO

THE GOVERNMENT OF AUSTRALIA

Australian Radiation Protection and Nuclear Safety Agency

Sydney, Australia
28 May - 8 June 2001

NSNI PROJECT
DEPARTMENT OF NUCLEAR SAFETY

INTERNATIONAL ATOMIC ENERGY AGENCY

END OF MISSION REPORT
by

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1. INTRODUCTION

1.1 BACKGROUND

Subsequent to a request from the CEO of the Australian Radiation Protection and Nuclear Safety Agency (ARPANSA), a peer review of the Preliminary Safety Analysis Report (PSAR) prepared for the Replacement Research Reactor (RRR) to be built in Lucas Heights, Australia, was organised by the International Atomic Energy Agency (IAEA). The review meeting was held at Australian Nuclear Science and Technology Organization (ANSTO) facilities in Lucas Heights as well as the ARPANSA offices between 28 May to 8 June 2001.

The review team consisted of two staff members (A. Gürpinar and M. Voth) and four outside experts; Messrs. H. AbouYehia (France), A. D'Arcy (South Africa), M. Hansen (Denmark) and D. Rorer (USA).

The PSAR was prepared by ANSTO as part of their application for a construction license. The main contractor is INVAP (from Argentina) and there are several Australian companies involved mainly in civil engineering as subcontractors to INVAP. The facility is called the Replacement Research Reactor because it is intended that it replaces the existing HIFAR research reactor at the same site. The proposed research reactor is of a pool type with a rated power of 20MW. A cold neutron source (CNS) in the reactor pool forms part of the facility; however, a separate safety analysis will be prepared for the experimental utilisation of the source. For this reason the CNS has not been reviewed in detail.

In August 1998, the IAEA was invited by Environment Australia (EA) to review the Environmental Impact Statement (EIS) for the Replacement Research Reactor (RRR). The review report prepared by the IAEA for the EIS included several recommendations that were to be taken into consideration in the PSAR. In this sense, a part of the present review constitutes a follow up of the review of the EIS.

The review concentrated on the identification and discussion of safety issues. A list of these issues (including any ensuing comments or recommendations) is given in Appendix I.

1.2 OBJECTIVES OF THE MISSION

The objectives of the mission were to:

- Advise ARPANSA on the adequacy of the PSAR,
- Advise ARPANSA on the safety of the proposed RRR on the basis of the PSAR,
- Follow up on the activities following the EIS recommendations.

1.3 REVIEW SCOPE

The review covered all chapters of the PSAR. While the review was comprehensive, it focused on a number of key areas important for the fundamental safe design of the reactor. Four working groups were established with designated specialists from the IAEA, ARPANSA and

ANSTO (as well as ANSTO contractors and subcontractors). The working groups are given in Appendix III.

The scope also included the follow up of recommendations from the review of the EIS which was conducted in August 1998.

1.4 BASIS AND REFERENCE FOR THE REVIEW

The review is based on the PSAR which was delivered electronically to the IAEA and the outside experts one week before the review meeting. Supporting documents were provided to the review team by ANSTO and INVAP during the meeting. A list of these additional documents is given in Appendix VII.

Discussions with counterpart specialists constituted a major part of the review basis. The interaction with the counterparts is summarized in the next section, Conduct of the Review.

The main references for the review are the IAEA safety standards related to research reactor safety as well as a selected number of standards related to nuclear power plant site and design safety. Of the latter use was made of the following:

- Safety Series No. 50-SG-D15, Seismic Design and Qualification for Nuclear Power Plants (1992)
- Safety Series No. 50-SG-S5, External Man-induced Events in relation to Nuclear Power Plant Siting (1981)
- Safety Series No. 50-SG-S1 (Rev. 1), Earthquakes and Associated Topics in relation to NPP Siting(1991)

1.5 CONDUCT OF THE REVIEW

Subsequent to the agreement reached between the IAEA and ARPANSA on the RRR PSAR peer review, ARPANSA made available to the IAEA background documents on previous work related to the site and EIS as well as ARPANSA rules and regulations on nuclear installation licensing. Then a preparatory meeting was arranged to decide on such matters as the scope, number and profile of experts, review meeting venue and dates, draft agenda of the review meeting, action plan and logistical matters. This meeting took place at ARPANSA offices in Miranda 26 to 28 March 2001 for which one IAEA staff member (A. Gürpinar) participated.

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

- Transmittal of the PSAR electronically to the IAEA and the external experts one week before the review,
- Written questions in Appendix IV (two sets on consecutive days) by the review team were answered by the counterparts (some of these answered in written form, Appendix V),
- Transmittal of issues and issue clarification by the review team to the counterparts,

- Discussion of the issues in working group and plenary meetings,
- Presentation of a preliminary draft report and discussion of the issues and main conclusions and recommendations with ARPANSA,
- Written responses (counterpart views and measures) from the counterpart to issues as described in the preliminary draft report,
- Preparation and submission of the draft report to ARPANSA on 8 June 2001.

The final Agenda of the meeting and milestones in the report preparation and submission are provided in Appendix II.

1.6 STRUCTURE OF THE REPORT

The main body of the report comprises the Introduction and the Main Conclusions and Recommendations. The latter is a summary of the main findings as well as some of the more pertinent recommendations that are elaborated in greater detail in Appendix I, Issues Identified.

Appendix I constitutes the main technical part of the report in which all the identified issues are discussed. Each issue is first clarified by the review team (ie. reasons are provided for why the team thinks that this is an issue). Then counterpart views and measures in response to the concern expressed are given. Finally, the review team either considers that the issue is resolved (ie. there is no issue) or provides comments and recommendations which may assist in addressing the issue.

Twenty two issues were identified during this review process. Several additional comments are made in Appendix VI which are of a more miscellaneous nature.

2. MAIN CONCLUSIONS AND RECOMMENDATIONS

The Main Conclusions and Recommendations of the review team are presented below. The first six items are general observations and conclusions whereas the remaining items include selected recommendations that are addressed more fully in Appendix I of this Report.

1. The review of the PSAR for the RRR, in part, constitutes the follow up of the IAEA review of the EIS performed in August 1998 for Environment Australia. It is confirmed that all the recommendations made during the EIS review have been addressed in the PSAR. Further recommendations related to some of the issues raised in the EIS review mainly concern the treatment of external events (Issues No. 2, 3, 4 and 6).
2. What has been already stated in the IAEA Review Report of the EIS can be confirmed as a result of the PSAR review, ie. the site for the proposed reactor has no negative characteristics which would make it unacceptable from a nuclear or radiological safety point of view. Furthermore, the 1.6 km exclusion zone around the reactor is a favourable feature which does not exist for a majority of research reactors or nuclear power plants in the world.
3. The review team appreciated the high level of professionalism and positive attitude of ANSTO, INVAP and their contractors and consultants during the review process which significantly enhanced the effectiveness of the review. Their awareness of pertinent safety issues and their dedication to deal with them in the design process provided the review team with confidence that the recommendations in this report will be followed up with attention and diligence.
4. In general, the design approach incorporates safety as a very important and an integral part of the process and provides the RRR with safety features having passive characteristics and high reliability. Potential radiological exposures to the population and workers have been kept at very low levels in line with good international practice and the principle of ALARA has been used effectively in the design process.
5. The PSAR reflects the present design of the RRR accurately, effectively and in considerable detail. It has been prepared using IAEA safety standards and reflects good current international practice. It provides an adequate basis for licensing purposes.
6. Both ANSTO and INVAP have Quality Assurance Programmes (QAP) and the work performed by the subcontractors is also covered under the project QAP. ANSTO has conducted audits to ensure the adequacy of the QA measures implemented at the contractor and subcontractor level.
7. The Probabilistic Safety Assessment (PSA) which is presented as an Appendix to the PSAR is a first attempt to quantify core damage probabilities (for Level 1 PSA) for both internally and externally initiated accident sequences. The total core damage (CD) frequencies result in very low values (between 10^{-7} and 10^{-8} CD per annum). It is recommended to continue to develop the PSA, integrating elements coming from detailed design and experience feedback from other installations and other initiators such as fire, in order to be able to use it as a realistic design tool.
8. While it is recognised that the seismic design basis proposed by ANSTO is a conservative choice when compared to nuclear installations (both research reactors and nuclear power plants) in intraplate seismotectonic regions of the world, it is recommended that an additional study 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

conservatism used in the design process should be demonstrated on the basis of a comparison with the results of this study. Alternatively, ANSTO may choose to demonstrate that the design basis adopted for the RRR is sufficiently conservative with respect to the most recent studies, specifically those taking account of the uncertainties. 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. All Seismic Class 1 items which are active or which have moving parts should be qualified by testing. The testing of the control rod drive mechanism for the First Shutdown System (FSS) is of particular importance because of the special significance of this system for seismic safety. 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.

9. The thermal-hydraulic analysis of reactor transients following anticipated initiating events has received some attention in this review. In general, the review team found the selection of initiating events and the conduct of the safety analyses to be a well presented and representative exercise. Numerous issues were clarified with the counterparts. The major issues, addressed in more detail in Appendix I, focus attention on the validation of codes used in the analysis and the presence of marginal boiling during the evolution of some of the transients. It would be useful if such behaviour predicted by the computer codes were supported by experimental tests in identified cases. In particular, it is suggested that, at the time of hot commissioning tests, it would be useful to make physical measurements of fuel clad temperature at various operating powers using an instrumented fuel plate in order to check the validity of the calculated values. This would be especially useful for transients where cooling is by natural convection (an important safety issue).

10. The PSAR reports a systematic review of potential failures of structures, systems or components and considers the consequence of each failure. The events with bounding consequences are evaluated in greater detail. While the list of analysed failures is extensive, the review team recommends that two additional scenarios be considered. First, a break of the reflector vessel dump line in the reactor pool has the potential to make the secondary shutdown system inoperable. Second, the side plates or ends of plates of a fuel assembly can be damaged during fuel handling so as to restrict coolant flow; this could result in excessive local heating in the high power density core. The applicant indicated the intention to include additional analyses of these scenarios in the final design and engineering phase.

APPENDIX I - ISSUES IDENTIFIED

In the course of the review of the PSAR the review team identified twenty-two specific issues which it considered in greater detail. The following list identifies the issues which are discussed in the pages that follow:

ISSUE No.	TITLE
1.	Application of ASME Codes and Standards to Aluminium and Zircaloy Vessels
2.	Seismic Hazard
3.	External Hazards (Other than Seismic)
4.	Seismic Design of Reactor Building
5.	Replacement/Repair of Embedded Components
6.	Seismic Design and Qualification of Systems and Components
7.	Mechanical Stresses in the Reflector Vessel
8.	Tritium
9.	Validation of Thermal-Hydraulic Codes used for the Reactor Core
10.	Material Surveillance Programme
11.	Independence of the Reactor Protection Instrumentation
12.	Cold Neutron Source Analysis Assumptions
13.	Break of Reflector Drain Line in Reactor Pool
14.	Treatment of the Onset of Nucleate Boiling (ONB) Condition
15.	Engineering Hot Spot Factors
16.	Bounding of Cold Water Injection by a Control Rod Withdrawal
17.	Channel blockage due to Fuel Assembly Damage
18.	Hazards Associated with High Energy Piping Systems
19.	Consequence Analysis for Internal Fire or Explosion
20.	Impact of Loss of Communications on the Safety of the Reactor
21.	QA on Design Calculations
22.	Probabilistic Safety Assessment

ISSUE NUMBER: 1

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

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

3. ISSUE CLARIFICATION:

The application of the ASME Codes to vessels and piping made of materials such as aluminium and Zircaloy may present cases for which clarification is required, since there may be no code case which is directly applicable and some interpretation may be necessary. ARPANSA, ANSTO, and the contractors for this project could differ in their justification for applying selected portions of the existing ASME code to the Reactor Pool, Service Pool, Reflector Tank, Cold Neutron Source and associated piping.

4. COUNTERPART VIEWS AND MEASURES:

The importance of the appropriate application of codes and standards within the project is acknowledged and the clarification suggestion is accepted. With respect to the Reflector Vessel and Cold Neutron Source, the design provisions are as follows:

a) The project is using appropriate ASME codes and Code cases as guideline for design, calculation, manufacturing (for example, welding) and examination. The design uses Finite Element Models and confirmatory comparisons with ASME code solutions in accordance with the design by analysis provisions of the codes.

b) Since ASME and ASTM are widely used and validated it is also an advantage for purchasing materials, screening, manufacturing suppliers, testing standards and acceptance criteria.

c) Materials properties are selected using ASTM standards and safety limits determined using codes as guidelines. Conservative margins are provided.

d) The Reflector Vessel is designed using ASME III as a guideline. This code has been applied in other similar engineering projects with success and results have been validated with years of experience.

e) Complementary design aspects are considered such as: technical advise of CNEA materials engineering technical divisions, international consultants and international raw materials suppliers. Additional information is drawn from the very extensive use of aluminium alloys in research reactor applications and Cold Neutron Sources, and Zirconium alloys in research reactors and nuclear power plant applications.

f) The use of aluminium alloys for the cold neutron source is to the requirements of the Code Case N519 of ASME III. This provides for the use of conservative materials properties for design and also requires the establishment of a surveillance program to confirm the changes in materials properties with time in a radiation environment. This is being developed as described in PSAR Section 5.9.7.

With respect to the other internal structures, the design provisions are as follows:

a) ASME and ASTM are used as guidelines for the design of internal structures.
b) ASME III NF codes are used for mechanical calculation design for all loading cases including seismic loads.
c) All components are designed to work within elastic limit in all loading cases; including SL-1 and SL-2 seismic cases.
d) Neutronic calculations are performed considering radiation damage, activation and shielding issues.

As indicated in PSAR Sections 5.2 and 5.9, attention is being paid to the variation of materials properties due to the effect of radiation and environmental conditions during the lifetime of each component.

5. COMMENTS AND 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 Zircaloy 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.

ISSUE NUMBER: 2

1. REVIEW AREA: PSAR Section 3.2.6.2

2. ISSUE TITLE: Seismic Hazard

3. ISSUE CLARIFICATION:

Several seismic hazard analyses have been performed for the HIFAR and the RRR at Lucas Heights in recent years. The study by the Institute of Geological Nuclear Science (IGNS) resulted in a peak ground acceleration (PGA) of 0.41g associated with a site specific response spectrum which has less amplification in the longer periods of vibration than standard response spectra (such as the USNRC R.G. 1.60). This study was reviewed by two independent reviewers (Robin McGuire of Risk Engineering and AGSO) who indicated that the methods used in the IGNS study were state of the art but some of the assumptions in the seismic source model were overly conservative. There was also a suggestion that some assumptions which had high uncertainty could not be evaluated as to their effect on the results (ie. whether they were conservative assumptions or not). In November 2000 a meeting was convened in Canberra which included Australian specialists and IGNS to discuss the uncertainties in the IGNS study. This meeting recommended some other values (or range of values) to be taken into consideration in a new study again to be undertaken by IGNS. This study is nearing completion.

ANSTO has decided to use a USNRC R.G. 1.60 type response spectrum anchored to a zero period acceleration value of 0.3g. The two response spectra have a cross over point at about 7 Hertz (ie. the site specific response spectrum anchored to 0.41g exceeds the standard spectrum anchored to 0.3g for frequencies over 7 Hertz). The reasons given for this decision are, (1) the expectation that the seismic hazard will substantially decrease as a result of the new study, (2) the standard spectrum anchored to 0.3g envelopes the site specific response spectrum developed by IGNS as a result of their probabilistic seismic hazard assessment (PSHA), (3) the work done over many years for the Lucas Heights site.

The issues related to seismic hazard are as follows:

- The assumption that the adopted (standard) response spectrum anchored to 0.3g envelopes the site specific response spectrum anchored to 0.41g is not valid for all frequency ranges of interest. The rock site transmits high frequencies and according to INVAP calculations all the components analysed within the pool have a frequency in excess of 7 Hertz. Even the reactor building itself has a natural frequency close to 7 Hz.
- ANSTO has not initiated detailed site investigations (similar to but not as detailed as those recommended in IAEA Safety Guide 50-SG-S1, Rev.1) which may have removed some of the uncertainties in the seismotectonic model. There is only one study relating to faults and fractures within ~3km of the site. It is therefore not clear the basis on which ANSTO anticipates a substantial decrease in the hazard values.
- The Operating Basic Earthquake (OBE) value is derived using a scaling of the whole hazard curve and it is not necessarily true that this will be the case even if there is a substantial reduction in the PGA value.

4. COUNTERPART VIEWS AND MEASURES:

The choice of a design value for the Reactor Facility was based on the numerous studies that have been done by ANSTO for the Lucas Heights region over many years, using the best available seismic experts. In 1995, Corran reviewed and compiled much of this information into a single report and proposed a seismic hazard curve for the SL-2 or SSE event of 0.23 g. with the Carbon Canyon spectrum. More recent advice from AGSO has confirmed the appropriateness of this choice.

Following the PSA of HIFAR in 1998, PLG commented that the seismic hazard curve quoted in Corran (1995) had a low uncertainty band. As a result of this and to assure conservatism for the replacement reactor, ANSTO increased the peak ground acceleration to 0.3g. This position was accepted by ARPANSA as a suitable design approach. The methodology used for the component assessment, by using 100% of both X and Y components effectively increases the 0.3g by a factor of 1.4. For the buildings, ANSTO will follow AS 1170 and also combine the horizontal components in a conservative manner to give a design value higher than 0.3g.

With respect to the IGNS study, the appropriate process is to design against the controlling earthquake, obtained by disaggregation of hazard curves. The controlling earthquake for the Lucas Heights site is the M5.5-6.5 event at around 10 km. From the IGNS work, this has a PGA of 0.33g. The design value effectively bounds this recommended value from IGNS.

The PSAR shows that the structure has considerable capacity beyond 0.3g, to a factor of two.

Site Investigation

The only recommendation from IGNS was for a near field assessment to confirm that there were no local features that could amplify a specific event on this site. That was done and is in the Coffey report. Other data were not available to do a more extensive study.

Operating Basis Earthquake

The counterpart agrees with the comment on the scaling. The process followed from their choice of the US NRC 1.6 approach. However, detailed analysis by Max Irvine confirms that the choice is conservative for the OBE.

5. COMMENTS AND RECOMMENDATIONS:

C1. The seismic design basis proposed by ANSTO is considered to be a conservative choice when compared to nuclear installations (both research reactors and nuclear power plants) in intraplate seismotectonic regions of the world. For example, nuclear power plants in Canada, Eastern USA, China, Russian Federation, Korea, Brazil, Argentina and Europe have design basis seismic values less than 0.3g. In fact, the only nuclear facilities (with possibly a few exceptions) with zero period acceleration values in excess of 0.3g are in California and Japan (which are interplate areas).

C2. The PSAR relies on several studies conducted and/or reviewed by ANSTO and other interested parties.

C3. There is a lack of site investigations for the RRR such as those recommended by the IAEA Safety Guide 50-SG-S1 (Rev. 1). Only local investigations for faults and fractures have been done out to a radius of ~3kms. Because of this and the fact that the site is in a predominantly intraplate tectonic environment, there is significant uncertainty related to seismotectonic parameters which are used in seismic hazard analyses (ie. source identification, maximum magnitudes, b values, attenuation relationships and their standard deviations, etc.). General lack of historical earthquake data because of the comparatively short period of recorded history in Australia adds to this uncertainty.

C4. There is considerable controversy in Australia regarding the results obtained by IGNS. It seems that a majority of the local specialists consider the results to be overly conservative. The comparatively high zero period acceleration value of 0.41g was largely driven by uncertainties resulting in the perceived over conservatism. However, to date, a quantification of the conservatism within a systematic framework is lacking. The results of the second study of IGNS (recommended by the November 2000 specialists' meeting in Canberra) are expected to be delivered soon.

C5. The high frequency (ie. short period) part of the design basis response spectrum is particularly important for the site geological conditions (ie. rock) and the components of the RRR.

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 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.

ISSUE NUMBER: 3

1. REVIEW AREA: PSAR Section 4.3.3

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

3. ISSUE CLARIFICATION:

Lucas Heights is located near the urban area of Sydney and consequently there is a multitude of sources for potential human induced hazards in the site vicinity. These are listed in the PSAR and can be summarised as; airports, military base, gas pipeline, and two small methane gas power plants. There is also road traffic and a railway although the latter is not very near.

All the hazards with a potential to originate from the above sources are screened either using the distance or the low probability of occurrence. The calculations are not provided in the PSAR and their conservatism may be questioned. The only human induced event which has been considered is the crash of a Cessna airplane on the reactor building. The effects from this event are taken into account in the design partially by a grillage structure over the reactor building. For other human induced events there is no provision in the design (e.g. impact from an artillery shell or blast from any source) due to the results of the screening process.

The issues are as follows:

- It is considered good practice to provide for an 'envelope' design for the reactor building of nuclear installations to resist impact and blast loads from a variety of sources. This seems not be the case for the RRR
- The potential burden that the grillage structure may bring for other loads such as wind, earthquake (especially overturning) and the increase in the overall fire hazard that it may cause, may need to be further discussed.

4. COUNTERPART VIEW AND MEASURES:

The counterpart pointed out that the information provided in the PSAR on human induced external events summarised matters addressed in the Facility Licence Application, Site Authorisation submitted to ARPANSA. The Application drew on studies of the impacts on the site of the military base, local industrial facilities and gas pipelines, and hazardous road and rail transport made for the EIS for the RRR and the HIFAR PSA. The Facility Site Licence F0001 issued by ARPANSA identified aircraft impact and bushfires as the non-seismic external events that required to be addressed in the Reactor Facility design. The other external events were accepted as being bounded by the aircraft crash or having no safety impact on the Reactor Facility. The Environment Australia recommendations from the EIS assessment required ongoing monitoring of the frequency and severity of external events to ensure that assessed risks to the Reactor Facility remain valid and acceptable. ANSTO has administrative control in the 1.6 km exclusion zone radius and does not intend to permit any increase of activities in that zone that could pose a hazard to the reactor. In addition, ANSTO will continue to monitor road and rail traffic usage to assess any changes in the frequency of hazardous transport. No significant change in the usage assessed in the earlier studies has been identified to date. Severe blast impacts assessed in the conduct of sabotage studies have also shown adequate resistance of the Reactor Facility building and structures with no damage to the reactor of safety systems. ANSTO considers that in total, these various measures address the principles of an envelope design of the facility. Nevertheless, ANSTO will undertake a further assessment of an appropriate blast event involving traffic on the local public roads.

Regarding the second point, it was indicated by the counterpart that increases in all loads related to the existence of the grillage was considered and incorporated into the design. The fire following an aircraft crash (in the grillage) would last 20 minutes and would not pose a threat to the reactor.

5. COMMENTS AND RECOMMENDATIONS:

C1. ANSTO has agreed to undertake a review of the impact on the facility of blast from traffic accidents.

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.

ISSUE NUMBER: 4

1. **REVIEW AREA: PSAR Section 4.4.3.3.2**

2. **ISSUE TITLE: Seismic Design of Reactor Building**

3. **ISSUE CLARIFICATION:**

The reactor building is founded on competent bedrock (V_s ~ 1300 m/s) and is embedded to a maximum depth of about 7 metres although the part under the grade is not in contact with the rock. The water table is very near the surface and buoyancy forces will affect the seismic analysis. The seismic analysis is being done using both the IAEA Safety Guides for NPPs (50-SG-D15) and IAEA TECDOC 348 which is applicable to nuclear installations with limited radioactive inventory including research reactors up to a few MWs. In particular, some questions were discussed relating to the combination of ground motion components (2 horizontal and one vertical).

For the SSE (SL-2) and OBE (SL-1) earthquakes, response spectra at 7% and 4% damping were used respectively. Elastic design principles were used throughout.

The issues are:

- A clear application of the standards used in the design is not given in the PSAR. It is not clear for example to what aspects of design 50-SG-D15 and IAEA TECDOC 348 apply. The compatibility of these as well as their applicability to different cases and items is not discussed in the PSAR.
- Somewhat linked to the above issue is a concern about the conservatism to be incorporated in to the building design. The perception that the hazard value is overly conservative seems to be influencing the decisions made for the conservatism used in the design of a reactor building.

4. **COUNTERPART VIEWS AND MEASURES:**

The PSAR cites the standards in chapter 2 and 4, but a new Table has been provided to the review team (Appendix 5) outlining the use of the various Codes and Standards used in the seismic design process.

Related to the second issue, ANSTO confirmed that the seismic hazard related topics (such as the perceived conservatism) will not in any way allow the designer to reduce seismic loads or use unjustified loading combinations. ANSTO and INVAP provided a written statement to the review team on the procedure for combining horizontal components. This results in a conservative choice for the design value, beyond the 0.3g. Details are given in Appendix 5. A report was also provided to the review team outlining the process used in the design of components to conservatively account for the horizontal components. This gives a value of 1.4 times the 0.3g design value.

5. COMMENTS AND 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.

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. ISSUE CLARIFICATION:

There is a concern that some components of the plant, such as the decay tanks, certain piping sections, and pool leak collection systems, are embedded in concrete or placed in other locations that are difficult or impossible to access. The potential impact of the failure of these components should be addressed in the PSAR, together with the possibility of their repair, replacement and/or the consequences of abandonment.

Specific items to be considered are: Reactor Pool, Service Pool, Pool Leak Collection Systems, Decay Tanks and piping embedded in concrete shielding.

4. COUNTERPART VIEWS AND MEASURES

The leak collection systems of the Reactor Pool and Service Pool will help to localise any leakage to a particular section of the wall or floor of the pool. Repairs can be made by draining the pool and re-welding the stainless steel pool tank at the location of the leak. The geo-membrane covering on the exterior surfaces of these pools is intended to guide the leaking water to the collection sensors. Failure of this membrane may reduce the leak collection efficiency but should not prevent the eventual determination of the location of the leak point.

The Decay Tanks are accessible for inspection and repair. There is space in the Decay Tank rooms to access the outside surface of the tanks, and the interior surfaces can be accessed via manholes.

No aluminium piping is embedded in concrete. Carbon steel piping embedded in concrete is protected from corrosion by cathodic protection systems. Embedded stainless steel piping is very unlikely to corrode or crack if proper water chemistry is maintained.

5. COMMENTS AND RECOMMENDATIONS

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.

ISSUE NUMBER: 6

1. REVIEW AREA: PSAR Section 4.5.15

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

3. ISSUE CLARIFICATION:

The reactor is designed using passive systems to reduce the probability of having accidents. In such cases the mechanical integrity of systems and components under extreme internal and external loads become significant. In particular, the seismic qualification of the pool including all the internals, penetrations, interactions with water and the concrete as well as the building structure itself is a major task to be undertaken. A global analytical Finite Element (FE) model would be needed for this. The studies which have been started already for the reflector pool openings and Cold Neutron Source (CNS) containment can be integrated to this global model. It is noted that some of these studies lead to stresses which are very near to allowable values (using ASME codes). The applicability of ASME codes to some types of materials used in the design (e.g. Zircaloy) is also an open question. (See Issue Number 1).

Active systems as well as systems whose parts need to move during the earthquake (such as control rods) are generally qualified by testing because analytical models do not adequately represent the dynamic behaviour of these systems. Given that the SSS requires 15 seconds to shut down the reactor, during a seismic event the FSS is needed for safe shutdown.

The issues are as follows:

- There is no clear indication in the PSAR related to the type of analysis which will be performed for the pool and all the associated structures and components.
- The applicability of the ASME codes to materials employed needs to be checked. (See Issue Number 1).
- There is no clear indication in the PSAR for the type of qualification (ie. testing) intended for active systems (or systems with moving parts) which are Seismic Class 1.

4. COUNTERPART VIEWS AND MEASURES:

Explanations on the intended analyses of the pool and the associated structures and components were provided to the review team.

A global dynamic analysis of the pool structure with the internals and penetrations will be made (with the possible exception of some isolated and well defined components).

The Pool is perfectly embedded in the Concrete Reactor Block because it is used as formwork during the fresh concrete cast. The welded rings and beams reinforcements will be located inside the concrete. This special construction ensures that the pool will have the same global deformation as the Concrete Block. In this case, the interaction between internal components and pool shell will be only local and in the support zones.

Two types of analysis will be carried out for the Pool.

- The pool will be modelled and subjected to the displacement of the Concrete Block during SSE. This model will give information of stresses mainly between the rings.
- The local zones of supports and connections will be analysed in order to provide enough safety margins at these joints. This is important for two reasons: to give appropriate supports to the components and to provide water tightness avoiding the pool shell failure.

The internal structures will be analysed isolated from the Reactor Pool and with their supports fixed to the pool. The support reactions obtained in these conditions will be used in the verification of stresses at the pool shell.

The second issue is discussed under Issue 1.

The counterpart agreed that all Seismic Class 1 active systems or systems with moving parts will be seismically qualified in accordance with applicable section of 50 SG D15 and a project Procedure that was presented to the review team. The notable exception in this agreement is the control rod drive mechanism for the FSS. The difficulty and the limited usefulness of the results of such test were pointed out. It was agreed that some testing (although probably not in a shaking table) would be foreseen to test the effectiveness of the control rods under an imposed deflection.

The moving parts of the FSS include: a robust component (the mechanism) and a less rigid component (control rod bar and absorber plate).

These moving parts with less rigid components move within a very robust and rigid structure: they are guided through the control rod guide box and the control rod penetration to the reactor pool bottom, through bushings and seals, that provide a solid support for an accurate linear perpendicular motion.

The fact is stressed that the bushings and seals are linked rigidly to very stiff components, namely the penetration channel through the pool bottom and the inlet plenum, so by geometry, thickness, and design, the deformations are negligible in these supporting components, ensuring a correct guiding along the whole displacement of the moving parts, even during seismic actions.

The stiff guiding components from the pool into the control drive room have been seismically evaluated and undergo minimum deformation under seismic events. Calculations demonstrate that seismic produced deformations are negligible and do not interfere with the safety function of the components.

The fall of the control rods is furthermore ensured by the action of the compressed air, supplied by storage tank located close to the CRDs, and designed to perform under seismic loads.

5. COMMENTS AND 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 rod 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 the 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.

ISSUE NUMBER: 7

1. REVIEW AREA: PSAR Section 5.2.6.5, Reactor

2. ISSUE TITLE: Mechanical Stresses in the Reflector Vessel

3. ISSUE CLARIFICATION:

The reflector vessel is a very complex structure that is difficult to design, construct, and analyse. The welds will be difficult to inspect after fabrication but particularly in service. The integrity of the structure is critical to maintain separation of the heavy water in the reflector from the light water in the pool. The reflector is subject to multiple stresses including residual stresses from welding, thermal expansion, irradiation-induced growth, interactions with interconnecting systems, bouyancy when the heavy water is dumped by actuation of the secondary shutdown system and potential accidents such as the drop of a heavy object, a detonation in the cold neutron source well or a seismic event. The reflector tank will be subjected to substantial fast and thermal neutron flux which will affect the material properties. Repair or replacement will likely involve significant personnel radiation exposure, down time and expense. The PSAR reports the stress due to heavy water replacement to be 63% of yield; however, this can be reduced procedurally by lowering the reactor pool level during this evolution. The margin to yield in some cases is very small considering the potential for error and uncertainty in analytical techniques.

4. COUNTERPART VIEWS AND MEASURES:

It is acknowledged that the Reflector Vessel is a complicated structure. However significant effort has been undertaken in evaluating the loads that the vessel will experience in normal operating and accident conditions. Notice has also been taken of the operation of similarly complicated structures used in Research Reactors such as those at HANARO (Korea), Maple (Canada), FRM-2 (Germany), ILL (France) and others.

A detailed finite element analysis has been used to compute the stress distribution under various conditions. The neutron fluence and heat loads have also been characterised extensively using Monte Carlo techniques. Reports were provided to the review team that summarised the stress analyses as well as the forces resulting from a detonation in the cold neutron source thimble. The applicant and designers acknowledge the critical nature of the vessel and the forces to which it is subjected. The intent is that it be a lifetime fixture monitored under the in-service inspection program.

The following Load Combinations were studied at the Preliminary Engineering phase:

Operational

Operating Pressure + Temperature Operation + Zy Growth (at 40 years)

Pressure Operation + Temperature SSS discharge + Zy Growth (at 40 years)

Pressure Heavy Water Replacement + Zy Growth (at 40 years)

Accident

Operating Pressure + Operating Temperature + Zy Growth (at 40 years) + Earthquake

Operating Pressure + SSS discharge temperature + Zy Growth (at 40yrs) + Earthquake

Operating Pressure + Operating Temperature + Zy Growth (at 40years) + CNS detonation

The value of 150 MPa is ≈63% of the Zircaloy yield strength and corresponds to the highest value that was found during the preliminary analysis. This maximum stress is located at the joints between the top flat plate and the internal columns. The load that produces this stress is that imposed by the Heavy Water Replacement Pressure (vacuum inside the Vessel and full pool-height head of water). It is noted that this is not an accident condition, but rather a planned maintenance operation.

During the Detail Engineering phase, the stresses will be re-evaluated taking into account the final geometry, internal supports, irradiation tubes and applied loads. If at this stage the stresses in the Reflector Vessel, during the replacement of Heavy Water, are of concern, the operation will be amended to ensure that stresses are within allowable limits. Post-weld annealing of the Reflector Vessel is an available option if the residual stresses in the vessel welds are of concern.

Reflector Vessel design criteria and ability to perform periodic inspections and maintenance:

The design of the Reflector Vessel takes into account fabrication and inspection issues, component lifetime, and operation and maintenance requirements. These design groups criteria and the engineering documentation were extensively reviewed by a large group of INVAP and ANSTO personnel during the Preliminary and Critical Design Reviews carried out during the Preliminary Engineering stage.

The work for the internal reactor structures includes manufacturing of a 1:10 scale model and a 3D-computer model to test and evaluate all tasks related to internal component movements and operations, including assembling sequences, feasibility of inspection and testing, start up and maintenance activities. The PSAR incorporates preliminary drawings and cutaways produced with the 3D-computer model.

During manufacturing, all internal components (Reflector Vessel, control rod guide box, core grid, irradiation facilities rigs, and other components) will be assembled in the workshop. A systematic confirmation of the adequacy of the components with respect to installation, operation, maintenance and operational issues will be undertaken. This methodology, in which the design engineers who undertook the design participate, has been used in other projects with success. The feedback of lessons learned is part of INVAP's approach to QA

Access for inspection to the Reflector Vessel

Much of the Reflector Vessel surface is accessible from outside, either from the reactor pool (for chimney, irradiation tubes, lateral wall, top and bottom plates, tubes holding pneumatic tubes and silicon rigs, and beam tubes necks), or through the beam tubes. Access for internal inspection of the Reflector Vessel is through three main access points: from the reactor pool through the flanges of the Cold and Hot Neutron Sources, and through the large heavy water drainage pipe from the heavy water room or the Control Rod Drive Room.

Presently available technology of endoscopes and remote operated tools make if feasible to carry out a set of reliable inspections. Both INVAP and ANSTO have relevant experience in this area.

5. COMMENTS AND RECOMMENDATIONS:

C1. The applicant recognizes the critical nature of the reflector vessel, the implications of repair or replacement and the difficulty in performing inservice inspection. Appropriate attention has been given during the preliminary design of the vessel.

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

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. ISSUE CLARIFICATION:

A significant quantity of tritium will be produced in the reflector tank during operation, and the tritium concentration will rise to approximately 10 Curie/l in 10 years. This tritium cannot be completely contained in sealed systems but will unavoidably migrate, albeit at significantly lower concentrations, to other systems within the facility. Various components in the reflector system are expected to require maintenance, and perhaps removal from service for repair in a workshop. The demand for opening of the system and the potential incidents could result in a release of tritiated water. Welding required on the system could also cause a release. Leakage from pumps, valves, seals, etc., must be taken into account and actions planned accordingly. During decommissioning the release of tritium and tritium-contaminated components must also be addressed.

4. COUNTERPART VIEWS AND MEASURES:

Two factors were analysed regarding this issue: human factors and plant design features.

With respect to human factors, ANSTO has extensive experience in operating HIFAR where components containing or having been in contact with tritiated water are routinely removed from the reactor.

The Reflector Vessel and its associated cooling and purification systems are sealed and not normally opened to the atmosphere. Consequently, anticipated releases of tritiated water to the atmosphere will be very much less than for HIFAR. Additionally, the reflector cooling and purification systems incorporate many features designed to minimise the release of tritiated water to the environment during routine maintenance or as a consequence of accidents. These include the ability to drain and vacuum dry parts of the systems without opening the systems to atmosphere.

The Reflector Vessel is sealed and of Zircaloy construction. Almost every joint is welded with the numbers of flanges being minimised. The rest of the heavy water circuitry is constructed of stainless steel and will also remain sealed during normal operation. The reflector purification system will maintain the purity of the heavy water such that corrosion of the components will be negligible. The cover gas of the heavy water circuit is helium.

Should there be a leak of heavy water from the circuit, it will be into a sealed room with its own ventilation system that incorporates molecular filters to extract the tritium prior to discharge or the need for personal entry to effect maintenance or repairs.

An intermediate cooling circuit is provided containing high purity demineralised water to form an additional barrier between the heavy water circuit and the secondary cooling system.

This circuit is also provided with monitors that detect any leakage of tritiated heavy water through the primary plate-type heat exchanger which itself is welded.

The heavy water plant room is also fitted with appropriate monitoring provisions and closed circuit TV to detect any leaks that may occur.

5. COMMENTS AND RECOMMENDATIONS:

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 of tritium in the reflector system, particularly personnel performing maintenance. The Emergency Planning and Preparedness should include special precautions to be taken when working with tritium-contaminated components.

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. ISSUE CLARIFICATION:

Section 5.8 of the PSAR presents the main aspects of the thermal-hydraulic design for the reactor and provides information that indicates that the core integrity will be preserved and fuel cladding will be properly cooled in all conditions including operational states and anticipated operational occurrences. PARET-PC and RETRAN02 thermal-hydraulic computational codes have been used to calculate respectively reactivity transients and accidents, and transient involving a design basis initiating event (DBIE) relating to cooling systems. The thermal – hydraulic calculations, which are used in the safety analysis presented in chapter 16, have significant uncertainties relating to the various parameters and modeling assumptions used in the codes. The fact that some margins were added to the input data used in the calculations may not mean, in the absence of a sensitivity check for the various parameters, that the calculated fuel cladding temperatures are conservative.

One possible solution to this issue could be the use, during the reactor commissioning tests, of a fuel element instrumented with thermocouples to measure the fuel cladding temperature and to compare the results of measurements with the calculated values. This experimental validation, which could be done by placing the instrumented fuel element at location corresponding to hot spots of the core, constitutes a means to confirm the presence or absence of critical phenomena that may cause damage to the core fuel elements.

4. COUNTERPART VIEWS AND MEASURES:

INVAP and ANSTO indicated that the thermal – hydraulic calculations for the Reactor Facility were made with conservative assumptions, so that worst case conditions were always considered in the analyses. Some sensitivity studies have been performed that study the effect of the uncertainties associated with the various parameters and the effect of combined uncertainties on the calculated values of fuel cladding temperatures. These studies include sensitivity analysis of (a) “friction factors” in various critical components of the Primary Cooling system (e.g. the flap valves); (b) thermal conductivity and heat capacity of the fuel; (c) non-functioning of selected key engineered components (e.g. the flap valves). The results of the analyses showed no significant changes to fuel temperatures. The counterpart indicates that the feasibility of the above-mentioned experimental validation with an instrumented fuel element will be examined. However, it is noted by the counterpart that there are practical difficulties with measuring clad temperatures accurately that may lead to significant experimental errors. The suggested experimental validation would be performed at normal operation conditions, where single-phase flow exists, and therefore the onset of critical phenomena would not be experimentally tested.

5. COMMENTS AND RECOMMENDATIONS:

C1. Direct measurement of the fuel cladding temperature with an instrumented fuel element, as a 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 especially useful for the evaluation of the transients in which the cooling is by natural circulation (important safety issue) and for which the validity of the various calculation codes is 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.

ISSUE NUMBER: 10

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

2. ISSUE TITLE: Materials Surveillance Program

3. ISSUE CLARIFICATION:

The PSAR lacks information on a materials surveillance program such as:

1. The need to obtain representative sample coupons of the aluminium, Zircaloy, and stainless steel actually used in the construction of the reactor and its components.
2. Placement of the coupons within the reactor, taking into account the expected sample environment (temperature, pressure, water chemistry, magnitude of the neutron flux, and ratio of fast to thermal neutron flux) at the proposed location.
3. Dimensions of the coupons required to meet ASTM requirements for the appropriate metallurgical tests; e.g. tensile, Charpy impact, etc.
4. Number of coupons needed to accommodate a testing program spanning the expected lifetime of components while leaving the remainder to accumulate exposure.

Examination of material coupons for neutron activation products (especially those arising from impurities in the materials) will provide valuable estimates of the buildup of radioisotope inventories. This information will be needed for decommissioning.

4. COUNTERPART VIEWS AND MEASURES

This issue has been addressed in the PSAR section 5.9.7. In addition, the review team was provided a copy of the Materials Surveillance Program prepared during the Preliminary Engineering stage.

5. COMMENTS AND RECOMMENDATIONS

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

ISSUE NUMBER: 11

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

2. ISSUE TITLE: Independence of Reactor Protection Systems.

3. ISSUE CLARIFICATION:

The rules governing protection channels, including their detectors and associated electronic equipment, specify that such channels must be dedicated. No other functions (such as control or monitoring) may be performed using the same inputs.

There are several instances in the PSAR where it is evident that such duplication of function is considered in the First Reactor Protection System (FRPS). One example is as follows:

Section 16.15.2.4 describes the use of an alarm initiated by an area monitor on the operating floor of the pool structure to warn operators when they have inadvertently lifted a radioactive object too high in the pool at risk of receiving an unacceptable dose of radiation – an activity that has nothing to do with the safety of the reactor core.

The same area monitor is connected to the FRPS and has the purpose of tripping the reactor via the FSS in the event that reactor operation leads to high radiation levels at the pool operating floor.

4. COUNTERPARTS VIEWS AND MEASURES:

ANSTO clearly agreed that the reactor protection systems are independent and isolated from the other systems. The design strictly follows IEEE 384 “Standard criteria for independence of class 1 E equipment and circuits” and IAEA Safety Series 50 SG D3 section 7.8.4. and IAEA TECDOC 973 section 2.2.3.5.

Where the First Reactor Protection System and the Second Reactor Protection System are activated by the same event, the two systems have their own three trains with three sensors dedicated to each system. This means that in some cases, both systems will be triggered at the same time, e.g. a seismic event. In cases where the First Reactor Protection System is said to be sufficient to shut down the reactor and the First Shutdown System fails, then the Second Shutdown System is triggered.

5. COMMENTS AND RECOMMENDATIONS:

C1. If the construction of the systems are designed 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 is therefore closed.

ISSUE NUMBER: 12

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

2. ISSUE TITLE: CNS Safety Analysis Assumptions

3. ISSUE CLARIFICATION:

The large surface area that is chilled to cryogenic temperatures provides pumping speeds for condensable gases that exceed that of any external pump. Potential accumulation of oxygen and nitrogen from small leaks and gas impurities is a hazard that should be considered, especially if the helium blanket is breached when any part of the system must be opened to the atmosphere from time to time for maintenance and repairs.

The use of adsorbers to trap gaseous impurities in the cold helium raises the question of off-gassing of these impurities when the system is operated in the warm gas circulation Stand-by Operation mode. There is a potential that these accumulated impurities will freeze out in places other than the adsorber when the helium gas is cooled down again.

The Cold Neutron Source Protection System is capable of generating a reactor trip in case the moderator chamber cooling cannot be guaranteed. This interaction with the Reactor Control System bears careful scrutiny.

4. COUNTERPART VIEWS AND MEASURES

The entire deuterium inventory is contained within a blanket of inert gas composed of either cryogenic purity helium gas or high purity nitrogen gas, depending upon the location (helium within the reactor pool and the in-pile portion of the system, nitrogen for the cold box, valve boxes and the buffer tanks outside the pool). This blanket forms a barrier between the oxygen in the atmosphere and the deuterium within the cold neutron source and is provided with monitors to detect the in-leakage of deuterium and/or oxygen.

Additionally, there is a vacuum surrounding the in-pile portion of the system, which is further enveloped by the heavy water in the Reflector Vessel, providing further protection against the intrusion of oxygen. Therefore, only minute quantities of oxygen should ever be present in the blanket and these should be of no concern, even when accumulated over extended periods of time.

Notwithstanding the low probability of mixing significant amounts of oxygen with the deuterium, the CNS Vacuum Containment is designed to withstand a hypothetical detonation of a stoichiometric mixture of hydrogen and oxygen. It is acknowledged that the effects of the resulting shock wave in the Reflector Vessel still need to be analysed.

5. COMMENTS AND 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 utmost importance to guarantee that there are no “sneak” paths for accumulation and mixing of oxygen with the deuterium.

R2. The Cold Neutron Source forms 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.

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. ISSUE CLARIFICATION:

PSAR Chapter 16 considers accidents from a series of initiating events, among them the loss of heavy water events (Section 16.12). Not considered is a double-ended break of the heavy water line that passes from the reflector vessel through the reactor pool floor. This line is an integral part of the secondary scram system used to dump the heavy water from the reflector vessel to a collection tank. Should the break occur in the reactor pool, one side of the line will drain pool water into the heavy water collection tank until the tank is full. At the reflector side of the break, where the heavy water is initially at a slightly lower pressure than the reactor pool at the core elevation, the pressure will quickly equalise by a small amount of pool water entering the reflector vessel dump line. There will no longer be a driving head to dump the heavy water from the reflector vessel, rendering the secondary shutdown system inoperable.

4. COUNTERPART VIEWS AND MEASURES:

Both the Reflector Vessel and the Reflector Tank Drain line are classified as Seismic Category 1, and they are designed to withstand the consequences of ground motions associated with the Safe Shutdown Earthquake (SSE) or SL-2 earthquake. The double guillotine break of the drain line during an earthquake is therefore considered a Beyond Design Basis Event.

A preliminary finite element evaluation of the Reflector Vessel has been done for the case of very large seismic actions (beyond SL-2). A report on this analysis was presented to the review team. The analysis shows that the Reflector Vessel remains in elastic behaviour during the SL-2. Thus, the boundary conditions on the drain line are mild, even for an earthquake of larger magnitude than the SL-2 earthquake. The drain line is a very stiff structure in itself (being a pipe) and it runs a relatively short length (≈1.5 m) between the Reflector Vessel wall and the reactor pool floor. During detail engineering, the drain line will be analysed to show that it can safely withstand the SL-2 earthquake and that it has a margin for withstanding seismic actions beyond the SL-2 event.

5. COMMENTS AND RECOMMENDATIONS

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.

ISSUE NUMBER: 14

1. REVIEW AREA: PSAR Chapter16, Safety Analysis

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

3. ISSUE CLARIFICATION:

In numerous instances the analyses show fuel clad temperatures increasing beyond the temperature for ONB. In one instance a clad temperature of 17 deg C above the ONB temperature is sustained for 9 seconds (PSAR section 16.8.7.3.3). The interpretation of such behaviour is not clearly described in the PSAR.

A further caution is the fact that the ONB temperature (obtained by the Bergles-Rohsenow equation) does not remain constant, but is dependent on the coolant bulk temperature and the heat transfer coefficient, and can decrease by as much as 10 deg in a reactivity induced power excursion. Further dependence on coolant pressure and saturation temperature produce a similar reduction during loss of flow transients.

4. COUNTERPART VIEWS AND MEASURES:

PARET and RETRAN are validated codes. The PARET analysis of the SPERT experiment compares very well with the experimental results. The input data to the codes contains numerous conservatisms that artificially elevate the clad temperature.

The counterpart accepts the conclusions of the analyses on this basis even though, in some cases as described above, a limited degree of boiling may be observed. This is partly based on results from the analysis for the bounding case accident - Inadvertent withdrawal of a control plate during Start-Up (PSAR section 16.8.7.3.3) with assumed FSS failure – which indicates that the minimum ratio of expected heat-flux to DNB is 1.6. This result combined with the conservative assumptions applied in the analysis indicates a large margin to the critical phenomena and hence to fuel damage.

The counterpart recognises that the predicted ONB temperature varies with core conditions during a transient and agrees that the initial value is not necessarily the best value to use for comparison with the fuel clad temperature at all times during transient evolution.

5. COMMENTS AND 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.

ISSUE NUMBER: 15

1. REVIEW AREA: PSAR Chapter16, Safety Analysis

2. ISSUE TITLE: Engineering Hot Spot Factors

3. ISSUE CLARIFICATION:

There is no evidence in the safety analyses that the anticipated manufacturing deviations (within specification) and other uncertainties related to the fuel assemblies have been taken into account in determining the peak clad temperatures.

4. COUNTERPART VIEWS AND MEASURES:

Implementation of the Engineering Factors in the thermal-hydraulic design of the core is discussed in section 5.8. Section 16.4.1 refers to chapter 5 in the context of core parameters used in the safety analyses.

5. COMMENTS AND RECOMMENDATIONS

C1. A discussion, in the context of the safety analyses and in sufficient detail to acquaint the reader with the nature of the treatment and the significance of all the geometrical tolerances, uncertainties and conservatisms incorporated in the treatment, would be very useful.

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.

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 Rod Withdrawal

3. ISSUE CLARIFICATION:

Section 16.8.4.1 (d) simply states that the reactivity insertion due to cold water injection following the start-up of the primary pumps during low power operation is low and bounded by the Control Rod Withdrawal DBIE. This is not justified in any way.

Basic information critical to the event includes the coldest possible temperature of the dormant water in the Primary Coolant System (PCS), how much is available (ie., how long the reactivity insertion can be sustained) and what the reactivity addition is compared with the Control Rod Withdrawal DBIE.

4. COUNTERPART VIEWS AND MEASURES:

The statement in this section that “the event is considered to be within the design basis” should read “the event is not considered to be within the design basis”. An interlock prevents the start-up of the pumps during low power operation mode.

The start-up of the PCS pumps would result in a reactivity insertion of less than 400pcm. This insertion is bounded by the withdrawal of a control plate during reactor start-up.

5. COMMENTS AND RECOMMENDATIONS

R1. It is recommended that the typographical correction be addressed and that a justification along the lines of the last sentence of the Counterpart Views and Measures above be provided to support the statement that the initiating event is bounded by the Control Rod Withdrawal DBIE.

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. ISSUE CLARIFICATION:

The discussion of mechanisms for fuel channel blockage in section 16.9.2.2.3 does not include the possible bumping of the fuel assembly during handling, which may deform an outer fuel plate and result in the narrowing of the adjacent coolant channel. Another operator "mishap" is the possibility for bending over the tops of several adjacent fuel plates when using an unauthorized tool. Fuel handling and the use of tools is administratively controlled and such an incident can not be dismissed on that basis. The analysis does not determine the extent of damage that can be afflicted in this way and the consequences if the reactor is operated with such a damaged fuel assembly undetected.

4. COUNTERPART VIEWS AND MEASURES:

Fuel damage during handling is an additional cause for channel blockage, and it will be added to the list of causes presented in PSAR Section 16.9.2.2.3. However, given the design of the fuel assemblies (stiff lateral plates, fuel handling pin that prevents the handling tool from reaching the plates), tool design (tool cannot fit in between plates, will not grab any other part of the fuel assembly other than the handling pin) and operational procedures (inspection of fuel assembly after suspected damage), the potential damage and its consequences in terms of fission product release are bounded by the melting of three fuel plates, analysed in Section 16.19 of the PSAR.

5. COMMENTS AND RECOMMENDATIONS:

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

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. **ISSUE CLARIFICATION:**

Section 16.14 claims that there are no high energy piping systems in the facility and that the associated hazards (pipe whip, jet impingement etc.) need not be considered. While it is true that there are no high temperature and pressure systems such as one finds in a nuclear power station, the question still begs as to whether there is sufficient energy in any of the piping systems to cause secondary damage in the event of a rupture. The potential for this is not studied and presented for all Primary Coolant System (PCS) pipes, as well as those carrying other mediums (nitrogen, helium, etc.) at any appreciable pressure. Apart from physical damage, the consequences could also involve the formation of ice, ice projectiles and oxygen deficient atmospheres.

4. **COUNTERPART VIEWS AND MEASURES:**

Even though high energy lines are not part of the project, there is a design strategy and methodology in place to identify potential hazardous equipment and plan the routing so as to place them in isolated places, placing equipment in isolated places, double jacketing equipment when necessary, performing all necessary calculations and also placing all required safety devices such as breathing air and others.

The matter will be fully addressed and analysed in the FSAR.

5. **COMMENTS AND RECOMMENDATIONS:**

C1. Given the counterpart views and measures above, the statement in section 16.14 is unclear.

C2. It is understood that the identification of “potential hazardous equipment” mentioned by the counterpart will include the high energy flywheels on the primary pumps.

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

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. ISSUE CLARIFICATION:

Section 16.14.1 states that no consequence analysis is necessary for internal fires and explosions, on the basis that the reactor will shut down safely. The treatment of internal fires is similarly absent from the PSA.

4. COUNTERPART VIEWS AND MEASURES:

The Defence in Depth approach was adopted for assessment of internal fires and explosions. Section 2.9 and 10.2 provided information on the prevention, detection, control and extinguishing of fires. Section 16.14.1 focussed on the challenge of any fire to the safety of the reactor, namely the ability to shutdown and continue cooling the core. This showed that the three main safety functions would be ensured in the event of internal fires or explosions. It is worth pointing out that all the Engineered Safety Features identified in Chapter 7 of the PSAR have features and redundancies that ensure their ability to function in the event of a fire or explosion.

The provisions put in place to protect against internal fires and explosions will be validated at the FSAR stage by the carrying out of room by room assessments of fire loadings and fire consequences. The results will be incorporated into a probabilistic fire assessment (Fire-PSA).

5. COMMENTS AND RECOMMENDATIONS:

C1. Since a Fire-PSA is in part a consequence analysis, the statement that "no consequence analysis is needed" seems to be contradicted by the intention. It would have been useful if the statement of the last paragraph under Counterpart Views and Measures had been made in the PSAR.

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

ISSUE NUMBER: 20

1. REVIEW AREA: PSAR Section 16.14.3.3, Loss of Communications Capabilities

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

3. ISSUE CLARIFICATION:

Section 16.14.3.3 implies that the loss of communication with the rest of LHSTC is not a safety issue. It furthermore makes no mention of loss of communications within the RRR facility.

The first point is contradicted in section16.17.5.2.6 where it is stated that an accident at HIFAR that could affect the RRR will be communicated to the RRR control room so that the necessary actions can be taken.

The second point misses the importance of communication within the reactor facility with regard to both reactor and radiological safety, as well as to the safety of personnel.

4. COUNTERPART VIEWS AND MEASURES:

Section 16.14.3.3 focussed on whether the loss of communications with the rest of the LHSTC site needed consideration as an initiating event challenging reactor safety and it was determined that it does not. All Engineered Safety Features identified in Chapter 7 of the PSAR will remain operable irrespective of whether communications with the rest of LHSTC are available.

Loss of communications with LHSTC or within the facility will affect operations and has the potential to influence the radiological and occupational health and safety of the operators. For this reason, a fault tolerant communication system is proposed, as described in Section 10.3.

In the highly unlikely event of an accident at HIFAR, its occurrence would be communicated to the Control Room of the Reactor Facility. However, no action is required to ensure the continued safety of the Reactor Facility, the necessary actions to be taken referring to ensuring the health and safety of operating personnel.

5. COMMENTS AND RECOMMENDATIONS:

C1. The statements of (and omissions from) section 16.14.3.2 are unclear. A reference to the description of the communications systems in Chapter 10 and elaboration such as provided by the counterpart above, would provide clarification.

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

ISSUE NUMBER: 21

1. REVIEW AREA: PSAR Chapter 18, Quality Assurance

2. ISSUE TITLE : QA on Design Calculations

3. ISSUE CLARIFICATION:

Chapter 18 of the PSAR presents the general quality system organisation relating to activities of ANSTO (Principal), INVAP (Contractor) and subcontractors, which are associated with INVAP for construction of the facility. Design calculations concerning the reactor facility are performed by the main contractor and its sub-contractors for the benefit of ANSTO. In this context, QA provisions for ensuring technical quality are very important. Additional relevant information that is important for discussion in the PSAR includes the following items:

- arrangements for ensuring good technical quality of the various studies relating to design of the reactor,
- procedure for approving corrective actions in case of non-conformances during construction of the reactor,
- main hold points relating to the reactor facility project.

4. COUNTERPART VIEWS AND MEASURES :

ANSTO presented in a document given to the review team (see Appendix V) the environment for the conduct of design related activities and described the existing quality management systems for the different organisations concerned by the RRR project. ANSTO performs its own studies in order to verify and ensure the good technical quality of INVAP's activities and studies relating to the design of the reactor. All proposed modifications, associated with identified non-conformances during the construction of the reactor, will be reviewed and approved by ANSTO both prior to and after the implementation of the corrective actions. The schedule of the construction and commissioning activities will include the hold and witness points required by the Principal, the Contractor and the regulatory authority.

5. COMMENTS AND RECOMMENDATIONS:

C1. According to information given by the counterpart, the design process for the reactor is fully transparent to ANSTO which has an active role for checking the technical quality of studies performed by the main Contractor and its sub-contractors.

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

ISSUE NUMBER : 22

1. REVIEW AREA : PSAR, Appendix A

2. ISSUE TITLE : Probabilistic Safety Assessment (PSA)

3. ISSUE CLARIFICATION:

The scope of the PSA presented in Appendix A of the PSAR is Level I with some Level III aspects. The Level I PSA results provide the frequency of events (including internal and external events) that may damage the core fuel or other sources of radioactive materials.

Level III PSA aspects include the consequence analysis of some accident sequences selected on the basis of making a significant contribution to the overall core damage frequency. The calculated value of this parameter (3.7×10^{-7} /year) is very low compared to the value adopted for Nuclear Power Plants (10^{-5}/year). There are important uncertainties in the input data used for the calculated values of core damage frequencies corresponding to the different event sequences. In fact, the data base concerning the failure rates of systems or components for research reactors may not be representative.

PSA studies will be useful as a complement to deterministic studies to identify weak points in the design of the reactor and to evaluate the impact of alternative modifications

4. COUNTERPART VIEWS AND MEASURES :

ANSTO and INVAP representatives expressed their agreement concerning the comments on the uncertainties used for the PSA calculations. They indicated that the uncertainties in the data have been considered explicitly, and are provided in the PSA for each component and equipment. These uncertainties have been propagated in order to evaluate their effect on the results. Chapter 6 of the Appendix of the PSAR presents the estimated CDF with confidence limits. They also indicated that the core damage frequency is indeed lower than in the case of a nuclear power plant, due to design characteristics of the facility which include many inherent safety features.

The PSA will be updated in order to take into account the evolution of the project.

5. COMMENTS AND RECOMMENDATIONS:

R1. PSA studies should continue to be updated in order to take into account the detailed 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.

APPENDIX II - SCHEDULE

The peer review project progressed on the following schedule, all dates referring to 2001:

26 to 28 March	Meeting to discuss peer review process	ARPANSA/IAEA
30 March	IAEA receives request for peer review	ARPANSA
30 March	Agreement on PSAR availability date	ARPANSA
6 April	Commitment to perform review and dates	IAEA
17 April	Information on review team provided	IAEA
18 May	PSAR sent to review team	ARPANSA
28 May	Joint meeting for presentation on: - Content and approach of peer review - Introduction of peer review team - RRR design, safety philosophy and PSAR	 EA/ARPANSA IAEA ANSTO*
28 May	Formation of working groups	IAEA
29 May	Discussion of mission expectations	IAEA/ARPANSA
29 May	Review of material , prepare question	IAEA
30 May	Submit written questions	IAEA
30 May	Discuss questions	ARPANSA/ANSTO /IAEA
31 May	Submit additional written questions	IAEA
31 May	Discuss questions	ARPANSA/ANSTO/IAEA
31 May	Identify potential issues	IAEA/ASNO (Observer)
1 June	Working groups clarify issues with counterparts	IAEA/ARPANSA/ANSTO
1 June	Status review	IAEA/ARPANSA
4 June	Response to potential issues	IAEA/ARPANSA/ANSTO
5 June	Prepare draft report of findings	IAEA
6 June	Continue drafting report	IAEA
6 June	Present draft report to ARPANSA; copy given to ANSTO	IAEA

7 June	Discuss presentation of counterpart views stated in draft report	IAEA/ARPANSA/ANSTO
7 June	Finalise draft report	IAEA
8 June	Present draft report to ARPANSA	IAEA/EA/ASNO
8 June	Discussion with Nuclear Safety Committee (Kidziak, Smith and McAneny)	IAEA/ARPANSA/EA/NSA
8 June	Scheduled press briefing (only attended by Michael Priceman, a concerned member of the public)	IAEA/ARPANSA/EA
23 June	ARPANSA comments on the draft report and ANSTO counterpart views and measures sent to IAEA	ARPANSA
29 June	Comments on draft report due to IAEA	ARPANSA
27 July	Final report due to ARPANSA	IAEA

* "ANSTO" is used in this appendix to reference the entire design team. This includes the applicant, ANSTO; their prime contractor, INVAP; and subcontractors.

APPENDIX III - LIST OF PARTICIPANTS

Participants

ARPANSA

Don Macnab	Director, Regulatory Branch
Vince Diamond	Manager, Nuclear Installations
Allan Murray	Manager, Policy & Licensing
Trevor Mountford-Smith	Senior Safety Engineer, Nuclear Installations
Michael Kerr	Safety Engineer, Nuclear Installations
Sergei Zimin	Physicist, Nuclear Installations
Kim Goodrick	Assessment Officer, Policy & Licensing

Environment Australia

Jon Millard	Acting Director, Environment Assessment Branch

Australian Safeguards and Non-Proliferation Office

Andrew Leach	Assistant Secretary

ANSTO

Ron Cameron	Director, Safety Division
Ken Horlock	Director, Nuclear Technology Division
Garry Seaborne	Project Manager, RRRP
Neil McDonald	Manager, Safety & Licensing, RRRP
Andy Willers	Manager, Facility Safety Unit, Safety Division
Mark Summerfield	Senior Engineer, Safety & Reliability, Safety Division
Charles Morris	Senior Instrumentation Engineer, HIFAR, Nuclear Technology Division
Doug Patterson	Structural Engineer, RRRP
Simon Bastin	Senior Reliability Engineer, Safety & Reliability, Safety Division
Warwick Payten	Senior Research Scientist, Materials Division
Greg Storr	Reactor Physicist, HIFAR, Nuclear Technology Division

INVAP

Pablo Abbate	Design Manager
Veronica Garea	Safety & Licensing Manager
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David Polkinghorne	Project Manager

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Boyne Schmidt	Structural Consultant
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IAEA

Aybars Gürpinar	IAEA Vienna
Marcus Voth	IAEA Vienna
Hassan Abou Yehia	France
Alan D'Arcy	South Africa
Mogens Bagger Hansen	Denmark
David Rorer	USA

IAEA Working Groups

Group 1:

Review area: External events including seismic, structures, EIS, follow-up

PSAR chapters: 1, 2, 3, 4, 14 and 18

Reviewer : Mr. Aybars Gürpinar

Group 2:

Review area: Safety systems, core coolant, containment, electrical

PSAR chapters: 1, 2, 5, 6, 7, 9, 10, 11, 13, 15, 18 and 20

Reviewers: Mr. Marcus H. Voth
Mr. David C. Rorer

Group 3:

Review area: Safety approach, categorisation, safety analysis, PSA

PSAR chapters: 1, 2, 12, 16, 17, 18, 20 and Appendix A

Reviewers: Mr. Hassan Abou Yehia
Mr. Alan J D'Arcy

Group 4:

Review area: I&C, reactor protection, software

PSAR chapter: 1, 2, 8, 18 and 19

Reviewer: Mr. Mogens Bagger Hansen

IAEA Review Team Curriculum Vitae

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Aybars Gürpinar was born in Turkey in 1944. He received his BS. in Civil Engineering from Princeton University in 1967 and PhD from University of New Mexico in 1971. He was a professor of Engineering Sciences at the Middle East Technical University in Ankara (1971-1979). He became a senior staff consultant to D'Appolonia S.A. in Brussels (1979-1981) mainly related to safety of nuclear installations from external events. He continued independent consulting in Brussels for the nuclear and petroleum industries (siting and external hazards) until 1986 when he joined the IAEA. as siting and external events specialist. At the IAEA he was always in the Division (then Department) of Nuclear Safety. He became the Unit Head for design in 1997 and Section Head for the Engineering Safety Section in 1998. His publications include three books and over 150 articles in international scientific journals and conference proceedings.

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Relevant Experience:

Station nuclear engineer for the startup of a BWR power plant and manager of licensing (Monticello), operations manager for a 5 MW research reactor and hot cell facility used for the production of radiopharmaceuticals (Union Carbide), director of a university radiation science and engineering center with a 1 MW TRIGA reactor (Penn State), received two USNRC senior reactor operator licenses, member of research and power reactor safety review committees.

Relevant Expertise:

Safety analysis preparation and review, reactor engineering, core analysis, accident analysis, integrated facility operation, ventilation and filtration systems, radioactive material handling systems, radioactive waste processing and shipping, quality assurance program implementation, radiation monitoring, decommissioning.

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Relevant Experience:

20 years experience in the field of the safety of research reactors, Head of the Bureau of Safety Analysis for Research Reactors, Deputy Head of the Division for the Safety Assessment of Gas-Cooled, Fast Breeder, Test and Naval Propulsion Reactors, IPSN expert team leader for the TACIS project relating to the decommissioning of BN-350 in Kazakstan.

Relevant Expertise :

Evaluation of safety documents, accident analysis and evaluation of radiological consequences, reactor physics, analysis of system design and operation (heavy and light water reactor cooling systems, ventilation and filtration systems, radiation monitoring system, etc.), analysis of safety files concerning the design, the operation and the decommissioning of installations.

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Relevant Experience:

Responsible for the revision of the SAFARI-1 Research Reactor operating conditions, including a full safety analysis to meet new operating requirements; Generation of a new SAR and OTS; Liaison with the regulating authority to establish a new operating licence; Participation in the PSA for SAFARI-1; Responsible for the safety evaluation of experimental facilities and rigs for isotope production and other safety related aspects of reactor operation and utilisation, for approval by the SAFARI-1 Reactor Safety Committee (SRSC); Responsible for administration and co-ordination of the SRSC; Responsible for core and fuel management and Safeguards accounting at SAFARI-1; Responsible for reactor operator training; Responsible for all ISO-9001 accredited Quality System documentation relating to reactor safety, core and fuel management, fuel manufacture, operator training and the SRSC; Carried out numerous safety analyses for the Koeberg Nuclear Power Station (PWR type) and transient studies for the validation of the Koeberg full scope simulator; Attended TCMs in Vienna on “Code on the Safety of Nuclear Research Reactors: Design and Operation” and “Safety of Core Management and Fuel Handling for Research Reactors”, the latter of which he chaired; he also attended a CS in Vienna to finalise the TECDOC on core management and fuel handling. Undertook two IAEA supported expert missions to Libya to assist with and review the safety analysis and SAR for the Research Reactor at Tajoura, Tripoli.

Relevant Expertise:

Safety analysis preparation and review; Thermal-hydraulic operational and accident analyses using the PARET and RELAP5 computer codes; Computer based reactor simulation; Reactor protection and specification of OLC; Licensing liaison and co-ordination; Reactor operator training; Quality System maintenance.

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Relevant Experience:

Working as a system engineer at a 10 MW research reactor undertaking the responsibility for the instrumentation and control systems. Also working as a duty officer. Later working as Assistant Reactor Manager and lately as Reactor Manager. Presently Head of Department of Risø Decommissioning (this includes all nuclear facilities at Risø). Participated in Technical Committee meetings concerning instrumentation and control systems and training of staff.

Relevant Expertise:

Renewing instrumentation and control systems, preparation of safety documentation and operational limits and conditions, training of staff, maintenance, heavy water treatment, preparation of emergency plans, building of organizational structures, shipping of spent fuel and decommission.

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Relevant Experience:

Manager of 60 MW research reactor (HFBR) and 3 MW medical research reactor (BMRR), chairman of Laboratory Safety Committee (BNL), member of National Research Council subpanel for review of research reactor operations (NIST), member of peer review teams for 3 research reactors (U. of Michigan, U. of Missouri, and Mass. Inst. of Technology), chairman of Cold Neutron Source Design review committees (NIST and HFIR), member of BWR Nuclear Review Board (Shoreham).

Relevant Expertise:

Safety analysis preparation and review, environmental impact statement preparation and review, cold neutron source design and operation, epithermal neutron beam design, reactor operator training, conduct of reactor operations, emergency planning, radiological protection and monitoring, accident analysis.

APPENDIX IV - QUESTIONS FROM REVIEW TEAM TO COUNTERPARTS

The working groups generated two set of questions requesting clarification or additional information from their ANSTO counterparts and contractors. The first set was submitted Wednesday morning, 30 May 2001, and discussed with counterparts later in the day. A second set of questions was submitted the morning of Thursday, 31 May 2001, and discussed later that day. Questions related to Issue Number 2 (Seismic Hazard), 3 (External Hazard) and 21 (Quality Assurance) were answered both verbally and in writing. The written responses are included as Appendix V. Working Groups 2 and 3 had two members each; their individual question sets are designated as 2a/2b and 3a/3b, respectively.

Question Set No.1, Working Group 1

Chapter 2

Page 2.3-5 (Section 2.3.5.4)

1. How is dependence avoided in seismic design, fire and floods ? Is interdependence taken into account, i.e. between seismic/fire, seismic/flooding, etc?

Page 2.4-11 (Section 2.4.9.1)

1. Is the MCR habitable during the SL-2 earthquake ? Is the ECC habitable during the SL-2 earthquake?
2. Which design basis accidents will result in the MCR being uninhabitable ?

Page 2.4-17 (Section 2.4.9.12)

1. In what way will the FRPS and the SRPS be qualified seismically? Will only the maximum acceleration values recorded and used as signal to the shutdown systems ?

Page 2.5-3 (Section 2.5.2)

(Item d-not applicable) There may be many items under this category which may need to be secured to avoid shifting or sliding (even becoming missiles) under seismic motions.

The description given for Seismic Class 2 (taken from IAEA Safety Guide 50-D-15) does not really cover items listed in Class 2 as given in Table 2.5/2 (e.g. System 35 Fire Protection Building).

The design basis for Seismic Class 2 and 3 seem to be very close (i.e. 0.09g for SL-1 and 0.08g for Seismic Zone in which the site is included). Is it worth carrying out two separate analyses for two values so close?

(Selected questions for table 2.5/2)

1. Why is the reflector (01-20) designated as N/A for seismic class ?
2. Why is the rod position indicator (02-25) Seismic Class 2, is this item not needed to activate SRPS?

In general support or control systems of Seismic Class 1 items are in Seismic Class 2. This may not be appropriate for all cases.

Pages 2.6-1 and 2.6-2 (Section 2.6.1)

IAEA Safety Guide 50-SG-S1 (Rev. 1) was used in a limited way. In particular, (last para) The first two sentences seem to contradict each other.

Page 2.6-2 (Section 2.6.1.1)

1. Does the standard AS1170.4 (applicable to SL-0) cover non structural items (such as mechanical or electrical equipment) ?

Page 2.6-2 (Section 2.6.1.2.1)

The first paragraph has no substantiation and is probably incorrect.

IAEA Safety Guide 50-SG-S2 has been superseded by 50-SG-D15 in 1992.

Page 2.6-3 (Section 2.6.1.3.1)

The first paragraph has contradictory statements. Clarification is needed.

Page 2.6-4 (Section 2.6.1.4)

1. Is SSI needed given the embedment depth and the competent rock expected for the foundation? If so, is rocking neglected?

Page 2.6-4 (Section 2.6.1.5.1)

1. How does the first paragraph of this section (i.e. μ=1) agree with the μ=1.25 from Calvi's equation (previous page)?

Page 2.6-5 (Section 2.6.1.6)

1. What method will be used to seismically qualify the control rod drive mechanism?

Page 2.6-9 (Section 2.6.4)

1. Why is a permanent dewatering system needed ? Will this be a SL-1 system?

Page 2.7-13 (Section 2.7)

Some of the Agency standards cited in the text have not been included in the List of References (e.g. 50-SG-S1, 50-SG-D15, 50-SG-D5).

Chapter 3

Page 3.2-10 (Section 3.2.5.1)

In the first paragraph of this page the existence of numerous faults is indicated for the near region of the site. It is also stated that “their location will probably require detailed surface mapping and application of seismic techniques”. 50-SG-S1 (Rev.1) recommends both of these. Why were these investigations not performed ?

Page 3.2-18 (Section 3.2.5.3.3.5)
1.
Which building is on piles and what Seismic Class is this building?

Page 3.2 – 19 (Section 3.2.6.1)

1. First paragraph states that there are no major faults within a radius of 35 kms of the site. What is the nearest ‘minor’ fault to the site and what are its characteristics?

Page 3.2 –24 (Section 3.2.6.3)

1. The first paragraph provides a slip rate for the Lapstone structural complex and then states that the evidence is for its being inactive. If there is slip on the structure why would the structure be inactive ? Are there no recorded earthquakes on this fault ?

The last paragraph of the section quotes the 10000 year acceleration from IGNS as 0.4 (whereas it is 0.41). Then it gives reduction percentages of 30 – 37% due to over conservatism (from McGuire’s review report). Are the under estimated parameters also considered (as cited in the same reference). Arithmetically the reduced numbers (even when 30 – 37% are taken) are greater than 0.26 to 0.2g. There may be a typographical error.

Page 3.2 – 24 (Section 3.2.6.4)

1. The first paragraph states that the DBE is set at 0.3g PHGA. In fact, the previous discussion does not lead to this value. Some clarification is necessary.
2. The 5% damping RS is provided, however, in Section 2.6.1.3.1 a damping of 7% is suggested for SL-2. What will be used for design ?

Page 3.2 – 30 (Section 3.2.8.2)

In the first paragraph of this page it is stated that the effects of the burning methane gas are bounded by aircraft crash effects. The provided design against aircraft crash is aginst impact loads, therefore it is not clear how it will provide protection against other types of man-induced events (such as blast, fire).

Page 3.2 – 31 (Section 3.2.8.3)

1. It seems that the artillery range is not further considered because it is under the administrative control of the military which is judged to be safe. Is this the case ?

Page 3.2 –31 (Section 3.2.9.1)

1. Which airport does the military aircraft use?

Page 3.2 – 32 (Section 3.2.9.2)

1. What is the distance from the ammo depot of the military base to the reactor and what is the maximum capacity of the depot?
2. Is any blast loading included in the design basis (military, railway, truck, etc)?

Figure 3.2/10-1

1. What is the design basis wind ? Does the tornado loading consider missile effects?

Chapter 4

Page 4.4 – 13 (Section 4.4.3.1.2)

1. IAEA Safety Guide 50-SG-D15 provide load combinations involving both process and accident loads with SL-1 and SL-2. Are these considered ?

Page 4.4 – 17 (Section 4.4.3.2.3)

For the OBE, how is the 0.09g obtained ?

1. For the SSE, it is stated that a uniform risk horizontal acceleration spectrum is adopted. Where is this spectrum ? Why is 7% damping chosen for the SSE?

The reference to IAEA Safety Guides 50-SG-S1 and S2 (which no longer exists) to justify the RS is misleading.

Page 4.4 – 17 (Section 4.4.3.2.4.1)

On the basis of the discussion in Chapter 3, the choice of the aircraft seems arbitrary and not well founded.

Page 4.4 – 18 (Section 4.4.3.2.4.4)

1. Does the fire duration consider burning of the steel grillage on top of the roof?

Page 4.4 – 21 (Section 4.4.3.3.1.1.)

1. Why are the rock anchors needed ?

Page 4.4 – 22 (Section 4.4.1.3.1.8)

1. Is the reactor block also analysed against LOCA pressure ? Where is this analysis described ?

Page 4.4 – 25 (Section4.4.3.3.2.1)

1. Why was the SSI not included in the FE model?

Page 4.4 – 26 (Section 4.4.3.3.2.2)

1. I could not find Figure 4 – 20 (referenced in the next to last paragraph).What criterion is used in developing artificial time histories?

Page 4.4 – 30 (Section 4.4.3.3.2.4)

1. How and at what point on the floor are the vertical FRS calculated?
2. What are the FRS at the four anchorage of the four legs of the grillage structure?

Page 4.4 – 31 (Section 4.4.3.3.2.5)

1. For the steel grillage, it is stated that plane impact loads are much bigger than those from the SSE. Is the total energy input considered in this comparison (i.e. the duration of the two events)?

Please provide a brief description of the uplift analysis (Footings).

1. A brief description of structural damage expected in the reactor building is given for the case of a seismic overload (~0.6g). What happens to the Seismic Class 1 components?

Page 4.4 – 33 (Section 4.4.3.3.3.1)

1. How is the grillage justified (i.e what load reduction does it provide) in the light of paragraph 4 of this section ?

Page 4.4 – 37 (Section 4.4.3.3.3.2)

It is preferable to use the metric units throughout. Please provide further explanation of the last paragraph on this page (adopted values for perforation thickness).

Page 4.4 – 40 (Section 4.4.3.3.3.5)

Please provide a sketch of the AISFG structure including supports and any potential interaction with the reactor, stack and other buildings.

1. Doesn't the rigid missile result in higher response? (first paragraph of 'Composite Roof Structure').
2. What is the permanent formwork?
3. Is there any impact protection for the part the building below 17 m? Has the probability of an impact to this area been considered ? Are other effects of an aircraft impact considered (e.g. vibration and fire)?

Page 4.4 42 (Section 4.4.3.4.1)

1. Is there a basemat for the reactor building?

Page 4.5 – 3 (Section4.5.1.5.1)

1. Does the table refer to load combinations?

Page 4.5 – 5 (Section 4.5.1.5.2.2)

1. Are the CRD room connection through concrete ? Is there any way of inspecting this piping? How is the seismic analysis performed (pipe – concrete interaction)?
2. How big is the CRD room ? Does it have a liner?

Page 4.5 – 10 (Section 4.5.1.5.5)

1. How does the Level 1/Level 2 approach described in this section agree with the Seismic Classes and required load combinations?
2. Was a complete seismic analysis of the pool (including internals, water, penetrations and piping/concrete interaction) performed?

Page 4.5 – 14 (Section 4.5.1.7.1)

1. Does the table refer to load combinations?

Page 4.5 – 15 (Section 4.5.1.8.1)

1. Does the table refer to a load combination?

Questions on the New Seismic Hazard Study

1. What is the main reason to renew the study?
2. Are there new data revealed since the last study in the following fields:
   - Regional tectonics (new literature)
   - Near regional and/or site vicinity tectonics (new literature or field work)
   - Seismological information (historical or instrumental)
3. Is there any change envisaged in the new study to treat uncertainties?
4. What particular comments of McGuire are being considered in the new study?
5. What particular comments of AGSO are being considered in the new study?
6. What are the main differences in the results obtained so far between the two studies?
7. Does the company have a QA Programme and was this implemented for the seismic hazard studies for Lucas Heights?
8. When will the new study be finalised?

Question Set No.1, Working Group 2a

6.2 Primary Cooling System

1. How are tritium levels measured in the PCS?
2. Piping embedded in concrete cannot be monitored for leaks?

6.6.4.1 Reflector Primary Cooling System

1. How would a small leak from the Reactor Pool into the Reflector Tank be detected?
2. Page 6.6-2 Stainless steel pipe embedded in concrete is covered with polyurethane rigid foam to allow for thermal expansion… What about leak detection? Any method of tracing leaks for particular pipe section?
3. How is maintenance performed on the canned centrifugal pumps? Is heavy water containment system opened? Expected worker tritium exposure?
4. How long will it take one peristaltic pump to re-fill the system after an SSS trip?
5. What reliability and maintenance experience is there for peristaltic pumps? Expected worker tritium exposure?
6. Is there an oxygen deficiency hazard associated with working in the Heavy Water Room? If so, what measures are taken to protect workers?
7. Is it really intended to allow tritium levels to build to 0.37 TBq/kg before changing out heavy water?

6.8 Secondary Cooling System

1. Secondary Water Activity Monitor (SAMO) -- Are there any plans to monitor secondary water for tritium?
2. Secondary Coolant Treatment System: -- What provisions are there to protect against spill of treatment chemicals? To provide containment for plastic tanks?
3. What plans are there for human surveillance of secondary water quality? (PSAR emphasis is on automated operation.)

4.5.1.5.2.5 Reactor Pool Leak Detection (also Service Pool Leak Detection)

1. Have the radiation resistance characteristics of the geo-membrane film and geo-textile bands been determined yet? What alternatives are there if these materials are not suitable?
2. Is there any possibility of false alarms due to condensation of moisture on the exterior surface of the pool container?

Question Set No.1, Working Group 2b

Chapter 5, Questions

1. Section 5.2.4.11 addresses access to reactor components for operations and maintenance. Section 5.2.5 states that inspection and testing of the reactor will be performed. What degree of testing, what frequency, etc. is anticipated and how will this be accomplished? (E.g. use of

remote devices, under full depth of water, with minimum amount of water for shielding, with vessel drained and temporary shielding.) How frequently will this be required, what will be inspected (welds, high stress areas, wear), what degree of disassembly is anticipated, and how long will this require? Are there anticipated changes that will require routine or one-time disassembly? (E.g. installation of the hot neutron source, installing new experimental rigs, adjusting flow to existing experimental rigs, replacement of components which are approaching an exposure limit) Should surveillance coupons be installed that can be removed periodically to track the accumulated fluence on critical components? Have flow-induced vibrations among core and reflector components been considered?

2. Section 5.2.6.5, page 5.2-15, reports a stress calculation for the reflector tank to be 63% of yield. What is considered a reasonable safety factor to use in light of the complex structure being annualized, residual stresses and heat effected zone property changes from welding the reflector assembly, the interaction of combined loads, the uncertainty in calculations, etc.?

Chapter 6, Questions

1. Page 6.2-2 and Figure 6.2/1 -1 suggest that the primary coolant line from the reactor to the decay tank is embedded in concrete. If this is correct, how is corrosion controlled between the concrete and piping? What provisions are made for inspection of the piping? What are the seismic implications?

2. Page 6.2-3, paragraph 7, states that the PCS flow diverted for chimney down-flow is controlled by a valve adjusted at commissioning. What determines the optimum down-flow and does this change over the life of the reactor? What assurance is there that the valve position remains the same for the life of the facility?

Chapter 9, Questions

1. Section 9.3.3, paragraph 3, states that the lines from the fuel tanks to the EDGs run in accessible trenches (except for conduits under roadways) so that they can be inspected. Should a fuel leak occur, where can the leakage travel in the trench and what additional areas could be contaminated and subjected to a fire hazard as a result of fuel intrusion? Will there be provisions to periodically empty the tanks or test the fuel to verify its quality, considering the fact that it may be a number of years before being consumed?

2. Section 9.3.7 addresses many issues in general terms in a half page. E.g. cables will be protected from radiation, low-smoke, halogen-free, fire-rated etc. Likewise in Section 9.3.8 it discusses earthing, grounding, and lightning in a similar manner. Are these subjects addressed elsewhere saying how these measures will be implemented and criteria used in specifying materials (e.g. low-smoke) and installation (e.g. radiation shielding.) Where is cathodic protection of underground equipment addressed?

3. Section 9.3.8.3 discusses the mesh of rebar and later in the same paragraph the metal structure provided for aircraft impact. It then talks of a Faraday cage effect that minimises the risk of internal side-flashing. Is the latter comment referencing the rebar, the exterior metal structure, or both?

4. Section 9.4.3.2 (last sentence) addresses hot re-strike features for discharge luminaries. What is hot re-strike and what is the significance of this statement?

5. Figure 9.3/1-1 shows that on loss of offsite power the cold neutron source has no electrical power supply. Is this assumed in the design of the CNS and analyzed in the LOOP event in Chapter 16?

6. In Section 9.7 the design numbers appear to be very close to the anticipated loads for breakers and transformers. Should there be more margin for error or for later additions of equipment that would increase the electrical load**?**

Question Set No.1, Working Group 3a

Chapter 2

1. How will be assembled the reflector vessel and the structures of the core? What type of control or test will be performed for the welds?
2. Taking into account the design of the reactor core and associated reflector vessel , how will be performed the regular inspection, maintenance and eventual reparation works for these structures?
3. P.2.4-3 and 2.4-4: What kind of validation was done concerning the thermal-hydraulic calculations?
4. How will be limited the rate of insertion of reactivity?
5. P.2.4.9.12: Where will be implemented the Free Field Strong Motion Accelerometers ?
6. P.4-27: Where will be stored the contaminated water of the reactor pool in case of reparation work in this pool ?
7. Table 2.5/1: Why the chosen safety category for safety functions H and K is 2 instead of 1?

Chapter 4

1. Does the verification of the pool resistance to earthquake take into account all the penetrations to this structure (such as neutron beam tubes) ?
2. The description of the membrane used at the exterior of the reactor pool (para. 4.5.5.2.5) seems to be missing.
3. Is it Possible to repare or replace the decay tank in case of leakage (difficulty of access) ?
4. How the surveillance and eventual reparation work could be ensured for the pipes embedded in the concrete ?
5. P. 4. 5.2 : It is indicated that the safety of the stored spent fuel is ensured during normal operation and after any accident situation . Does this cover the case of earthquake ?
6. P.4.5.8 (para. 4.5.1.5.2.5.) : drawings should be added for the reactor pool leak detection . The ageing of Geo-membrane film and Geo textile band should be assessed. Is it possible to replace them in case of significant degradation?

Chapter 17 – OLC

1. Table 17. 2/2-1: The parameter " Core power (Kw)" is considered as a safety limit for low power mode with natural circulation cooling. How do you determine and check the value of this parameter periodically?
2. Table 17.4/1-2: " Limits on the shutdown margin during reactor shutdown " It should be indicated : " If the control rod having the most important worth does not insert " instead of " if one control rod does not insert " .

Chapter 18 – QA

This chapter should indicate :

- the provisions for ensuring the good quality of the various studies relating to the design of the reactor,
- the procedure for approving the corrective actions in case of non-conformances during the construction of the reactor.

It should also indicate the principal "holding points" for the realisation of the project.

<u>Question Set No.1, Working Group 3b</u>

GENERAL

1. FSS operation

    1. Why is it necessary to accelerate the fall of the CRs during a scram by using compressed air?
    2. What happens if the air supply is not available?
    3. What is the difference in drop time with and without the compressed air?
    4. What is the difference in drop time with and without primary flow?
    5. If the shutdown of the reactor by the FSS is adequate without the air supply, why the additional expense and complexity?

2. Low power mode

How is core power (400 kW max) determined and maintained under natural flow conditions, considering that the nuclear instrumentation reference to the thermal power may have been lost due to:

- a core reload,
- maintenance work on the instruments?

SPECIFIC

1. Section 16.8

1.1 The entire section is missing from the Table of Contents.

1.2 Cold Water Injection.

Consider a scenario where the reactor is operating at full power (with two primary pumps operating) and a developing problem is detected with one of the pumps (vibration, say). The operator decides that he will change over to the standby pump without interrupting reactor operation. He intends to bring the standby pump on line and, once the flow is fully established, switch off the affected pump. He can do this without violating any of the operating limits (??).

However, in doing this he introduces a cold slug into the primary system from the standby pump, lines and heat exchanger. This situation should be analysed.

1.3 Section 16.8.4.1 (d)

It is stated that the reactivity insertion (due to start-up of the primary pumps during low power operation) is "Low and bounded by the CR withdrawal".

- Where is this demonstrated?
- What is the lowest possible temperature of the dormant water in the PCS?
- How much is available (ie, how long can the reactivity insertion be sustained)?
- What is the reactivity worth of the temperature difference?

2. Section 16.8.7.3.3 (6^{th} and 7^{th} paragraphs)

With a clad temperature 17 deg C above the temperature for ONB, further quantitative justification is needed for the statement "The presence of nucleate boiling does not imply any damage to the clad. . ", with consideration for the following:

- Channel instabilities occur very rapidly after ONB,
- The presence of voids in the hot channel will have less of an effect on the overall reactivity feedback than they will have on the heat transfer in the hot channel,
- Do codes such as PARET and RETRAN adequately predict the behaviour of thin rectangular channels in this regime?

3. Section 16.9.2.2.3 Core blockage

You could add to the list of possible causes of fuel channel blockage:

d. Mechanical damage to fuel plates due to

- Foreign object dropping on exposed fuel assembly,
- Mishap (or operator negligence) during fuel handling.

Also

Administrative procedures (such as visual inspection of fuel assemblies before loading) do not make good arguments for “rendering a channel blockage beyond design basis. . .”

Experience has shown that damage to fuel elements by the operators themselves occurs with sufficiently high frequency to give it serious consideration.

I agree that complete blockage of a fuel assembly (or a significant part of the core) is not feasible, but even a partial blockage of several adjacent channels can lead to fuel clad failure.

4. Section 16.11

Consider a guillotine break of the primary cold leg line, somewhere between the primary pump discharge and the point where it enters the reactor pool structure concrete (which includes the heat exchanger).

- This could lead to air being drawn directly into the core, since the pumps can continue operating, drawing from the substantial volume of the decay tank. Probably a loss of core delta-P would scram the reactor (needs to be checked), but if it doesn’t, the air in the core will be rapidly replaced by water down the chimney as soon as the primary pumps draw air from the decay tank and stop delivering.
- If, however, the natural circulation valve in the affected line opens (due to a drop in the PCS pressure following the break), the air may not reach the core, but cooler water from the pool will then enter the core via the valve.

Such a possibility needs to be looked at a bit more closely.

5. Section 16.13

What is the intended procedure following an unscheduled reactor shutdown (eg, due to a loss of offsite power) and the inability to restart due to Xe-135 poisoning?

- Wait about 2 days for Xe-135 decay?
- Immediately reload the reactor with Xe-free fuel?

Considering that there will be a strong commercial aspect to reactor operation, extended unscheduled outages may not be contracturally tolerated.

If this is the case and the second option is intended, this section does not allow for it.

<u>Question Set No.1, Working Group 4</u>

Chapter 8, Instrumentation and Control

1. What is the major advantage of using a computerised (digital) system for the first shut down system.
2. When using the computerised system, how is it assured that a systematic error isn’t compiled into all three independent trains and computers?

3. Why are the sensors for the Reactor Protection System (RPS) also used for other purposes? (ex. in the Post Accident Monitoring System, Section 8.6)
4. How critical is it if there is no compressed air for the pressure vessel serving the First Shutdown System? (8.2-17)
5. How is it assured that the computer based system can be maintained in 40 years? (spare parts, skilled people) (8.2-3 /29.)
6. Is it correctly understood that the containment is only closed automatically when the sensors in the stack indicate high radiation level? (8.2/19-1) And the reactor might not be tripped.
7. The RPS is based on 2 out of 3 logic. Has 2 out of 4 logic been considered?
8. How can the operator distinguish between the initiating parameter and the follow up when a trip is performed. (8.2-38 h)

Question Set No.2, Working Group 1

Page 1-5, (Section 1.2)

Is this report considered to be the comprehensive PSA document in its present form?

GENERAL

Why is level 2 PSA not considered explicitly?

Why was fire/flood PSA not considered?

Page 3-18, (Section 3.2.4)

On what basis is the probability of a shell hitting the Reactor Building calculated to be less than 10^{-7} ?

Page 3-51, (Section 3.4.2.1)

What are the assumptions behind decreasing the hazard curve uniformly (ie., for all return periods?)

Page 3-34, (Section 3.4.6)

Is the UPS needed for the SSS to function?

Page 3-51, (Section 3.4.21)

It is normal practice to include uncertainties (see eg Agency documents). Why were they ignored in the seismic PSA?

Page 4-10, (Section 4.2.6)

Why is the 1 sec. Drop time not needed for the seismic trip?

Page 4-43, (Section 4.6)

Please clarify.

Page 5-13, Section 5.2.4.1)

For Event B2 (when seismically initiated), what are the fragilities assumed for all the items (active/passive) in the CD path?

Section 6.2

It is normal practice in seismic PSA to take the whole hazard curve (independent of the design value) into account. Why is this considered a conservatism?

GENERAL

Have you compared the CDF (for seismic event) to those of nuclear installations in regions of similar seismicity worldwide?

Have you compared the ratio of seismic CDF to the overall CDF value to those of nuclear installations in regions of similar seismicity worldwide?

Question Set No.2, Working Group 2a

Chapter 13, Conduct of Operations

1. How are environmental protection concerns addressed?
2. Are there any formal requirements for job safety analysis or hazard services before proceeding with any job?
3. Are requirements for personal protective equipment identified as part of each procedure?

Chapter 15, Commissioning

1. How is environmental protection factored into the commissioning process?
2. Is there a requirement for an Environmental Monitoring Program to be in place?
3. What are the “Pre-Operational Tests” listed by different number in Part 2 of the Table in Section 15.3.1.5.2? (needs more description)

Chapter 19, Decommissioning

1. Are there any design features that might facilitate dismantling of the Reactor and Service Pools? How is this envisaged to be accomplished?
2. Why isn’t Co-60 mentioned as an impurity in Section 19.8.1.1.2.1? (for the fuel element aluminium, it is listed at 60 ppm)
3. In Section 19.8.1.2.2 it is stated that “No materials containing Co in their formulation will be used”. This will rule out, for example, Hastalloy wear rings in pumps. Considering the amount of Co already present as impurities, is this really necessary?

<u>Question Set No.2, Working Group 2b</u>

Chapter 10

1. **Section 10.1.6.1** says the building is designed from spent fuel cask loading and off site shipment. What is the crane capacity? What shipping cask weight is assumed? What floor loadings are assumed where a cask might be set down? What crane braking provisions assure a soft set down? What is the potential for a brake failure?

2. **Section 10.2.5** does not include the cooling tower as a fire zone. To what extent do other analyses assume the cooling tower is available as an ultimate heat sink?

3. **Section 10.4.2.8.2** describes two air intakes for Emergency Control Centre emergency ventilation, one being from the Neutron Guide Hall in the event outside air is contaminated. What air enters the Guide Hall to provide an ongoing supply of uncontaminated makeup air.

4. **Figure 10.4 / 2-2** shows two parallel trains of hot cell exhaust filters. What shielding is provided for the filter banks? If there is a major hot cell release, can the room be accessed safely until the filters decay? What provisions are made for recovery from such an event?

Chapter 11

5. Is the numbering scheme shown in Section 11.3.9 a matter of good human factors? It appears inconsistent with Page 1.2-14. Beam Assembly #1 feeds TG-1, TG-2 and TG-3. Assembly #4 feeds TG-4. Assembly #5 feeds HB-1 and HB-2. Consider a system like numbering beam assemblies #10 thru #50 with #10 feeding TG-11, TG-12 and TG-13, #50 feeding HB-51 and HB-52 etc.

6. Page 11.4.3 lists limiting powers required to prevent onset of boiling for irradiation capsules. There appears to be a large number of combinations, each one being a potential source of human error; eg: container shape, irradiation location, size of cooling flow orifice, specimen material, specimen mass, time or irradiation, etc. Are there physical limits (eg: square tubes for square samples), limited combinations (eg: a bank of positions with the same flux and orifice size) colour coding, or other means to reduce the potential for error? How close to limiting conditions (heat, internal pressure) are targets irradiated? Is there a design basis accident for a target failure that would bound all target failure events? (eg: misloading a fueled target resulting in a specimen failure and release of fission products).

7. References section 11.4.6. When performing sample transfers pneumatically, is there verification when a transfer is properly executed? Is it possible to over-irradiate a sample due to a single equipment failure and a single human error?

8. Are beam shutters discussed in section 11.5 high maintenance items? What provisions are made in the design to reduce exposure when repairing or servicing them?

Chapter 5

9. **Section 5.8.12.1** shows the model used for heat transfer calculations of a fission product moly target but the shape is different. Why is this?

Chapter 18

10. What QA practices are in place at this time? For the calculations referenced in the previous question, do calc sheets exist that are signed by the originaters and reviewed by an independent reviewer? Are all parties/sub-contractors doing this and placing them in archives?

<u>Question Set No.2, Working Group 3a</u>

Chapter 20, Emergency Planning and Preparedness

Para 20.12: For emergency planning it is advisable to consider the design basis accidents considered in the Safety Analysis Report, cumulated with some degradation of the filtration system efficiency?

Para 20.2.3 – Figure 20.2/8: It should be justified that the evacuation routes in case of radiological accident are outside the zone concerned by the plume.

Para 20.3.2: The range of measurement associated with emergency equipment should be consistent with the accident conditions.

Appendix A (PSA)

- Many uncertainties could be associated to the data used in the calculation. However, the total core damage frequency taking into account the contribution from internal and external events is very low (3.7 x 10^{-7}/year to be compared with 10^{-5}/year adopted for NPPs).
- This study will be useful to determine the effect of some modifications on the core damage frequency. It should be updated in order to take into account the evolution of the project.

<u>Question Set No. 2, Working Group 3b</u>

Chapter 16

16.14: "*There are no high energy piping systems in the facility …*" Is this true? There are surely compressed air, liquid N_2, liquid CO_2, liquid He etc pipes (sometimes inflexible pipes?). In addition, the primary piping, with two primary pumps delivering 1900 m^3/h contains a substantial amount of dynamic energy.

16.14.1 last paragraph: I do not understand the statement that no consequence analysis is necessary for internal fires and explosions. There are surely very serious consequences.

- propagation of poisonous smoke and fumes by ventilation systems
- blockage of ventilation filters by smoke particles (to name a few).

16.14.3.2 Loss of compressed air: Are there no actions required to be carried out in mitigation of consequences of accidents from the MCC or ECC that require compressed air?

16.14.3.3 Loss of communications: This should refer not only to communications with the rest of LHSTC, but communications within the reactor building as well, the loss of which could seriously impact the safety of the reactor and of personnel.

In **16.14.3.3** communications with the rest of LHSTC (which presumably includes HIFAR) was written off as having no safety impact on RRR.

16.14.3.4 Loss of lighting: The list of places provided with emergency lighting should include escape routes. Loss of lighting here could jeopardise the safety of personnel.

16.14.5 Improper access: I imagine that access to the decay tank room has additional control, similar to the heavy water room and the CRD room.

16.15.2.3 Rigs Exchange: The first sentence is badly phrased. It gives the impression that such mistakes <u>can</u> occur. The subsequent argument, however, states that such a possibility is designed out. Try "The potential for incorrect positioning of ….. rigs due to human error has been addressed in this design".

16.15.2.4 Staff irradiation: Why is the FRPS concerned with target handling outside the reactor? Shutting down the reactor does not remove the problem.

16.15.3.1 Excessive target activity: Add to the list

d) Targets containing materials other than indicated on irradiation requests (either as an impurity or due to mistakes during packaging of the targets).

16.15.3.2 Excessive target heating power (→ 16.15.2): A similar discussion of can failure due to excessive heating and/or inadequate heat transfer <u>within</u> the can in those positions cooled by water should be given. Spillage of the contents of such a can (especially if in powdered form) into the RSPCS and consequently into the pool water can lead to significant contamination problems.

16.15.3.5 Can failure inside Pneumatic System:

- There is some repetition here of 16.15.3.2
- Failure of both cans in a double can arrangement is <u>not</u> incredible. It can arise from:
  - Mistakes in the calculation of the volumetric expansion of the target material with temperature;
  - Mistakes in the estimation of gas generation;
  - Faulty can sealing equipment affecting both cans etc.

16.15.3.6 Can Rupture in the Hot Cell: A much sought after isotope is ^{125}I which is obtained by irradiating ^{124}Xe (*I think*). Xe is not retained in the hot cell filters and will be released up the ventilation stack. In other words, it is not bounded by the iodine event.

16.15.4.1 Failure of Ventilation System: What if ventilation fails during a can opening or target processing procedure? (Murphy's law!)

16.15.4.2 Failure of Cell Elevator: Are the "wagon" and the "carrier" two separate pieces of equipment?

16.16.1 Spurious Trigger of FSS (Also 16.16.2 –SSS): Add to the list of possible causes, "An inadvertent action by an operator in the control room".

What will be the rules governing operation of the reactor with one FRPS channel shut down (for test, maintenance or due to an instrument fault)? FSS protection would then in a ½ configuration. A spurious fluctuation in only one trigger signal will then lead to FSS actuation.

16.17.5.2.6 Accidents in HIFAR: This discussion refers to communications systems between HIFAR and RRR which will allow early notification to RRR of any need to evacuate following an accident at HIFAR.

In 16.14.3.3, communications with the rest of LHSTC (which presumably includes HIFAR) was written off as having no safety impact on RRR.

16.18 Human error: One of the most important factors minimising the impact of human failure is not mentioned, namely automatic control and protection of the reactor and its facilities.

Chapter 17

General Comments:

1. A Safety Limit on total neutron fluence for the most important core components (core grid, reflector tank walls, core box).
2. A Safety Limit on fuel and absorber plate burn-up.
3. A Safety Limit on coolant flow velocity in the fuel assemblies.
4. The document is devoid of any quantities/values/settings for SL, SSS and LOC. Perhaps it is premature to supply these values in the PSAR (although they were used extensively in the safety analysis), but they should appear in the FSAR as the precursor to the OTS.
5. Chapter 17 should include a section on the procedure to be followed (recording and reporting) should an OLC (especially SL and SSS) be violated.

Question Set No.2 Working Group 4

Chapter 19, Decommissioning

1. Has it been considered to take samples of the material that is foreseen to be in a neutron flux?
2. Has it been considered that there might be tritium in some materials?
3. How is the heavy water to be deposited?

Chapter 5, Reactivity Worth of the Reflector, Sections 5.5.4.9 and 5.7.6

The reflector vessel has a complicated geometry. The model for the vessel is then to be used in a code, which is calculating the reactivity worth of the reflector. Under what circumstances are these calculations made? What is the reliability and sensitivity of this calculation?

APPENDIX V - WRITTEN COUNTERPART VIEWS

In response to the review team questions in Appendix IV, ANSTO/INVAP chose to submit additional information in writing on Issue Number 2 (Seismic Hazard), 3 (External Events) and 21 (Quality Assurance). These written responses constitute Appendix V.

ISSUE NUMBER: 2

1. REVIEW AREA: SAR Section 3.2.6.

2. ISSUE TITLE: Seismic Hazard

Counterpart Response

Background

The choice of a design value for the Reactor Facility was based on the numerous studies that have been done for the Lucas Heights region over many years. It was not just a response to IGNS but was based on the knowledge that the LHSTC is located on a sandstone plateau in the Sydney Basin, the current earthquake hazard map of south-eastern Australia (SAA, 1993), shows the Sydney Basin to lie in a low intensity seismic zone. While there are a number of geological features in the Sydney Basin indicative of past earthquake activity, no seismically active geological structures have been identified, and there are no major faults within 35 km of LHSTC.

The seismic hazard at Lucas Heights has been assessed by various authors and organisations including Mumme (1976), the Bureau of Mineral Resources (1989, now the Australian Geological Survey Organisation or AGSO), and the Department of Transport and Construction (1982). In 1995, Corran reviewed and compiled much of this information into a single report and proposed a seismic hazard curve (best estimate and upper bound). The SL-2 or SSE for the HIFAR reactor had been previously determined in consultation with the then Nuclear Safety Bureau (now ARPANSA) as 0.20 g PHGA, although HIFAR Engineering have in recent times worked to a level of 0.23 g. This was consistent with the seismic hazard assessed by Corran as 0.17 g with an uncertainty of ±0.06 g.

In conjunction with this, Sommerville (1989) had assessed a wide range of possible spectra and concluded that the Carbon Canyon value was the most appropriate for Lucas Heights. More recent advice from AGSO has confirmed the appropriateness of this choice. Max Irvine's analysis for the PSAR was also taken into account in justifying this position.

When Pickard, Lowe and Garrick (PLG, 1998) undertook the independent PSA of HIFAR in 1998, they commented that the seismic hazard curve quoted in Corran (1995) had a low uncertainty band. As a result of this they recommended that ANSTO have the seismic hazard, and in particular, its associated uncertainty, reassessed. For the purpose of the PSA, PLG broadened the standard deviation. This resulted in an increase in the mean value to about 0.24 g and the 85% value to 0.34 g.

Following the recommendation of PLG, the Department of Industry, Science and Resources conducted a PHSA independent of ANSTO. This was done by IGNS and arrived at a recommendation of 0.41g and a different spectrum. ANSTO was sufficiently concerned at the conservatism of this study to commission two reviews, one by Robin McGuire of the US and one by AGSO. Both studies (attached) confirmed our concern over the conservatism in the IGNS results.

Controlling Earthquakes

For design purposes, the design earthquake, following US guides, is adopted from the analysis of the controlling earthquakes. Controlling earthquakes are determined by disaggregation of hazard curves. The IGNS results showed that a bimodal distribution of magnitude-distance combinations dominated the hazard at Lucas Heights. For all return times and spectral periods, a modal magnitude and distance group is observed at about M5.5-6.5 (the mode increases with increasing spectral period) and 10 km distance, and a second group is observed at large magnitudes (from M7 to 8.1) and distance of 30 to 50 km. These correspond to the East Sydney Basin source zone and the Lapstone Structural Complex. . The question of the Lapstone Complex has already been discussed but this has less impact at short spectral distance and so the recommended controlling earthquake for design purposes is the M5.5-6.5 event at around 10 km. This is very similar to the Michael-Leiba M5.75-6.25 event at 15 km used in previous studies.

Choice of Design Value

ANSTO was concerned in the design specification to choose a value for pga and a spectrum that built on previous studies and which was consistent with Australian information. Hence the decision was taken to raise the previous value of 0.23g with the Carbon Canyon spectrum to 0.3g and Carbon Canyon. This, we believed was based on the best appropriate data and was consistent with the recommendation of PLG. This position was accepted by ARPANSA as a suitable design approach. We are aware that it is necessary to show that the design can withstand the safe shutdown level earthquake and we have shown that it has considerable capacity beyond 0.3g, to a factor of two.

However, given the work of IGNS, the design value was also shown to be at a level that would be close to or bound the extreme value for IGNS, which we believed would certainly decrease with further assessment of the assumptions.

In addition, the comparison of the 0.3g pga with the Carbon Canyon spectrum with the disaggregated East Sydney controlling earthquake of 0.33g, as in Figure 1 below, shows that the design value effectively bounds this recommended value from IGNS.

In addition, the comparison with a reduced IGNS recommendation at 0.3g also shows that the design earthquake bounds this curve over all spectral values.

In addition preliminary analysis of the 0.3g and Carbon Canyon with the IGNS even at 0.41g shows that the former is more challenging.

Conclusion on SSE

1. The design seismic value for the Reactor Facility is based on the best appropriate information on Eastern Australian earthquake studies. This was accepted by ARPANSA as a suitable basis for design.

2. The choice effectively bounds the disaggregated IGNS results for the controlling earthquake.
3. It also is expected to fully bound the revised IGNS values. A scaled 0.3g pga and spectrum compared to the design value is also shown in Figure 2.

Site Investigation

We believe that we do know the area well after many years of study and that a further assessment would have no new data to work with. The only recommendation from IGNS was for a near field assessment to confirm that there were no local features that could amplify a specific event on this site. That was done and is in the Coffey report.

Operating Basis Earthquake

We agree with the comment on the scaling. The process followed from our choice of the US NRC 1.6 approach. However we have detailed analysis by Max Irvine that confirms that the choice is conservative for the OBE.

Figure 1 – Comparison between design value and disaggregated IGNS value.

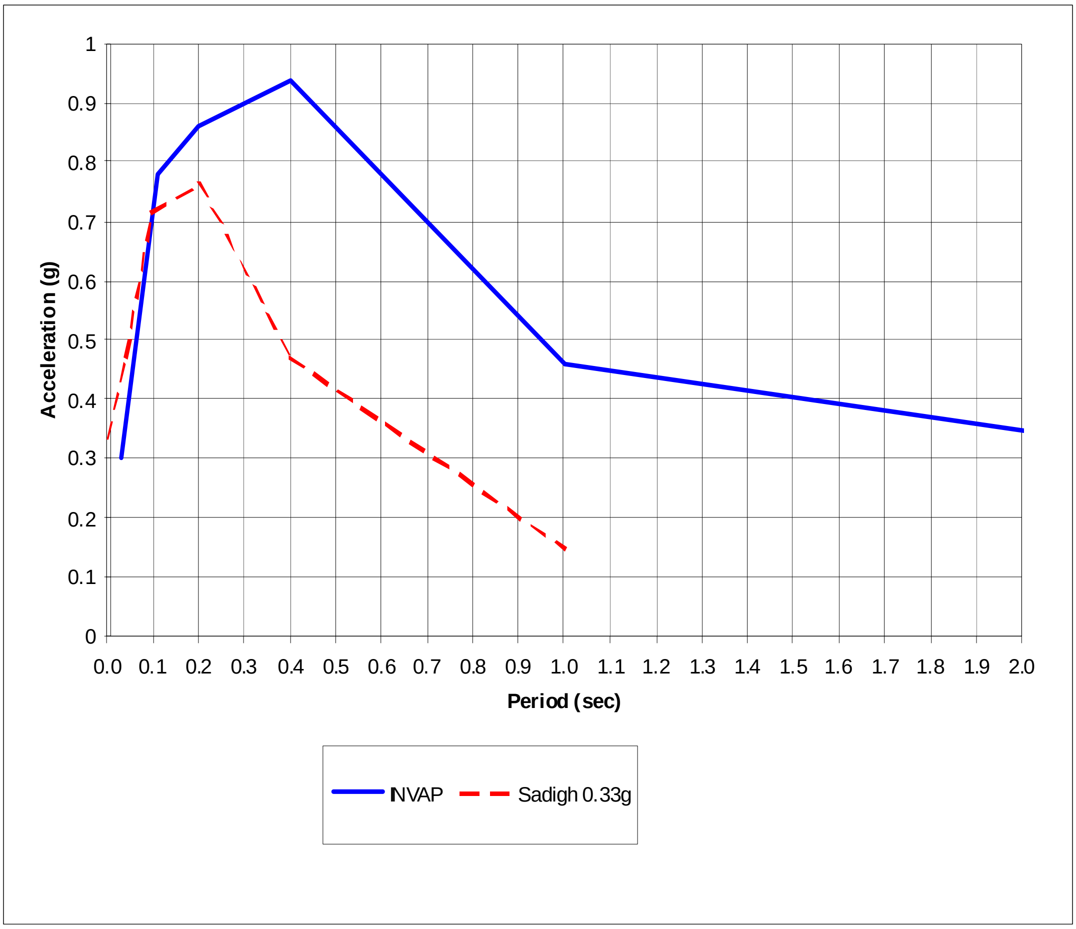
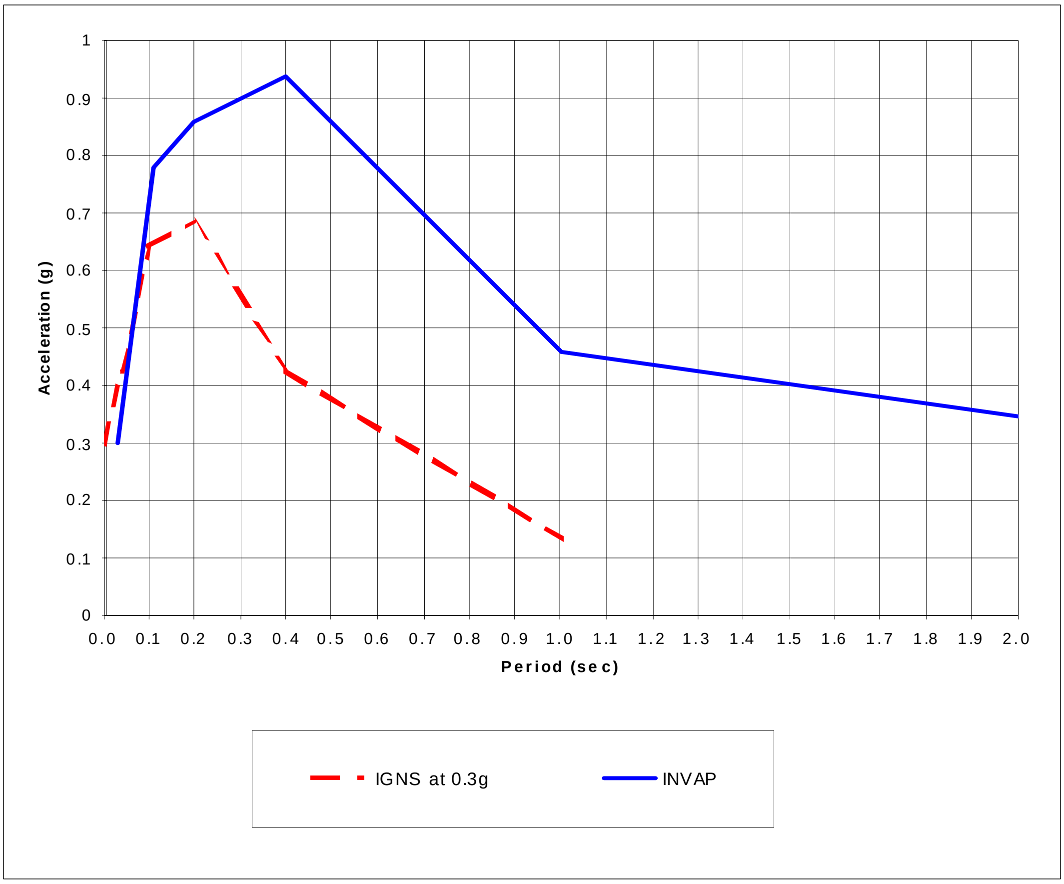


Figure 2 – Comparison between design value and reduced IGNS value.



ISSUE NUMBER: 3

1. Review Area: PSAR Section 4.3.3

1. Issue Title: External Hazards (other than seismic)

Counterpart Response

The screening of external events was extensively addressed in the Application for a Facility Licence, Site Authorisation, in the EIS and in the HIFAR PSA. These documents are attached. The conclusions on these analyses were submitted for public and peer review comment and were extensively reviewed by ARPANSA as part of their evaluation of the site licence application. The peer reviewers were CH2M Hill (Aust), Parkman (UK) and the IAEA. In addition, ANSTO used DNV (Technica) to review the impacts of industrial accidents, including chemical explosions. The conclusions of these studies were:

1. there are no current likely explosive or chemical hazards associated with the facility at Lucas Heights that have potential significant blast impact on the facility.
2. the type of road traffic does not pose a significant hazard.
3. the rail hazards do not present a significant hazard to the facility.
4. the likelihood of aircraft crash greater than that assumed or military strike is very low, as estimated by both ANSTO and independently by PLG.
5. the major issues to be considered in design are light aircraft crash and bushfires.

On these bases, ARPANSA issued a site licence to ANSTO.

Nevertheless, the structure has significant strength against high wind, tornadoes and aircraft crash that does provide blast resistance. In addition, ANSTO controls the nearby land uses and will not permit development that could threaten the facility.

ISSUE NUMBER: 21

1. Review Area: PSAR Chapter 18

2. Issue Title: QA on Design Calculations

Counterpart Response

Replacement Research Reactor Project

Preliminary Safety Analysis Report

Peer Group Review

Identification of Issues – 1 June 01

11. QA on Design Calculations

11.1 Environment for the conduct of design related activities

Contract No. CC 193338 between ANSTO (Principal) and INVAP (Contractor), required INVAP to:

(1) provide evidence of an existing, certified quality management system, and

(2) implement a specific Project Quality Assurance Program (PQAP) for the duration of the Contract, at which time ANSTO would operate the new facility in accordance with the Facility Quality Management System developed under contract by INVAP in close cooperation with ANSTO, taking into account the existing engineering and operations quality management systems applying to HIFAR.

INVAP's design based operations are undertaken in accordance with a quality management system certified to ISO 9001:1994. Whilst the certification covers both its nuclear and space program related activities (in conjunction with NASA), the certification for the nuclear related activities covers the design, construction and commissioning of nuclear research reactors, low power nuclear reactors and auxiliary nuclear facilities. The Contract PQAP operates within this environment.

11.2 Provisions for ensuring good technical quality of the various studies relating to the design of the reactor

The provisions include the following:

(a) INVAP's project quality management system, and that of its principal subcontractor, John Holland Evans Deakin Joint Venture (JHEDI) and its consultants including Connell Wagner, are all certified and are subject to regular internal audits, external surveillance audits and re-assessment audits in accordance with best international practice. In addition, the Contract provides for ANSTO to undertake its own annual (or more frequently if it feels necessary), surveillance audits of INVAP's Contract activities, particularly those related to the project phase currently under way at the time of the audit. The first of these audits was undertaken in February this year by Det Norske Veritas concentrating on design process control and design records. INVAP passed the audit with one non-conformance noted and a small number of observations. ANSTO formally required evidence under its ISO 9001 certified project quality management system that action had been taken in regard to each item noted in the auditor's report; the evidence was provided by INVAP, and all documentation relating to the audit and its outcome has been provided to ARPANSA.

(b) INVAP determined prior to contract, that they would benefit from the introduction of internationally recognised specialist consultants into their team to add value and quality to the various design studies and subsequent design deliverables provided to ANSTO. These consultants are identified in the PSAR. It is important to note that the consultants are working as part of an integrated INVAP design team, eg. they have been present at every Preliminary Design Review (PDR) and Critical Design Review (CDR) attended by ANSTO in their particular fields of interest – see below.

(c) all design activities and functions are being undertaken by INVAP in accordance with the Contract Design Plan agreed with ANSTO prior to contract. The design process embedded in the Design Plan is fully transparent to ANSTO, and is based on the tried and proven US military standard (Mil. Std.) system for the planned and orderly conduct of hardware and software based design programs. The program has included:

   (i) the conduct of a Systems Requirements Review (SRR) by INVAP immediately following contract award in July 01,

   (ii) the conduct of Preliminary Design Reviews (PDR) by INVAP at notionally 70% completion of the following design activity groupings:

   - reactor systems,
   - reactor processes,
   - instrumentation & control,
   - neutron sources,
   - irradiation facilities, and
   - balance of plant.

   (iii) the conduct of Critical Design Reviews (CDR) by INVAP at 100% practical completion of the above mentioned design activities.

Each of these activities (I), (ii) & (iii), has been preceded with the provision of design documentation by INVAP for familiarisation and review before the design meeting. ANSTO has been represented on each occasion on a full participatory basis, including by its staff involved in the contracted technology transfer program where ANSTO works within INVAP's design team's under its supervision. The INVAP and JHEDI consultants have also been present for the duration of the relevant meetings. The meetings have typically extended over a week and all outcomes have been documented and issued to each participating organisation. The meetings have concentrated on INVAP providing a demonstration of compliance with the Contract Specification in all respects.

(a) the Contract requires ANSTO to review, verify and accept each design deliverable prepared by INVAP under the Design Plan. This includes ANSTO undertaking its own studies to ensure that it has verified to its satisfaction that the details provided by INVAP are soundly based. Examples of ANSTO's direct involvement in studies to verify the technical quality of INVAP's activities include:

   (i) neutronics modelling and calculations
   (ii) thermo-hydraulics calculations (including technology transfer)
   (iii) seismic modelling and calculations
   (iv) energy management system and containment design and functionality
   (v) fuel technology (technology transfer)
   (vi) instrumentation and control (technology transfer)

All of this activity has been undertaken to date, and will continue to be undertaken by ANSTO staff because they have demonstrated that they have the necessary skills to perform the verification functions properly. This does not preclude ANSTO using any other resource that it sees fit at any stage of the design process.

In relation to INVAP's responsibilities, the Contract says:

<u>Clause 3.3: Contractor's Warranties</u>

"The Contractor warrants that -------- the design will be fit for its intended purpose and in accordance with the standards specified in the Contract -----"

<u>Clause 3.4: Warranties Unaffected</u>

"The Contractor agrees that the warranties given --- shall remain unaffected and that it shall bear and continue to bear full liability and responsibility for the design and construction of the Works notwithstanding ---- b) any comment or direction upon, review or acceptance of, consent to proceed with, or request to vary the Contract Design Documents (or any part thereof) by the (ANSTO) Project Manager".

In summary, all of these measures ensure the good technical quality of the various studies relating to the design of the reactor. In this regard, ANSTO will not accept any Contract Design Deliverable, including design studies, from INVAP, and subsequent work will not proceed, until such time as ANSTO is satisfied through its own verification activities that the design is both safe and fit for the intended purpose.

11.3 Procedure for approving the corrective actions in case of non-conformance during the construction of the reactor

The procedures are referenced in the Construction Inspection & Test Plan (CITP) Sec. 8 – Non-Conformance Control.

A copy of the CITP is provided.

11.4 Main hold points relating to the project

The project is broken down into a number of discrete Work Packages (WP) which define the total scope of the project at a relatively high level of the work breakdown structure. This description is the basis of the Contract Master Schedule (CMS) and it effectively integrates contract cost, schedule and performance throughout the period of the contract.

The manner of the development and implementation of the project construction inspection & test program is defined in the CITP at the level where individual WP procedures and instructions will be developed. These WP procedures and instructions will be provided to ARPANSA during their development for review and comment as to their adequacy, and to enable ARPANSA to identify its requirements in terms of hold and witness points which in turn will be incorporated into the CMS.

In a global sense however in terms of the life cycle of the development of a structure, system or component from design to operation, the following hold points are not uncommon:

- successful conduct of a systems requirements review
- successful conduct of a preliminary design review
- successful conduct of a critical design review
- identification of the configured status of the design including the configured items (CI), following the conduct of the critical design review
- approval of procurement specifications
- approval of inspection & test procedures for sourcing materials, manufacture, PHS&T, receipt into store or at site, installation, installed equipment prior to cold testing, cold testing and commissioning throughout the various stages, and performance demonstration testing, all including hold & witness points required by the Principal and the Contractor as well as those required by the regulatory authority.

Attachments:

Construction Inspection & Test Plan [RRRP-7033-EDEJH-001-C]

Contract Master Schedule – Rev 003

gjs 2/6/01

APPENDIX VI - MISCELLANEOUS COMMENTS

In the course of the review the team members identified information that would benefit those who will be revising the current version of the PSAR. These matters are included below:

- **Chapter 4 (Para.4.5.1.5.2.5):** Drawings for the reactor pool leak detection system should be included . An assessment should be made concerning the aging of Geo-membrane and Geo textile. This assessment, which should take into account the effect of irradiation, is very important because it will be difficult to replace the membranes in case of deterioration.

- **Chapter 20 (Para. 20.3.2):** It should be checked that the range of measurement associated with emergency equipment is adapted to the considered accident conditions.

- **Chapter 16:** As a general remark, many events (such as core blockage, failure of the vacuum vessel of the cold neutron source) are considered to be unlikely and beyond design basis, without sufficient justifications.

- **Chapter 16 (Para.16.12, item j):** The indication " any failure of the reflector vessel …would lead solely to isotopic degradation of the heavy water" is not strictly correct because there will be in this case contamination by the tritium of the reactor pool water.

- **Chapter 16 (Table 16.19/3-1):– partition fraction in nuclide transportation):** The value of 30% considered for the release of iodine and cesium from the melted fuel to pool water is very low compared to the value of about 80% observed for these fission products during the partial meltdown accident of 6 fuel plates, which occurred in1967 in SILOE reactor.

- **Chapter 16 (Para. 16.8.4.1):** The sub-paragraph numbering follows the sequence a), d), e). Presumably it should be a), b), c).

- **Chapter 8 (Instrumentation and Control):** A digital computer system is to be used in First Reactor Protection System compared to the Second Shutdown System which is hard wired. This implies a wish of introducing modern technology and gives a kind of diversity. The technology of modern digital or computerised control systems is now so advanced that it is also used in the nuclear field. However, especially the testing of the programming is very important and independent test programs based on different philosophy to minimise the likelihood of systematical errors should be used during the whole phase of constructing the protection system. The test phase should be very well documented and carried out by independent groups as mentioned in the PSAR.

When constructing the digital computer system, protection against intervention from unauthorised personnel who can introduce changes in the system should also be considered.

When constructing electronic equipment it is normal practice to assure it is protected against electro magnetic radiation which certainly is very important in this case.

The subjects addressed above are to be carried out with Best Practices in mind, according to ANSTO.

APPENDIX VII - LIST OF ADDITIONAL DOCUMENTS

The following documents were provided to the review team during the course of the review as part of the counterpart responses to questions raised by the team.

AGSO Report "Review – Seismic Hazard Analysis, Lucas Heights."

Coffey Report No. S20251/1-AQ "ANSTO RRR, Lucas Heights Report on Near Site Investigations," 8 October 1999.

INVAP Report RRRP-0170-2BEIN-015-A, Rev. A, "Reactor Pool Openings Structural Integrity," 10 April 2001.

INVAP Report RRRP-0410-2BEIN-010-A, Rev. A, "Reflector Vessel – Structural Analysis," 9 February 2001.

INVAP Report RRRP-0410-2BEIN-036-B, Rev. B, "Surveillance Program," 30 May 2001.

INVAP Report RRRP-0410-2BEIN-037-A, Rev. A, "Neutron Beams and CNS Containment Preliminary Structural Analysis," 5 April 2001.

INVAP Report RRRP-6230-2BEIN-008-A, Rev. A, "CNS Containment Detonation Analysis," 22 February 2001.

INVAP Report RRRP-6230-3BEIN-016-A, Rev. A, "CNS Extreme Energy Release Background," 1 June 2001.

INVAP Report RRRP-7205-EDEIN-004-A, Rev. A, "Methods and Criteria for Structures and Components Seismic Qualification," 26 March 2001.

Meeting Notes, "Review of Parameter suitability for a PSHA at Lucas Heights for ARPANSA," Olim's Canberra, 6-7 November 2000.

Mumme, I. "Review of the ALLIANCE Report – Seismic Hazard Analysis – Lucas heights Site of the High Flux Australian Reactor," (performed for ARPANSA), April/June 2000.

Risk Engineering Report, "Review of Seismic Hazard Analysis Lucas Heights Site of the High Flux Australian Reactor," 26 January 2000.

APPENDIX VIII – LIST OF ACRONYMS

AGSO	Australian Geological Survey Organisation
ALARA	As Low As Reasonably Achievable
ANSTO	Australian Nuclear Science & Technology Organisation
ARPANSA	Australian Nuclear Safety & Radiation Protection Agency
ASME	American Society of Mechanical Engineers
ASNO	Australian Safeguards and Non Proliferation Office
ASTM	American Society For Testing and Materials
CD	Core Damage
CEO	Chief Executive Officer
CNS	Cold Neutron Source
CR	Control Rod
DBIE	Design Basis Initiating Event
EA	Environment Australia
EIS	Environmental Impact Statement
FE	Finite Element (Analysis)
FSAR	Final Safety Analysis Report
FSS	First Shutdown System
FRPS	First Reactor Protection System
HIFAR	High Flux Australian Reactor
IAEA	International Atomic Energy Agency
IGNS	Institute of Geological & Nuclear Sciences
INVAP	(Argentina based company)
JHEDI	John Holland Evans Deakin Industries
LHSTC	Lucas Heights Science & Technology Centre
MW	Megawatt
OBE	Operating Basis Earthquake
ONB	Onset of Nucleate Boiling
PARET	Computer software used for predicting core thermal-hydraulic behaviour
PCS	Primary Cooling System
PGA	Peak Ground Acceleration
PLG	(US based consulting company)
PSA	Probabilistic Safety Assessment
PSAR	Preliminary Safety Analysis Report
PSHA	Probabilistic Seismic Hazard Assessment
QA	Quality Assurance
QAP	Quality Assurance Plan
RETRAN	Computer software for thermal-hydraulic analysis of coolant systems
RRR	Replacement Research Reactor
SPERT	Special Power Excursion Reactor Test
SSC	Systems, Structures, and Components
SSS	Second Shutdown System
USNRC RG	United States Regulatory Commission Regulatory Guideline