Evaluation of Proliferation Resistance of a Nuclear Fuel Fabrication Facility

by

Suresh Gangotra

Enrolment No.: stra01201004001

A thesis submitted to the

Board of Studies in Strategic Studies

In partial fulfilment of requirements

for the Degree of

DOCTOR OF PHILOSOPHY

of

Homi Bhabha National Institute

December, 2015

STATEMENT BY AUTHOR

This dissertation has been submitted in partial fulfilment of requirements for an advanced degree at Homi Bhabha National Institute (HBNI) and is deposited in the Library to be made available to borrowers under rules of the HBNI.

Brief quotations from the dissertation are allowable without special permission, provided that accurate acknowledgment of source is made. Request for permission for extended quotation from or reproduction of this manuscript in whole or in part may be granted by the Competent Authority of HBNI when in his or her judgment the proposed use of the material is in the interests of scholarships. In all other instances, however, permission must be obtained from the author.

(Suresh Gangotra)

DECLARATION

I, hereby declare that the investigation presented in the thesis has been carried out by me. The work is original and has not been submitted earlier as a whole or in part for a degree / diploma at this or any other Institution / University.

(Suresh Gangotra)

CERTIFICATE

I hereby certify that I have read this dissertation prepared under my direction and recommend that it may be accepted as fulfilling the dissertation requirement.

Place: Signature

Date: Name: R.B. Grover

Thesis Supervisor

List of Publications Arising from the Thesis

PEER REVIEWED JOURNALS

1. Gangotra, S., Grover, R. B., Ramakumar, K. L., Kamath, H. S., and Panakkal, J. P., "Safeguards-by-Design (SBD) Concepts for Thorium-Based Fuel Fabrication Facilities" , Journal of Nucl. Mat. Management, **41 (1)**, 43-51 (2012).

2. Gangotra, S., Grover, R. B. and Ramakumar, K. L., "Comparison for thorium fuel cycle facilities of two different capacities for implementation of safeguards", Nuclear Engineering and Design, **262**, 535-543 (2013),

3. Gangotra, S., Grover, R. B., and Ramakumar, K. L., "Analysis of Measures to Enhance Safeguards, and Proliferation Resistance in Thorium Based Fuel Fabrication Plants", Progress in Nuclear Energy, **77**, 20-31 (2014).

CONFERENCE PROCEEDING

1. Gangotra, S., "Emerging Trends in Safeguards and NUMAC Practices", Nuclear Material Accounting and Control: Current Practices and Future Perspectives (NUMAC-PP2013), Mumbai, 101 – 111 (2013).

DEDICATION

This thesis is dedicated to my parents, family, guide, teachers, colleagues and friends who have been a constant source of inspiration.

ACKNOWLEDGMENTS

I would like to express my deep gratitude to my guide, Dr. R.B. Grover, Vice-Chancellor, HBNI, for his continuous encouragement, valuable contribution and critical evaluation & advice at all stages of my work.

I would also like to place on record my sincere appreciation and gratitude to Mohd. Afzal, S. Anantharaman, K.R. Anilkumar, M. Anuradha, U.K. Arora, Arun Kumar, P.G. Behre, K.M. Danny, P.V. Hegde, M.M. Hussain, Kaushal Jha, S.K. Jha, H.S. Kamath, K.B. Khan, P.B. Kharat, P.V. Kumar, Ranajit Kumar, T.R.G. Kutty, Vaibhavi Lad, K.N. Mahule, S. Majumdar, S.K. Malhotra, Sudhir Mishra, R.K. Mittal, Latha Nair, A.K. Nema, S. Padmakumar, J.P. Panakkal, G.J. Prasad, Ravishankar, R.S. Prasad, K.L. Ramakumar, G.V.S. Hemantha Rao, K.C. Sahoo, R.P. Singh, C. Uma Shankar, Garima Sharma, Sheela, V. Shivakumar, S. Vedamoorthy, V. Venugopal and C.S. Viswandham, who have helped me in various capacities to accomplish this academic endeavor.

I am also thankful to my wife Rajshree and daughter, Riddhima for their constant support.

(Suresh Gangotra)

List of Contents

SYNOPSIS

Evaluation of Proliferation Resistance of a Nuclear Fuel Fabrication Facility

Introduction

Electricity generation in India during the fiscal year ending on 31st March 2015 was about 1100 TWh and it is estimated that by the middle of century electricity demand in India will be about 8000 TWh [Grover and Chandra, 2006]. Considering that India is not endowed with large reserves of natural energy resources, all available resources need to be exploited and that includes nuclear. Nuclear power offers a safe, reliable, affordable and environment friendly source of energy. A recent study ranks nuclear energy as number one amongst seven different electricity generating options based on key sustainability and economic-environment indicators [Brook and Bradshaw, 2014]. Nuclear fuel cycle can be once through or closed fuel cycle. Once through fuel cycle consists of mining of uranium, conversion, enrichment (in case of use in light water reactors), fuel fabrication, irradiation in reactor, discharge of spent fuel and permanent storage of spent fuel. In a closed fuel cycle, the spent fuel discharged from the reactor is reprocessed for separation of plutonium, which is then refabricated and used for generation of power either in thermal or fast reactors. India has an ambitious programme to generate electricity using nuclear technology for meeting its growing demand and its policy framework calls for utilizing full energy potential of nuclear fuel by following a closed fuel cycle approach [Grover, 2013]. Considering its large thorium reserves, India has formulated a three stage nuclear programme with the objective of utilizing vast reserves of thorium available in India in the third stage.

Nuclear energy, though a clean and green source of energy, requires responsible governance. Soon after the demonstration of nuclear energy, it was realized that the technology for harnessing atom for power generation can also be exploited to manufacture nuclear weapons. This is well documented in Acheson-Lilienthal authored ***Report on the International Control of Atomic Energy*** published in 1946, which states, "The development of atomic energy for peaceful purposes and the development of atomic energy for bombs are in much of their course interchangeable and interdependent". The report proposes that "A system of inspection superimposed on an otherwise uncontrolled exploitation of atomic energy by national governments will not be adequate safeguard." While the first quote has scientific basis, the second quote is a statement of value judgment by the authors and is debatable. In any case the report and subsequent developments led to the formation of the International Atomic Energy Agency (IAEA). The Statute of the IAEA, which entered into force in July 1957, requires that IAEA safeguards be applied to nuclear plant and material furnished by the IAEA and to other nuclear activities assisted, sponsored, supervised or controlled by the IAEA. [Fischer, 1997]. With the passage of time, the IAEA safeguards system has matured into what could be described as a comprehensive set of internationally approved technical and legal measures, applied by the IAEA to verify that the political undertakings of States to not use nuclear material to manufacture nuclear weapons are being honoured by them.

Proliferation Resistance

Proliferation resistance is a characteristics, which is closely related to safeguards and is generally defined as, "..that characteristic of a nuclear energy system that impedes the diversion or undeclared production of nuclear material or misuse of technology by States in order to acquire nuclear weapons or other nuclear explosive devices."[IAEA, 2002]. Barriers that increase proliferation resistance are characterized as either intrinsic or extrinsic. While intrinsic features are those that are inherent to a particular fuel cycle system and depend on scientific characteristics of fissile material used, extrinsic features are the administratively-added security features such as physical protection and international safeguards [Bari et al., 2004]. The facilities can be developed to enhance the proliferation resistance by incorporating appropriate measures in design and operation. The degree of proliferation resistance thus results from a combination of, inter alia; technical design features, operational modalities, institutional arrangements and safeguards measures.

Implementation of Safeguards and Thorium Fuel Cycle

Nuclear material can be in the form of sealed fuel elements or in bulk and this has implications on safeguards implementation. When nuclear material is in the form of sealed fuel elements, safeguards implementation involves item counting. Implementing safeguards in item counting facilities is comparatively easy. In once through fuel cycle, bulk handling facilities are needed up to the stage of pellet fabrication, while in a closed fuel cycle facility, they are needed even afterwards as the spent fuel has to be reprocessed and made into pellets. Nuclear material in the form of gases, powders or

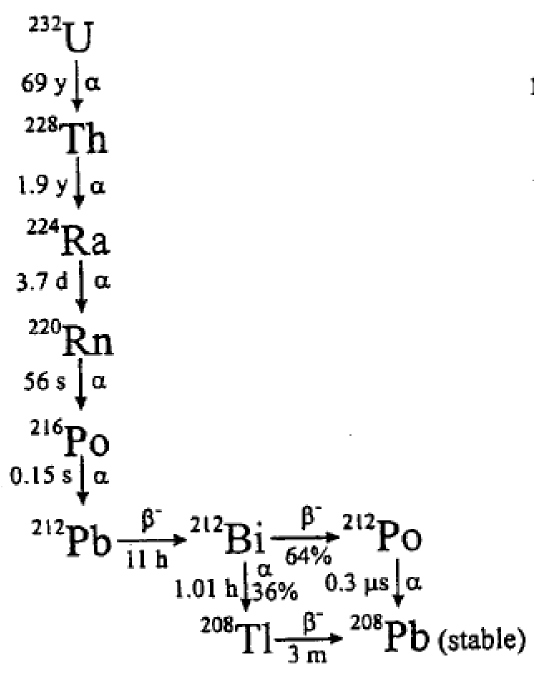


Fig. 1: Decay Chain of U^{232}

solutions is present in a reprocessing plant and in a plutonium fuel fabrication plant. India has large thorium deposits and has an ambitious programme to exploit thorium. This includes development of an Advanced Heavy Water Reactor [Sinha and Kakodkar, 2006]. Thorium based fuels and fuel cycles have intrinsic proliferation-resistance due to the formation of U^{232} due to interaction of fast neutrons with Th^{232} and subsequent reactions [IAEA, 2005]. The half-life of U^{232} is 73.6 years and the daughter products have very short half-life and some like $Bi^{212}$and Tl^{208} emit strong gamma radiations. Fig. 1 shows the decay chain of U^{232} [Kang et al., 2001].

Implementation of safeguards involves aggregation of several measures. Till recently, measures commonly implemented included nuclear material accounting and containment and surveillance. Measures added in recent times include unattended remote monitoring systems, satellite imagery, near real time monitoring, dynamic nuclear material accounting, safeguards-by-design etc. [JNMM, 2009]. In this doctoral work, additional measures have been proposed to enhance the safeguardability of thorium based fuels in the fuel fabrication facilities. Most of these measures are being proposed for the first time [Gangotra et al., 2014].

Classification of Safeguards Measures

These can be classified as A) Conceptual, B) Design Related, C) Engineering Related and D) Operational as given below:

Conceptual

1) Safeguards-By-Design
2) Co-location of facilities

3) Provision of nuclear material storage during physical inventory verification

4) Isolation of services

Design Related

1) Systems for plant imaging

2) Measurement of nuclear material inventory at every box / cell

3) Nuclear material tracking systems using RFID and bar codes

4) Overall reduction in the total number of items of process equipment

5) Footprint reduction of the plant

6) Reduction in ventilation ducting length

Engineering Related

1) Implementation of automation in the plant

2) Incorporation of efficient process powder recovery systems

3) Integration of process equipment

4) Integration of QC equipment with main process equipment

5) Improvements in equipment design

Operational

1) Implementation of near real time monitoring

2) Implementation of dynamic nuclear material accounting

3) Optimization of overall manpower deployment

4) Computerized tracking of nuclear material in the plant

5) Optimization of nuclear material flow in fabrication lines

Assessment of Proliferation Resistance

A quantitative assessment of the features proposed in this study has been carried out for enhancing the effectiveness of safeguards in fuel cycle facilities. The influence of either the presence or the absence of a safeguards feature on the overall safeguardability of facility has been judged based on the responses received to a questionnaire circulated among experts. For the purpose of overall assessment, all opinions have been combined using Multi Attribute Utility Analysis (MAUA) [Chirayath et al., 2010]. Sensitivity analysis has also been carried out to assess the relative importance of various features on the overall proliferation resistance. In addition to the MAUA method, the assessment of proliferation resistance (PR) has been carried out using a variation of the Japan Atomic Energy Agency (JAEA) method [Inoue et al., 2003; Inoue et al., 2004]. The JAEA method gives an arbitrary value for PR. In the current study, the JAEA methodology has been modified by adding a normalizing step to give PR values between 0 and 1 as is in the case of MAUA. The study has resulted in the assessment of PR in powder pellet type of fuel fabrication facility utilizing the safeguards measures proposed. These measures have different impact on the overall value of PR. Though the safeguards measures have been evaluated to enhance the PR in the context of thorium based fuel fabrication plants, the measures are general in nature and are applicable equally to facilities handling fuels other than thorium

Co-location: Possible Configurations and Comparison Thereof

A typical thorium fuel cycle facility has a number of plants including a fuel fabrication plant for initial and equilibrium core, a reprocessed U^{233} fuel fabrication plant, a reprocessing plant, a fuel assembly / disassembly plant and associated waste handling and management plants. A dedicated thorium fuel cycle facility can be set up to serve reactors at a single site. Alternatively, one can follow a hub and spoke approach with a large thorium fuel cycle facility acting as a hub, catering to the requirements of reactors at several sites as spokes. These two concepts have their respective merits and shortcomings in terms of engineering and economics. This doctoral work also includes a comparison of both the configurations on the basis of merits and challenges for implementation of safeguards on the two concepts viz. a large fuel cycle hub catering to reactors at several sites versus a small fuel cycle facility dedicated to reactors at a single site [Gangotra et al., 2013].

Chapter Wise Description

Chapter 1 describes briefly the importance of nuclear energy and its importance with respect to energy scenario in India. This Chapter briefly describes the various types of nuclear reactors most of which are in commercial operation around the world. It also introduces the concept of proliferation resistance and safeguardability. The chapter also describes importance of having a system of governance and governance system in India. This chapter also highlights the significance of the work carried out and brief description of the objectives of the present study.

Chapter 2 describes the methodologies for evaluating proliferation resistance. It also contains a description of the different fuel cycles and their proliferation resistance. India's approach to fuel cycle from the perspective of proliferation resistance is explained. Different methods for Proliferation Resistance (**PR**) evaluation are given along with their merits and limitations. Selection of method for **PR** assessment used in this study is also covered.

Chapter 3 describes the thorium fuel cycle and thorium fuel fabrication. A layout to facilitate safeguards measures is explained. Details of proposed safeguards measures in fuel fabrication facilities are given. Merits of implementing safeguards at design stage are also explained.

Chapter 4 describes two possible configurations for a thorium based power programme and comparison of resulting thorium fuel cycle facilities of two different capacities for implementation of safeguards.

Chapter 5 presents the compilation of data related to impact of safeguards measures, collected from the experts. This data has been used as input for analysis by MAUA and JAEA methodologies to assess overall proliferation resistance. The criteria for selection of experts, design of questionnaire and summary of responses are also detailed.

Chapter 6 presents the mathematical analysis of the impact of proposed safeguards measures on proliferation resistance by MAUA and Modified JAEA methodology for its implementation and impact on proliferation resistance in fuel fabrication facilities.

Chapter 7 lists the conclusions that could be arrived at on the basis of the studies carried out in this work. This chapter also includes some suggestions regarding future scope of work.

Appendix -1 describes the three-stage nuclear power programme of India.

REFERENCES

1. Bari, R., Peterson, P., Roglans, J., and Mladineo, S., Report on "Proliferation Resistance Modeling", ESARDA/INMM Workshop; Como, Italy, **39** (2004).

2. Brook, B. W. and Bradshaw C. J. A., "Key role for nuclear energy in global biodiversity conservation", Conservation Biology, 29 (3), 702-712 (2014).

3. Chirayath, S., "Multi-Attribute Utility Analysis for Proliferation Resistance Assessment", Presentation at the INMM Workshop on Proliferation Assessment, Texas A&M University, (2010).

4. Fischer, D., "History of the International Atomic Energy Agency- the First Forty Years", IAEA, Pub: International Atomic Energy Agency, ISBN: 9201023979 (1997).

5. Gangotra, S., Grover, R. B. and Ramakumar, K. L., "Comparison for thorium fuel cycle facilities of two different capacities for implementation of safeguards", Nuclear Engineering and Design, **262**, 535-543 (2013).

6. Gangotra, S., Grover, R. B., and Ramakumar, K. L., "Analysis of Measures to Enhance Safeguards, and Proliferation Resistance in Thorium Based Fuel Fabrication Plants", Progress in Nuclear Energy, **77**, 20-31 (2014).

7. Grover, R. B., and Chandra, S., "Scenario for Growth of Electrical Energy in India", Energy Policy, 34, 2834-2847 (2006).

8. Grover, R.B., "Green Growth and Role of Nuclear Power: A Perspective from India", Energy Strategy Review, (2013).

9. IAEA-STR-332, Report on "Proliferation Resistance Fundamentals for Future Nuclear Energy Systems" (2002).

10. IAEA-TECDOC-1450, Thorium Fuel Cycle – Potential Benefits and Challenges, ISBN: 9201034059 (2005).

11. Inoue, N., Hori, M., Hori, K. and Takeda, H., "Methodologies of Nuclear Proliferation Resistance Assessment for Nuclear Fuel Cycle Options," Proc. 44th Annual Meeting of the Institute of Nuclear Materials Management, Phoenix, Arizona (2003).

12. Inoue, N., Kurakami, J. and Takeda, H. “Review of JNC’s Study on Assessment Methodology of Nuclear Proliferation Resistance,” Proc. 45th Annual Meeting of the Institute of Nuclear Materials Management, Orlando, FL (2004).

13. JNMM, Special Issue: The Next Steps in International Safeguards, Journal of Nuclear Materials Management, **37(4)**, Ed. Mangan, D., Pub: Institute of Nuclear Materials Management Inc., ISBN: 9780750686730 (2009).

14. Kang, J. and Hippel, F., “U-232 and the Proliferation Resistance of U-233 in Spent Fuel”, Science & Global Security, **9**, 1-32 (2001).

15. Sinha, R. K. and Kakodkar, A., “Design and Development of AHWR- the Indian Thorium Fueled Innovative Nuclear Reactor”, Nuclear Engineering and Design, **236(7-8)**, 683-700 (2006).

List of Publications Arising from the Thesis

PEER REVIEWED JOURNALS

1) Gangotra, S., Grover, R. B., Ramakumar, K. L., Kamath, H. S., and Panakkal, J. P., "Safeguards-by-Design (SBD) Concepts for Thorium-Based Fuel Fabrication Facilities" , Journal of Nucl. Mat. Management, 41 (1), 43-51 (2012).

2) Gangotra, S., Grover, R. B. and Ramakumar, K. L., "Comparison for thorium fuel cycle facilities of two different capacities for implementation of safeguards", Nuclear Engineering and Design, 262, 535-543 (2013),

3) Gangotra, S., Grover, R. B., and Ramakumar, K. L., "Analysis of Measures to Enhance Safeguards, and Proliferation Resistance in Thorium Based Fuel Fabrication Plants", Progress in Nuclear Energy, 77, 20-31 (2014).

CONFERENCE PROCEEDING

1) Gangotra, S., "Emerging Trends in Safeguards and NUMAC Practices", Nuclear Material Accounting and Control: Current Practices and Future Perspectives (NUMAC-PP2013), Mumbai, 101 – 111 (2013).

List of Figures

List of Tables

Chapter 1

INTRODUCTION

1.1 Role of Nuclear Energy in India's Energy Mix

India is the seventh largest and the second most populous country in the world. A significant segment of this population does not have access to electricity and those who have, face regular shortages. As energy availability is vital for human development and is the prime mover of economic growth, the demand for energy will further grow. The present resources in the country for producing electricity are mainly coal, petroleum, nuclear and large hydro. To a limited extent, renewable resources also contribute to electricity generation. Coal is the main resource being used at present and coal-fired plants will continue to be the primary source of electricity production in the country for quite some time to come. The Integrated Energy Policy of the Government of India indicates that at a growth rate of 5% in domestic production, economically extractable coal resources will be exhausted in about 45 years [Planning Commission, 2006]. Nearly 80 per cent of the petroleum is imported in the country, which is an area of concern for the Government with regard to energy security for the present and in the near future. India is working towards exploiting full potential of large hydro and other renewable sources. However, renewable energy sources such as solar and wind are intermittent in nature. An intermittent energy source is an energy that is not continuously available due to factors outside direct control of the consumer. Hydro power has the limitation as it is dependent on rainfall and topography of the location of water sources. Solar energy presents an inexhaustible energy source for a tropical country like India but solar output varies throughout the day and through the seasons, and is affected by cloud cover. Technologies to store energy for use when the sun is not available, are very expensive.

Other renewable energy sources like wind have similar limitations arising from itermittency. Effective use of intermittent sources in an electric power grid usually rely on using the intermittent sources to displace fuel that would otherwise be consumed by non-renewable power stations, or by storing energy in battery based storage systems or in the form of renewable pumped storage, compressed air or ice, for use when needed, or as electrode heating for district heating schemes. In comparison to these intermittent energy sources, nuclear energy offers the most potent means for long term reliable energy security. Currently, the share of nuclear energy in electricity generation is about 3% in India. The nuclear share in total primary energy mix is expected to grow, as the installed nuclear power capacity grows.

As per statistics published by International Energy Agency, in 2012 India, with a total electricity generation of 1128 TWh, is the third largest producer of electricity [IEA, 2014]. While total electricity generation looked impressive, this was not so when one looked at it in per capita terms as India's population has now crossed 1.25 billion. Average per capita electricity consumption in the world is about five times of that in India. In India there is shortage of about 2% in peak demand and 5% in energy availability [CEA, 2014 – 15]. When India is in midst of rapid economic transformation, electrical installed capacity infrastructure needs attention. Electricity growth requirements have been a matter of concern and to look at the future, and to delineate role of nuclear energy, a study was conducted by Department of Atomic Energy (DAE) to forecast growth over the next 50 years. This study was completed in 2004 and the beginning of 10^{th} plan i.e. 2002-03 was taken as the base year for the study. The study was based on available projections about growth of economy and population and forecasts that by the middle of the century electrical installed capacity in India would be close to 1400 GW, which is about eight times the present installed capacity [Grover and Chandra, 2006]. The study concluded that full potential of all energy sources including nuclear should be exploited to ensure a robust growth of Indian economy.

This study was followed by another study by Planning Commission of India which was published as a report titled Integrated Energy Policy [Planning Commission, 2006]. Study by Planning Commission projected electricity requirements based on two scenarios for growth rates for Indian economy covering a horizon of three decades. There were differences in the forecasts made by the two studies, but both these forecast projected a massive increase in installed capacity. While the total electricity generation by the middle of the century as per DAE study looks very large. The per capita availability in the year 2052-53 is projected to be quite modest that is 5,300 kWh. This may be compared with present average per capita in OECD countries, which is about 8,089 kWh [IEA, 2014]. India has a long way to go and has to aggressively plan to increase per capita electricity generation by exploiting all sources including nuclear energy.

1.2 Types of Nuclear Reactors

Globally nuclear energy contributes ~11 % of electricity and presently, 438 nuclear reactors are operating in 30 countries. In addition, 69 reactors are under construction in 14 countries and 184 reactors are under planning in 27 countries [WNA, 2015]. Quite a few designs of nuclear Reactors have been developed and these are classified based on features like type coolant, moderator, fuel, neutron energy spectrum etc. A brief description of some of the commonly used reactor types is given below.

Heavy water reactors use heavy water as coolant and moderator. The fuel is natural uranium dioxide and neutron energy spectrum is thermal. In the Pressurised Heavy

Water Reactors the fuel is loaded in hundreds of horizontal pressure tubes. The fuel is cooled by pumping heavy water through the tubes (under high pressure to prevent boiling) and then to steam generator to produce steam which is then used for driving the turbine for generation of electricity. Such type of reactors are in operation in India, Canada, Argentina, Republic of Korea and few other countries.

Light water reactors are of two types. Pressurised Water Reactors (PWR) and Boiling Water Reactors (BWR). They use light water as coolant and moderator. The fuel is enriched uranium dioxide with enrichment upto ~5% in the isotope U^{235}. The fuel, is arranged in arrays of fuel pins and interspersed with movable control rods. The fuel assemblies are loaded in vertical pressure vessel through which light water is circulated at high pressure. The high-pressure water then passes through a steam generator, which produces steam to run turbine for generating electricity. These reactors are widely used in over 20 countries. The second type of water cooled and moderated reactor is BWR, which does away with the steam generator and, by allowing water within the reactor to boil, produces steam directly for driving the turbine for electrical power generation. Absence of steam generators in BWRs leads to some radioactivity in the steam circuit and the turbine, which then requires shielding of these components in addition to that surrounding the reactor. Such reactors are in use in some ten countries throughout the world.

Of the main commercial reactor types around the world, two (Magnox and Advanced Gas Cooled Reactor) owe much to the very earliest reactor designs in that they are graphite moderated and gas cooled. The Magnox reactor is named after the magnesium alloy used to encase the fuel, which is natural uranium metal. Fuel elements consisting of fuel rods encased in Magnox cans are loaded into vertical channels in a core constructed of graphite blocks. Further vertical channels contain control rods (strong neutron absorbers) which can be inserted or withdrawn from the core to adjust the rate of the fission process and, therefore, the heat output. The whole assembly is cooled by blowing carbon dioxide gas past the fuel cans, which are specially designed to enhance heat transfer. The hot gas then converts water to steam in a steam generator. Early designs used a steel pressure vessel, which was surrounded by a thick concrete radiation shield. In later designs, a dual-purpose concrete pressure vessel and radiation shield was used. In order to improve the cost effectiveness of this type of reactor, it was necessary to go to higher temperatures to achieve higher thermal efficiencies and higher power densities to reduce capital costs. This entailed increases in cooling gas pressure and changing from Magnox to stainless steel cladding and from uranium metal to uranium dioxide fuel. This in turn led to the need for an increase in the proportion of U^{235} in the fuel. The resulting design, known as the Advanced Gas-Cooled Reactor, or AGR, still uses graphite as the moderator and, as in the later Magnox designs, the steam generators and gas circulators are placed within a combined concrete pressure-vessel / radiation shield.

Water cooled, graphite moderated reactors are commonly known by the Russian acronym 'RBMK'. The RBMK have a large graphite core containing vertical channels, each containing enriched uranium dioxide fuel (~ 2% enriched U^{235}). Heat is removed from the fuel by pumping water under pressure up through the channels where it is allowed to boil, then to steam drums, further for driving electrical turbo-generators. Many of the major components, including pumps and steam drums, are located within a concrete shield to protect operators from the radioactivity in the steam.

All of today's commercially successful reactor systems as described above are "thermal" reactors, using slow or thermal neutrons to maintain the fission chain reaction in the U^{235} fuel. It is however, possible to use fast neutrons in reactors to cause fission in the fuel. These reactors do not have a moderator, and use less-moderating coolants. The physics of this type of reactor dictates a core with a high fissile concentration, with plutonium upto 30%. In order to make it breed, the active core is surrounded by material (largely U^{238}) left over from the thermal reactor enrichment process. This material, being fertile converts to fissile material when irradiated during operation of the reactor. Due to the absence of a moderator, and the high fissile content of the core, heat removal requires the use of a high conductivity coolant, such as liquid sodium. Sodium circulated through the core heats a secondary loop of sodium coolant, which in turn heats water in a steam generator to produce steam. Otherwise, design practice follows established lines, with fuel assemblies clad in cans and arranged together in the core, interspersed with movable control rods. The reactor is largely unpressurised since sodium does not boil at the temperatures experienced, and is contained within steel and concrete shields.

All the reactors outlined above are based on nuclear fission reaction. It is also possible to have reactors based on nuclear fusion reaction. In fusion reactors, energy is produced by fusing together the nuclei of light elements. This is the process which provides the energy source in the sun and other stars. The idea of releasing large amounts of energy by the controlled fusion of the nuclei of atoms such as deuterium and tritium is very attractive because deuterium occurs naturally in seawater. Unfortunately, controlled fusion has turned out to be an extraordinarily difficult process to achieve. For the reaction to proceed, temperatures in excess of one hundred million degrees must be obtained and high densities of deuterium and tritium must be achieved and retained for a sufficient length of time. So far, it has not proved possible to sustain these requirements simultaneously in a controlled way. Fusion reactors are now in experimental stages at several laboratories around the world. A consortium from the United States, Russian Federation, India, European Union, China, Republic of Korea and Japan is building a fusion reactor called the International Thermonuclear Experimental Reactor (ITER) in Cadarache, France, to demonstrate the feasibility of using sustained fusion reactions for making electricity.

1.3. Nuclear Fuel Cycle and Proliferation

The nuclear fuel cycle is the progression of nuclear material through a series of steps including mining of uranium, conversion, enrichment, fuel fabrication and loading in nuclear reactors and discharge as spent fuel. If the spent fuel is not reprocessed, the fuel cycle is referred to as an open fuel cycle (or once-through fuel cycle); if the spent fuel is reprocessed, it is referred to as a closed fuel cycle. In a once through fuel cycle, the fissile material is used to produce electricity in a power reactor and the spent fuel discharged is cooled and stored in deep geological repositories. Spent fuel contains uranium, plutonium, other minor actinides and fission products which are produced in the reactor. In case of thorium *(thorium is a fertile material hence throughout in the dissertation thorium fuel means thorium as fertile material)* based nuclear fuels, spent fuel also contains U^{233} which is produced from Th^{232} in the reactor. In case of a closed fuel cycle, the plutonium or U^{233} in the spent fuel is separated by reprocessing and refabricated to be charged back in the reactors for generation of electricity. U^{235}, Pu^{239} and U^{233} are the three species of nuclear material that undergo fission by thermal neutrons, which are used for generation of heat and electricity in nuclear power plants. Of the three species, U^{235} is the only naturally occurring fissile isotope. Pu^{239} and U^{233} are generated in nuclear reactors by the nuclear reactions and subsequent radioactive decays. By following the closed fuel cycle, the fissile and fertile material in the spent fuel is ploughed back in the reactors to generate additional electricity, thereby extracting more energy per kilogram of mined nuclear material.

The three fissile species U^{235}, Pu^{239} and U^{233} are also the nuclear material which can be used for the manufacture of nuclear explosive devices. Soon after the demonstration of nuclear energy, it was realized that the technology for harnessing atom for power generation can also be exploited to manufacture nuclear weapons. This is well documented in Acheson-Lilienthal authored ***Report on the International Control of Atomic Energy*** published in 1946 [Acheson-Lilienthal, 1946], which states, "The development of atomic energy for peaceful purposes and the development of atomic energy for bombs are in much of their course interchangeable and interdependent". The report proposes that "A system of inspection superimposed on an otherwise uncontrolled exploitation of atomic energy by national governments will not be adequate safeguard." While the first quote has scientific basis, the second quote is a statement of value judgement by the authors and is debatable. In any case the report and subsequent developments led to the formation of the International Atomic Energy Agency (IAEA). The Statute of the IAEA, which entered into force in July 1957, required that IAEA safeguards be applied to nuclear plant and material furnished by the IAEA and to other nuclear activities assisted, sponsored, supervised or controlled by the IAEA [Fischer, 1997].

Nuclear safeguards, which had been relatively rare and crude in the early years of the IAEA, came into their own with the advent of the Treaty on The Non-Proliferation of Nuclear Weapons (NPT), which imposed an obligation on the Non-Nuclear Weapon States (NNWS) to declare all of their nuclear materials, facilities and activities and place them under IAEA safeguards- hence the term "full-scope" or "Comprehensive" safeguards [Findlay, 2012]. Besides 'Comprehensive Safeguards' there are 2 other types of safeguards, voluntary offer agreement and facility specific safeguards. IAEA safeguards seek to provide reasonable assurance of the timely detection of a "Significant Quantity" of declared "Special" nuclear material being diverted from peaceful uses to nuclear weapons production. Administering safeguards is one way of addressing the issue of nuclear proliferation, the other being the ongoing initiatives to develop fuel and fuel cycles that have high proliferation resistance (PR) [Mourogov et al., 2002].

Proliferation resistance is generally defined as, "..that characteristic of a nuclear energy system that impedes the diversion or undeclared production of nuclear material or misuse of technology by States in order to acquire nuclear weapons or other nuclear explosive devices [IAEA, 2002]. The degree of proliferation resistance results from a combination of, inter alia; technical design features, operational modalities, institutional arrangements and safeguards measures. Intrinsic proliferation resistance features are those features that result from the technical design of nuclear energy systems, including those that facilitate the implementation of extrinsic measures. Extrinsic proliferation resistance measures are those measures that result from State's decisions and undertakings related to nuclear energy systems." The extrinsic proliferation resistance measures also called as safeguards measures play an important role in ensuring proliferation resistance. Safeguards consist of measures for nuclear material accounting (NUMAC), verification of nuclear material and containment and surveillance measures, which ensure the continuity of knowledge with regard to the location of the nuclear material.

Some researchers feel that the term "proliferation resistance" has fallen out of favour in non-proliferation circles. The Proliferation Resistance and Physical Protection Evaluation Methodology Working Group of the Generation IV International Forum has documented an evaluation methodology for Proliferation Resistance and Physical Protection (PR&PP) of Generation IV Nuclear Energy Systems (NESs)" [GIF/PRPPWG/2011/003, 2011]. The document defines proliferation resistance as well as physical protection in the context of PR & PP goals set for future NESs. This document lists six measures for proliferation resistance and three measures for physical protection, which are the high-level PR&PP pathway characteristics of the NES. As per the document;

(a) Proliferation resistance is that characteristic of an NES that impedes the diversion or undeclared production of nuclear material or misuse of technology by the host state seeking to acquire nuclear weapons or other nuclear explosive devices. The six measures for Proliferation Resistance are as follows:

1) Proliferation Technical Difficulty – The inherent difficulty, arising from the need for technical sophistication and materials handling capabilities, required to overcome the multiple barriers to proliferation.
2) Proliferation Cost - The economic and staffing investment required to overcome the multiple technical barriers to proliferation, including the use of existing or new facilities.
3) Proliferation Time – The minimum time required to overcome the multiple barriers to proliferation (i.e., the total time planned by the host state for the project)
4) Fissile Material Type – A categorisation of material based on the degree to which its characteristics affect its utility for use in nuclear explosives.
5) Detection Probability – The cumulative probability of detecting a proliferation segment or pathway.
6) Detection Resource Efficiency – The efficiency in the use of staffing, equipment, and funding to apply international safeguards to the NES.

(b) Physical Protection (robustness) is that characteristic of an NES that impedes the theft of materials suitable for nuclear explosives or Radiation Dispersal Devices (RDDs) and the sabotage of facilities and transportation by sub-national entities and other non-host state adversaries. The three measures for Physical Protection are as follows:

1) Probability of Adversary Success – The probability that an adversary will successfully complete the actions described by a pathway and generate a consequence.

2) Consequences – The effects resulting from the successful completion of the adversary's action described by a pathway.
3) Physical Protection Resources – The staffing, capabilities, and costs required to provide physical protection, such as background screening, detection, interruption, and neutralisation, and the sensitivity of these resources to changes in the threat sophistication and capability.

Safeguardability in the document is defined as the ease with which a system can be effectively (and efficiently) put under international safeguards.

The above three definitions have been formulated in the context of objectives of the Generation IV International Forum (GIF). If one looks at proliferation outside the context of GIF, one would see it as a much broader issue consisting of continued production of weapons grade material by states possessing nuclear weapons as a consequence of no progress towards disarmament, production of weapon grade material (including attempts to do so) by states who have legally forsaken such production by signing NPT as a non-nuclear weapon state, withdrawal by a state from NPT after having signed it, and attempts to acquire weapon grade material by sub-national groups or non-state actors or rogue states.

Shorn of the context of GIF literature, the term proliferation resistance can be seen as encompassing physical robustness, intrinsic features (including critical mass, dose rate), technological barriers, operational and safeguards measures.

As per the document "Guidance on International Safeguards and Nuclear Material Accountancy at Nuclear sites in the UK 2010 Edition" [Guidance Document, 2010], proliferation resistance are intrinsic measures or technical barriers which either inherently impede potential for misuse of nuclear material, reduce nuclear material attractiveness or give high diversion detection capability. Safeguardability has also been described variedly in literature. Bari et al. [2011] describe “safeguardability as “the degree of ease with which a nuclear energy system can be effectively and efficiently placed under international safeguards. Effective IAEA safeguards are a key element contributing to proliferation resistance, in addition to other intrinsic (design) and extrinsic (institutional) features. Bjornard et al. [2010] state that safeguardability refers to the extent to which the design of the facility readily accommodates and facilitates, the effective and cost-efficient safeguards for the facility.

As per Zentner [2011], the fundamental objective of international safeguards is to detect in a timely manner:

1) The diversion of significant quantities of nuclear material from peaceful to non-peaceful uses,

2) Possible misuse of nuclear facilities for undeclared purposes.

How well and how efficiently a Nuclear Energy System (NES) meets this objective is defined as its safeguardability. Safeguardability can be understood as the extent to which the facility design readily accommodates and facilitates effective and cost-efficient safeguards, that is, effectively integrating a nuclear facility technical design features with required safeguards measures. An important use of the results of proliferation resistance studies is to evaluate and if necessary improve the safeguardability of an NES by:

1) identifying, evaluating, and optimizing intrinsic barriers in the system design;
2) reviewing and evaluating safeguards measures for cost and effectiveness; and
3) ensuring that safeguards goals can be met.

Safeguardability is thus defined as the ease with which a system can be effectively and efficiently put under international safeguards.

Bathke et al. [2012] have used the term material attractiveness to study the proliferation resistance. The primary factors of material attractiveness are the bare critical mass, internal heat generation and the radiation dose rate. The metric used by these authors is a term "Figures of Merit" (FOM) which are intended to explain the attractiveness or preferences for a range of nuclear material across a span of credible nuclear adversaries. The authors have also made distinction between the state as an adversary and also non-state actors as adversaries.

To sum up, **proliferation** can be seen in a much broader sense consisting of:

a) continued production of weapons grade material by states possessing nuclear weapons as due to failure to achieve disarmament,

b) production of weapon grade material (including attempts to do so) by states who have legally forsaken such production by signing Non Proliferation Treaty (NPT) as a non-nuclear weapon state,

(c) withdrawal by a state from NPT after having signed it, and

(d) attempts to acquire weapon grade material by sub-national groups or non-state actors or rogue states.

The degree of **proliferation resistance** results from a combination of, inter alia; technical design features, operational modalities, institutional arrangements and safeguards measures. **Proliferation risk** can be expressed in terms of a risk triplet [Kaplan et al., 1981];

1) What can go wrong?
2) How likely is it? and
3) What are the consequences?

For proliferation risk, technical proliferation resistance studies answer the first and the third question; the likelihood of the deliberate act of proliferation is a difficult calculation most suited to State level proliferation studies.

With respect to the current study, the emphasis is on physical robustness which is a part of broader definition of proliferation resistance. With this background, the term “proliferation resistance” has been used in this study.

The fuels used in the power reactors are mostly in the form of sintered pellets of the oxides of uranium, plutonium or thorium. These fuels are fabricated using the conventional powder pellet type of processes. The fuel elements in the reactor are charged in the form of sealed and welded fuel pins, bundles or assemblies. In reprocessing facilities, the processes involve chopping and leaching of spent fuel in acids, followed by wet chemical processes using organic solvents where uranium, plutonium and fission products are separated as nitrate solutions. The nitrate solutions are subsequently converted to obtain the uranium or plutonium dioxide powders. The nuclear material in the complete fuel cycle is present in various forms like solids, liquids, powders, green pellets, sintered pellets etc. with regard to safeguards implementation, nuclear facilities can be broadly classified in two types: item counting and bulk handling facilities. Item counting facilities are the reactors and parts of fuel assembly plants where the nuclear material is present in sealed and welded fuel pins, bundles or assemblies. The bulk handling facilities are the conversion plants, enrichment plants, fuel fabrication plants, fuel reprocessing plants and the waste handling facilities. The item counting facilities pose lesser challenges for safeguards implementation as compared to the bulk handling facilities.

The concepts of proliferation, proliferation resistance and safeguardability have been introduced here. The details of proliferation resistance are given in section 1.4.

The institutional arrangements for implementation of safeguards have to be provided by a State through an appropriate system of governance and this aspect with respect to India is discussed in section 1.5.

1.4 Proliferation Resistance

Metcalf [2009] has studied the relative importance of non-proliferation factors. The methodologies to determine the **PR** of nuclear facilities rely on either expert elicitation, a resource-intensive approach without easily reproducible results, or numeric evaluations, which can fail to take into account the institutional knowledge and expert experience of the non-proliferation community. In an attempt to bridge the gap and bring the institutional knowledge into numeric evaluations of PR, a survey was conducted of 33 individuals to find the relative importance of a set of 62 non-proliferation factors, sub sectioned into groups under the headings of Diversion, Transportation, Transformation, and Weaponization. One third of the respondents were self-described non-proliferation professionals, and the remaining two thirds were from secondary professions related to non-proliferation, such as industrial engineers or policy analysts. The factors were taken from previous work which used multi-attribute utility analysis with uniform weighting of attributes and did not include institutional knowledge. In both expert and non-expert groups, all four headings and the majority of factors had different relative importance at a confidence of 95%. Metcalf concluded that such analysis and survey demonstrates that institutional knowledge can be brought into numeric evaluations of PR, if there is a sufficient investment of resources made prior to the evaluation.

Bean et al. [2007] have proposed a Simulation Enabled Safeguards Assessment Methodology (SESAME) which is a software package to provide capability for nuclear reprocessing facilities. The software architecture has been designed for distributed computing, collaborative design efforts, and modular construction to allow step improvements in functionality. Drag and drop wireframe construction allows the user to select the desired components from a component warehouse, render the system for 3D visualization, and, linked to a set of physics libraries and/or computational codes, conduct process evaluations of the systems designed. Virtual engineering has been applied to the facility design, as well as to the safeguards system design which will reduce total project cost and improve efficiency in the design cycle.

Kang [2005] in his analysis of nuclear proliferation resistance analysis has laid stress on country specific proliferation risks. Summarising the various methods adopted for the analysis of PR, Kang notes the lack in interpretation of country specific proliferation risk that is imposed by major nuclear weapons states, even though the countries are members of the Treaty on Non-Proliferation of Nuclear Weapons (NPT). His study outlines the assessment of **PR** by TOPS, INPRO etc., and points out the country specific risks and calls for further study to increase the **PR** of the civil nuclear energy systems in the specific NPT countries such as South Korea. The study concludes that:

a) Combined protection of intrinsic barriers and institutional measures is essential to effective proliferation resistance, although effective proliferation resistance measures depend on the proliferation threats.
b) New approaches are necessary to measure country-specific proliferation risk among the Non-Nuclear Weapons States (NNWS) under the NPT, to clear the discrimination of peaceful use of nuclear energy among the NNWS and
c) Multinational approaches to the implementation of the sensitive civil nuclear energy systems could be a prominent institutional measure of reinforcing non-proliferation.

Skutnik [2011] in his doctoral thesis has proposed a methodology for enhancing nuclear fuel cycle proliferation resistance analysis. While robust probabilistic risk assessment (PRA)-based methods for **PR** evaluation have been developed by experts at the national laboratories, such methods are generally resource intensive and often rely upon sensitive, non-public data to perform their analyses. In as much, there remains a strong need for open-source alternative **PR** models which can be used by the academic and policymaking communities, particularly for such tasks as scoping analysis of novel fuel cycles. An alternative to PRA has been in attribute-based models, such as attribute analysis (AA) and multi-attribute utility analysis (MAUA), which characterize **PR** through the use of multiple independent "barriers" to a proliferation attempt. Using one such method his study describes a methodology for enhancing **PR** evaluation using such models. These enhancements include the exploration of system **PR** dynamics via direct coupling with nuclear materials characterization analysis and methods to reduce the inherent subjectivity of attribute weighting. A wide variety of nuclear fuel cycle configurations were evaluated using this methodology. These fuel cycles fall into three categories: "open cycles" with no actinide recycling, "modified open cycles" which consist of limited actinide recycling (e.g., separating plutonium for single-recycle in mixed-oxide fuels), and "fully closed" cycles consisting of the recovery of all transuranic materials in spent nuclear fuel for use in fast-spectrum reactors. The characteristics of system **PR** were explored for each of these fuel cycle classes, including the dynamics of system **PR** in response to the fuel cycle parameters identified. The dynamics of system **PR** showed the strongest response for parameters which show a sustained "cascade" throughout the fuel cycle, such as uranium fuel burnup (impacting the plutonium composition) in partially-closed and full-closed fuel cycles, affected also by the choice of actinide recovery strategy. The technique of Adversary Pathway Analysis (APA) was also developed in this study as an additional means of enhancing AA/MAUA methods for fuel cycle **PR** analysis. APA involves the characterization of fuel cycle **PR** as a function of assumed adversary capabilities and final target material. This technique can be used to refine **PR** evaluation carried out in AA/MAUA methods by providing an analysis of the convergent pathways evaluated in PRA based techniques, thus providing a "bridge" between the methodologies. An evaluation was made as to the effect of simplifications in the nuclear fuel depletion calculation as well as cross-section uncertainty effects upon the material attractiveness calculation used for **PR** analysis.

Kiriyama and Pickett [2000] have studied the non-proliferation criteria for nuclear fuel cycle options. They compared the implied meaning of proliferation resistance in proposals regarding the nuclear fuel cycles. In their opinion there are discrepancies among the proposals regarding the technical definition of proliferation resistance, however there is a consistent focus on the importance of "physical form" as a key in determining a proliferation resistance fuel cycle. The reviewed proposals make little explicit mention of the importance of the time to process the material to a construct a nuclear weapon. While certain proposals discuss the importance of non-separation of plutonium from actinides, there are other proposals, which clearly do not view this aspect as vital in defining proliferation resistance. Recognizing that there are numerous political and infrastructure measures that may also be taken to guard against proliferation risks, Kiriyama and Pickett have focused here on the definition of proliferation resistance in terms of technical characteristics. The overall conclusion that they draw are:

a) Reduction of plutonium

b) Less separation of material

c) Fewer steps in the fuel cycle.

Kuno et al. [2009] have proposed measures for nuclear proliferation-resistance and safeguards for future nuclear fuel cycle. According to them, corresponding to the world nuclear security concerns, future nuclear fuel cycle (NFC) should have high proliferation-resistance and physical protection, while promotion of the peaceful use of the nuclear energy must not be inhibited. In order to accomplish nuclear non-proliferation from NFC, a few models of the **PR** systems should be developed so that international community can recognize them as worldwide norms. To find a good balance of 'safeguard-ability (so-called extrinsic measure or institutional barrier)' and 'impede-ability (intrinsic feature or technical barrier)' will come to be essential for NFC designers to optimize civilian nuclear technology with nuclear non-proliferation, although the advanced safeguards with high detectability can still play a dominant role for **PR** in the states complying with full institutional controls. The proposals made by Kuno et al. are:

1) Ratification of Additional Protocol
2) High detection capability
3) No presence of separated-Pu
4) No presence of weapon-grade Pu

Chirayath et al. [2008] have presented assessment of proliferation resistance requirements for fast reactor fuel cycle facilities. As per them, inclusion of intrinsic safeguards in fast reactor systems could lower one of the barriers to a closed fuel cycle. Their project aimed to locate and evaluate the proliferation concerns in a generic fast reactor fuel cycle: plutonium driver fuel from LWR or CANDU spent fuel with a depleted uranium blanket and PUREX reprocessing. Quantitative estimates for the material flow in a fast cycle were developed. The GEN IV International Forum suggested Multi Attribute Utility Analysis methodology for its semi-quantitative approach. From these reviews, estimates, and suggestions, this project applied a multiplicative MAUA methodology of assessing **PR** to establish the points in the fuel cycle of most interest. These areas of interest have been analyzed in more detail and a methodology for the inclusion of intrinsic safeguards in these areas has been being developed. Threat-scenario selection and metrics used were are along GEN IV PR&PP methodology guidelines and allow for quantitative trade-off studies as envisaged by GEN IV International Forum.

Kimura et al. [2011] have presented work on evaluation of **PR** based on decay heat of plutonium. Proliferation resistance of plutonium can be enhanced by increasing the decay heat of plutonium. For example, it can be enhanced by increasing the isotopic fraction of Pu ^{238} which has the largest decay heat among plutonium isotopes. In the study, the proliferation resistance of plutonium was evaluated based on decay heat with a physical assessment model. New criteria were proposed to evaluate the proliferation resistance based on isotopic compositions of plutonium from the viewpoint of decay heat. The criteria were applied to evaluate the proliferation resistance of plutonium produced in typical Light Water Reactor and Fast Breeder Reactor based on an evaluation function ‘‘Attractiveness’’ as case studies. The effects of Pu^{240} and Pu^{242} on the proliferation resistance of Pu were also considered in the evaluation. Technical difficulty against the misuse of Pu for a Hypothetical Nuclear Explosive Device (HNED) is enhanced by both the large decay heat of Pu^{238} and the large mass required for HNED of Pu^{240} and Pu^{242}, and there is a certain Pu^{238} isotopic fraction to make HNED technically unfeasible due to the critical temperature of high explosives inside HNED. A new criteria was proposed to evaluate proliferation resistance based on the isotopic compositions of plutonium from the viewpoint of decay heat. Plutonium with >15% Pu^{238} in high technology, >6% Pu^{238} in medium technology, and >2% Pu^{238} in low technology is technically unfeasible for the misuse of Pu for HNED. The present criteria were applied to evaluate the proliferation resistance of Pu in typical LWR and FBR blankets based on an evaluation function ATTR as case studies. The concept ATTR is defined as the ratio of potential explosive yield factor of fissionable materials to characteristics of technical difficulty factor in manufacturing Nuclear Explosive Device (NED) : Two types of Pu were discussed, the typical Pu and protected Pu by transmutation of Minor Actinides (MAs). It was confirmed that technical difficulty against the misuse of Pu for HNED considerably increases owing to the Pu^{238} produced by the transmutation of MAs.

Artisyuk et al. [2008] have developed a methodology to assess proliferation resistance of nuclear heavy metals. The study deals with comparison of proliferation resistance of essential fissile/fissionable compositions emerged in potential fuel cycles oriented on production, use and storage of denatured plutonium. The main focus is made on elaboration of associated criteria to bring this comparison on a quantitative base. New evaluation function is introduced that largely relies on a ratio of function of a-Rossi (that reflects energy yield) to characteristics reflecting technological difficulties in bringing nuclear material to supercritical state. The lack of explanation of physical background of currently used numbers for characterization of uranium (border line of 20% enrichment) and plutonium (Pu^{240} enrichment) has been stressed. The main focus was made on elaboration of evaluation function to be useful for comparison of proliferation resistant properties of advanced fuel compositions. Such a function in terms of attractiveness for potential proliferators that represents the ratio of characteristic of explosive yield to characteristics of technological difficulties to achieve sufficient yield has been proposed. The particular form of attractiveness depends on class of fissile materials. For the case of low-decay heat and low-neutron source materials the mathematical expression of attractiveness is translated to ratio of cubic a-Rossi to the product of compression (ratio of density under compression to that at normal condition) and mass of fissionable material. For the case of plutonium (class of materials with inherent decay heat and neutron source) the attractiveness is translated to ratio of cubic a-Rossi to product of decay heat and neutron source. Special emphasis was given to analysis of attractiveness of Np^{237} and U^{234} which are essential to produce Pu^{238} and represent its decay product. It was shown that in terms of attractiveness Np^{237} is close to uranium of 50% enrichment and U^{234} is to that of 30% enrichment. For plutonium case it was shown that attractiveness of compositions with 6% Pu^{238} doping is of the same level as plutonium with 30 - 60% of $240Pu^{240}$ (depending upon assumed model of fission energy release).

Greneche et al. have reported on proliferation resistance assessment: an illustration through the French fuel cycle [Greneche et al., 2004]. Technical or intrinsic measures as well as institutional and extrinsic measures provide concrete and effective barriers. Proliferation resistance barriers have already been implemented in nuclear energy systems operating today, and they will be continuously deployed and possibly strengthened in the nuclear energy systems (reactors, associated fuel cycle, safeguards and verifications, export control, security of supply, ….) which are designed for the near future and the more distant future. The paper introduced barriers and gave illustration through real life examples, borrowing on the existing comprehensive French fuel cycle. What remains a challenge is to assess the strength and robustness of the combination of those barriers. Assessing the value of each barrier for a given component of the fuel cycle is a first step, integrating the value of one barrier for the whole fuel cycle from cradle to grave is more difficult, integrating the value of all barriers and find the best compromise between sometimes conflicting indicators is even more difficult.

Yue et al. [2006] have reported work on proliferation resistance for advanced nuclear energy systems. The work presents an approach, which is based on Markov modeling, to the evaluation methodology for Generation IV nuclear energy systems being developed for PR&PP. Using the Markov model, a variety of proliferation scenarios have been constructed and the proliferation resistance measures quantified, particularly the probability of detection. To model the system with increased fidelity, the Markov model was further developed to incorporate multiple safeguards approaches. Evaluations of diversion scenarios for an example sodium fast reactor (ESFR) energy system have been used to illustrate the methodology. The Markov model is particularly useful because it can provide the probability density function of the time it takes for the effort to be detected at a specific stage of the proliferation effort.

Sleaford et al. [2010] have made an assessment of the attractiveness of material mixtures containing special nuclear materials (SNM) associated with reprocessing and the thorium-based LWR fuel cycle. The study examined the attractiveness of SNM associated with the reprocessing of spent light water reactor (LWR) fuel by various reprocessing schemes and the recycle of plutonium as a mixed oxide (MOX) fuel in LWR. Thorium-based reactors produce very attractive materials. The U^{233} that is produced has a substantial amount of U^{232}. The presence of U^{232} increases the dose of the material particularly at ages of about 10 years after irradiation. This is due to the in growth of Tl^{208} which has an intense high energy gamma-ray emission. In terms of weapons utility or material attractiveness this dose rate is only a nuisance to the adversary. It is not anywhere near sufficient to incapacitate a dedicated adversary. So if long term health and safety is not a concern to the adversary, U^{233} is one of the most attractive of all nuclear materials. Even though U^{233} is very attractive, like reactor-grade Pu, it is not normally attractive when it is contained within used nuclear fuel. The high dose rate of the used fuel in combination with the large mass of the used fuel assembly and the low concentration of SNM makes the material self-protecting for many years. Like used LWR and Heavy Water Reactor (HWR) fuels, however, the material eventually becomes attractive as the dose rate decays with age. Consistent with other studies of fuel cycles, the Th-based materials and processes need high levels of safeguards and moderate to high levels of security. Full safeguards would be needed on all facilities handing greater than 8 kg of U^{233} and Pu. However, security can be reduced for the used fuel while the dose rate is high enough for it to be self-protecting, but security needs to be high in the recycling and fuel fabrication facilities and moderate to high in any fresh fuel handing facilities.

Coles et al. [2009] have published a report on a project that investigated the use of social and cultural information to improve nuclear proliferation assessment, including non-proliferation assessment, proliferation resistance assessments, safeguards assessments, and other related studies. These assessments often use and create technical information about the State's posture towards proliferation, the vulnerability of a nuclear energy system to an undesired event, and the effectiveness of safeguards. Based on the literature search authors concluded that there are opportunities to use social models to improve understanding and assessment of proliferation-related problems. In fact, for decades analysts have theorized about the factors that dictate whether a State pursues the development of nuclear weapons–these factors are primarily social factors or are factors that are intimately related to social factors (e.g., national identity, leadership, politics, domestic security, economic capability). Yet another opportunity for social modeling was the area of non-State proliferation, particularly as it related to what some analysts call the supply-side. The supply-side substructure of nuclear proliferation might be considered to include manufactures, scientists, middlemen, transporters, opportunists, and violent groups who contribute to proliferation by supplying technology, knowledge, and material to the world. The interconnection of these groups is of interest because globalization has produced a large number of organizations that operate across State borders. The authors concluded that opportunities exist for social modeling in proliferation assessment.

Charlton et al. [2007] have reported on proliferation resistance assessment methodology for nuclear fuel cycles. A methodology, based on the multi-attribute utility analysis, for evaluating the proliferation resistance of nuclear fuel cycles and systems was developed and the details of its implementation have been presented. This methodology is intended for application to potential advanced fuel cycle systems, allowing for relative comparisons of different systems and technologies. It can also be used to assess safeguards effectiveness throughout a complete fuel cycle. The assessment methodology generates a relative proliferation resistance measure as a function of time for the history of a unit mass input into the fuel cycle. Examples of the implementation of this methodology for a variety of simple single-process systems and two multiprocess, long-term systems were given to demonstrate the methodology's viability as an assessment tool and its capability in discriminating diverse fuel cycle options.

Cleary et al. [2007] have reported on robust and reliable quantitative proliferation assessment tools which have the potential to contribute significantly to a strengthened non-proliferation regime and to the future deployment of nuclear fuel cycle technologies. Efforts to quantify proliferation resistance have thus far met with limited success due to the inherent subjectivity of the problem and interdependencies between attributes that lead to proliferation resistance. Authors have suggested that these limitations flow substantially from weaknesses in the foundations of existing methodologies – the initial data inputs. In most existing methodologies, little consideration had been given to the utilization of varying types of inputs, particularly the mixing of subjective and objective data or to identifying, understanding, and untangling relationships and dependencies between inputs. To address these concerns, a model set of inputs has been suggested that could potentially be employed in multiple approaches. Authors have also presented an input classification scheme and the initial results of testing for relationships between these inputs.

This paper documented the list of attributes and inputs developed to date and demonstrated approach to testing the list for the ability to associate numbers with inputs, the completeness of the set, the method of obtaining information, and the relationships between data inputs. While additional testing will be required to reach conclusions which can be used to revise the list, these examples suggested that this draft set of inputs and attributes substantially – though not completely – fulfil the performance targets developed.

Larry et al. [2004] have presented a study on the complexities in gauging time-dependency of proliferation resistance. To a considerable extent, policy decisions on nuclear fuel cycle issues depend upon how decision makers recognize and weigh “long-term” and “short-term” nuclear proliferation risk factors. Priorities and structures of advanced fuel cycle and safeguards research and development programs are affected similarly. There exists a diversity of understanding of the precise meanings of these proliferation risk terms, leading to lack of precision in their usage. In addition, proliferation risk evaluation fundamentally involves value judgments on the relative importance of time-dependent risks. Poor communication and diverse conclusions often result. The study explores some complexities in gauging “long-term” and “short-term” proliferation risk in the context of advanced nuclear fuel cycles. A convenient vehicle for this purpose is a commonly used notional plot of some proliferation resistance attribute of spent fuel or separated plutonium versus years from reactor discharge, often overlain with similar notional curves denoting multiple fuel irradiation and recycle. A common basis for misuse of such plots is failure to clearly define the range of proliferation threats being evaluated, as illustrated by several common examples of such omissions. Partial arguments of this type can be misleading and provide a disservice to policy makers who must have a clear picture of the trade-offs being made. This paper concludes with a call for much greater care to avoid overly simplistic interpretations of notional proliferation-related concepts and greater precision in general in use of proliferation-related terminology.

In the field of safeguards, a number of techniques have gained significance in recent times [JNMM, 2009]. Lockwood has reported about performing real-time process monitoring and surveillance in unattended mode. Bjornard et al. [2009] have emphasized the importance of Safeguards-by-Design and its impact on safeguards. Wallace et al. [2009] have presented the importance of information from open sources on safeguards. Such information can be publicly available such as that is provided by news media, or fee based or that is available on the internet. Niemeyer [2009] in his paper has discussed about information from satellite imagery and the possibilities and limits of gathering such information. He has also described the various types of imageries that can be used to verify the correctness and completeness of the IAEA member states' declarations, and to provide preparatory information for inspections, complimentary access and other technical visits.

1.5 Importance of a system of Governance and Governance System in India

For any country to exploit nuclear energy, it is necessary to have a sound system of governance at the national level to ensure that nuclear material is used for its intended purpose and is well secured. Additionally public is concerned with safety including waste management. India has an ambitious programme to generate electricity from nuclear for meeting its growing demand and to achieve that has in place a sound governance system. Its policy framework calls for utilizing full energy potential of nuclear fuel by following a closed fuel cycle approach [Grover[1], 2013]. Currently, India has 21 nuclear reactors in operation generating approximately 5,780 MWe and six other reactors are under construction which is expected to generate an additional 4,300 MWe. More details of India's three-stage nuclear power programme is given in Appendix – 1.

To provide sound governance for all issues related to nuclear power, Atomic Energy Act, 1948 was the first legislation, which was later replaced by Atomic Energy Act, 1962, framed. Thereafter, several rules, orders and notifications have been promulgated under this act for ensuring proper governance. For regulation of nuclear facilities, Atomic Energy Regulatory Board was set up in 1983 by an executive order issued under the Atomic Energy Act, 1962. A bill to set up a Nuclear Safety Regulatory Authority was introduced but lapsed due to change in the government after the election. It is expected that it will soon be re-introduced in the Indian Parliament. The Bill aims to convert de facto autonomy of the regulatory body into de jure autonomy and will be a step in allaying public apprehension in the regulation of nuclear power in India. India is a Member State of International Atomic Energy Agency and has signed various conventions including Convention on Nuclear Safety, Convention on the Physical Protection of Nuclear Material including its 2005 amendments and Convention on Early Notification. Nuclear Power Corporation of India is a member of World Association of Nuclear Operators (WANO), which is an association of nuclear power plant operators and carries out peer-reviews of all power plants. All nuclear power plants in India have gone through a peer review by WANO by experts drawn from all over the world. IAEA also carries out peer review of nuclear power plants under Operational Safety Review Team (OSART) mission. OSART missions have been carried out at Indian pressurized heavy water reactors [Grover[2], 2013].

Considering planned expansion of nuclear installed capacity, it was felt necessary to have legislation in India for civil liability for nuclear damage and first steps towards this were taken in late 1990s. Based on studies commissioned from experts and discussions within the DAE, it was decided to have legislation based on accepted international principles and join Convention on Supplementary Compensation for Nuclear Damage (CSC) [IAEA, 2004]. The job started in late 1990s was completed with the enactment of ‘Civil Liability for Nuclear damage Act, 2010’ and its notification on November 11, 2011 [CLND-2010]. The act is based on the principle of no-fault liability and liability is channelled to operator. Liability is limited in time as well as in amount. A nuclear operator must maintain financial security or an insurance cover to meet his liability. The Act conforms to the accepted international norms for civil liability for nuclear damage. Overall, it is consistent with the provisions of the CSC. Rules under the Act have also been framed [CLND Rules-2011]. India has also signed CSC and is expected to ratify it soon. All this establishes a sound liability regime in the country.

India has a well formulated strategy for nuclear waste management and has achieved important successes in developing needed technologies. As already explained, India is pursuing a closed fuel cycle and intends to go in for multiple recycling so as to utilise full energy potential of uranium and fully exploit thorium. This approach will generate minimum waste per unit of electricity produced. With the nuclear power profile on the verge of an exponential increase, it becomes imperative to consider and adopt cross-cut technologies that would not only lead to a substantial reduction in repository capacity both in terms of volumes and thermal loads but also lead to a reduction in radiotoxicity of the waste forms. High level nuclear waste arising from reprocessing plants contains minor actinides and fission products. Minor actinides have a long half life and need to be stored for a very long time. Partitioning of high level waste to recover minor actinides from high level waste is a step towards achieving the objective of reducing radiotoxicity of high level waste in a reasonable time frame. Towards these objectives, an engineering scale demonstration facility for partitioning of actual high level liquid waste (HLLW) from reprocessing of PHWR fuel has been set up in India. This facility is also being used to address routine recovery of residual uranium from HLLW leading to higher waste loading in glass, and also serve as a test facility for partitioning of minor actinides from uranium lean

HLLW. This is a significant achievement towards induction of partitioning technology for radioactive HLLW. Eventually minor actinides will be fabricated into fuel elements and incinerated in fast reactors. This will lead to fully closing the fuel cycle and waste will require storage for only about 300 years. Technologies for vitrification of HLW have been developed and deployed. Vitrified waste is packed in stainless steel canisters and a facility for interim storage of vitrified waste has been established. Work towards setting up a repository for long term storage is also progressing [Manohar et. al., 2013].

An important aspect of governance is export controls. Towards this object, rules have been framed under the Atomic Energy Act to control exports of nuclear items, including export of technologies. Related dual use technologies are controlled by rules framed under the Foreign Trade (Development & Regulations) Act, 1992 and overall export control regime has been further strengthened by the Weapons of Mass Destruction and their Delivery Systems (Prohibition of Unlawful Activities) Act, 2005. India, thus, has a law based export control regime with an impeccable record of implementation of its legal obligations and has taken measures to prevent any horizontal proliferation. India also has in place a system for physical security of nuclear materials and while it pursues a closed fuel cycle, taking a cue from 'just in time' principle followed by the manufacturing industry, it follows the principle of 'reprocess to reuse' and has not built any stockpile of plutonium which can cause concerns related to physical security [Grover, 2014].

India has put in place a comprehensive material protection control and accounting programme comprised of three basic elements: (1) the legislative and regulatory framework; (2) an integrated physical protection programme for facilities and materials; and (3) a comprehensive "Nuclear Material Accounting and Control System" (NUMAC). A Nuclear Control & Planning Wing (NC&PW) was set up in the Department of Atomic

Energy in 2013 by integrating DAE's safeguards, export controls, and nuclear security related activities. IAEA safeguards are being implemented in some nuclear facilities in India for over three decades. India has an impeccable record and has never been found in violation of its safeguards agreements with the IAEA.

1.6 Significance and a Brief Description of the Doctoral Work

Engineering proliferation resistance in bulk handling facilities is a challenging task. In view of the recent development and importance of proliferation resistance, a study on the assessment of proliferation resistance in powder pellet type of fuel fabrication facility by proposing novel safeguards measures in addition to existing measures have been undertaken. The current study is dedicated mainly to the implementation of safeguards in Indian nuclear facilities specifically the thorium based fuel. Each of these measures have different impact on the overall value of proliferation resistance. Though the safeguards measures to enhance the proliferation resistance have been proposed and evaluated in the context of thorium based fuel fabrication plants, the measures are general and are equally applicable to facilities handling fuels other than thorium. This study would not only be useful from the application point of view but also would be of academic interest.

The objectives of the work are a) proposing technological measures to enhance safeguards thereby improving proliferation resistance, b) bringing out merit in implementing safeguards from the conceptual and design stage itself and c) evaluating importance of various safeguards measures based on expert opinion and ranking them based on their importance towards enhancing proliferation resistance.

1.7 Layout of the Thesis

This dissertation has been divided into seven chapters, including the present chapter (chapter 1) which provides an introduction to this dissertation.

Chapter 2 describes the methodologies for evaluating proliferation resistance. It also contains the different fuel cycles and their proliferation resistance. India's approach to fuel cycle from perspective of proliferation resistance is explained. Different methods for Proliferation Resistance (**PR**) evaluation are given along with their merits and limitations. Selection of method for **PR** assessment used in this study is also covered.

Chapter 3 describes the thorium fuel cycle and thorium fuel fabrication. A layout to facilitate safeguards measures is explained. Details of proposed safeguards measures in fuel fabrication facilities are given. Merits of implementing safeguards at design stage are also explained.

Chapter 4 describes two possible configurations for a thorium based power programme and comparison of resulting thorium fuel cycle facilities of two different capacities for implementation of safeguards.

Chapter 5 presents the compilation of data related to impact of safeguards measures, collected from the experts. This data has been used as input for analysis by MAUA and JAEA methodologies to assess overall proliferation resistance. The criteria for selection of experts, design of questionnaire and summary or responses are also detailed.

Chapter 6 presents the mathematical analysis of the impact of proposed safeguards measures on proliferation resistance by MAUA and Modified JAEA methodology for its implementation and impact on proliferation resistance in fuel fabrication facilities.

Chapter 7 lists the conclusions and that could be arrived at on the basis of the studies carried out in this work. This chapter also includes some suggestions regarding future scope of work.

Appendix -1 describes the three-stage nuclear power programme of India.

Chapter 2

Evaluating Proliferation Resistance

2.1 Studies comprising doctoral work

The different types of fuels and fuel cycles have different intrinsic proliferation resistance features. Proliferation resistance of these fuel cycles can be enhanced by extrinsic measures including safeguards. To enhance the safeguardability of thorium based fuels in the fuel fabrication facilities a number of measures has been proposed in this study. Some of the measures are based on the recent developments in the field of safeguards and many of them are the original ideas proposed in this thesis and are presented in chapter 3. This work is mainly concerned with using technology to facilitate implementation of safeguards and thereby enhancing proliferation resistance.

Safeguards implementation in bulk handling facilities like fuel fabrication facilities as compared to item counting type of facilities is more challenging. Moreover, fabrication of fuel in glove box or alpha tight hot cells type of facilities requires intricate and challenging measures for safeguards due to complexity in remote handling, material hold up in ventilation systems, process hold ups, manipulation and constraints of access. Effective implementation of safeguards in such fuel fabrication facilities, calls for novel extrinsic measures. It is best to incorporate all such measures at the design stage itself and this has led to the concept of SBD.

The uranium, plutonium and thorium fuels have different requirements of handling. While uranium may be handled in open atmosphere, plutonium fuel needs hermetically sealed glove boxes. U^{233}in the thorium fuel cycle has inherent proliferation resistance due to the presence of high energy gamma radiation on account of U^{232}, which is associated with it. This calls for safeguards measures to be implemented as per the type of facility and the nuclear material handled. A typical thorium fuel cycle facility has a number of plants including a fuel fabrication plant for initial and equilibrium core, a reprocessed U^{233} fuel fabrication plant, a reprocessing plant, a fuel assembly / disassembly plant and associated waste handling and management plants. A thorium fuel cycle facility can be set up to serve reactors at a site. Alternatively, one can follow a hub and spoke approach with a large thorium fuel cycle facility acting as a hub, catering to the requirements of reactors at several sites as spokes. These two concepts have their respective merits and shortcomings in terms of engineering and economics. The present study has attempted to compare the merits and challenges for implementation of safeguards on the two concepts viz. a large fuel cycle hub catering to reactors at several sites versus a small fuel cycle facility dedicated to reactors at a single site.

To compare relative merits of various measures, one has to rely on expert opinion, which has to be analysed based on available methodologies. Accordingly, the following section covers concepts used for **PR** assessment, various fuel cycles and their PR, methodologies used for **PR** assessment and methodologies selected.

2.2 Proliferation Resistance Assessment

Proliferation resistance has been studied and evaluation of **PR** has been done by various researchers in different ways. Bari et al. have reported work on proliferation resistance modelling [Bari et al., 2004]. The National Nuclear Security Administration of US had been developing methods for non-proliferation assessments. A working group on Non-proliferation Assessment Methodology (NPAM) had assembled a toolbox of methods for various applications in the non-proliferation arena. One application of this methodology is to the evaluation of the proliferation resistance of Generation IV nuclear energy systems. The paper first summarizes the key results of the NPAM program and then provides results obtained thus far in the ongoing application In NPAM, a top-level measure of proliferation resistance for a fuel cycle system is developed from a hierarchy of metrics. The problem is decomposed into; metrics to be computed, barriers to proliferation, and a finite set of threats. The analyst models the process undertaken by the proliferant to overcome barriers to proliferation and evaluates the outcomes. In addition to proliferation resistance (PR) evaluation, the analysis also addresses physical protection (PP) evaluation against sabotage and theft. The Generation IV goal for future nuclear energy systems is to assure the public that they are very unattractive and the least desirable route for diversion or theft of weapons-usable materials, and provide increased physical protection against terrorism. The Proliferation Resistance and Physical Protection (PRPP) Evaluation Methodology Working Group of the Generation IV International Forum, developing this methodologies, has identified six high-level measures for the **PR** goals (six measures have also been identified for the PP goals). Combined together, the complete set of measures provides information for program policy makers and system designers to compare specific system design features and integral system characteristics and to make choices among alternative options. The Group has developed a framework for a phased evaluation approach to analyzing **PR** and PP of system characteristics and to quantifying metrics and measures. This approach allows evaluations to become more detailed and representative as system design progresses. Three sequential levels of detail are anticipated: qualitative, semi-quantitative and quantitative evaluation of the metrics and measures.

2.2.1 Methods of Analysis

Two general categories of methods have been used historically as the basis for non-proliferation assessments: attribute analysis and scenario analysis.

Attribute analysis. In this approach, attributes of the systems being evaluated (often fuel cycle systems) that affect their proliferation potential are identified. For a particular system under consideration, the attributes are weighted subjectively. Typically, these studies are more qualitative than the scenario analysis studies. There is an extensive history of the use of formal methods of decision theory (such as multi-attribute utility theory) to assist in policy development using this type of approach.

Scenario analysis. In these studies, hypothesized scenarios of pathways to proliferation are examined. The analyst models the process undertaken by the proliferant to overcome barriers to proliferation and estimates the likelihood of success in achieving a proliferation objective. Typically, these studies use logic modeling techniques (often probabilistic techniques). The results are quantitative but rely, in some respects, on subjective judgments of experts. The NPAM working group identified two additional categories with potential to support non-proliferation assessment: two-sided methods and dynamic modeling.

Two-sided methods. These methods examine the interplay between opponents. War-gaming is a two-sided approach that has been used extensively in other applications. A war-game is a role-playing exercise where human participants, often with opposing goals, make sequential decisions to allow a scenario to unfold. War-games appear to have promising potential to provide policy insights for non-proliferation issues that are not addressed effectively by other methods.

Dynamic modeling. Dynamic modeling predicts the evolution of hypothesized future states, usually by the solution of differential equations. Dynamic modeling provides the basis for a number of geopolitical models in which proliferation is examined as one of a number of future geopolitical outcomes.

2.2.2 Barriers to Proliferation

One of the strategies that is typically taken in non-proliferation analysis is to identify barriers to proliferation and to determine how effective these barriers are to deterring proliferation. This strategy is used both by scenario-based and attribute-based approaches. However, the manner in which they assess the effectiveness of barriers differs. Barriers are typically characterized as either intrinsic, features that are inherent to a particular fuel cycle system, or extrinsic, administratively-added security features such as physical protection and international safeguards. The nature of the proliferation threat can impact the relative effectiveness of intrinsic and extrinsic barriers. If a nation state decides to remove its facilities from IAEA safeguards and to use its commercial nuclear facilities to produce weapons material, extrinsic barriers would become completely ineffective in deterring the production of weapons material but intrinsic barriers could still be in place.

2.2.3 Threat Description

Another standard strategy for the decomposition of non-proliferation problems is to define a set of threats and to evaluate the proliferation resistance of the option under consideration for each threat separately. Consider, for example, a fuel cycle facility that is under IAEA safeguards. One threat could be a country with a high level of technical competence that decides to divert material covertly. Another threat is a small sub-national group that attacks the facility and attempts to escape with weapons material. The relative resistance to these different proliferation threats varies depending on the alternative fuel cycle system under consideration.

2.2.4 Metrics

After the objectives of the study have been clearly defined, the analyst must determine the metrics or measures (high level metrics) that will be used to characterize the proliferation resistance of the alternatives being evaluated. The guidelines review metrics that have been used in previous studies. For non-proliferation studies that compare the proliferation characteristics of one fuel cycle with an alternative fuel cycle, the analyst should develop high-level measures that are representative of the characteristics of the fuel-cycle or part of a fuel cycle, rather than mixing characteristics of the fuel cycle and the proliferator. The analyst should also develop metrics for evaluation in a manner to minimize dependencies between the metrics as they affect the high level measures. A typical top-level measure is either proliferation resistance, which is a characteristic of a fuel cycle system, or proliferation risk, which also includes aspects of the proliferator.

2.2.5 System Segmentation

A non-proliferation issue relates to some type of system composed of facilities, processes, and controls. Frequently the system is an element or multiple elements of a fuel cycle system (for example, the element could be an enrichment facility). It is general practice to subdivide the system into discrete segments. The subdivision often occurs at the facility level. However, depending on the detail of the analysis, it may be necessary to further subdivide these facilities to the level of a distinct process line. For example, within the nuclear power plant, the accessibility and characteristics of fuel are different in the fresh fuel storage area, reactor core, and spent fuel storage pool. Thus, the nuclear power plant can be subdivided into three elements. Similarly, at the ultimate storage facility, accessibility of material is different in the surface facilities than in the subsurface facilities. Once again, this facility is subdivided into two subunits for analysis. In contrast, the facilities at the front end of the fuel cycle involve only natural uranium, which is not a key target of proliferation. A number of facilities have been aggregated. Transportation between facilities can also be a point of diversion. Important transportation links can be identified as segments of the fuel cycle system in the same manner as facilities.

2.3 Nuclear Fuel Cycles – Proliferation Resistance

The different types of fuels and fuel cycles have different intrinsic proliferation resistance features. Before comparing various fuel cycles, it is necessary to explain certain definitions. Significant quantity is the approximate amount of nuclear material for which the possibility of manufacturing a nuclear explosive device cannot be excluded. Significant quantities take into account unavoidable losses due to conversion and manufacturing processes and are distinct from critical masses.

Table –2.1: Significant Quantities of Fissile Material

Material	SQ
Direct use nuclear material	
Pu[a]	8 kg of Pu
U^{233}	8 kg of U^{233}
HEU ($U^{235} \geq 20\%$)	25 kg of U^{235}
Indirect use nuclear material	
U ($U^{235} < 20\%$)[b]	75 kg U^{235} (or 10 t natural U or 20 t depleted U)
Th	20 t Th
[a] For Pu containing less than 80% Pu^{238}. [b] Including low enriched, natural and depleted uranium.	

Direct use nuclear material is that nuclear material that can be used for the manufacture of nuclear explosive devices without transmutation or further enrichment. Indirect use nuclear material refers to all nuclear material except direct use material. It includes depleted, natural and low enriched uranium and thorium, all of which must be further processed in order to produce direct use material. Table -2.1 shows the significant quantities for the fissile material [IAEA, 2001].

2.3.1 Once Through Uranium Fuel Cycle – Proliferation Resistance

The natural uranium obtained after mining has U^{235} content of 0.7%. The significant quantity for natural uranium is 10 t. This is a large quantity and not easy to divert or steal. Pressurised heavy water reactors use natural uranium. The fuel in these reactors is in the form of sintered uranium dioxide pellets encapsulated in zirconium alloy tubes. The fuel is fabricated in fabrication facilities and is handled in the open. These are the powder pellet type of facilities and are bulk handling facilities. The safeguards measures for natural uranium fuel fabrication are relatively easier to implement. These measures are nuclear material accounting, verification of nuclear material, and containment and surveillance measures. However, the fuel in LWRs like the pressurized water reactors and the boiling water reactors is enriched uranium. The enrichment of U^{235} in these fuels is upto 5%. This enrichment of uranium is carried out in enrichment facilities. Though the significant quantity for U^{235} is 75 kg, the LEU is also vulnerable to enrichment to higher range, which then poses proliferation risk. For a uranium fuel cycle consisting of only natural uranium fuel, the proliferation resistance is higher and the safeguards measures are comparatively easier to implement. However, the uranium fuel cycle, involving enriched uranium has lower proliferation resistance. This calls for stringent safeguards measures. One of the novel methods for detecting undeclared enrichment is by the analysis of environment samples.

2.3.2 Uranium Plutonium Closed Fuel Cycle – Proliferation Resistance

Uranium plutonium fuel cycle is a closed fuel cycle. The spent fuel discharged from the thermal reactors is cooled and reprocessed to separate plutonium and depleted uranium. The spent fuel from the fast reactor contains plutonium and depleted uranium which can again be refabricated to make MOX fuel for use in LWRs or in fast reactors. The significant quantity for plutonium is 8 kg. Plutonium is handled in leak tight glove boxes because of the radiological hazard due to inhalation of airborne particles. The fuel fabrication plants for manufacture of plutonium based fuels are generally powder pellet type of facilities. All the process and quality control equipment are housed in trains of leak tight glove boxes. The fuel pins are handled in open, only after they are hermetically sealed and are free of loose contamination. The reprocessing plants contain the nuclear material in the form of liquids. The conversion plants in the reprocessing units have plutonium dioxide powder and is also handled in leak tight glove boxes. These bulk handling facilities containing plutonium pose challenges to safeguards and have lower proliferation resistance. The ventilation system of the plutonium handling facilities consists of exhaust fans, ventilation ducts and filters. Since powder is generated in number of operations in fuel fabrication, some amount of nuclear material gets deposited in the ventilation ducts. It also gets deposited in blind spots of the glove boxes and in boxes carrying out operations that are prone to higher powder generation like centreless grinding of powders and crushing of scrap for recycle. The MUF (material unaccounted for) and hold up in these facilities is difficult to estimate [Beckers et al., 2004]. For facilities having large throughput, the quantities of MUF may exceed the significant quantity of plutonium. Thus the bulk handling facilities of the plutonium fuel cycle pose greater challenges for safeguards. The item counting facilities of the plutonium fuel cycle are comparatively easier to safeguard. In the case of LWRs, the (U,Pu) MOX fuel assemblies are present in the reactor as fresh fuel in the fresh fuel stores. It is also present in the reactor pressure vessel and the spent fuel storage pools. The proliferation resistance of plutonium is similar to that in the thermal reactor of uranium fuel cycle for the reactor vessel and the spent fuel storage pools. However, the fresh fuel store houses the assemblies having sealed rods containing up to 5% plutonium. In case of fast reactors, the (U,Pu) MOX fuel contains plutonium up to 30 %. Thus the proliferation risk increases in the fresh fuel stores of fast reactors. In addition to challenges in safeguarding plutonium based fuel in the different type of facilities, there is an added risk of proliferation during the transport of plutonium from reprocessing facilities to fabrication facilities and also transport of fabricated fuel from fabrication facilities to the reactors. It may be beneficial to have integrated plants containing all the facilities handling nuclear material at one site [Gangotra et al., 2013].

2.3.3 Uranium Thorium Closed Fuel Cycle – Proliferation Resistance

Uranium thorium fuel cycle has number of advantages [IAEA, 2005]. Significant among them is the intrinsic proliferation resistance of U^{233}. This is due to the association of U^{232} with U^{233}, which is a high energy gamma emitter. This also necessitates fabrication of fuel in sealed and shielded hot cells. India is developing an Advanced Heavy Water Reactor (AHWR) [Sinha at al., 2006], whose fuel cycle is shown in Fig. 2.1. The fuel cycle comprises of AHWR and the associated fuel cycle facilities comprising of fuel fabrication plants, reprocessing plants, refabrication plants and waste handling facilities. The fuel cycle is described in detail in Chapter 3. The AHWR is based on two concepts, in terms of the driver fuel. One using (Th-U^{235}) and other using (Th-Pu) mixed oxide fuel as the driver fuel for the initial and the equilibrium core. The irradiated fuel containing Pu and U^{233} will be separated by aqueous reprocessing and refabricated to make (Th-U^{233}) mixed oxide fuel. This will be used as fuel in the subsequent cycles. The fuel cycle consists of a fuel fabrication plant, fuel assembly / disassembly plant, fuel reprocessing plant, waste management plant, post irradiation examination unit, pools side inspection unit and nuclear material stores.

The fresh fuel for initial and equilibrium core will be fabricated in the powder pellet type of glove based plant in case of (Th-Pu) oxide fuel. (Th-U^{235}) oxide fuel does not require glove box type of facility, and can be handled in open. The fuel assemblies made in the fuel assembly plant will be sent to the fuel building in the reactor for loading in the core. The irradiated fuel from the reactor, after adequate cooling, will be transported to the fuel assembly plant, which also has facilities for fuel disassembly. The fuel assembly / disassembly will be carried out in either hot cells and / or underwater. The segregated fuel pins will then be sent to reprocessing plant, which will have different streams based on aqueous route. The product of the reprocessing plant will be sent to reconversion laboratory for conversion to oxide powders of Pu, U (reprocessed) and U^{233}. The oxide powders of reprocessed U, Pu and Th will be sent for storage and U^{233} oxide will be sent to refabrication plant. This refabrication plant is alpha tight hot cells based, due to the presence of high gamma radiation. Fabricated and quality checked (Th-U^{233}) oxide fuel pins will be then sent to the assembly plant for fuel assembly formation in hot cells or underwater. The fuel assemblies will be sent to the AHWR for irradiation, thereby closing the fuel cycle. The difficulties in the thorium fuel cycle pose great technological challenges in aqueous reprocessing of highly inert thoria fuel, remote fuel fabrication inside shielded hot cells, assembly / disassembly under water and remote handling of highly radioactive fresh fuel.

The fresh fuel fabrication plant for the manufacture of (U,Pu) MOX needs safeguards implementation similar to the glove box type of manufacturing plants of the plutonium fuel cycle. However, the (Th-U^{235}) MOX plant will need greater attention due to the presence of enriched U^{235}, which could be as high as 19%. In case of the nuclear material at AHWR, the fuel assemblies for the initial core will be containing either U^{235} or Pu. These assemblies can be handled without shielding in the fresh fuel store. This means that they will have lower proliferation resistance and will need appropriate safeguards measures. However, the fuel assemblies for the equilibrium core will contain fuel pins of U^{233}. Since these will have high gamma radiation, even the fresh fuel will need storage under water. Nuclear material accounting will be easier on account of these being as item counting units. The fuel assembly and disassembly will be carried out in hot cells or under water in the pool. For operations in the hot cells, the proliferation risk is lower since the cells are shielded and operations are carried out by remote handling. The amount of powder generation is not much since only the failed fuel pins may leach out the nuclear material. Similarly for assembly and disassembly in the pool, the nuclear material accounting is as item counting, except for leached nuclear material from the failed fuel. The nuclear material in the reprocessing plant is mainly in the form of liquid, which is contained in the piping and vessel systems. The proliferation risk is similar to the reprocessing facilities in plutonium fuel cycle. The U^{233} powder in the conversion plant will need shielding due to gamma. For the same reason, the fuel refabrication plant for the manufacture of ($Th\text{-}U^{233}$) oxide will have to be carried out in hot cells. Since the nuclear material is inside shielded cells, the risk of proliferation is low. However, there arises difficulty in estimation of material hold up and MUF in the cells, equipment and ventilation ducting.

2.3.4 India’s Approach to Fuel Cycle and Proliferation Resistance Perspective

India follows a three stage approach for its nuclear programme. This envisages the use of all the three nuclear materials, viz. uranium, plutonium and thorium (U^{233}). The first stage involves setting up thermal reactors with uranium as fuel. The plutonium generated after irradiation of fuel in the reactors of first stage is separated and refabricated as MOX fuel to be charged back in fast reactors being set up under the second stage. The policy of “reprocess to reuse” advocated by India involves optimizing the schedule for reprocessing, refabrication and reloading of fissile material in reactors with the objective of minimizing stocks of fissile material. This ensures minimum inventory of plutonium and U^{233}. A prototype fast breeder reactor marking the launch of the second stage is nearing completion. Since the burn up of the fast reactors can be only around 200,000 MWD/ton, the plutonium has to be recycled multiple times to fully utilise the heavy metal. Accumulation of the minor actinides in the reprocessed Pu can make reactor physics complex. If these minor actinides can be separated, the gains are in terms of efficient use of recycled plutonium as well as the management of the remaining waste. Towards these objectives, India has set up an engineering scale demonstration facility for partitioning of high level liquid waste (HLLW) arising from reprocessing of PHWR fuel. This facility is also being used to address routine recovery of residual uranium from HLLW which if not recovered leads to higher waste loading in glass, and also serve as a test facility for partitioning of minor actinides from uranium lean HLLW. Multiple recycling and portioning of actinides for subsequent transmutation in fast reactors fully closes the fuel cycle. Also it generates minimum waste per unit of electricity produced.

The third stage will involve setting up of reactors to use India's vast thorium reserve. Further details about the three stage programme are given in the Appendix-1.

All three nuclear materials, U, Pu and U^{233} are used in India's nuclear programme. For the first stage utilising uranium, the safeguards are easier to implement since uranium is used either as natural uranium or with enrichment of U^{235} upto a maximum of 5%. For the second stage, the fast reactors will use MOX fuel with plutonium upto a maximum of 30%. In future, there are plans to use metallic fuels as well. Thus the reactors and facilities that will handle plutonium in oxide and metallic forms, will pose challenges in terms of safeguards. For the third stage, the facilities will handle U^{233} in addition to uranium and plutonium. U^{233} being inherently proliferation resistant, will be easier to safeguard, however, the interference of hard gammas due to presence of U^{232} will make nuclear material accounting challenging. A number of extrinsic measures including continued emphasis on the policy of "reprocess to reuse" will have to be incorporated to safeguard the fuel cycle facilities and reactors of the three stage of India's nuclear power programme.

2.4 Proliferation Resistance (PR) Evaluation

Proliferation resistance assessment have been done by various researchers in different ways. These reported methods have their own merits and shortcomings. The major methods used for evaluation of proliferation resistance are described below.

2.4.1 Gen IV International Forum – PRPP

The Proliferation Resistance and Physical Protection (PRPP) Evaluation Methodology Working Group of the Generation IV International Forum has developed a methodology for evaluating the **PR** and PP of the future nuclear energy systems [GIF/PRPPWG/2011/003, 2011]. The methodology considers a set of alternative systems and evaluates their resistance or robustness to a collection of potential threats. For the challenges considered, the response of the system to the challenges are assessed and expressed in terms of outcomes. Both technical and institutional characteristics are used to evaluate threats by states as well as sub-national adversaries. The outcomes of the system response are expressed in terms of PR & PP measures [Bari et al., 2006]. There are six PR and three PP measures. However, it is not clear if the method could demonstrate the ability to capture the proliferation resistance of a fuel cycle over a multi-year time period. While this methodology allows users to assess the probability, cost and consequences of a diversion, it does not suggest what an acceptable **PR** value might be.

2.4.2 International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) Methodology

An objective of the INPRO was to "develop the tools to analyse the role and structure of Innovative Nuclear Energy Systems (INS) required to meet sustainable energy demands and to develop the methodology to assess the INS [IAEA, 2004]. The methodology seeks to avoid attribute correlations and dependencies. The methodology can be used to assess individual facility and process within a given fuel cycle independently. Different INS will have different strengths and weaknesses with regard to **PR.** Assessment can identify the strengths and weaknesses to aid with decision making, but assessments cannot generally render a judgment as to which system is stronger with regard to **PR.** Aggregation methods to generate a single score for **PR** based on strengths and weaknesses identified in an assessment can be misleading [Zentner et al., 2011]. The method does not readily lend itself to sensitivity analysis. The method also does not provide quantitative analysis.

2.4.3 Technological Opportunities to Increase the Proliferation Resistance of Global Civilian Nuclear Power System (TOPS) Methodology

The TOPS barrier method analysis defines a framework that can be applied to compare the relative proliferation resistance of "mining-to-disposal" for civilian nuclear fuel cycles. The goal of the TOPS was to define a framework for **PR** assessment that could be applied to any system [TOPS, 2001]. TOPS method identifies the intrinsic barriers against proliferation from a given nuclear system, attempts to evaluate their effectiveness against potential proliferators and identifies where extrinsic barriers need to be added [TOPS, 2001]. TOPS is a well developed approach that lends itself to supporting tasks such as ranking and comparing technologies and identifying research needs. It is built on qualitative attribute assessment through expert opinion surveys. The reproducibility is difficult and sensitivity analysis and comparison of systems, processes and facilities is of limited value. Many of the recent methods for assessment of proliferation resistance are based on the experience of the TOPS methodology.

2.4.4 Japan Atomic Energy Agency (JAEA) Methodology

JAEA methodology is a quantitative assessment methodology for nuclear proliferation resistance assessment. The goal was to improve the development strategy for the commercial fast reactors systems [Inoue et al., 2003; Inoue et al., 2004]. This method is similar to the TOPS method. While this method allows users to arrive at a single **PR** value for an entire fuel cycle, it can also be easily broken down to give independent values for each stage in a cycle, using same set of consistent attributes for each of them. This method incorporates measurements of mass, volume, radiation field, isotopic and chemical composition into the attributes, but the quantitative scheme involves significant subjectivity. This method has limitations similar to the TOPS method.

2.4.5 Brookhaven National Laboratory (BNL) Markovian Method

BNL developed a Markovian probabilistic framework useful for evaluating pathways associated with a specific proliferation scenario by representing possible event sequences and characterizing **PR** on the basis of other factors. Events are characterized in terms of transition, detection, and failure rates from one state to the next. Thus, for application to proliferation resistance, requisite analogs to characterize failure rates on the basis of process physical parameters are required to be developed. This method assumes that proliferation activities are sequential, which is not consistent with real-world restrictions on proliferation efforts. The assumption of the model is that if detection probability is high, so is **PR.** The BNL method uses a two-fold approach in evaluation of **PR.** It uses intrinsic properties of nuclear material and fuel cycle facilities to give probabilities as a function of a time that a proliferators can successfully divert material out of a given fuel cycle stage. It then considers safeguards inspections as the sole extrinsic measure of detecting and stopping proliferation. This is expressed as a detection rate for each type of inspection. The detection rates are combined with the probability density functions to give the overall likelihood that a proliferator could successfully divert material from each stage [Yue et al., 2005]. There is a great deal of subjectivity in the analysis and thus reproducibility is difficult. The probability of detecting diversion is given as a function of time and it is highly dependent on the safeguards system in place. The consideration of physical protection measure is not properly defined as well. The BNL method provides an excellent way to analyse the impact of safeguards on **PR** and to provide time-dependent results.

2.4.6 Multi Attribute Utility Analysis (MAUA)

MAUA has been shown to provide a viable means for assessing systems with diverse attributes. Diversity includes some attributes lending themselves to scientific judgment, while others to value judgment. It also includes conflict between attributes. MAUA has been used in several areas related to nuclear energy such as public participation in siting a waste management facility, analysis of perception about nuclear energy and in recent years in assessing proliferation resistance of the nuclear fuel cycle [Charlton et al., 2007; Cleary et al., 2007]. Texas A&M University has developed MAUA method for proliferation resistance assessment for the Advanced Fuel Cycle Initiative (AFCI). The method includes multiplicative and additive forms [Chirayath et al., 2010]. The methodology yields a numeric **PR** value. Each of the attributes use definitions for their values and weighting factors use subjective determinations, and although the explanation of each attribute's utility function outlines whether it is objective, subjective, or both, there is no way to quantify how much subjectivity is involved in the analysis. Measurable quantities are included in the analysis, and while some of the attributes may be fully independent, others may be physically dependent even when measured separately. Expert elicitation is used to provide weighting factors for the method. The method lends itself to sensitivity analysis. MAUA method meets a great number of the desired characteristics for a **PR** assessment tool. It establishes one short list of critical attributes that can be used to assess any system, facility or vehicle that contains nuclear material and it makes excellent progress toward eliminating subjectivity from the analysis, but it demonstrates that there may be some attributes that are inherently subjective. The method could also be made to consider threat characteristics by adjusting the weighting factors for each attribute according to how they would effect each type of proliferator [Giannangeli, 2007].

2.4.7 Risk-Informed Probabilitstic Analysis (RIPA)

RIPA method for proliferation resistance assessment has been developed by Sandia National Laboratory. The goal of RIPA was to create a process capable of conducting a dynamic analysis to compare and outline probable outcomes of feasible proliferation pathways and forecast those pathways by creating likely scenarios [Blair et al., 2002; Greneche et al., 2006]. RIPA uses quantifiable information for considering potential proliferation pathways and introduces deductive reasoning to visualize the proliferation process. The result provides a quantitative analysis with uncertainties, allowing reviews and reproducibility of the outcomes. The calculation of the consequences is not focused and probabilities are difficult to estimate. The probabilities do not present much insight into the proliferation risk.

2.4.8 Simplified Approach for Proliferation Resistance Assessment (SAPRA) Methodology

SAPRA is being developed by a working group of French institutional and industrial experts, including AREVA, Inc. on proliferation resistance and physical protection. SAPRA is based on the TOPS methodology, with two important distinctions: the multiple barriers analysis is extended beyond diversion to the whole proliferation pathway and the specific "state characteristics" (e.g., skill level, existing facilities, non-proliferation commitments, etc.) are introduced as an important factor in the assessment. SAPRA assumes four stages to acquisition of a nuclear weapon by diversion, a) diversion of nuclear material, b) transportation of nuclear material to a second site, c) transformation of the material into a weapons-usable form, and d) weaponisation of the material by adding a physics package [Bari et al., 2004]. During each of these stages of proliferation, there are barriers which inhibit the progress of the proliferators to obtain a successful weapon. Each of the potential barriers to proliferation is then rated by a panel of experts on a scale of 0 to 4 with z being no barrier at all and 4 being an extremely resistant barrier. Like TOPS, this method suffers from heavy dependence on expert opinion.

2.5 Selection of methods for Assessment of PR

The intrinsic features of various fuel cycles including the thorium fuel cycle have been described in Section 2.3. The rest of the thesis describes the extrinsic measures that can be implemented for enhancing the proliferation resistance. As discussed in the previous section, many of the methods for **PR** evaluation, use subjective assessment to make qualitative judgment. However, some of the methods also yield quantitative values. A comparison of all these methods is given in Table – 2.2.

MAUA and JAEA methodologies can be suitably applied to evaluate **PR** of fuel cycles or any segment of the fuel cycle using attributes. The evaluation of **PR** in the current study has been carried out for the powder pellet type of fuel fabrication plant. Such a plant is one segment of the complete fuel cycle. Hence for evaluation of **PR** in the current work, the two methods that have been employed are the MAUA and JAEA methodologies. While the MAUA method yields the values between 0 and 1 to quantify the overall **PR**, the JAEA methodology results in arbitrary value for **PR.** The JAEA methodology has been modified in the current study by normalizing, so that the resulting **PR** value is between 0 and 1. Additionally, the MAUA method lends itself to sensitivity analysis. Based on the sensitivity analysis, and importance factor has been defined. The importance factor helps to rank the 20 proposed measures in order of their impact on the overall **PR** value. It thus helps the policy makers, designers and operators of such facilities to incorporate such measures to enhance overall **PR.**

2.6 Nuclear Security and Safeguards

Nuclear Security is the prevention and detection of, and response to, theft, sabotage, unauthorized access, illegal transfer or other malicious acts involving nuclear material, other radioactive substances or their associated facilities. The methods involved to ensure nuclear security are;

a) Protection of nuclear and other radioactive material against theft during use, storage or transport
b) Retrieval and return of lost material
c) Protection of facilities, location and transport against acts of sabotage.

Safeguards is a system of inspection and verification of the peaceful uses of nuclear materials. The objective of the safeguards is the timely detection of diversion of significant quantities of nuclear material from peaceful nuclear activities to the manufacture of nuclear weapons, or of other nuclear explosive devices or for purposes unknown, and deterrence of such diversion by early detection. Safeguards implementation involves nuclear material accounting and containment and surveillance measures.

In the context of safeguards, as described in Section 2.3, significant quantity is defined as the approximate amount of nuclear material for which the possibility of manufacturing a nuclear explosive device cannot be excluded. This does not imply that a loss of this quantity is acceptable from the safeguards implementation consideration. In fact the safeguards procedures, including frequency of inspections at a facility are decided on this basis so that based on the **timely** detection of diversion, it is ensured that the agency diverting the material doesn't have enough time to accumulate nuclear material to add up to make a SQ. In safeguards, no amount, howsoever small, of nuclear material can be allowed to be unaccounted.

A similar classification is made for nuclear security from the physical protection criterion, where the materials are categorized in 3 categories [IAEA, 2011]. This categorization is the basis for a *graded approach* for protection against *unauthorized removal* of *nuclear material* that could be used in a nuclear explosive device, which itself depends on the type of nuclear material (e.g. plutonium and uranium), isotopic composition (i.e. content of fissile isotopes), physical and chemical form, degree of dilution, radiation level, and quantity. This categorization also decides the level and type of physical protection measures to be adopted, both during storage and transport. Here too, no amount of loss of nuclear material is acceptable.

In comparing the technical objectives of nuclear security and safeguards, there are different points. For safeguards, the focus is on the State and its nuclear activity compliance in accordance with the legal obligations under relevant safeguards agreements. Accordingly, the safeguards activities are aimed at State's nuclear activities. The scope of the safeguards is focused on nuclear material. The timeliness concern for the detection of diversion and misuse is set of the order of a month or more based on a possible conversion time to the nuclear explosives.

For security, the concern of threat is more on non-state actors, criminals, terrorists and acts of sabotage by insiders. Scope of material is broader and covers all nuclear material and radiological substances. The timeliness concern is much shorter to be real time or immediate concern.

In safeguards, legally binding systems of nuclear material accountancy and control, and verification are the two key components of the effective governance. On the other hand, most of the international norms and guidance for security are not legally binding, but states are expected to conform to its contents through establishing appropriate security structures. Nuclear material accountancy and control is an essential element in the implementation of safeguards and is also an important factor for successful security undertakings. An important aspect of nuclear security is that it is the responsibility of a State.

In considering synergy between nuclear security and safeguards, there some common points; they are both aimed to deter and detect unauthorized removal of nuclear material, to provide assurance that all nuclear material is accounted for, to provide a timely detection of material loss or diversion, and to determine amount and location of any missing material. There are areas where safeguards and security can interact to improve effectiveness and efficiency in achieving their objectives.

Table- 2.2: A summary of the different proliferation resistance analysis methods [Giannanageli, 2007, Chirayath, 2015].

Desired Characteristics	**PRPP**	**INPRO**	**TOPS**	**JAEA**	**BNL**	**MAUA**	**RIPA**	**SAPRA**
Lends itself to analysis	Yes	No	No	No	Maybe	Yes	Yes	No
Independence between attributes	No	Yes	No	No	No	Maybe	Maybe	No
Amenable to quantitative analysis	Yes	No	No	Maybe	Yes	Yes	Yes	Maybe
Uses measured parameters from facilities	Yes	Yes	No	Yes	Yes	Yes	Yes	Maybe
Provides uncertainty or confidence level for results	Yes	No	No	No	No	No	Yes	No
Produces a time-dependent analysis	Maybe	Yes	Maybe	Maybe	Yes	Yes	Maybe	Maybe
Considers physical protection measures	Yes	Yes	Yes	Yes	Maybe	Yes	Yes	Yes
Considers threat characteristics	Yes	No	Yes	Yes	No	Yes	Yes	Yes
Avoids subjective determinations	Maybe	No	No	No	No	No	No	No
Ability to assess multiple facility types with consistent set of metrics	Yes	Yes	Yes	Yes	Maybe	Yes	Yes	Yes
Considers transportation of material	Yes	Maybe	Yes	Yes	Maybe	Maybe	Yes	Yes
Considers geological storage of material	Yes	Yes	Yes	Yes	Maybe	Yes	Yes	Yes
Allows for discrimination between different facilities/technologies	Yes	Maybe	Yes	Yes	Maybe	Yes	Yes	Yes

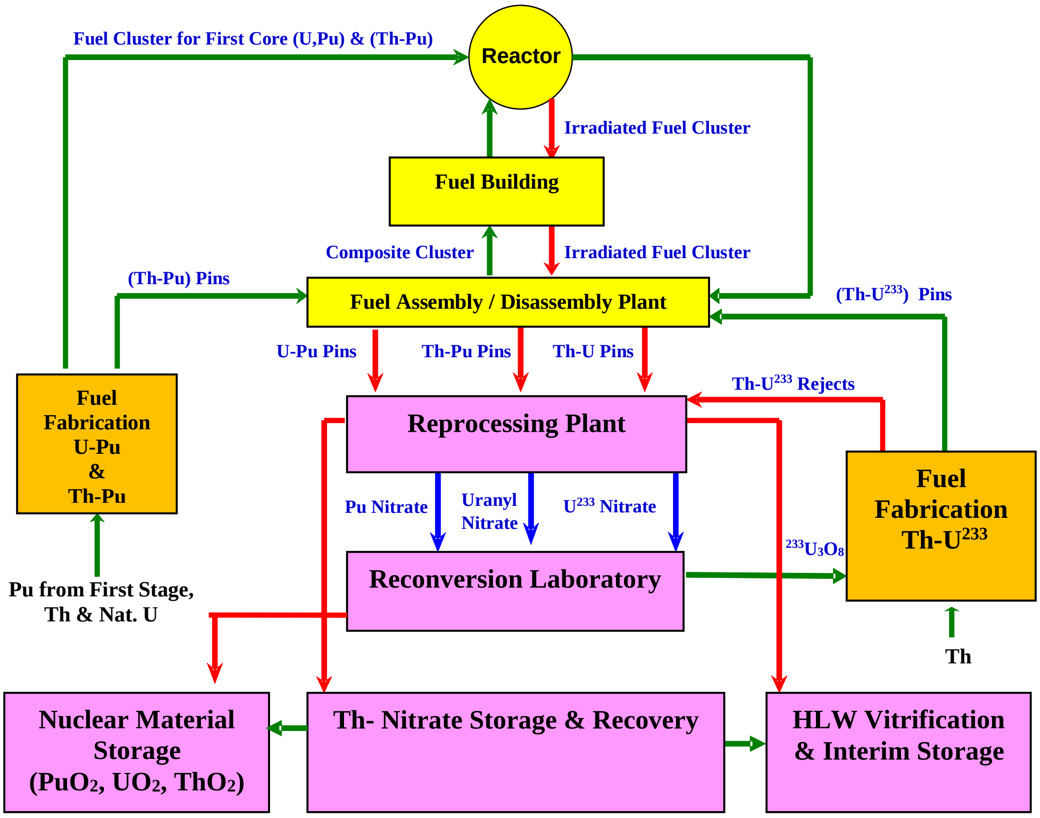


Fig. 2.1 Thorium Fuel Cycle

Chapter 3

Safeguards Measures for Enhanced Proliferation Resistance

3.1 Introduction

The thorium fuel cycle is of importance to India as abundant thorium reserves available in the country provide a potent means to energy security. While thorium fuel cycle has characteristics that make it intrinsically proliferation resistant to some extent, extrinsic measures have been proposed in the thesis that further increases the proliferation resistance of the thorium cycle. India has experience in fabrication of uranium, plutonium and thorium based fuels by the powder pellet route. The thorium fuel cycle and the fuel fabrication flow sheet for thorium based fuels are explained in this chapter. For fuel fabrication lines, two concepts are described, viz. the linear layout and the hybrid layout. The hybrid layout, designed as a part of the present work, helps to improve the safeguardability.

In this chapter, thorium fuel cycle is described followed by a hybrid layout which facilitates implementation of safeguards measures and then additional safeguards measures for powder pellet type of thorium fuel fabrication facilities are proposed and discussed. Merits of implementing safeguards in fuel fabrication plants at design stage are also discussed.

3.2 Thorium Fuel Cycle

Thorium is more abundant than uranium and is widely distributed in nature. During the early years of nuclear energy development, there was considerable interest in developing thorium fuel cycles. However, the interest in thorium fuel cycles ebbed due to improved availability of uranium. Countries like India never abandoned pursuing research and development on thorium systems due to vast reserves available in the country. The increasing demand for energy globally has brought thorium back on the agenda of several nations. Unlike natural uranium, which contains ~ 0.7% fissile U^{235} isotope, natural thorium (Th^{232}) does not contain any fissile isotope and therefore, thorium based fuels are utilized in combination with fissile U^{235} or Pu^{239} in reactors for conversion to U^{233} which is fissile. In addition, thorium based fuels and fuel cycles have intrinsic proliferation-resistance due to the formation of U^{232} due to interaction of fast neutrons with Th^{232} and subsequent reactions [IAEA, 2005]. The half-life of U^{232} is 73.6 years and its daughter products have very short half-life and some daughter products like Bi^{212} and Tl^{208} emit strong gamma radiations. As a result, there is significant build-up of radiation dose with storage of spent Th-based fuel or separated U^{233}. This necessitates remote and automated reprocessing and refabrication in shielded hot cells [IAEA, 2005].

One major limitations of thorium fuel cycle is that being only a fertile material, Th cannot achieve criticality on its own and it is necessary to use driver fuel to make reactor critical. Rapid advances in accelerator technology have made Accelerator Driven Systems (ADS) based on thorium a possibility in near future. Thus thorium fuels can complement not only uranium fuels thereby ensuring long term sustainability of nuclear power.

Overall, the major benefits of thorium fuel cycles are:

a) better thermo-physical properties like thermal conductivity ensuring better in-pile performance, and better chemical stability of ThO_2 as compared to UO_2 ensuring a more stable waste form,
b) lesser generation of long lived minor actinides and plutonium than the traditional uranium fuel cycle,

c) for countries looking for disposition of stocks of Pu, superior Pu incineration in $(Th\text{-}Pu)O_2$ fuel as compared to $(U\text{-}Pu)O_2$,

d) possibility of breeding even in thermal and epithermal flux environment in addition to fast neutron flux environment, and

e) Intrinsic proliferation resistance of U^{233} due to association with U^{232}, a high energy gamma emitter.

However, there are few challenges in deployment of thorium fuel cycles like

a) higher melting point of ThO_2 leading to requirement of higher sintering temperatures in fuel fabrication,

b) difficulty of dissolution of ThO_2 due to being a chemically inert matrix, and

c) need for developing remote and shielded facilities for U^{233} fuel fabrication due to presence of high gamma emission on account of U^{232}.

3.2.1 Thorium Fuel Cycle Developments in India

India has vast reserves of thorium and modest deposits of uranium. Thus there has been a strong incentive for development of thorium fuels. Since the inception of nuclear programme in India, many advances in the thorium fuel cycle have been achieved. ThO_2 pellets have been manufactured and used as blanket in fast breeder test reactor. Fuel bundles containing thoria pellets encapsulated in pins have been used for flux flattening in the pressurized heavy water reactors. Experimental thoria fuel pins have been irradiated and evaluated and also reprocessed for the recovery of U^{233}. The separated U^{233} has been used as Al cladded Al-20%U^{233} plate fuel assemblies in a 30 kWt research reactor KAMINI. A number of studies have been carried out on experimental fuel pellets and pins [Kakodkar, 2004].

An Advanced Heavy Water Reactor (AHWR) is under development which is a light boiling water cooled reactor of 300 MW (e) capacity, having vertical pressure tube type of design. It is being developed to obtain industrial scale experience in handling thorium and gain experience of safety features being proposed for GEN IV reactors. It has passive safety feature and is designed for on-power fuelling, using (Th, Pu)O_2 and (Th, U^{233})O_2 as the driver fuel [Sinha et al., 2006].

Thorium fuel cycle for the AHWR is shown in Fig. 2.1. The initial core of the AHWR will be loaded with mixed oxide (MOX) pin assemblies of (U, Pu) and (Th, Pu). These pins and assemblies will be fabricated in the fuel fabrication plant, which is a glove box type of facility. The discharged fuel from the reactor will consist of pins having U, Pu and U^{233}, besides Th and fission products. The cooled assemblies will be transported to the fuel assembly plant where the dismantling of the assemblies will be carried out either in the shielded hot cells or under water in a pool. The segregated (U, Pu), (Th, Pu) and (Th, U) pins will be sent to the reprocessing plant where the three stream aqueous reprocessing will yield nitrates of Pu, U and U^{233}. Th nitrate will be sent for storage and recovery, and the high level waste (HLW) will be sent for vitrification and interim storage. The nitrates of Pu, U and U^{233} will be sent to the reconversion laboratory for conversion to oxides. The oxide of U^{233} will be sent to the fuel refabrication plant which will fabricate (Th, U^{233}) MOX fuel pins in shielded and sealed hot cells. These pins will be sent to the fuel assembly plant in shielded flasks, to form the fuel assemblies for the equilibrium core consisting of MOX pins of (Th, Pu) and (Th, U^{233}). This assembly will be carried out in the hot cells or under water in the pool. The recharge assemblies will be sent to the fuel building of the AHWR for charging in the reactor. It is easier to implement safeguards in a reactor as the nuclear material is present in the form of sealed fuel elements. Implementation of safeguards in other plants is cumbersome as the nuclear material is in the form of solutions, powders, green pellets and sintered pellets.

3.2.2 Thorium Fuel Fabrication

A typical MOX fuel fabrication flow sheet is shown in Fig. 3.1. This flow sheet is for the powder-pellet route of manufacture, which is the preferred route for oxide fuels. The process starts by the blending of two or more powders. Blending is also an essential step to control the percentage of fissile isotopes and to obtain specified composition in the finished fuel. Often Clean Reject Oxide (CRO) is added at the beginning. The CRO is generated at various stages of fabrication, and has the chemical composition similar to the finished pellets. The CRO does not contain waste or impurity elements. CRO recycle helps in reduction of material hold up and also efficient recovery of fissile material during the entire fabrication process. After blending, the powders are mixed and milled in attritors. The next step is precompaction and granulation. This makes the powder free flowing. The mixed powder is then subjected to final compaction in a press.

Green pellets are formed in the stage of final compaction. These green pellets are then subject to sintering at high temperatures in reducing atmospheres. Till the stage of sintering, there is a lot of powder generation and these are the process areas responsible for higher material hold up and MUF. Such areas need special attention at the design stage underpinning the relevance of SBD. The sintered pellets are measured for diameter and oversized pellets are ground to final dimensions by the centreless grinding. Centreless grinding is another area where attention is to be paid for implementing safeguards measures, since a lot of dust and slurry is generated. The right sized sintered pellets are degassed and sent for stack formation and loading. They are loaded into bottom end plug welded zircaloy tubes. After loading of sintered pellets, along with the blanket pellets and other hardware, the top plug is welded under helium pressure. The welded and sealed pins are decontaminated and sent for appendage welding and assembly. It may be noted that quality control (QC) checks are carried out at a number of stages. At the intermediate stages of fabrication, the tests carried out on sintered pellets are dissolution test, dimensional measurement, linear mass, O/M (oxygen to metal ratio) assessment, total gas content, metallic and non-metallic impurities and autoradiography. For the finished pin, the QC checks are visual examination, helium leak testing, gamma scanning, cover gas analysis, metrology and X-radiography. The finished fuel pins are also checked for surface and fixed contamination after the step of decontamination [Gangotra et al., 2012]. The fuel fabrication is carried out in the glove box type of facilities in case of plutonium based fuels and in sealed and shielded hot cells in case of Th-U^{233} based fuels.

3.3 Proposing a Layout to Facilitate Implementation of Additional Safeguards Measures

The focus of the doctoral work is proposing and evaluating additional safeguards measures for powder pellet type of thorium fuel fabrication facilities. In powder pellet type of fuel fabrication facilities, the fabrication lines are generally laid out in a linear manner. The study includes proposing hybrid type of layout which has a number of advantages. Such layouts are also amenable for implementation of the new safeguards measures being proposed in this study. The two types of layouts for such fuel fabrication facilities are discussed in detail below.

3.3.1 Different Layouts for Thorium Fuel Fabrication Facilities

The conventional powder-pellet MOX fuel fabrication facilities are arranged in a linear layout, which houses the equipment according to the process steps as shown in Fig. 3.2. It shows two lines in a linear layout. The two lines are meant for movement of two different batches. Such types of production lines have inherent limitations. The linear layout is not easy to automate, due to limited space in each box for a conveyor system to be installed. Any mechanization is difficult to incorporate. In a linear type of layout, breakdown of any equipment down the line impairs the entire production. It is so because the process steps are sequentially laid out and bypassing of any individual box is difficult. Isolation and termination of any glove box housing the broken down equipment is difficult, if not impossible. A new layout has been designed in this study which is hybrid in nature mainly to overcome the shortcomings of a conventional linear layout. The hybrid layout shown in Fig. 3.3 is a layout with a common material transfer line in the central tunnel, having bifurcations connecting it with individual glove boxes / cells.

This layout also has two lines for different batches, but due to interconnectivity, the layout offers the movement of material of any batch to any of the boxes. The central tunnel is about 300 mm x 300 mm, having service ports at various intervals. The process material moves in the entire production lines in standardized stainless steel containers. These containers move on electromagnetic channels, such that inside of the tunnel has minimum of motorized or electrical installations. The hybrid layout offers a number of advantages over the linear layout. The central tunnel provides free movement of material in a manner that the material in the containers can be moved between any two boxes, without affecting the movement and operations in other boxes. This offers great flexibility in rerouting of material, and as a result, the total number of process equipment can be reduced. This is made possible due to sharing of equipment between the two batches moving in two lines. The hybrid layout is more adaptable to automation. The central tunnel is automated using electromagnetic mechanisms for container movement. The individual boxes can be isolated, if required either for maintenance or introduction of new equipment in the line. It can be done without stoppage of production since movement and operation in other boxes is independent and is not affected. Due to sharing of equipment by the two lines amalgamated in one hybrid layout, the overall redundancy for manufacturing can be achieved by lesser number of equipment. This also reduces the overall footprint of the fabrication line, in addition to reduction in total length of exhaust ventilation ducting. These have implications in reduction of material hold-up and total MUF in the plant. The hybrid layout is also more amenable for implementation of any fabrication flow design modification and introduction of new equipment for improvement.

3.3.2 Features of Hybrid Layout that enhance safeguardability

The hybrid layout for powder-pellet type of MOX fuel fabrication facility has inherent features that improve the safeguardability. As described above, the hybrid layout has a central tunnel which is helpful in automated movement of nuclear material in standard stainless steel containers. Higher level of automation limits manual intervention, thus improving physical security of the nuclear material by increasing challenge to theft of the nuclear material. Throughout the thesis, the term theft of nuclear material has been mentioned. Though the term used is "theft", it implies diversion by state or theft by non-state actors. The overall manpower requirement of the fabrication plant having hybrid layout with higher level of automation and lesser number of total process stations can be optimized such that the safeguardability is enhanced. To maximize safeguardability, level of manpower deployed in a nuclear facility needs to be optimized. A very low deployment would mean that there are areas in the plant which are deserted making theft easier. Conversely a large deployment of manpower could also reduce safeguardability due to exposure of nuclear material to a larger number of personnel.

To improve plant availability, it is necessary to incorporate adequate redundancy for critical processing equipment like attritors, pre-compactors, sintering furnaces, welding machines, decontamination set ups etc. In case of hybrid layout, where two batches are laid out with interconnectivity, the redundancy can be maintained with overall lesser number of equipment as equipment across the two lines can be easily shared. As a result, the linear layout requires more number of process stations compared to hybrid layout. Its overall impact is that in a hybrid plant the number of equipment come down, the footprint of the plant is reduced and the total length of the ventilation system is shortened. Lesser number of overall fabrication stations would provide lesser areas for presence of nuclear material, thus lesser chance of theft. If the CCTV cameras are installed for surveillance monitoring, the smaller footprint would need less number of cameras for overall coverage.

The reduced length of the exhaust ventilation ducting is beneficial for reducing the in-process material hold up and MUF. In-process material hold up and MUF is also reduced due to overall reduction in number of process stations, since the powder has a tendency to deposit at the walls of the glove boxes / hot cells and also to get lodged in the inaccessible and blind areas.

3.4 Proposing Additional Safeguards Measures for Powder Pellet Type of Thorium Fuel Fabrication Facilities

The powder pellet type of fuel fabrication facilities are either the glove box type (for (Th-Pu) MOX) or alpha tight hot cells (for ($Th-U^{233}$) MOX). In addition to inherent proliferation resistance of the thorium fuel cycle, there are extrinsic measures that can be implemented in the two type of facilities for improving the safeguardability of the facilities. Various measures can be employed in such type of facilities. Advancements based on Dynamic Nuclear Material Accounting (DNMA), Near Real Time Monitoring (NRTM) and Safeguards-by-Design (SBD) have been included in the study [Bjonard et al., 2010]. The author of this doctoral work has professional experience in the field of nuclear fuel fabrication, design of fuel fabrication facilities for plutonium based (glove box type) and thorium fuels (alpha tight hot cell type), operation in hot cells, design of hot cells for irradiated fuels and reactor components and implementation of IAEA safeguards in Indian facilities. Therefore, measures proposed in this study are a result of adaptation of experience of author's professional work [Gangotra et al., 2014]. These can be classified as A) Conceptual, B) Design Related, C) Engineering Related and D) Operational as given below:

Conceptual

1) Safeguards-By-Design
2) Co-location of facilities
3) Provision of nuclear material storage during physical inventory verification
4) Isolation of services

Design Related

1) Systems for plant imaging
2) Measurement of nuclear material inventory at every box / cell
3) Nuclear material tracking systems using RFID and bar codes
4) Overall reduction in the total number of items of process equipment
5) Footprint reduction of the plant
6) Reduction in ventilation ducting length

Engineering Related

1) Implementation of automation in the plant
2) Incorporation of efficient process powder recovery systems
3) Integration of process equipment
4) Integration of QC equipment with main process equipment

5) Improvements in equipment design

Operational

1) Implementation of near real time monitoring
2) Implementation of dynamic nuclear material accounting
3) Optimization of overall manpower deployment
4) Computerized tracking of nuclear material in the plant
5) Optimization of nuclear material flow in fabrication lines

Some of the above measures have common details and therefore, they have been combined in the detailed description given below:

3.4.1 Implementation of Safeguards-by-Design

SBD enhances the safeguardability of the facility by the incorporation of various safeguards measures right at the stage of design and has gained importance in recent times [Gangotra et al., 2012]. Merits of implementing safeguards at design stage are described in detail in 3.5. SBD implementation is not unique to fuel fabrication facilities and it can be implemented in any facility that needs to be safeguarded. The SBD implementation ensures optimal design of the plant, having integral systems for nuclear material accounting and containment and surveillance. A better control of nuclear material holdup and MUF is possible, which contributes towards enhancing proliferation resistance and criticality safety.

3.4.2 Co-location of fuel fabrication facility with reactor and reprocessing facilities

By reducing or eliminating the need for transportation of the nuclear material in the public domain, the proliferation resistance is greatly enhanced as the threat of diversion or loss of nuclear material is reduced. As mentioned earlier, a thorium fuel cycle facility has a number of plants including a fuel fabrication plant for the initial and the equilibrium core, a reprocessing plant, a fuel assembly/disassembly plant and associated waste handling and management plants. All these facilities can be located at one site or at different sites. Further, one thorium fuel cycle facility set up at a site can serve several reactors at the site. Alternatively, one can follow a hub and spoke approach with a large thorium fuel cycle facility acting as a hub, catering to the requirements of reactors at several sites as spokes. These two concepts have their respective merits and shortcomings in terms of engineering and economics as detailed in section 3.6 [Gangotra et al., 2013]. The ideal situation is the co-location of all, viz. reactor, fuel fabrication plant for initial and equilibrium core, fuel assembly/disassembly plant, fuel reprocessing facility and waste management facility at one site. This can greatly enhance the overall safeguardability of the nuclear material, by avoiding transport of the nuclear material in the public domain.

3.4.3 Provision of nuclear material storage during physical inventory verification

National safeguards authorities and the IAEA are the agencies responsible for the implementation of safeguards in nuclear facilities. Safeguards implementation procedures followed by the inspectors include annual Physical Inventory Verification (PIV). A day prior to PIV, Physical Inventory Taking (PIT) is performed. PIT involves moving the nuclear material present at various stages of fabrication to their respective Key Measurement Points (KMPs). It is easier to carry out such an exercise at item counting facilities. However, in a bulk handling type of facility, performing a PIT/PIV is tedious and can be facilitated by making provision of extra boxes, storage wells or vaults at specified locations nearer to KMPs in the plant. In addition to reducing the time for a PIT, such measures also give the plant operator designated storage spaces for moving the nuclear material when production is halted for any reason. As an added measure, all such areas can be provided with load cells for weighing of the material and also assay systems for estimation of the nuclear material. This can greatly enhance the overall proliferation resistance of the facility.

The hot cells in operation in India have shielded wells integrated with table tops. When the wells are covered with shield plugs, the space becomes table top for working. These wells are used as storage for nuclear and radioactive material during man-entry into hot cells for maintenance. This concept was incorporated while designing fuel fabrication facilities to move, secure and store intermediate nuclear material. This is also very useful during PIV, thus enhancing proliferation resistance.

3.4.4. Isolation of services from the main plant

The isolation of services enhances safeguardability by restricting access by personnel carrying out maintenance to the nuclear material. The glove box type or the hot cells type of facilities need more services due to the requirement for leak tightness and remote handling operations. Services include electric supplies, ventilation, compressed air, helium, argon and water supplies, and the waste management system. The helium and argon gas services are supplied in pipes with gas banks kept out of the plant. Ventilation systems include ducts for supply and exhaust air for both area ventilation and glove box/cell ventilation. The exhaust ventilation is connected to exhaust pumps and finally to the stack. The low level liquid waste is collected in the floor drains, which are connected to a liquid waste sump. Provisions are made for the collection of intermediate level liquid waste and high level liquid waste. The solid waste is generally collected and compacted for near-surface-waste-disposal.

Traditionally equipment providing various services such as diesel generator sets, battery banks, compressors, ventilation blowers, breathing air reservoirs, sump tanks etc.

are located in the near vicinity to reduce the length of pipes and wires. For enhanced safeguardability, a concept of isolation of the plant from services is proposed. In this configuration, the main plant is in a double fenced enclosure while the services are out of this enclosure in a nearby services area. This restricts entry of the maintenance and auxiliary staff to the main plant containing the nuclear material, thereby enhancing the safeguardability.

While designing fast reactor fuel cycle facilities, the main plants and service buildings were segregated in two separate areas. The idea was to outsource manpower for operation and maintenance of services. The entry of such personnel to designated areas would be easier for security agencies to implement. This concept was found to enhance proliferation resistance as well, and was thus added as a measure in this doctoral study.

3.4.5 Systems for plant imaging

Imaging using satellites has been used as a means of obtaining information about nuclear facilities [Wallace, 2009]. While satellite imaging relies on the images captured by the satellites [Niemeyer, 2009], the concept of plant imaging is based on cameras installed within the plant and is akin to the surveillance measures used for safeguards by the IAEA. The major areas in a powder-pellet type of fuel fabrication facility where the nuclear material is present are; the powder handling area, the pellet fabrication area, the pin fabrication area, the pin assembly area and the pin and assembly stores. There are other smaller areas where nuclear material is present in lesser quantities such as solid waste handling areas. Surveillance cameras installed in areas containing nuclear material help in tracking material movement and also detection of any theft. Different types of cameras used include continuous recording cameras, still cameras, motion detection cameras and night vision cameras. These cameras are used in combination in such a manner that complete coverage is assured during unattended periods or durations when such areas are not occupied.

A less extensive system would cover only the fabrication areas and storage areas for general viewing. Extensive systems would use cameras in every processing cell/box to completely cover areas where nuclear material is present. A hybrid layout would need fewer cameras compared to a linear layout as it has a smaller footprint. Judicious deployment of plant imaging systems for hybrid layout can greatly enhance the proliferation resistance in a fuel fabrication facility.

IAEA is using satellite imagery for observing the nuclear facilities. Similarly CCTV surveillance is being increasingly employed in industry, commercial establishment and even housing complexes. During the design of MOX fuel fabrication plants in India, CCTV surveillance was incorporated for security purpose. This concept enhances safeguardability as well.

3.4.6 Measurement of nuclear material inventory at every box/cell

Measurement of the inventory of the nuclear material at all the places where it is being handled in a plant greatly enhances its safeguardability. The nuclear material in a powder-pellet fuel fabrication facility starts in the form of a powder and ends up as a finished fuel pin or an assembly. It undergoes changes in the shape, the size and the form at different fabrication stations. In addition to the assay methods, the measurement of inventory at all the possible locations in the fabrication facility can be achieved by weighing at the starting and the ending point in each single cell/box. The movement of the nuclear material is carried out in standard containers. The first activity in any box/cell upon their receipt is weighing for which the provision can be made in the form of load cell based systems. The measurements can be directly coupled to the computerised material tracking system. Similarly, the last step in any unit box/cell is the weight measurement. By treating every single box/unit as an inventory monitoring station, the material can be tracked in real time, and a better estimate of material holdup and MUF can be made.

Post welding, the material is handled as an item counting unit. The weight of the encapsulated and sealed pin/assemblies can be measured using similar weighing systems at all subsequent processing stations. Such systems need space in every box/cell at the start as well as at the end. They also need provisions for electrical and electronics services with associated wiring for measurements and data transfer and a computer interface for integration with the main material handling system. The load cells also need to be designed to be radiation resistant since the nuclear material emits high gamma radiation. By making provisions for these systems, nuclear material accounting can be made more accurate and updated, both in terms of time and spatial location. Hence provision for measurement at each box/cell enhances the proliferation resistance of the nuclear material in such fuel fabrication facilities.

While designing the glove box process line of a MOX fuel fabrication plan, it was a requirement to track the material for safety (avoiding accumulation of nuclear material at one location), and process control. This was found to improve safeguardability and was thus included as another measure.

3.4.7 Nuclear material tracking systems using radio-frequency identification (RFID) and bar codes, and computerised material tracking

Bar codes and RFID chips are finding greater use in many applications both in the industry and consumer products [Liu et al., 2009]. This concept can be incorporated in the fuel fabrication plant for enhancing the safeguardability. The challenge, however, is to deploy tags which are resistant to gamma radiation and also to the harsh environment of a powder processing plant. Small stainless steel containers or units are used for storage and movement of the nuclear material. The containers holding the nuclear material are scattered all over the process areas of the fabrication facility. An RFID based container monitoring and tracking system helps in tracking the nuclear material. All the containers are provided with RFID tags. The tracking of such RFID tags needs detectors, amplifiers, receivers, and re-transmitters. For encapsulated and sealed fuel pins and fuel assemblies, the tracking can be done using bar codes, which can be engraved on the plugs by laser etching. Both the RFID systems and bar code systems can be integrated with the master computerised system for material tracking and nuclear material accounting. These measures for safeguards help track the nuclear material and reduce the probability of theft and thus increase the proliferation resistance.

During design of a MOX fuel fabrication plant in India, it was thought that if radiation resistance RFID tags could be sourced, they would be useful in tracking the containers in fuel fabrication plants. The etching of bar codes is already being practised for fabrication of fuels and assemblies of fast reactor fuel in India. It was realized that the advantage of both RFID and bar codes could be extended to improving proliferation resistance. Once it was decided to use RFID and bar codes for identification of containers of nuclear material and fuel pins and assemblies, it was logical to integrate it with the system for inventory control at every box/cell.

3.4.8 Overall reduction in the total number of items of process equipment

Overall lesser number of items of process equipment is desirable from a proliferation resistance point of view. To ensure a high plant availability factor, redundant items of equipment are installed for critical processes. In the case of hybrid layout explained earlier, the redundancy is maintained with an overall lesser number of items of equipment. If CCTV cameras are installed for surveillance monitoring, the smaller footprint will need fewer cameras for overall coverage. The reduction in length of the exhaust ventilation ducting helps in reducing the in-process material holdup and MUF.

The process powder has a tendency to get deposited at the walls of the glove boxes / hot cells, and also get lodged in the inaccessible and blind areas. Reduction in the number of process stations helps to reduce in-process material holdup and (MUF). Reduction in the total number of critical items of equipment such as sintering furnaces also has a bearing on the reduction of associated systems like furnace water cooling systems and filters. A lesser number of sintering furnaces drastically reduces the requirement for furnace water cooling which in turn reduces the overall footprint of the facility. Sharing of equipment like centreless grinders reduces the material holdup and thus results in reduction of MUF.

During design of a MOX fuel fabrication facility, the designers aimed at reducing the overall number of process equipment and thus the number of glove boxes. When this study on safeguards was taken up, it was realized that this reduction in total number of process equipment would also enhance proliferation resistance.

3.4.9 Footprint and ventilation ducting reduction of the plant

For identical production capacity, a fuel fabrication facility having a smaller footprint has better proliferation resistance. The number of surveillance cameras required is less for a smaller plant. The overall footprint of a fuel fabrication facility is dependent on the number of items of equipment and the number of glove boxes/cells. The footprint reduction for a plant can be achieved by a number of means such as having a hybrid layout and integration of some of the items of process equipment.

Lesser number of glove boxes and cells also leads to reduction in the requirement of services, especially the length of ventilation ducting. This is significant as the exhaust ventilation ducting is an area for material holdup and MUF. The hybrid layout has a smaller sintering furnace cooling water system. While efforts should be made to design the facility with a minimum footprint, the footprint as well as the length of the ventilation ducting of the plant cannot be drastically reduced.

During the design of a MOX fuel fabrication plant for plutonium fuels, the space available for the plant had constraints of footprint. The entire plant had to be accommodated in a limited area. There were also concerns of the operational cost since the plant was being designed to have once through ventilation with conditioned air. Similarly while designing the facilities for thorium based fuels (alpha tight hot cells type), the layout was so designed that the total length of ventilation ducting would be as short as possible. This was done so that the requirement for exhaust blowers could be optimized since longer lengths would need higher capacity blowers, which in turn would need larger air handling areas and larger exhaust filter banks. Longer length of ventilation would also increase the operational cost since the exhaust blowers are the critical equipment in such facilities and are required to operate continuously with adequate redundancy. It was realized that shorter length of ventilation ducting would mean lesser MUF and thus advantageous from the proliferation resistance perspective.

3.4.10 Implementation of Automation in Plant

Most of the operations in a powder-pellet type of MOX fuel fabrication plant are carried out manually. Incorporation of automation reduces the risk of theft by an insider by minimising access to nuclear material by operators. The central tunnel in a hybrid layout (described earlier) is useful for automated movement of nuclear material. The nuclear material is stored and transported in standard stainless steel containers. Systems for automatic dosing and blending of feed powders can be designed to draw different powders into the process straight from the storage canisters. The fabrication steps for attrition, pre-compaction, granulation and final compaction can also be automated. Remote loading of green pellets in boats into sintering furnaces can also be easily automated. The subsequent movement of boats containing pellets after sintering can be automated to deliver pellets to the pellet inspection and quality control systems. Manual inspection of the pellets can be replaced by vision based computerised systems, which accept the right size pellets, deliver undersized pellets to recycle bins and route oversize pellets to the centreless grinding station. From the pellet inspection station, the accepted pellets are transferred to the pin loading system and finally gas filling and end cap welding machine, preferably by laser welding. The finished pins are then sent to the automated decontamination station for laser decontamination. Systems can be designed to handle encapsulated pins for transfer to QC check stations and finally to the assembly area for formation of fuel assemblies. The task of making a fuel assembly from individual pins can also be automated. In addition to the above steps where automation can be implemented, the intermediate operations can also be automated to avoid manual handling. Such operations include laser engraving on the encapsulated pins, identification markings on the finished assemblies, sampling from powder for analysis, inspection of pellets by a pellet inspection machine, measurement of weight of bulk material in containers and item counting of finished pins. The implementation of automation in any plant restricts the access to nuclear material by the manpower operating the plant. However, large scale automation also requires frequent maintenance. The maintenance staff has access to the processing areas, only after the nuclear material has been removed to secured interim storage locations.

As will be shown in Chapter 6 automation is one of the features in a bulk handling facility that significantly improves safeguardability. An added advantage of automation is the reduction in the radiation exposure to the operators.

At the time of designing of the MOX fuel fabrication plants, a lot of effort was made to automate the process to the extent possible. This was required to reduce the radiation exposure to the operators and also to achieve higher throughput. Automation turned out to be highly beneficial when evaluated from the proliferation resistance view point.

3.4.11 Incorporation of efficient process powder recovery systems

To improve proliferation resistance, efforts must be made to reduce both the material holdup and MUF in the facility [Beckers et al., 2004]. The process stages with higher occurrence of powder generation and airborne activity are the blenders, attritors, precompactors, final compactors, centreless grinders and crushers for recycling of CRO. As an enhanced measure to recover the nuclear material, dedicated systems for powder recovery can be provided in these areas and boxes/cells. This can be achieved by having closed recirculatory systems consisting of suction devices, powder filters and collectors and pumps that help in an efficient recovery of the process powder. By having additional High-Efficiency Particulate Absorption (HEPA) filters in the exhaust ventilation of such areas, the airborne powders can be arrested significantly before they travel far. As discussed earlier, if more than one item of equipment is integrated into a single module, the overall powder generation and loss is reduced leading to reduction in material holdup and MUF. These provisions can be made both for glove box type and hot cell type facilities. It may be noted that the reduction in powder generation and enhanced powder recovery also help in reducing the risk of criticality hazard.

In the older fuel fabrication plants currently in operation in India, it was seen that powder generation in certain operations was a drawback. The decontamination of these boxes and equipment always posed difficulties. The designers studied this problem and were tasked with finding effective solutions which could be incorporated in the green field projects. Such systems were designed and demonstrated to be included in the new plants. Powder recovery turned out to be major factor in improving proliferation resistance.

3.4.12 Integration of process equipment

The overall numbers of glove boxes or the hot cells required for fabrication are based on the total number of process steps, capacity of the facility and redundancy of equipment required for fabrication. Some of the operations can be integrated in a manner that leads to replacing several machines or items of equipment by a single integrated work station. Lesser number of items of equipment implies that fewer glove boxes or cells are required for fabrication. This can reduce marginally the overall footprint of the plant and also the total length of the ventilation system. Manpower needed to operate this equipment will also be less. Moreover, the material holdup and Material Unaccounted For (MUF) are lower as the material that would have been spread over multiple stations is now confined to a single station. Hence, it is desirable to integrate as many process operations as possible. Some possible examples of integration of process equipment are given below.

The fuel fabrication process begins with the weighing of various feed powders, along with recycle scrap, and subsequent blending. These steps can be integrated by having a single station comprising of the powder dosing system along with blender. The next steps of pre-compaction, granulation and final compaction can also be combined in a single station performing all these operations. The operations of boat unloading, pellet inspection and stack formation can be integrated. The integration of stack loading, cover gas filling and top plug welding into a single unit can also be engineered. This also includes the fabrication step of laser engraving on top plugs for identification. It has been estimated that though integration of process equipment is desirable, it is not possible to achieve a very high level of integration, due to the fact that the various operations are diverse and the material transforms from bulk handling to item counting during the course of fabrication.

Integration of process equipment was taken up during design of fabrication equipment, both for plutonium fuels and thorium fuels. The intent was to use gravity for feed for multiple operations and also facilitate automation. It was later found useful for increasing proliferation resistance as well.

3.4.13 Integration of QC equipment with main process equipment

The QC samples are drawn for characterization of feed and blended powder, pre-compacts, green pellets, sintered pellets and welded pins. Samples, intermediate products as well as finished products, are drawn at various stages of fabrication for chemical analysis, spectrometric analysis, micro-structural analysis and non-destructive evaluation like leak testing, radiography, ultrasonic testing and metrology. In a conventional fabrication facility, quality control equipment is separated from the main process equipment. If the QC equipment is integrated with the main process equipment, it will eliminate the need to withdraw samples for analysis. This will reduce the number of exit points for the nuclear material in the line.

In the case of glove boxes, sampling for QC involves bag-in and bag-out operations and in the case of hot cells material transfer. These operations need manual intervention. The hybrid layout is amenable to integration of all QC equipment and boxes/cells within the central tunnel. Boxes/cells housing the QC equipment are placed at locations closer to their stage of fabrication. This ensures no withdrawal of nuclear material in any form from the fabrication line for the purpose of QC sampling. Though the total amount of material withdrawn for QC sampling is very small, complete elimination of this operation can greatly enhance the safeguardability of the plant. In the case of hybrid lines, there are only two points of material transfer, one for entry and the other for exit of the finished product. Integration of the QC equipment with the main line can enhance the proliferation resistance of the nuclear material handled in such a fabrication facility by reducing the total number of exit points. However, integrating all QC equipment with main process line will be challenging from scientific and engineering point of view.

The idea to integrate QC boxes with process boxes emanated from the need to avoid the operations of bag-out and bag-in of samples for Quality Control in plutonium fuel fabrication facilities in India. The designers initially decided to have a pneumatic rabbit transfer system using shuttles carrying samples. Later it was decided to integrate the QC boxes with main process boxes. This led to the idea of increased proliferation resistance since the nuclear material was confined to the process and QC lines.

3.4.14 Improvements in equipment design

The design features of the glove boxes, hot cells, ventilation ducting and the fabrication equipment have a bearing on the MUF and material holdup. The nuclear material flowing through these has a tendency of deposition and getting lodged in recesses, corners and blind spots making its recovery difficult. The surfaces where the powder is likely to come into contact can be given a smooth finish so that the deposition is minimal and it is easier to recover the nuclear material during clean up. The equipment can be designed in such a manner that all holes, tappings, crevices, openings etc. are covered and any difficult to clean sharp corners and blind spots are avoided. Each and every equipment installation should be assessed from the point of view of accumulation of nuclear material and powder recovery and relevant design changes should be made to minimise the nuclear material loss.

The accumulation of nuclear material and radioactive material has always been a concern in fabrication facilities and hot cells. As a designer, the author of this study has always kept the need to improve design of such equipment to avoid blind spots, sharp corners, grooves, etc. This has been found to improve proliferation resistance by reducing the MUF.

3.4.15 Implementation of Dynamic Nuclear Material Accounting (DNMA) and Near Real Time Monitoring (NRTM)

DNMA and NRTM systems can help in improving proliferation resistance by an early detection of any theft of the nuclear material [Ninagawa et al., 2010]. An on-line nuclear material accounting system consists of measuring equipment, its placement in specific locations, data acquisition and analysis systems and data storage and transfer systems. Unattended Non-Destructive Analysis (NDA) has been developed by some designers, which collects and transmits data of nuclear material movement. Additionally, systems have been developed for nuclear material accounting for a glove box assay, a fuel pin assay and a waste drum monitoring system. As a part of measures for safeguards, such monitoring systems based on neutron and/or gamma measurements can be customised in a manner that all relevant areas for measurements are covered. An important area for placement of such systems is the exhaust piping of the glove boxes/cells where the powder generation is large, such as the blending station, the attrition, pre-compaction and centreless grinding station. An adequate number of such systems, placed at designated locations can help in efficient material accounting and thus improve the proliferation resistance.

3.4.16 Optimisation of overall manpower deployment

A fuel fabrication facility needs manpower for various operations like fabrication, quality control, health physics, physical protection, safety, maintenance and services. From the viewpoint of reduction in possibility of theft of the nuclear material, an optimum number is desirable for a given size and capacity of the facility. Too small a number of plant personnel can pose higher risk of theft due to the fact that several areas of the plant, where the nuclear material is present either for processing or storage, are left unattended for long durations. This increases the vulnerability of theft of the nuclear material. Too large a manpower also increases the possibility of theft, due to the presence of more personnel who might be tempted to steal the nuclear material. There are a number of ways in which the requirement for manpower can be reduced. Automation in different segments of the plant greatly reduces the manpower needed for fabrication. However, it does increase the manpower needed for the maintenance. But the maintenance in automated lines is initiated only after the nuclear material is removed to secure locations. Integration of various items of process equipment also reduces the number of personnel needed for fabrication. Fewer items of equipment need less manpower. The hybrid layout needs less manpower due to lesser overall number of items of equipment for fabrication. As described earlier, the isolation of services from the main process plant reduces the possibility of maintenance staff gaining access to areas containing the nuclear material. Thus a judicious mix of various methods can be employed in the fabrication facility, so that optimum number of personnel is present in the designated areas. Such optimisation can greatly enhance the proliferation resistance of nuclear material in the fuel fabrication facility.

It is a part of design of fabrication facilities to identify the manpower needed for operation and maintenance. This is required to decide whether the plant would be operated round the clock or in two shifts. The criterion at the time of design was to keep minimum staff since the manpower required to operate such facilities is highly skilled. It became an interesting point of this study later, since the proliferation resistance is maximum at an optimum level of manpower.

3.4.17 Optimisation of nuclear material flow in fabrication lines

Optimisation of nuclear material flow can help in reducing the risk of theft by restricting the amount of the nuclear material present in process lines to the minimum. One way of achieving this is to adopt a hybrid layout, where the nuclear material moving in the central tunnel can be routed to any box/cell. This helps simultaneous processing of more than one batch in the fabrication line. However, different operations of fabrication need different time durations and therefore, the throughput of the plant is determined by the slowest step in the fabrication which is the sintering of green pellets [Danny et al., 2007]. Processing of multiple batches is facilitated by computerised simulation. Simulation also helps in re-routing of material flow in case of an unexpected failure or malfunction of equipment at any station [Chakraborty et al., 2009]. By such optimisation, the nuclear material in the fabrication lines is so controlled that just the right amount is present in the lines at any given time. This greatly reduces the risk of theft and hence improves the overall proliferation resistance of the fuel fabrication facility.

While designing plutonium and thorium fuel fabrication facilities, the total number of processing equipment had to be optimized. This was required since lesser number would mean longer fabrication time and hence lower plant capacity. On the other hand if the number of equipment were more, the penalty would be on space, operational cost, maintenance and decontamination. The design team was tasked with the goal of working out optimized material flow so that maximum utilization of process equipment could be achieved without affecting the overall capacity of the plant. It was realized during this study that this factor would improve proliferation resistance by releasing the nuclear material in the process lines for minimum time thereby reducing the risk of theft.

3.5 Merits of Implementing Safeguards Measures for Fuel Fabrication Plants at Design Stage

A part of the thesis work is to identify and study safeguards measures that can be implemented as SBD in thorium based powder-pellet fabrication plants. SBD has been described in Section 3.4.1. The current section details these SBD measures specifically identified for such fabrication facilities.

Safeguards-by-Design helps in enhancing the safeguardability of a nuclear facility [Ninagawa et al., 2010; IAEA, 2003; INL/EXT-09-17085, 2010; DeMuth et al., 2010]. Safeguards implementation in bulk handling facilities like fuel fabrication facilities is a challenging task, as compared to item counting type of facilities. Moreover, fabrication of fuel in glove box or alpha tight hot cells type of facilities requires extensive measures for safeguards due to complexity in remote handling, material hold up in ventilation systems, process hold ups, manipulation and constraints of access. Effective implementation of safeguards in such fuel fabrication facilities, calls for novel measures, both intrinsic and extrinsic. It is best to incorporate all such measures at the design stage itself and this has led to the concept of Safeguards-by-Design (SBD). The SBD concept involves incorporation of safeguards measures from the stage of conceptual planning of the facility leading their integration with the plant processes on the drawing board stage itself. This reduces cost of safeguards implementation by avoiding retrofitting at a later stage. Improvised methods using dedicated instruments for nuclear material accounting and material balance can be engineered to provide data required for safeguards. SBD can also be designed to obtain safeguards data in near real time monitoring mode. Early investment in SBD helps reduce hold up inventories and MUF. The added advantage of such measures is close control of inventories and avoidance of criticality due to buildup of fissile material in ducting, blind areas of the fabrication lines and equipment. Safety, security and safeguards are an essential part of any nuclear facility. SBD can help integrate these three aspects, resulting in reduction of total equipment inventory and overall cost. In fuel fabrication facilities for thorium based fuels, the proposed SBD concepts can be implemented in the following;

a) Incorporation of process powder recovery systems (E)

b) Integration of quality control equipment with main processing equipment (E)

c) Systems for dynamic nuclear material accounting / near real time monitoring

d) Incorporating systems for plant imagery (D)

e) Isolation of services from the plant (C)

f) RFID and bar codes based systems for material tracking (D)

g) Provision of inventory measurement at every cell / glove box (O)

h) Provision of dedicated equipment for measurement of material hold-up and MUF (D)

i) Provision of systems for material storage during physical inventory verification (D)

j) Installation of portal monitors for personnel scanning (D)

k) Integration of safety, security and safeguards systems (C)

(C: Concept; D: Design; E: Engineering; O: Operational)

A good practice would be to implement SBD measures at these facilities right from the stage of conceptual design. This will go a long way in efficient implementation of the safeguards measures.

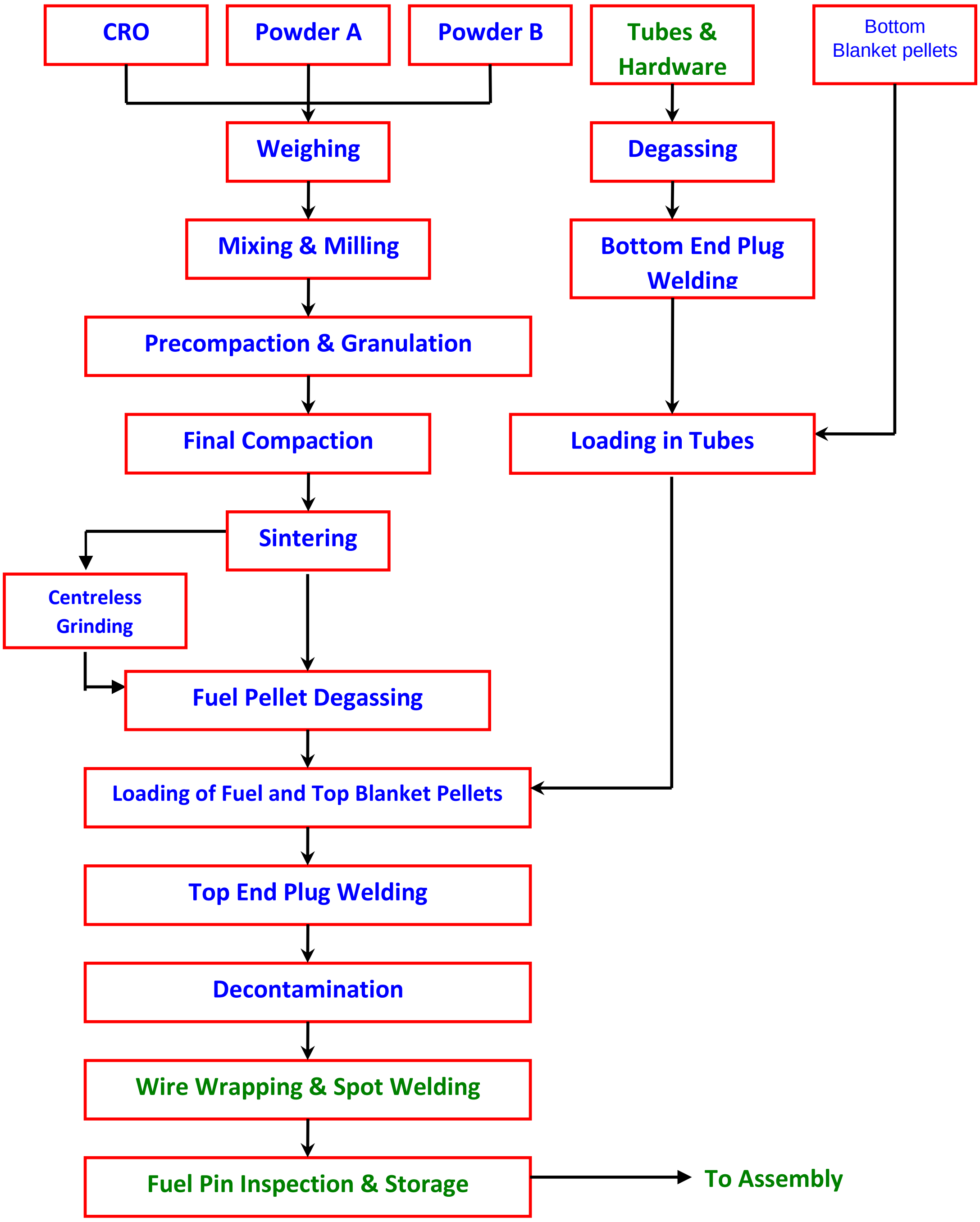


Fig. 3.1: The figure shows the MOX Fuel Fabrication Flow Sheet

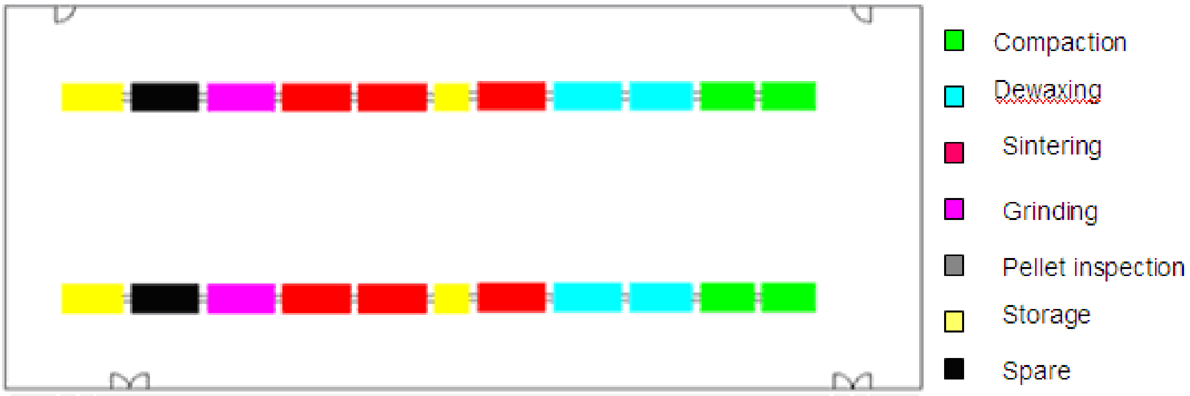


Fig. 3.2 : The figure shows Linear Layout of Powder Pellet MOX Fuel Fabrication Facility

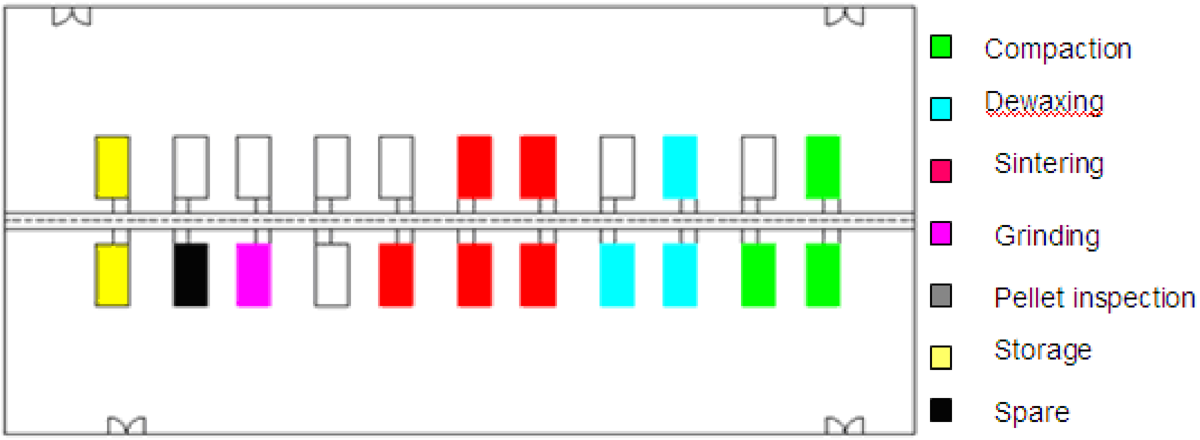


Fig.3.3: The figure shows Hybrid Layout of Powder Pellet Fuel Fabrication Facility

Chapter 4

Co-location: Possible Configurations and Comparison Thereof

4.1 Co-location: Possible Configurations

It is envisaged that when India decides to set up thorium reactors, there will be many reactors at a site and several sites at different places over the country. Fuel cycle facilities can be arranged in various configurations. The configuration as shown in Fig. 2.1 is for a small fuel cycle facility dedicated to reactors at a single site. Such a concept envisages several small sized fuel cycle facilities each associated with reactors located at different sites. This configuration will make a site self-contained and is referred to as self-contained concept. Alternately, a large fuel cycle hub can be designed, which could cater to several thorium based reactors in the country as shown in Fig. 4.1. This is referred to as hub and spoke concept. Decision to adopt a particular configuration for deployment of thorium fuelled reactors on a large scale with their associated fuel cycle systems will depend on a number of factors like engineering challenges, economics and power requirement of various regions of the country. Safeguardability and proliferation resistance will also be two of the deciding criteria, since large amounts of nuclear material will be handled at the reactors and the fuel cycle facilities [Cojazzi et al., 2008].

The two concepts have been studied in terms of application of safeguards measures [Gangotra et al., 2013]. These are described below:

4.2 Automation

The advantage of having automation in fuel fabrication facilities is described in section 3.4.10. Both, in case of a large fuel cycle facility set up as a part of hub and spoke concept and small fuel cycle facilities set up as part of self-contained concept, the level of automation and its impact on safeguardability are similar. An important process to be automated is the transfer of nuclear material in the form of small quantities of samples from various areas of the fabrication plant to the on-site safeguards laboratory. It is envisaged to have one safeguards laboratory in India which will cater to requirements of all the safeguarded facilities. In a hub and spoke concept, this transfer can be engineered using pneumatic rabbits. These systems exclude the intervention and handling by plant personnel, thus enhancing safeguardability. In case of few dedicated fuel cycle fuel facilities, such transfers will have to be made using transport in public domain, thus reducing the safeguardability of nuclear material during transit. It may be added that the total amount of such nuclear material will be small, since these will only be samples for measurements and calibration. Generally the weight of such process monitoring samples is in the range of a few grams.

Hence, it can be concluded that in terms of automation and its effect on safeguardability, a hub and spoke concept has a slight edge over self-contained concept, due to automated transfer of samples from the fabrication lines in a plant to the safeguards laboratory.

4.3 Integration and Reduction of Process Equipment

Section 3.4.8 describes the merit in reduction of process equipment and section 3.4.12 describes the advantage of integration of process equipment in fuel fabrication facilities. In the self-contained concept, the dosing and blending unit for feed powders are required for every separate facility. In case of the hub and spoke concept, one dosing and blending unit can cater to different plants and lines. In such a plant, the overall MUF and material hold up will be much less than in case of smaller facilities. Moreover if one compares the overall footprint and ventilation ducting length, it would be smaller for the hub and spoke concept as compared to the self-contained concept. This translates to lower MUF and material hold up in the process lines.

Another impact will be on the manpower. Integrated equipment and lesser number of equipment need lesser manpower compared to manpower requirement in a plant having no integration and large number of process equipment. Smaller number of equipment also need reduced load on near real time monitoring systems, and implementation of dynamic nuclear material accounting system would be easier.

Thus, it can be said that, from the consideration of equipment integration and overall equipment reduction, it is beneficial to adopt a hub and spoke concept rather than a self-contained concept.

4.4 Effect of Quality Control Operations

In the self-contained concept, every fuel cycle facility will have its own quality control laboratory, preferably a single facility catering to the requirements of all the plants in the facility. The samples from various fabrication plants and process areas of the plant will be transferred to the laboratory by dedicated pneumatic rabbit transfer systems. However, in the hub and spoke concept, a similar single laboratory for the entire complex will suffice. On comparison of the two concepts, it can be observed that in the case of a self-contained concept, the proliferation resistance of the plant is inferior as compared to the case of the hub and spoke concept, since nuclear material in the form of sample will be handled in just one single laboratory, having associated sample transfer systems. There will not be many points of exit for the nuclear material either from the process lines or the fuel fabrication plants.

4.5 Overall Footprint of the Facility

Self-contained concept spread over several sites will require larger total area compared to a hub and spoke concept of equivalent capacity. In case of hub and spoke concept, the plants in the complex can be designed to have multiple lines for different fuels, both for reprocessing and fabrication. Moreover, plants like fuel assembly / disassembly plant, can cater to the requirement of all the different reactors. In the case of self-contained concept, each site will require its own fuel assembly / disassembly plant. Thus a hub and spoke concept will have just one assembly plant while the self-contained concept will require as many fuel assembly plants as the number of sites. One single quality control laboratory can serve the requirement of all the plants and their products in a hub and spoke configuration, while the self-contained concept will require quality control laboratory at every site. For smaller facilities, different lines will be housed in different plants, complete with their associated buildings for services. By having all plants at one location, all the services can be shared. Hence the overall footprint will be smaller. An important gain in terms of proliferation resistance will be in the reduction of total MUF and also material hold up, because of comparatively smaller ventilation ducting length.

Hence, from the consideration of footprint alone, it is advantageous to adopt a hub and spoke concept rather than self-contained concept.

4.6 Dynamic Nuclear Material Accounting / Near Real Time Monitoring Systems

Importance of implementing DNMA/NRTM is described in section 3.4.15. While both the concepts of thorium fuel cycle facilities would need comparative systems and equipment for DNMA and NRTM, a hub and spoke concept could provide for one on-site laboratory, exclusive for the purpose of safeguards. Such laboratory can have equipment and systems for destructive and non-destructive analysis of different samples from all the plants.

The laboratory will also be useful for calibration of various on-line equipment installed inside the plants and in process and fabrication lines for nuclear material accounting.

Hence, a hub and spoke concept would offer better proliferation resistance of nuclear material compared to self-contained concept, by incorporation of on-site safeguards laboratory.

4.7 Isolation of Services

The concept of isolation of services is described in section 3.4.4. On comparison, a hub and spoke concept will have higher proliferation resistance compared to self-contained concept. This is due to the fact that the single facility will have just one double fenced island containing all the nuclear material, while the self-contained concept will require one island per site.

4.8 RFID, Bar Code Readers, Transmitters and Receivers

The merits of using RFID and barcodes in fuel fabrication facilities is given in section 3.4.7. For comparison, a hub and spoke concept will have one common system while self-contained concept will require separate systems for individual sites. The two systems will help in improving proliferation resistance, but a single system will be compact and complete in itself, covering the entire nuclear material of the safeguarded thorium facilities out of reactors. Thus, a hub and spoke concept may have a marginal advantage over self-contained concept.

4.9 Plant Imagery

Concept for plant imaging in fuel fabrication facilities is given in section 3.4.5. In a hub and spoke concept since individual plant size would be large, it would be easier for the satellite cameras to detect any change, as compared to a self-contained concept.

Thus a larger facility would offer better proliferation resistance, since a smaller facility could camouflage constructions and expansions, as the footprint of the buildings for expansion will be much smaller for such plant. In respect of in-plant surveillance imagery, both the concepts will provide similar safeguardability, though the total number of cameras needed may be lesser for one single facility, without compromising the equivalent overall coverage.

4.10 Co-Location

The concept of co-location and its impact on fuel cycle facilities is described in section 3.4.2. In a self–contained facility, the movement of nuclear material is between fuel fabrication plant, fuel assembly / disassembly plant, reactor, reprocessing plant, fuel refabrication plant and characterization laboratory. In a hub and spoke concept, there will be requirement of transporting fresh fuel to various reactor sites and transporting spent fuel from the reactors back to the fuel cycle facility. It is due to the fact that the single fuel cycle facility will operate as a hub, catering to various reactors located at different sites. Thus the transport in public domain will be a necessity and cannot be avoided. The security of nuclear material while in transport is of great importance. The transport of the nuclear material by surface, involves multiple agencies and has implications for safety, security and cost. During transport, the safeguardability has to be ensured by escort services provided by law enforcing agencies, containment and surveillance measures as well as internal remote monitoring using satellites. For this reason, the self-contained concept, will offer higher proliferation resistance as compared to a hub and spoke concept catering to different reactors.

4.11 Manpower

Section 3.4.16 describes the usefulness in optimising the manpower in a fuel fabrication plant. On a comparative basis, a hub and spoke concept will warrant relatively lesser manpower than the self-contained concept for equivalent capacity. It is possible since the total number of equipment will be less in a larger plant, and many services like quality control, fuel assembly, and disassembly can be shared by various fuel configurations. Thus from the consideration of proliferation resistance of nuclear material, it will be advantageous to adopt a hub and spoke concept rather than self-contained concept.

4.12 Integration of Safety, Security and Safeguards Systems

A hub and spoke concept can be designed with one common integrated system for safety, security and safeguards. In a self-contained concept, number of integrated systems will equal number of sites. However, it is expected that these facilities may not be integrated with one another, due to separation in sites. Thus there will be multiple systems in case of self-contained concept. It may be beneficial to adopt a hub and spoke concept having an integrated system for safety, security and safeguards, since all the facilities will be covered by single system. The physical protection systems will also be more robust and easier to design.

4.13 SBD Implementation

The usefulness of SBD measures for fuel fabrication facilities is given in sections 3.4.1 and 3.5. Both the concepts can greatly benefit by incorporation of SBD measures. In the case of self-contained concept, the SBD measures will have to factor in the requirements of additional reactors at the same site. It may be difficult to implement if the schedules of construction of additional reactors at the same site and the fuel cycle facilities are vastly different, or construction of additional reactors at a given site is not envisaged in the beginning. The requirement for additional reactors may emerge much after the fuel cycle facilities have been constructed. In a hub and spoke concept, the SBD measures will be unique to the facility itself and are less likely to change. Any provision for capacity expansion would be factored in at the time of design of the integrated facility itself. Hence, from a viewpoint of SBD implementation, a hub and spoke concept has significant advantages.

4.14 Consolidated MUF and Material Hold Up

MUF and material hold up in any plant or a facility is of great concern from the point of view of safeguards and safety. It is desirable to keep the total MUF and material hold up in a facility as low as possible. The nuclear material in the form of fine powder has a tendency to deposit on the walls of equipment and cells. Similarly in case of liquids in reprocessing plant, there is piping, valves, vessels where the estimation of nuclear material is difficult. In fuel disassembly plant, nuclear material from the failed fuel may leach out to water in the pools, thus making its estimation difficult. For a single facility of a large capacity, the total MUF and the material hold up will be large, since the total nuclear material handled is large This is because there will be more number of equipment, longer piping, vessels, areas and ducting where nuclear material can be lodged, which is difficult to recover or estimate. However, for different smaller facilities, the total MUF and the material hold up for each plant will be individually lower.

The MUF and material hold up is specified as a percentage of the throughput of the plant as a whole. In a hub and spoke concept, the total throughput will be large and consequently the absolute value of MUF and material hold up will be large and may exceed the values of significant quantities. Thus from the consideration of total MUF and material hold up, it would be advantageous to adopt a self-contained concept rather a hub and spoke concept.

4.15 External Events

External events that may cause damage to structures, emergency services, fuel storage pools, critical utilities can affect safety and security of nuclear facilities and have become focus of attention post Fukushima event. While Fukushima event was caused by a tsunami that followed an earthquake, other events that can have serious consequences include flooding, fires and tornadoes.

In a hub and spoke concept, if the complex is stuck by such a disaster, then the damage can be extensive and the nuclear material may be lost. However, for a self-contained concept, the damage will be restricted to only the site affected, and is unlikely to have effect on other facilities situated elsewhere. Hence, from the viewpoint of safeguard of nuclear material alone, disregarding the safety and security concerns, the nuclear material in a hub and spoke concept has higher vulnerability as compared to self-contained concept, in the event of large scale calamity. Thus, it may not be advantageous to adopt a hub and spoke concept.

4.16 Summary of Merits and Limitations of Hub and Spoke Configuration

A thorium fuel cycle facility can be set up to serve reactors at a site. Alternatively, one can follow a hub and spoke approach with a large thorium fuel cycle facility acting as a hub, catering to the requirements of reactors at several sites as spokes. The two concepts have their respective merits and shortcomings in terms of engineering and economics. In summary, the factors which favour the hub and spoke concept, catering to all the thorium based power reactors in the country are;

1) Implementation of automation in the plants of facility
2) Integration and overall reduction in number of process equipment
3) Integrated quality control
4) Footprint reduction
5) Implementation of Dynamic Nuclear Material Accounting / Near Real Time Monitoring

6) Isolation of services from the main plant

7) Nuclear material tracking systems using RFID and bar codes

8) Systems for plant imagery

9) Optimisation of overall manpower deployment

10) Integration of safety, security and safeguards systems

11) Implementation of SBD measures

However, in the case of a single thorium fuel cycle facility for all the thorium based power reactors in the country, there are factors which will reduce the overall proliferation resistance. These factors which will affect the safeguardabilty adversely are;

1) Total MUF and material hold up in the facility

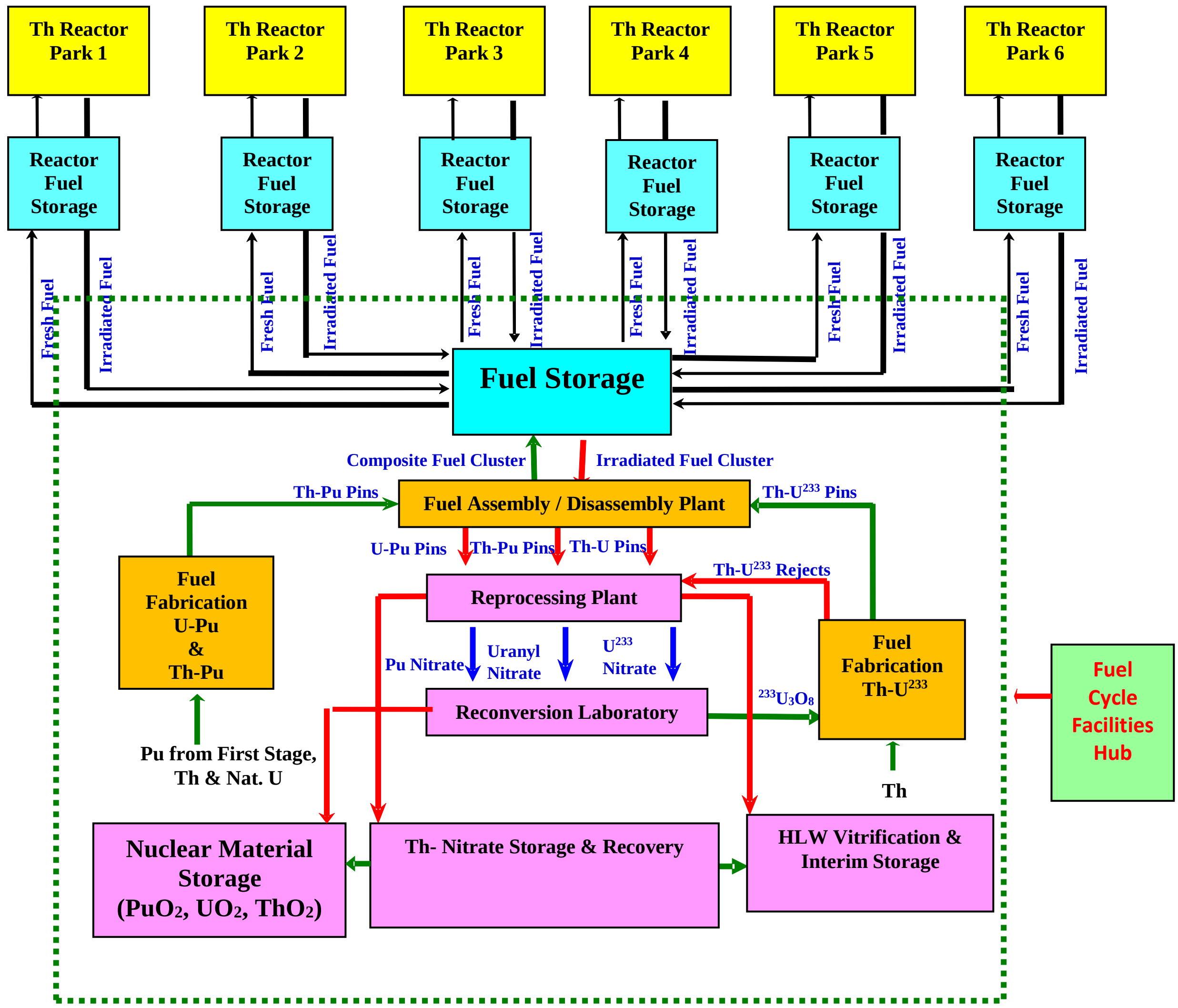
2) Vulnerability to nuclear material loss in case of a severe external event

3) Transportation of nuclear material in public domain due to location of reactors away from the fuel cycle facility

It has been seen that having a hub and spoke concept for thorium fuels will have relatively more merits than the self-contained concept. Deployment of large scale automation in various plants, integration of equipment, quality control integration, islanding by isolation of services, plant imagery systems, optimisation of manpower, integration of safety, security and safeguards system and SBD play an important role in increasing proliferation resistance for both the concepts. They are easier to implement and have more impact in case of a hub and spoke concept. One area of emphasis is the incorporation of on-site safeguards laboratory.

While considering a hub and spoke concept, efforts will have to be made, both in terms of design and operation so that overall MUF and material hold up is kept to a minimum. On the other hand, while considering a hub and spoke concept, efforts will have be made both in terms of design and operation, that overall MUF and material hold up is kept to a minimum; appropriate measures taken while designing to take care of nuclear material safety and security during severe external event; and the security and safety during transport of both fresh fuel and spent fuel in the public domain.

There are other factors like safety, economics, technology etc. that play an important role in deciding the scheme to be adopted. The assessment presented in the study will be useful in final conceptualization, design and deployment of thorium based fuel cycle facilities in India.



The fuel storage is in two parts: One for fresh fuel and the other for irradiated fuel

Fig: 4.1 The figure shows schematic for Hub and Spoke configuration of Thorium Fuel Cycle Facilities

Chapter 5
Expert Opinion

5.1 Selection of Experts

Twenty measures for enhancing proliferation resistance in nuclear fuel fabrication facilities for thorium based fuels were proposed in Chapter 3. The next step is to analyse the relative contribution of various measures towards proliferation resistance of a facility. The incorporation of these measures will have a significant bearing on the overall proliferation resistance. Expert opinion was sought on the impact of these measures on the overall proliferation resistance. India has a qualified and trained manpower in the field of nuclear fuel fabrication for uranium, plutonium and thorium based fuels. The country has research scale, pilot plant scale and industrial scale fabrication facilities for different fuels. Expertise is available in such diverse fields of nuclear fuel cycle like operation and maintenance, quality control, design of facilities and implementation of safeguards in fuel fabrication facilities. The experts for feedback were carefully chosen based on the knowledge and experience of a particular expert in the related area of specialization. It is significant to note that many experts have knowledge and experience in more than one areas of fuel fabrication as a result of their job assignments in the past. The experts, included eight designers, six operators, three quality control managers and seven implementers of safeguards in fuel fabrication facilities. Some of them have had previous assignment in a category that is different from their present assignment. In this study, some experts have composite expertise, e.g. a senior designer also has an expertise on plant operation and an expert on implementing safeguards has experience as quality control manager. A total of 24 experts were selected and board areas of their expertise is as follows:

a) Design of glove box and hot cell based fuel cycle facilities.

b) Fabrication of plutonium and thorium based mixed oxide fuels.

c) Fabrication of natural uranium based fuels.

d) Quality control of uranium and plutonium based oxide and metallic fuels.

e) Implementation of safeguards in safeguarded facilities

f) Designers of equipment for fuel fabrication facilities

5.2 Problem Formulation and Design of Questionnaire

To elicit expert opinion, two hypothetical facilities, one based on the conventional design, and the other incorporating measures for safeguards enhancement were proposed. To give an idea of plant size, the assumed capacity of the plants is 5 ton/year of MOX fuel. The measures proposed for enhancing safeguards are of three types.

a) In the first type either the measures available or not available and this can result in two utility values.

b) For the second type, the proposed measures could be available in different degrees, e.g., automation from 0% to 100%.

c) In the third type, the measure is available in different configurations. As described later, co-location of the fuel fabrication facility, the reactor and the reprocessing plant can be in various combinations.

Expert opinion was sought to obtain utility values and weighting factors for every proposed safeguards measure. Questionnaire incorporating all above is given in Table–5.1. While, for the purpose of analysis, various safeguards measures are assumed to be independent, there is some inter-dependency between different measures. For example, implementation of automation is treated as independent of safeguards-by-design, but one can argue that they are not truly independent. Similarly, many measures are closely linked to each other, like integration of process equipment, overall reduction in the total number of items of process equipment, footprint reduction of the plant and reduction in ventilation ducting of the plant etc.

5.3 Summary of Responses

Table-5.2 to Table-5.5 contain the feedback of the 24 experts for evaluation of **PR** by MAUA. For data of the **PR** assessment of proposed measures by JAEA methodology, the feedback of 21 experts has been collected as per the Table-5.6. As in the case of evaluation of **PR** by MAUA methodology, two plants have been compared. In the old plant, none of the measures are present, while in the new plant all measures are present in full extent. The values of the attributes are between 0 and 5 and the weightage is between 0 and 1.0. The feedback of the 21 experts is presented in Table-5.7 to Table-5.13.

A detailed analysis of the data presented in this chapter for evaluation of **PR** by both MAUA and JAEA is described in Chapter 6.

Table-5.1 : Sample Feedback Form for MAUA

No.	Feature	Description	Weightage	Utility Value	Remarks
1	Near Real Time Monitoring	Not available			
		Yearly			
		Monthly			
		Weekly			
		Daily			
		Every Shift			
		Continuous			
2	Dynamic Nuclear Material Accounting	Not available			
		All entry and exit			
		Every Processing Area			
		Every Box			
3	Safeguards by Design	Not available			
		Concept Stage			
		Design Stage			
		Construction Stage			
		Post Construction			
4	Co-Location	No Co-location			
		With Reactor			
		With Reprocessing			
		With Reactor & Reprocessing			
5	Surveillance Imagery	No Imagery			
		Plant Imagery			
		Processing Areas			
		Individual boxes			
6	Equipment Number Reduction	No reduction			
		5 % Reduction			
		10% Reduction			
		15 % Reduction			
7	Footprint Reduction	No Reduction			
		5% Reduction			
		10 % Reduction			
		15 % Reduction			
8	Ventilation System	No Reduction			
		5% Reduction			
		10 % Reduction			

		15 % Reduction			
9	Automation	0%			
		10%			
		20%			
		30%			
		40%			
		50%			
		60%			
		70%			
		80%			
		90%			
		100%			
10	No. of Operators	50			
		75			
		100			
		200			
		250			
		300			
11	Integration of Equipment	0%			
		5%			
		10%			
		15%			
12	Ease of Isolation	Conventional			
		Provisions made			
13	Optimisation of Material flow	No optimisation			
		Optimal			
14	QC equipment integration	No Integration			
		25% Integration			
		50 % Integration			
		75 % Integration			
		100 % Integration			
15	Laser engraving / RFID	Not available			
		RFID on containers			
		Marking on individual pins			
		Marking on Assemblies			
16	Computerised Tracking	No Tracking			
		Tracking only in process area			
		Tracking in individual box			

17	Powder Reduction	0%			
		20%			
		40%			
		60%			
		80%			
		100%			
18	Measurement at each station	No Measurement			
		Each Room			
		Each line			
		Each box			
19	Improved Design	Old design			
		Smooth surface finish			
		Less blind pockets			
		No sharp corners			
20	Isolation of Services	No Provision			
		Provisions made			

Table-5.2 : Feedback Data (1 to 6)

No.	Feature	Description	1		2		3		4		5		6	
			UV	Wt	UV	Wt	UV	Wt	UV	Wt	UV	Wt	UV	Wt
1.		Not available	0.05		0.15		0.10		0.04		0.04		0.03	
		Yearly	0.20		0.25		0.20		0.30		0.40		0.20	
		Monthly	0.30		0.10		0.20		0.20		0.20		0.30	
	Near Real Time Monitoring	Weekly	0.50	0.10	0.40	0.080	0.25	0.100	0.40	0.090	0.60	0.080	0.45	0.100
		Daily	0.80		0.20		0.30		0.70		0.90		0.80	
		Every Shift	0.90		0.50		0.40		0.90		0.80		0.90	
		Continuous	0.95		0.30		0.60		0.80		0.90		0.90	
2		Not available	0.10		0.15		0.10		0.20		0.30		0.20	
	Dynamic Nuclear Material Accounting	All entry and exit	0.25	0.080	0.30	0.100	0.20	0.070	0.30	0.080	0.20	0.050	0.25	0.090
		Every Processing Area	0.60		0.25		0.30		0.70		0.50		0.60	
		Every Box	090		0.50		0.40		0.80		0.80		0.90	
3		Not available	0.10		0.20		0.10		0.10		0.10		0.20	
		Concept Stage	0.40		0.25		0.20		0.30		0.50		0.40	
	Safeguards by Design	Design Stage	0.60	0.010	0.35	0.080	0.30	0.050	0.70	0.070	0.50	0.080	0.60	0.080
		Construction Stage	0.85		0.70		0.60		0.80		0.80		0.80	
		Post Construction	0.90		0.80		0.70		0.85		0.90		0.90	
4	Co-Location	No Co-location	0.30		0.20		0.10		0.30		0.20		0.20	
		With Reactor	0.60		0.50		0.20		0.60		0.50		0.70	
		With Reprocessing	0.50	0.100	0.30	0.040	0.30	0.050	0.80	0.060	0.40	0.050	0.60	0.040
		With Reactor & Reprocessing	0.90		0.60		0.40		0.70		0.90		0.90	
5		No Imagery	0.20		0.20		0.10		0.10		0.10		0.30	
	Surveillance Imagery	Plant Imagery	0.60	0.100	0.50	0.080	0.20	0.090	0.50	0.070	0.50	0.070	0.60	0.080
		Processing Areas	0.80		0.30		0.40		0.70		0.70		0.80	
		Individual boxes	0.90		0.60		0.50		0.80		0.90		0.90	
6		No reduction	0.20		0.70		0.10		0.10		0.20		0.20	
	Equipment Number Reduction	5 % Reduction	0.30	0.010	0.80	0.010	0.70	0.020	0.40	0.040	0.40	0.020	0.30	0.030
		10% Reduction	0.40		0.90		0.80		0.30		0.30		0.40	
		15 % Reduction	0.50		0.75		0.90		0.50		0.50		0.50	
7		No Reduction	0.10		0.15		0.10		0.20		0.10		0.20	
	Footprint Reduction	5% Reduction	0.20	0.010	0.30	0.030	0.10	0.010	0.30	0.050	0.30	0.010	0.30	0.005
		10 % Reduction	0.30		0.25		0.20		0.40		0.20		0.40	
		15 % Reduction	0.50		0.40		0.30		0.50		0.50		0.50	
8	Ventilation System	No Reduction	0.10	0.010	0.20	0.005	0.10	0.020	0.10	0.005	0.10	0.030	0.30	0.005

		5% Reduction	0.20		0.35		0.20		0.30		0.30		0.40	
		10 % Reduction	0.30		0.25		0.30		0.20		0.40		0.50	
		15 % Reduction	0.50		0.30		0.40		0.40		0.60		0.60	
9	Automation	0%	0.25	0.090	0.20	0.080	0.10	0.100	0.20	0.070	0.20	0.100	0.20	
		10%	0.30		0.20		0.10		0.30		0.40		0.30	
		20%	0.40		0.30		0.10		0.45		0.50		0.35	
		30%	0.50		0.30		0.10		0.50		0.50		0.45	
		40%	0.55		0.30		0.10		0.60		0.55		0.50	
		50%	0.60		0.30		0.20		0.65		0.66		0.60	
		60%	0.65		0.30		0.20		0.70		0.65		0.65	
		70%	0.70		0.35		0.25		0.80		0.70		0.70	
		80%	0.75		0.40		0.25		0.85		0.80		0.80	
		90%	0.80		0.40		0.10		0.90		0.85		0.85	
		100%	0.90		0.40		0.10		0.90		0.90		0.90	
10	No. of Operators	50	0.40	0.070	0.15	0.080	0.10	0.050	0.30	0.010	0.30	0.070	0.40	0.080
		100	0.50		0.45		0.40		0.50		0.40		0.50	
		150	0.80		0.60		0.50		0.70		0.70		0.70	
		200	0.60		0.20		0.30		0.80		0.50		0.60	
		250	0.50		0.30		0.20		0.40		0.60		0.50	
		300	0.40		0.40		0.10		0.60		0.40		0.40	
11	Integration of Equipment	0%	0.20	0.050	0.15	0.030	0.10	0.050	0.10	0.040	0.20	0.030	0.20	0.050
		5%	0.30		0.10		0.20		0.30		0.30		0.30	
		10%	0.40		0.40		0.30		0.20		0.40		0.40	
		15%	0.50		0.60		0.50		0.50		0.50		0.50	
12	Ease of Isolation	Conventional	0.30	0.005	0.15	0.020	0.10	0.010	0.40	0.030	0.30	0.050	0.40	0.020
		Provisions made	0.60		0.70		0.60		0.70		0.50		0.60	
13	Optimisation of Material flow	No optimisation	0.30	0.040	0.20	0.030	0.10	0.050	0.30	0.070	0.30	0.060	0.30	0.060
		Optimal	0.70		0.70		0.60		0.60		0.70		0.70	
14	QC equipment integration	No Integration	0.10	0.050	0.20	0.060	0.10	0.040	0.20	0.030	0.10	0.010	0.10	0.060
		25% Integration	0.15		0.20		0.20		0.15		0.20		0.20	
		50 % Integration	0.20		0.30		0.10		0.10		0.30		0.25	
		75 % Integration	0.25		0.40		0.30		0.25		0.20		0.30	
		100 % Integration	0.30		0.50		0.30		0.35		0.30		0.35	
15	Laser engraving / RFID	Not available	0.20	0.040	0.45	0.060	0.10	0.050	0.15	0.030	0.30	0.040	0.20	0.050
		RFID on containers	0.50		0.55		0.50		0.45		0.50		0.50	
		Marking on individual pins	0.60		0.70		0.60		0.65		0.70		0.60	
		Marking on Assemblies	0.80		0.80		0.70		0.80		0.80		0.80	

16	Computerised Tracking	No Tracking	0.20	0.060	0.20	0.050	0.10	0.040	0.30	0.060	0.25	0.070	0.25	0.020
		Tracking only in process area	0.40		0.30		0.20		0.35		0.40		0.50	
		Tracking in individual box	0.80		0.40		0.30		0.90		0.90		0.80	
17	Powder Reduction	0%	0.10	0.060	0.20	0.080	0.10	0.090	0.20	0.090	0.10	0.080	0.20	0.100
		20%	0.30		0.30		0.20		0.30		0.20		0.30	
		40%	0.40		0.40		0.30		0.40		0.40		0.40	
		60%	0.50		0.50		0.40		0.60		0.50		0.50	
		80%	0.80		0.60		0.50		070		0.70		0.80	
		100%	0.90		0.70		0.50		0.90		0.90		0.90	
18	Measurement at each station	No Measurement	0.20	0.080	0.70	0.060	0.60	0.070	0.30	0.070	0.25	0.060	0.20	0.080
		Each Room	0.30		0.70		0.60		0.40		0.30		0.40	
		Each line	0.40		0.80		0.70		0.50		0.40		0.60	
		Each box	0.60		0.80		0.80		0.60		0.70		0.80	
19	Improved Design	Old design	0.20	0.030	0.20	0.020	0.10	0.030	0.20	0.030	0.25	0.020	0.20	0.010
		Smooth surface finish	0.30		0.30		0.20		0.25		0.30		0.30	
		Less blind pockets	0.20		0.30		0.40		0.30		0.20		0.20	
		No sharp corners	0.20		0.60		0.50		0.20		0.20		0.20	
20	Isolation of Services	No Provision	0.20	0.005	0.20	0.005	0.10	0.010	0.25	0.005	0.20	0.020	0.20	0.030
		Provisions made	0.50		0.70		0.60		0.50		0.40		0.50	

Table-5.3: Feedback (7 to 12)

No.	Feature	Description	7 UV	7 Wt	8 UV	8 Wt	9 UV	9 Wt	10 UV	10 Wt	11 UV	11 Wt	12 UV	12 Wt
1.	Near Real Time Monitoring	Not available	0.20	0.100	0.10	0.080	0.02	0.090	0.01	0.100	0.20	0.070	0.10	0.100
		Yearly	0.40		0.20		0.10		0.09		0.40		0.20	
		Monthly	0.60		0.40		0.40		0.38		0.60		0.30	
		Weekly	0.75		0.60		0.60		0.68		0.75		0.50	
		Daily	0.80		0.70		0.80		0.86		0.80		0.70	
		Every Shift	0.85		0.80		0.85		0.91		0.85		0.80	
		Continuous	0.90		0.90		0.90		0.95		0.90		0.90	
2	Dynamic Nuclear Material Accounting	Not available	0.20	0.080	0.10	0.060	0.02	0.080	0.01	0.080	0.20	0.100	0.10	0.020
		All entry and exit	0.60		0.40		0.30		0.30		0.60		0.20	
		Every Processing Area	0.80		0.50		0.55		0.70		0.80		0.80	
		Every Box	0.90		0.85		0.90		0.85		0.90		0.90	
3	Safeguards by Design	Not available	0.20	0.090	0.10	0.080	0.05	0.060	0.01	0.080	0.20	0.080	0.20	0.100
		Concept Stage	0.40		0.50		0.40		0.40		0.40		0.50	
		Design Stage	0.50		0.70		0.60		0.51		0.50		0.70	
		Construction Stage	0.70		0.80		0.80		0.78		0.70		0.80	
		Post Construction	0.80		0.90		0.90		0.92		0.80		0.90	
4	Co-Location	No Co-location	0.20	0.050	0.10	0.020	0.20	0.100	0.20	0.090	0.20	0.050	0.10	0.050
		With Reactor	0.40		0.50		0.55		0.55		0.40		0.50	
		With Reprocessing	0.60		0.50		0.55		0.60		0.60		0.50	
		With Reactor & Reprocessing	0.80		0.90		0.90		0.92		0.80		0.90	
5	Surveillance Imagery	No Imagery	0.20	0.100	0.10	0.080	0.05	0.050	0.10	0.080	0.20	0.090	0.10	0.030
		Plant Imagery	0.40		0.50		0.50		0.55		0.40		0.50	
		Processing Areas	0.60		0.60		0.70		0.80		0.60		0.80	
		Individual boxes	0.80		0.90		0.85		0.90		0.80		0.85	
6	Equipment Number Reduction	No reduction	0.20	0.010	0.05	0.050	0.10	0.020	0.10	0.020	0.20	0.040	0.05	0.020
		5 % Reduction	0.40		0.20		0.25		0.30		0.40		0.10	
		10% Reduction	0.60		0.30		0.30		0.48		0.60		0.20	
		15 % Reduction	0.70		0.50		0.45		0.50		0.70		0.40	
7	Footprint Reduction	No Reduction	0.20	0.020	0.05	0.010	0.10	0.010	0.10	0.005	0.20	0.010	0.10	0.040
		5% Reduction	0.40		0.20		0.20		0.22		0.40		0.20	
		10 % Reduction	0.60		0.30		0.30		0.25		0.60		0.30	
		15 % Reduction	0.70		0.50		0.40		0.30		0.70		0.50	
8	Ventilation System	No Reduction	0.30	0.010	0.10	0.030	0.10	0.010	0.10	0.005	0.30	0.010	0.10	0.020
		5% Reduction	0.50		0.20		0.20		0.22		0.50		0.20	

		10 % Reduction	0.60		0.30		0.30		0.28		0.60		0.30	
		15 % Reduction	0.80		0.50		0.40		0.30		0.80		0.50	
9	Automation	0%	0.20	0.060	0.05	0.080	0.01	0.090	0.10	0.070	0.20	0.060	0.00	0.100
		10%	0.245		0.10		0.10		0.45		0.25		0.10	
		20%	0.29		0.20		0.20		0.50		0.29		0.15	
		30%	0.335		0.30		0.30		0.55		0.34		0.20	
		40%	0.425		0.40		0.40		0.60		0.43		0.25	
		50%	0.38		0.50		0.50		0.65		0.38		0.40	
		60%	0.47		0.60		0.60		0.70		0.47		0.45	
		70%	0.515		0.80		0.70		0.75		0.51		0.80	
		80%	0.56		0.85		0.80		0.80		0.56		0.85	
		90%	0.74		0.90		0.85		0.85		0.70		0.90	
		100%	0.80		0.95		0.90		0.90		0.80		0.95	
10	No. of Operators	50	0.30	0.060	0.50	0.050	0.50	0.070	0.40	0.060	0.30	0.070	0.50	0.080
		100	0.40		0.60		0.60		0.50		0.40		0.70	
		150	0.50		0.80		0.65		0.70		0.50		0.80	
		200	0.45		0.80		0.75		0.75		0.45		0.85	
		250	0.40		0.60		0.55		0.50		0.40		0.60	
		300	0.20		0.30		0.45		0.40		0.20		0.40	
11	Integration of Equipment	0%	0.45	0.040	0.10	0.050	0.10	0.040	0.20	0.020	0.50	0.030	0.10	0.040
		5%	0.50		0.20		0.20		0.40		0.60		0.20	
		10%	0.55		0.40		0.35		0.50		0.60		0.30	
		15%	0.60		0.50		0.45		0.60		0.60		0.40	
12	Ease of Isolation	Conventional	0.20	0.010	0.02	0.020	0.30	0.010	0.25	0.030	0.20	0.010	0.10	0.005
		Provisions made	0.50		0.80		0.70		0.60		0.50		0.40	
13	Optimisation of Material flow	No optimisation	0.50	0.040	0.20	0.050	0.30	0.050	0.30	0.010	0.50	0.060	0.10	0.060
		Optimal	0.80		0.80		0.90		0.75		0.80		0.80	
14	QC equipment integration	No Integration	0.20	0.040	0.10	0.040	0.01	0.030	0.10	0.040	0.20	0.020	0.10	0.010
		25% Integration	0.30		0.20		0.20		0.15		0.30		0.15	
		50 % Integration	0.40		0.40		0.30		0.18		0.40		0.20	
		75 % Integration	0.45		0.50		0.40		0.20		0.45		0.30	
		100 % Integration	0.50		0.60		0.50		0.25		0.50		0.50	
15	Laser engraving / RFID	Not available	0.20	0.060	0.10	0.050	0.10	0.060	0.40	0.040	0.20	0.030	0.10	0.060
		RFID on containers	0.50		0.50		0.50		0.40		0.50		0.50	
		Marking on individual pins	0.60		0.50		0.70		0.45		0.60		0.50	
		Marking on Assemblies	0.80		0.50		0.70		0.45		0.80		0.50	

16	Computerised Tracking	No Tracking	0.40	0.060	0.10	0.060	0.10	0.050	0.20	0.070	0.40	0.070	0.10	0.060
		Tracking only in process area	0.60		0.50		0.60		0.40		0.60		0.50	
		Tracking in individual box	0.80		0.80		0.85		0.85		0.80		0.70	
17	Powder Reduction	0%	0.20	0.090	0.50	0.080	0.05	0.070	0.10	0.050	0.00	0.080	0.00	0.100
		20%	0.25		0.60		0.15		0.25		0.20		0.20	
		40%	0.50		0.70		0.30		0.40		0.25		0.40	
		60%	0.55		0.80		0.60		0.60		0.50		0.60	
		80%	0.80		0.85		0.85		0.80		0.80		0.80	
		100%	0.85		0.90		0.90		0.90		0.95		0.90	
18	Measurement at each station	No Measurement	0.20	0.010	0.10	0.080	0.10	0.050	0.10	0.060	0.20	0.070	0.10	0.070
		Each Room	0.30		0.50		0.50		0.30		0.30		0.20	
		Each line	0.40		0.80		0.70		0.50		0.40		0.40	
		Each box	0.50		0.90		0.90		0.70		0.50		0.80	
19	Improved Design	Old design	0.20	0.050	0.10	0.020	0.10	0.030	0.10	0.040	0.20	0.020	0.10	0.030
		Smooth surface finish	0.30		0.30		0.20		0.30		0.30		0.30	
		Less blind pockets	0.40		0.30		0.20		0.25		0.40		0.30	
		No sharp corners	0.50		0.30		0.20		0.25		0.50		0.30	
20	Isolation of Services	No Provision	0.40	0.020	0.10	0.010	0.20	0.030	0.10	0.050	0.40	0.030	0.10	0.005
		Provisions made	0.50		0.60		0.55		0.60		0.50		0.80	

Table-5.4: Feedback (13 to 18)

No.	Feature	Description	13		14		15		16		17		18	
			UV	Wt	UV	Wt	UV	Wt	UV	Wt	UV	Wt	UV	Wt
1.	Near Real Time Monitoring	Not available	0.02	0.070	0.025	0.100	0.00	0.050	0.020	0.200	0.50	0.100	0.05	0.090
		Yearly	0.10		0.11		0.20		0.50		0.30		0.20	
		Monthly	0.40		0.39		0.40		0.55		0.40		0.40	
		Weekly	0.60		0.65		0.70		0.70		0.60		0.50	
		Daily	0.80		0.83		0.80		0.80		0.70		0.75	
		Every Shift	0.85		0.89		0.85		0.85		0.80		0.85	
		Continuous	0.90		0.95		0.90		0.90		0.90		0.90	
2	Dynamic Nuclear Material Accounting	Not available	0.10	0.060	0.025	0.100	0.20	0.080	0.10	0.100	0.10	0.080	0.20	0.100
		All entry and exit	0.20		0.25		0.25		0.20		0.30		0.40	
		Every Processing Area	0.70		0.60		0.70		0.50		0.70		0.80	
		Every Box	0.85		0.90		0.80		0.80		0.90		0.90	
3	Safeguards by Design	Not available	0.10	0.100	0.05	0.080	0.10	0.090	0.01	0.080	0.10	0.080	0.20	0.100
		Concept Stage	0.20		0.40		0.30		0.50		0.20		0.30	
		Design Stage	0.40		0.60		0.50		0.70		0.70		0.70	
		Construction Stage	0.80		0.85		0.80		0.80		0.80		0.80	
		Post Construction	0.90		0.90		0.90		0.85		0.90		0.85	
4	Co-Location	No Co-location	0.10	0.080	0.20	0.010	0.20	0.050	0.10	0.020	0.10	0.040	0.20	0.080
		With Reactor	0.60		0.60		0.60		0.50		0.60		0.50	
		With Reprocessing	0.60		0.55		0.60		0.50		0.60		0.50	
		With Reactor & Reprocessing	0.85		0.90		0.80		0.85		0.90		0.85	
5	Surveillance Imagery	No Imagery	0.00	0.070	0.10	0.080	0.10	0.060	0.10	0.080	0.20	.0100	0.10	0.040
		Plant Imagery	0.20		0.60		0.50		0.50		0.50		0.40	
		Processing Areas	0.40		0.75		0.80		0.70		0.80		0.70	
		Individual boxes	0.90		0.90		0.90		0.85		0.90		0.80	
6	Equipment Number Reduction	No reduction	0.10	0.010	0.10	0.010	0.10	0.020	0.10	0.030	0.20	0.010	0.10	0.020
		5 % Reduction	0.15		0.20		0.15		0.15		0.40		0.20	
		10% Reduction	0.20		0.30		0.20		0.20		0.50		0.30	
		15 % Reduction	0.40		0.45		0.30		0.30		0.60		0.50	
7	Footprint Reduction	No Reduction	0.10	0.020	0.10	0.010	0.10	0.020	0.10	0.010	0.10	0.020	0.05	0.050
		5% Reduction	0.20		0.20		0.15		0.15		0.20		0.10	
		10 % Reduction	0.25		0.30		0.20		0.20		0.30		0.20	
		15 % Reduction	0.40		0.45		0.30		0.30		0.50		0.40	
8	Ventilation System	No Reduction	0.10	0.010	0.10	0.030	0.10	0.010	0.10	0.020	0.05	0.005	0.10	0.010
		5% Reduction	0.20		0.20		0.20		0.15		0.10		0.15	

		10 % Reduction	0.30		0.30		0.30		0.20		0.20		0.30	
		15 % Reduction	0.35		0.45		0.40		0.30		0.30		0.50	
9	Automation	0%	0.00	0.090	0.001	0.080	0.00	0.100	0.10	0.100	0.10	0.080	0.05	0.090
		10%	0.15		0.35		0.10		0.30		0.15		0.10	
		20%	0.20		0.40		0.20		0.40		0.20		0.20	
		30%	0.25		0.42		0.30		0.45		0.40		0.30	
		40%	0.30		0.50		0.40		0.50		0.45		0.40	
		50%	0.40		0.60		0.50		0.55		0.50		0.50	
		60%	0.45		0.65		0.60		0.60		0.60		0.60	
		70%	0.50		0.75		0.70		0.65		0.65		0.70	
		80%	0.60		0.80		0.80		0.7		0.70		0.80	
		90%	0.80		0.85		0.90		0.75		0.80		0.90	
		100%	0.90		0.90		0.95		0.80		0.90		0.95	
10	No. of Operators	50	0.10	0.040	0.50	0.080	0.10	0.070	0.50	0.060	0.20	0.080	0.50	0.050
		100	0.40		0.60		0.20		0.70		0.30		0.60	
		150	0.80		0.65		0.60		0.75		0.50		0.70	
		200	0.85		0.80		0.70		0.80		0.80		0.80	
		250	0.70		0.60		0.20		0.50		0.60		0.40	
		300	0.20		0.50		0.10		0.30		0.10		0.20	
11	Integration of Equipment	0%	0.10	0.050	0.10	0.040	0.10	0.050	0.10	0.040	0.10	0.030	0.05	0.040
		5%	0.20		0.30		0.20		0.20		0.30		0.20	
		10%	0.30		0.40		0.30		0.30		0.40		0.30	
		15%	0.50		0.50		0.40		0.40		0.60		0.50	
12	Ease of Isolation	Conventional	0.10	0.010	0.26	0.010	0.20	0.030	0.15	0.010	0.10	0.005	0.05	0.010
		Provisions made	0.70		0.68		0.80		0.70		0.70		0.80	
13	Optimisation of Material flow	No optimisation	0.10	0.060	0.23	0.050	0.20	0.050	0.20	0.040	0.10	0.050	0.10	0.060
		Optimal	0.70		0.70		0.85		0.70		0.70		0.80	
14	QC equipment integration	No Integration	0.00	0.050	0.01	0.040	0.10	0.050	0.10	0.070	0.10	0.040	0.05	0.030
		25% Integration	0.10		0.05		0.15		0.50		0.20		0.10	
		50 % Integration	0.20		0.10		0.20		0.55		0.30		0.20	
		75 % Integration	0.30		0.15		0.25		0.60		0.40		0.30	
		100 % Integration	0.40		0.21		0.30		0.70		0.50		0.40	
15	Laser engraving / RFID	Not available	0.10	0.050	0.15	0.040	0.20	0.060	0.10	0.050	0.10	0.060	0.10	0.050
		RFID on containers	0.50		0.50		0.60		0.50		0.30		0.30	
		Marking on individual pins	0.50		0.60		0.60		0.50		0.40		0.40	
		Marking on Assemblies	0.50		0.70		0.60		0.80		0.45		0.40	

16	Computerised Tracking	No Tracking	0.10	0.050	0.10	0.060	0.10	0.010	0.10	0.070	0.10	0.060	0.05	0.060
		Tracking only in process area	0.60		0.40		0.40		0.30		0.40		0.50	
		Tracking in individual box	0.70		0.80		0.80		0.70		0.90		0.80	
17	Powder Reduction	0%	0.10	0.070	0.10	0.090	0.20	0.080	0.10	0.090	0.10	0.080	0.10	0.070
		20%	0.20		0.30		0.40		0.20		0.50		0.40	
		40%	0.30		0.40		0.60		0.30		0.60		0.70	
		60%	0.40		0.60		0.80		0.40		0.70		0.75	
		80%	0.70		0.80		0.85		0.70		0.80		0.80	
		100%	0.90		0.90		0.90		0.80		0.90		0.85	
18	Measurement at each station	No Measurement	0.10	0.080	0.10	0.060	0.10	0.080	0.10	0.070	0.10	0.060	0.10	0.070
		Each Room	0.30		0.30		0.20		0.20		0.30		0.50	
		Each line	0.50		0.40		0.60		0.30		0.60		0.70	
		Each box	0.80		0.60		0.90		0.40		0.80		0.85	
19	Improved Design	Old design	0.10	0.020	0.10	0.010	0.10	0.030	0.10	0.020	0.10	0.010	0.05	0.030
		Smooth surface finish	0.40		0.30		0.40		0.20		0.40		0.30	
		Less blind pockets	0.40		0.20		0.40		0.20		0.40		0.30	
		No sharp corners	0.40		0.20		0.40		0.20		0.40		0.60	
20	Isolation of Services	No Provision	0.10	0.010	0.20	0.020	0.10	0.010	0.10	0.020	0.10	0.010	0.05	0.005
		Provisions made	0.70		0.50		0.90		0.60		0.80		0.70	

Table-5.5: Feedback (19 to 24)

No.	Feature	Description	19		20		21		22		23		24	
24			UV	Wt	UV	Wt	UV	Wt	UV	Wt	UV	Wt	UV	Wt
1.	Near Real Time Monitoring	Not available	0.50	0.070	0.05	0.100	0.00	0.090	0.00	0.100	0.02	0.100	0.02	0.080
		Yearly	0.10		0.10		0.10		0.10		0.12		0.10	
		Monthly	0.50		0.40		0.30		0.50		0.40		0.40	
		Weekly	0.60		0.60		0.50		0.70		0.63		0.60	
		Daily	0.80		0.85		0.70		0.80		0.84		0.80	
		Every Shift	0.90		0.90		0.90		0.90		0.90		0.90	
		Continuous	0.95		0.95		0.98		0.90		0.95		0.95	
2	Dynamic Nuclear Material Accounting	Not available	0.10	0.080	0.05	0.090	0.00	0.080	0.00	0.060	0.025	0.070	0.00	0.080
		All entry and exit	0.35		0.30		0.40		0.40		0.25		0.30	
		Every Processing Area	0.60		0.60		0.90		0.60		0.60		0.70	
		Every Box	0.90		0.90		0.97		0.90		0.93		0.95	
3	Safeguards by Design	Not available	0.10	0.080	0.05	0.070	0.00	0.090	0.00	0.080	0.05	0.060	0.05	0.080
		Concept Stage	0.20		0.25		0.10		0.40		0.38		0.40	
		Design Stage	0.60		0.65		0.40		0.60		0.65		0.70	
		Construction Stage	0.85		0.80		0.80		0.80		0.85		0.80	
		Post Construction	0.90		0.90		0.95		0.90		0.90		0.90	
4	Co-Location	No Co-location	0.05	0.060	0.20	0.040	0.00	0.040	0.20	0.050	0.20	0.070	0.10	0.050
		With Reactor	0.50		0.70		0.25		0.60		0.62		0.50	
		With Reprocessing	0.50		0.55		0.60		0.60		0.54		0.60	
		With Reactor & Reprocessing	0.80		0.90		0.90		0.90		0.90		0.90	
5	Surveillance Imagery	No Imagery	0.20	0.100	0.10	0.080	0.00	0.070	0.10	0.050	0.10	0.100	0.10	0.080
		Plant Imagery	0.50		0.60		0.30		0.60		0.60		0.50	
		Processing Areas	0.70		0.70		0.70		0.80		0.75		0.80	
		Individual boxes	0.90		0.90		0.90		0.90		0.90		0.95	
6	Equipment Number Reduction	No reduction	0.20	0.050	0.20	0.040	0.00	0.020	0.20	0.010	0.10	0.010	0.10	0.020
		5 % Reduction	0.40		0.25		0.25		0.30		0.20		0.20	
		10% Reduction	0.50		0.30		0.40		0.40		0.30		0.30	
		15 % Reduction	0.60		0.40		0.60		0.50		0.45		0.40	
7	Footprint Reduction	No Reduction	0.10	0.010	0.05	0.010	0.00	0.005	0.10	0.010	0.10	0.005	0.10	0.010
		5% Reduction	0.20		0.10		0.20		0.20		0.20		0.20	
		10 % Reduction	0.50		0.25		0.25		0.30		0.30		0.30	
		15 % Reduction	0.60		0.40		0.30		0.40		0.45		0.50	
8	Ventilation	No Reduction	0.10	0.005	0.05	0.020	0.00	0.010	0.00	0.030	0.10	0.005	0.10	0.010

	System													
		5% Reduction	0.20		0.15		0.20		0.20		0.20		0.20	
		10 % Reduction	0.30		0.25		0.25		0.30		0.30		0.30	
		15 % Reduction	0.45		0.30		0.30		0.40		0.40		0.50	
9	Automation	0%	0.10		0.20		0.20		0.00		0.01		0.01	
		10%	0.30		0.25		0.30		0.30		0.32		0.30	
		20%	0.50		0.30		0.35		0.40		0.38		0.40	
		30%	0.55		0.35		0.45		0.50		0.42		0.45	
		40%	0.60		0.40		0.50		0.60		0.50		0.50	
		50%	0.65	0.080	0.45	0.090	0.60	0.100	0.60	0.090	0.60	0.090	0.60	0.080
		60%	0.70		0.50		0.65		0.70		0.62		0.65	
		70%	0.75		0.55		0.70		0.70		0.72		0.70	
		80%	0.80		0.75		0.80		0.80		0.80		0.75	
		90%	0.85		0.85		0.90		0.80		0.85		0.80	
		100%	0.90		0.95		0.97		0.90		0.90		0.95	
10	No. of Operators	50	0.10		0.40		0.35		0.50		0.50		0.50	
		100	0.50		0.55		0.45		0.60		0.60		0.60	
		150	0.90	0.080	0.65	0.080	0.60	0.070	0.60	0.060	0.65	0.080	0.70	0.070
		200	0.85		0.90		0.75		0.70		0.80		0.80	
		250	0.70		0.80		0.40		0.70		0.60		0.60	
		300	0.30		0.50		0.25		0.40		0.50		0.50	
11	Integration of Equipment	0%	0.10		0.20		0.20		0.20		0.10		0.10	
		5%	0.30	0.020	0.30	0.030	0.30	0.050	0.30	0.050	0.25	0.020	0.20	0.040
		10%	0.50		0.40		0.45		0.40		0.50		0.30	
		15%	0.60		0.50		0.60		0.50		0.55		0.40	
12	Ease of Isolation	Conventional	0.10	0.005	0.30	0.005	0.25	0.010	0.30	0.020	0.26	0.010	0.25	0.005
		Provisions made	0.75		0.70		0.70		0.80		0.68		0.65	
13	Optimisation of Material flow	No optimisation	0.05	0.030	0.25	0.050	0.20	0.040	0.10	0.040	0.23	0.050	0.20	0.070
		Optimal	0.70		0.75		0.75		0.80		0.70		0.80	
14	QC equipment integration	No Integration	0.10		0.01		0.20		0.20		0.01		0.05	
		25% Integration	0.15		0.02		0.30		0.30		0.05		0.10	
		50 % Integration	0.20	0.020	0.15	0.040	0.40	0.060	0.30	0.050	0.10	0.060	0.20	0.040
		75 % Integration	0.25		0.20		0.45		0.40		0.15		0.25	
		100 % Integration	0.30		0.25		0.50		0.40		0.21		0.30	
15	Laser engraving / RFID	Not available	0.05		0.20		0.10		0.10		0.15		0.10	
		RFID on containers	0.50	0.050	0.50	0.010	0.40	0.040	0.50	0.060	0.50	0.020	0.60	0.060
		Marking on individual pins	0.50		0.70		0.50		0.60		0.60		0.70	

		Marking on Assemblies	0.70		0.80		0.65		0.40		0.70		0.80	
16	Computerised Tracking	No Tracking	0.10	0.070	0.15	0.050	0.20	0.060	0.10	0.040	0.10	0.070	0.10	0.070
		Tracking only in process area	0.50		0.45		0.45		0.60		0.40		0.40	
		Tracking in individual box	0.75		0.90		0.70		0.90		0.85		0.70	
17	Powder Reduction	0%	0.05	0.080	0.05	0.100	0.10	0.080	0.00	0.090	0.10	0.090	0.10	0.080
		20%	0.30		0.35		0.30		0.10		0.20		0.20	
		40%	0.50		0.60		0.45		0.30		0.40		0.40	
		60%	0.60		0.70		0.55		0.50		0.60		0.60	
		80%	0.70		0.80		0.70		0.60		0.60		0.80	
		100%	0.80		0.90		0.90		0.90		0.80		0.90	
18	Measurement at each station	No Measurement	0.05	0.060	0.15	0.080	0.20	0.070	0.10	0.050	0.10	0.060	0.10	0.050
		Each Room	0.60		0.35		0.45		0.40		0.30		0.20	
		Each line	0.75		0.50		0.60		0.60		0.50		0.40	
		Each box	0.80		0.70		0.80		0.90		0.70		0.70	
19	Improved Design	Old design	0.10	0.040	0.20	0.010	0.15	0.010	0.10	0.050	0.10	0.010	0.10	0.020
		Smooth surface finish	0.50		0.40		0.27		0.40		0.30		0.20	
		Less blind pockets	0.50		0.30		0.30		0.30		0.20		0.30	
		No sharp corners	0.50		0.25		0.35		0.20		0.20		0.30	
20	Isolation of Services	No Provision	0.05	0.010	0.20	0.005	0.15	0.005	0.20	0.010	0.20	0.020	0.30	0.005
		Provisions made	0.80		0.40		0.40		0.50		0.50		0.50	

Table-5.6: JAEA Sample Feedback Form

No.	Feature	Weightage	Old Plant (max 5.0)	New Plant (max 5.0)	Remarks
1.	NRTM				
2.	DNMA				
3.	SBD				
4.	Co-Location				
5.	Imagery				
6.	Equipment Reduction				
7.	Footprint Reduction				
8.	Ventilation Reduction				
9.	Automation				
10.	Manpower Reduction				
11.	Integration of Equipment				
12.	Ease of Isolation				
13.	Optimisation of Material Flow				
14.	QC Integration				
15.	RFID / Laser Engraving				
16.	Computerised Material Tracking				
17.	Powder Reduction				
18	Measurement at Each Station				
19.	Equipment Design				
20.	Isolation of Services				

Table-5.7: JAEA Feedback Form (1 – 3)

	Description	**1**			**2**			**3**		
		Wt	**UV Old**	**UV New**	**Wt**	**UV Old**	**UV New**	**Wt**	**UV Old**	**UV New**
1.	NRTM	1.0	1.0	4.5	0.9	0.3	2.0	0.8	1.0	4.6
2.	DNMA	0.7	2.0	4.2	0.6	0.2	2.5	0.7	2.0	4.5
3.	SBD	0.7	1.25	4.2	0.7	0.1	3.0	0.8	1.15	4.3
4.	Co-Location	0.7	1.5	4.1	0.4	0.2	3.5	0.8	1.75	4.2
5.	Imagery	0.8	1.0	4.5	1.0	0.2	4.0	0.7	1.0	4.1
6.	Equipment Reduction	0.4	2.7	3.5	0.1	0.3	4.5	0.3	3.0	3.5
7.	Footprint Reduction	0.2	2.4	3.4	0.1	0.35	4.5	0.3	2.5	3.3
8.	Ventilation Reduction	0.1	2.4	3.0	0.2	0.4	2.0	0.2	2.6	3.4
9.	Automation	0.9	1.0	4.6	0.8	0.06	4.0	0.8	2.4	4.5
10.	Manpower Reduction	0.7	1.5	4.0	0.7	0.1	4.0	0.6	1.5	4.6
11.	Integration of Equipment	0.7	1.5	3.5	0.4	0.2	3.5	0.8	1.7	3.6
12.	Ease of Isolation	0.4	2.0	2.5	0.2	0.1	2.33	0.6	2.5	3.4
13.	Optimisation of Material Flow	0.8	2.0	4.5	0.9	0.3	2.5	0.3	2.0	3.5
14.	QC Integration	0.3	1.5	3.2	0.1	0.2	2.6	0.4	1.5	4.4
15.	RFID / Laser Engraving	0.3	1.5	4.3	0.4	0.1	3.5	1.0	1.0	4.5
16.	Computerised Material Tracking	0.9	1.0	4.2	0.5	0.2	2.5	0.2	1.5	4.2
17.	Powder Reduction	0.8	1.3	4.6	0.9	0.1	2.5	0.8	1.2	4.3
18	Measurement at Each Station	0.3	1.2	4.7	0.8	0.5	2.0	0.9	1.3	4.7
19.	Equipment Design	0.3	1.0	3.7	0.2	0.6	2.7	0.3	1.0	4.6
20.	Isolation of Services	0.2	1.0	3.0	0.1	0.6	2.8	0.4	1.2	3.0

Table-5.8: JAEA Feedback (4-6)

	Description	4			5			6		
		Wt	UV Old	UV New	Wt	UV Old	UV New	Wt	UV Old	UV New
1.	NRTM	1.0	1.0	4.6	0.7	1.0	4.4	0.8	1.5	4.5
2.	DNMA	0.8	2.0	4.3	0.8	2.0	4.1	0.9	2.5	4.5
3.	SBD	0.6	1.3	4.2	0.6	1.25	4.1	1.0	1.5	4.5
4.	Co-Location	0.7	1.5	4.1	0.2	1.5	4.2	0.5	3.5	4.5
5.	Imagery	0.5	1.0	4.5	0.6	1.0	4.6	0.7	2.0	3.5
6.	Equipment Reduction	0.5	2.7	3.5	0.4	2.7	3.6	0.4	2.0	3.5
7.	Footprint Reduction	0.1	2.5	3.5	0.1	2.4	3.7	0.3	2.0	3.5
8.	Ventilation Reduction	0.3	2.4	3.0	0.2	2.4	3.2	0.1	1.0	4.5
9.	Automation	0.9	1.0	4.7	0.9	1.0	4.6	1.0	1.5	4.0
10.	Manpower Reduction	0.6	1.5	4.0	0.4	1.5	4.0	0.9	1.5	4.0
11.	Integration of Equipment	0.5	1.5	3.5	0.2	1.5	3.6	0.3	1.5	3.5
12.	Ease of Isolation	0.2	2.0	2.5	0.3	2.0	2.4	0.1	1.5	3.5
13.	Optimisation of Material Flow	0.8	2.0	4.5	0.9	2.0	4.6	0.4	1.5	3.5
14.	QC Integration	0.6	1.5	3.2	0.9	1.5	3.1	0.8	1.5	4.0
15.	RFID / Laser Engraving	0.8	1.6	4.4	0.6	1.5	4.2	0.1	1.5	4.0
16.	Computerised Material Tracking	0.4	1.0	4.2	0.5	1.0	4.3	0.6	1.0	4.5
17.	Powder Reduction	0.9	1.4	4.7	0.7	1.3	4.8	0.7	1.0	4.5
18	Measurement at Each Station	0.2	1.2	4.7	1.0	1.25	4.6	0.9	1.5	3.0
19.	Equipment Design	0.4	1.0	3.7	0.2	1.3	4.0	0.3	1.5	4.0
20.	Isolation of Services	0.1	1.0	3.0	0.4	1.2	3.5	0.2	1.5	4.0

Table-5.9: JAEA Feedback (7 – 9)

	Description	7			8			9		
		Wt	UV Old	UV New	Wt	UV Old	UV New	Wt	UV Old	UV New
1.	NRTM	1.0	1.75	4.5	0.9	1.5	4.6	0.7	2.29	3.57
2.	DNMA	0.8	2.0	4.25	0.7	2.3	4.3	0.6	2.25	3.5
3.	SBD	1.0	1.5	4.0	0.9	1.5	4.4	0.8	1.42	3.1
4.	Co-Location	0.5	1.5	4.5	0.8	2.0	4.2	0.6	2.6	3.0
5.	Imagery	0.8	3.0	4.0	1.0	1.5	4.1	0.9	2.0	3.0
6.	Equipment Reduction	0.1	2.5	3.5	0.3	2.1	3.7	0.1	1.95	2.9
7.	Footprint Reduction	0.1	2.5	3.5	0.2	1.2	3.5	0.1	1.4	3.2
8.	Ventilation Reduction	0.1	2.5	3.5	0.1	1.5	3.5	0.2	1.66	3.2
9.	Automation	0.8	1.5	4.5	0.9	1.5	4.5	0.8	1.36	2.8
10.	Manpower Reduction	0.6	1.5	4.25	0.4	1.45	4.1	0.5	2.2	2.5
11.	Integration of Equipment	0.1	1.5	3.5	0.4	1.4	3.8	0.1	2.1	3.2
12.	Ease of Isolation	0.3	1.5	2.5	0.4	1.5	3.0	0.1	1.7	2.4
13.	Optimisation of Material Flow	0.3	2.0	4.0	0.6	2.1	4.2	0.9	2.3	3.6
14.	QC Integration	0.5	1.5	3.5	0.7	1.2	3.2	0.4	1.7	2.48
15.	RFID / Laser Engraving	0.5	1.5	4.0	0.4	1.2	4.2	0.6	1.47	3.1
16.	Computerised Material Tracking	0.5	1.5	4.0	0.6	1.8	4.2	0.3	1.48	3.4
17.	Powder Reduction	1.0	1.0	4.5	0.6	1.6	4.7	0.9	1.6	1.8
18	Measurement at Each Station	0.8	1.5	4.5	0.4	1.4	4.6	0.6	1.7	2.4
19.	Equipment Design	0.1	1.5	3.5	0.2	1.2	3.8	0.4	1.28	2.4
20.	Isolation of Services	0.3	1.5	3.0	0.3	1.1	2.9	0.2	1.36	2.8

Table-5.10 : JAEA Feedback (10 -12)

	Description	10			11			12		
		Wt	UV Old	UV New	Wt	UV Old	UV New	Wt	UV Old	UV New
1.	NRTM	1.0	1.5	4.5	0.8	1.0	4.0	0.9	1.0	4.5
2.	DNMA	0.8	2.0	4.2	1.0	2.0	4.2	0.7	2.0	4.2
3.	SBD	0.6	1.5	4.1	0.8	1.25	4.25	0.8	1.0	4.2
4.	Co-Location	0.9	1.5	4.1	0.4	1.5	4.5	0.6	1.5	4.5
5.	Imagery	0.8	1.5	4.6	0.6	1.5	4.5	0.5	1.0	4.8
6.	Equipment Reduction	0.2	2.5	4.0	0.5	2.5	4.2	0.3	3.0	3.8
7.	Footprint Reduction	0.1	2.5	3.5	0.2	2.5	3.8	0.1	3.0	4.0
8.	Ventilation Reduction	0.2	2.0	3.5	0.2	2.5	3.0	0.1	2.5	3.5
9.	Automation	0.9	1.0	4.8	0.9	1.5	4.8	0.8	1.0	4.8
10.	Manpower Reduction	0.5	1.5	4.5	0.8	1.5	4.5	0.8	1.5	4.5
11.	Integration of Equipment	0.7	1.5	3.0	0.4	1.5	3.5	0.3	1.5	3.5
12.	Ease of Isolation	0.1	1.5	3.0	0.3	1.5	2.5	0.4	1.5	4.0
13.	Optimisation of Material Flow	0.4	2.0	4.0	0.3	2.0	4.0	0.5	1.5	4.5
14.	QC Integration	0.1	1.5	3.0	0.1	2.0	3.0	0.5	1.0	3.5
15.	RFID / Laser Engraving	0.5	1.5	3.0	0.7	1.5	3.0	0.5	2.0	3.5
16.	Computerised Material Tracking	0.4	1.5	3.0	1.0	1.5	3.5	0.4	1.2	4.0
17.	Powder Reduction	0.9	1.2	4.8	0.4	1.2	4.8	1.0	1.0	4.7
18	Measurement at Each Station	1.0	1.5	4.9	0.5	1.3	4.6	0.6	1.0	4.8
19.	Equipment Design	0.3	1.0	3.5	0.4	1.0	3.5	0.1	1.0	3.7
20.	Isolation of Services	0.5	1.5	4.0	0.1	1.5	4.2	0.4	1.0	4.8

Table-5.11 : JAEA Feedback (13 – 15)

	Description	13			14			15		
		Wt	**UV Old**	**UV New**	**Wt**	**UV Old**	**UV New**	**Wt**	**UV Old**	**UV New**
1.	NRTM	0.7	1.5	4.0	1.0	1.0	4.5	0.8	1.5	4.0
2.	DNMA	0.8	2.5	4.0	0.8	2.0	4.5	1.0	1.5	3.5
3.	SBD	0.8	1.5	3.5	1.0	1.5	4.0	0.7	1.0	3.5
4.	Co-Location	0.4	1.5	4.0	0.6	1.0	3.8	1.0	1.5	3.8
5.	Imagery	0.7	1.0	4.6	0.9	1.0	3.5	0.8	1.0	4.0
6.	Equipment Reduction	0.1	3.0	3.2	0.2	1.5	3.0	0.1	2.0	3.5
7.	Footprint Reduction	0.1	2.5	3.0	0.2	1.0	3.5	0.1	1.5	3.6
8.	Ventilation Reduction	0.2	2.0	3.0	0.2	1.0	3.8	0.1	1.0	3.6
9.	Automation	1.0	1.0	4.0	0.8	1.0	4.8	0.9	1.5	4.5
10.	Manpower Reduction	0.9	2.0	3.5	0.7	1.5	4.6	0.9	1.5	4.5
11.	Integration of Equipment	0.5	2.5	3.0	0.6	1.5	3.5	0.6	1.0	3.0
12.	Ease of Isolation	0.3	1.0	2.0	0.3	1.0	3.0	0.6	1.0	3.5
13.	Optimisation of Material Flow	0.6	2.5	4.0	0.5	1.5	4.0	1.0	1.0	4.0
14.	QC Integration	0.5	1.0	3.0	0.2	1.0	3.5	0.5	1.0	3.8
15.	RFID / Laser Engraving	0.3	1.0	4.0	0.2	1.0	3.8	1.0	1.0	3.5
16.	Computerised Material Tracking	0.2	1.0	4.0	0.4	1.0	3.5	0.4	1.0	4.0
17.	Powder Reduction	0.9	1.0	4.0	0.9	1.0	4.8	1.0	1.0	4.5
18	Measurement at Each Station	0.3	1.0	4.1	0.9	1.0	4.5	0.5	1.0	4.8
19.	Equipment Design	0.3	1.0	3.5	0.1	1.0	3.8	0.3	1.5	3.5
20.	Isolation of Services	0.1	1.0	4.2	0.4	1.0	4.2	0.2	1.5	4.5

Table-5.12 : JAEA Feedback (16-18)

	Description	16			17			18		
		Wt	UV Old	UV New	Wt	UV Old	UV New	Wt	UV Old	UV New
1.	NRTM	1.0	1.0	4.5	1.0	0.75	4.5	1.0	1.5	4.75
2.	DNMA	0.7	2.0	4.0	0.8	2.0	4.25	0.7	1.9	4.5
3.	SBD	0.7	2.0	4.5	0.8	1.5	4.5	0.9	1.1	4.3
4.	Co-Location	0.2	1.0	4.2	0.6	1.5	4.0	0.9	1.15	4.7
5.	Imagery	0.7	1.5	4.0	0.8	1.0	4.5	0.6	1.0	4.4
6.	Equipment Reduction	0.4	2.8	3.8	0.6	2.53	3.5	0.1	2.0	3.5
7.	Footprint Reduction	0.1	1.5	3.8	0.2	2.5	3.5	0.1	2.0	3.4
8.	Ventilation Reduction	0.2	1.5	3.5	0.1	2.25	3.0	0.1	2.1	3.6
9.	Automation	0.9	1.0	4.8	1.0	1.1	4.5	0.9	1.1	4.7
10.	Manpower Reduction	1.0	1.5	4.0	0.6	1.5	4.0	0.9	2.0	3.8
11.	Integration of Equipment	0.1	1.5	3.5	0.4	1.5	3.5	0.8	2.1	3.9
12.	Ease of Isolation	0.2	1.0	3.0	0.3	0.5	2.5	0.1	1.5	2.8
13.	Optimisation of Material Flow	0.7	1.5	3.8	0.9	2.0	4.5	0.3	1.7	4.2
14.	QC Integration	0.4	1.5	4.0	0.4	1.5	3.0	0.7	1.8	3.2
15.	RFID / Laser Engraving	0.4	1.0	3.8	0.3	1.5	4.25	0.3	1.2	4.6
16.	Computerised Material Tracking	0.5	1.0	3.5	0.9	1.75	4.5	0.4	1.1	4.7
17.	Powder Reduction	0.7	1.5	4.5	1.0	0.5	4.5	0.8	1.2	4.6
18	Measurement at Each Station	0.6	1.0	4.5	0.3	1.25	4.6	0.9	1.2	3.0
19.	Equipment Design	0.1	1.5	3.0	0.5	1.0	3.75	0.1	1.3	3.5
20.	Isolation of Services	0.1	1.5	4.0	0.5	0.75	3.0	0.4	1.1	2.8

Table-5.13 : JAEA Feedback (19 – 21)

	Description	19			20			21		
		Wt	UV Old	UV New	Wt	UV Old	UV New	Wt	UV Old	UV New
1.	NRTM	0.7	1.2	4.9	0.8	0.74	4.55	0.7	1.3	4.0
2.	DNMA	0.8	2.1	3.0	1.0	2.01	4.25	0.9	2.0	4.2
3.	SBD	0.6	1.8	4.8	0.7	1.22	4.25	0.8	1.5	4.0
4.	Co-Location	0.6	2.0	4.6	0.5	1.52	4.15	0.9	2.0	4.3
5.	Imagery	0.8	2.0	4.3	0.6	1.5	4.5	1.0	1.2	4.2
6.	Equipment Reduction	0.4	2.5	4.8	0.4	2.78	3.6	0.5	3.0	3.0
7.	Footprint Reduction	0.1	2.1	4.5	0.1	2.49	3.45	0.3	3.0	3.0
8.	Ventilation Reduction	0.2	2.3	2.9	0.1	2.46	3.08	0.2	2.5	3.0
9.	Automation	0.9	1.2	4.8	0.8	1.05	4.65	0.8	1.0	4.5
10.	Manpower Reduction	0.5	1.2	4.0	0.4	1.45	4.1	0.6	2.0	4.1
11.	Integration of Equipment	0.5	1.6	4.8	0.5	1.46	3.45	0.6	2.5	3.2
12.	Ease of Isolation	0.5	1.6	3.8	0.1	1.49	2.5	0.3	2.6	3.0
13.	Optimisation of Material Flow	0.4	1.8	4.6	0.7	2.05	4.45	0.3	2.0	4.1
14.	QC Integration	0.2	1.5	2.2	0.2	1.95	3.15	0.9	2.0	3.0
15.	RFID / Laser Engraving	0.8	1.2	4.5	0.4	1.5	4.35	0.3	2.5	4.2
16.	Computerised Material Tracking	0.8	1.2	4.9	1.0	1.45	4.22	0.6	2.4	4.5
17.	Powder Reduction	0.7	1.3	4.6	0.8	1.27	4.61	0.7	0.5	4.6
18	Measurement at Each Station	0.8	1.2	4.9	0.9	1.5	4.75	0.8	2.4	4.8
19.	Equipment Design	0.1	1.7	4.8	0.4	1.2	3.6	0.2	1.5	3.5
20.	Isolation of Services	0.1	1.9	3.8	0.1	1.5	3.0	0.3	0.5	3.0

Chapter 6

An Assessment of the Proliferation Resistance

6.1 Proliferation Resistance (PR) Evaluation

Proliferation resistance evaluation has been carried out based on expert opinion analysed using MAUA and JAEA methodology. In the present Chapter, the data collected and presented in Chapter 4 has been analysed in detail using these two methods. Additionally, the data has also been analysed using modified JAEA methodology.

When multiple alternatives are present, decision making is a complex process. Complexity is enhanced when attributes are a mix of scientific and value judgments. As detailed in section 2.4, various tools have been developed for identifying and evaluating options and these include Multi Attribute Utility Analysis (MAUA), SWOT (Strengths, Weaknesses, Opportunities and Threats) Analysis, Cost-benefit Analysis and Cost-performance Analysis, Risk Analysis and others. MAUA has been shown to provide a viable means for assessing systems with diverse attributes. [Charlton et al., 2007; Cleary et al., 2007].

As described in section 2.5, MAUA can be used to assess different facilities or processes of a fuel cycle independently and the results can be integrated over the complete fuel cycle [Giannangeli, 2007]. MAUA provides a logical method for making choices based on multiple factors [Chirayath, 2010]. A fundamental premise of the utility theory is that each attribute must be "utility independent" of all the others. This means that if all other attributes are held constant, regardless of their value, a change in the value of the attribute will cause a corresponding change in the overall utility value. The general form of the multi-attribute function is given in eq. (1):

$$u(x) = \sum_{i=1}^{n} k_i u_i (x_i) + k\sum_{i=1}^{n} k_i k_j u_i (x_i) u_j(x_j) + \ldots\ldots. k^{n-1} k_1 k_2 \ldots. k_n u_1(x_1) u_2(x_2) \ldots u_n(x_n), \quad \ldots..(1)$$

where the functions u_i are utility functions for the individual attributes, normalised between 0 and 1, the constant k_i are weighting factors for each attribute which indicate an attribute's importance relative to the others, and the constant k is a scaling parameter and that is a solution to

$$1+k = \prod_{i=1}^{n} (1+kk_i). \qquad \ldots..(2)$$

$$u(x) = \sum_{i=1}^{n} k_i u_i (x_i). \qquad \ldots\ldots(3)$$

When the sum of all individual weighting factors k_i is equal to unity, then the scaling parameter $k=0$ and eq. 1 reduces to what is known as the additive utility function:

The additive utility function works out to be a weighted average of all the individual attributes. It is reiterated that each attribute has a utility value $u_i(x_i)$ between 0 and 1 and their weighting factors k_i are also between 0 and 1.

In this study, **PR** assessment by MAUA has been used to compare two facilities, with respect to the influence of twenty measures listed in the Chapter 3. The evaluation does not consider the complete scenario of proliferation threat encompassing nuclear material theft to the manufacture of a nuclear explosive device. The additive MAUA methodology yields value of **PR** between 0 for something that is completely vulnerable to proliferation, and 1.0 for perfect proliferation resistance.

6.2 Analysis of Expert Opinion Using MAUA

For each of the twenty safeguards measures, the opinion of 24 experts was sought. As mentioned in Chapter 5, the measures proposed for enhancing safeguards are of three types. In the first type either the measures available or not available can result in two utility values. For the second type, the proposed measures could be available in different degrees, e.g., automation from 0% to 100%. For such cases, the relation between the degree or the extent to which the measure is available and the utility value has been assessed. The data has been plotted and tabulated. In the third type, the measure is available in different configurations. As described later, co-location of the fuel fabrication facility, the reactor and the reprocessing plant can be in various combinations.

6.2.1 Automation

Automation in fuel fabrication plants is described in section 3.4.10. The results of the assessment of the impact of automation on the utility value of proliferation resistance are plotted in Fig. 6.1. The experts have assigned utility values to various levels of automation beginning from 0% to 100%. The average utility value has been calculated and all points fall on a straight line. As will be shown later in this chapter, automation is one of the features in a bulk handling facility that significantly improves safeguardability. As mentioned earlier, an added advantage of automation is the reduction in the radiation exposure to the operators.

6.2.2 Integration of Process Equipment

Advantages of integration of process equipment in fuel fabrication facilities are described in section 3.4.12. Fig. 6.2 shows the variation of **PR** utility value with implementation of integration of process equipment, based on expert opinion. It has been estimated that though integration of process equipment is desirable, it is not possible to achieve a very high level of integration, due to the fact that the various operations are diverse and the material transforms from bulk handling to item counting during the course of fabrication. About a maximum of 15 % of all process fabrication steps can be integrated. Designing the plant with integrated equipment, wherever feasible, will lead to enhanced proliferation resistance.

6.2.3 Reduction in the Number of Items of Process Equipment

The advantage of reducing the total number of process equipment is described in section 3.4.8. Fig. 6.2 shows the effect of percentage reduction in total items of equipment on the **PR** utility value. Similar to the case of integration of equipment, the overall reduction in total number of items of process equipment may not exceed 15%.

6.2.4 Integration of Quality Control (QC) Equipment in the Main Fabrication Line

Section 3.4.13 describes the usefulness of integrating the QC equipment in the main fabrication line for fuel fabrication facilities. Fig. 6.1 shows the effect of integration of QC equipment with the main fabrication line on the **PR** utility value. Integration of the QC equipment with the main line can enhance the proliferation resistance of the nuclear material handled in such a fabrication facility by reducing the total number of exit points. However, the impact on the utility value of **PR** is low due to a lower quantity of nuclear material in the samples.

6.2.5 Footprint Reduction of the Plant

Advantage of having a lower footprint of a fuel fabrication facility is described in section 3.4.9. For identical production capacity, a fuel fabrication facility having a smaller footprint has better proliferation resistance. While efforts should be made to design the facility with a minimum footprint, the footprint of the plant cannot be drastically reduced. Fig. 6.3 shows the variation of the **PR** utility value with footprint reduction and reduction in the length of ventilation ducting.

6.2.6 Dynamic Nuclear Material Accounting (DNMA) and Near Real Time Monitoring (NRTM) Systems

Usefulness of implementation of DNMA/NRTM in fuel fabrication facilities is described in section 3.4.15. Table - 6.1 gives the variation in utility values due to the incorporation of DNMA and Table - 6.2 based on the incorporation of NRTM, both based on expert opinion. Data has also been plotted in Fig. 6.4. The variation is a non-linear growth function and once-a-shift monitoring is sufficient to arrive at an efficient system from the point of proliferation resistance.

6.2.7 Process Powder Recovery Systems

The usefulness of efficient process powder recovery systems is described in section 3.4.11. To improve proliferation resistance, efforts must be made to reduce both the material holdup and MUF in the facility. It may be noted that the reduction in powder generation and enhanced powder recovery also help in reducing the risk of criticality hazard. Fig. 6.1 shows the relation between the process powder recovery (%) and the utility value of **PR** based on expert opinion.

6.2.8 Isolation of Services in a Fuel Fabrication Facility

The concept of isolation of services is described in section 3.4.4. Isolation of services in fuel fabrication plants has an effect on the overall **PR.** Table - 6.3 shows the effect of isolating services from the main plant on the **PR** utility value.

6.2.9 Nuclear Material Inventory Measurement at Every Box / Cell

The measurement of nuclear material inventory at every box/cell is described in detail in section 3.4.6. Measurement of the inventory of the nuclear material at all the places where it is being handled in a plant greatly enhances its safeguardability. Table - 6.4 shows the values of the **PR** utility function when provision for measurement is made at different areas of the plant.

6.2.10 Material Tracking using RFID, Bar Code Readers, Transmitters and Receivers

As mentioned in section 3.4.7, measure of using bar codes and RFID tags can be incorporated in the fuel fabrication plant for enhancing the safeguardability. Table - 6.5 shows the **PR** utility value resulting from the incorporation of RFID/Laser engraving in the fuel fabrication plants. Table - 6.6 shows the variation in the **PR** utility values due to the provision of computerised material tracking.

6.2.11 Provision of Plant Imaging

Section 3.4.5 describes the systems for plant imaging. Plant imaging can be used to enhance safeguardability of nuclear material by offering larger and real time coverage of the plant using overlapping cameras. Table - 6.7 shows the variation in **PR** utility value when plant imaging is implemented in different areas of the plant.

6.2.12 Provision for Material Storage during Physical Inventory Verification (PIV)

Section 3.4.3 describes the importance of making provision for material storage during PIV in fuel fabrication facilities Table - 6.8 shows the variation in the **PR** utility value for the two cases viz., one where such provision does not exist and the other where such a provision is made.

6.2.13 Co-location of Fuel Fabrication with Reactor and Reprocessing Facilities

The usefulness of co-location is described in section 3.4.2. By reducing or eliminating the need for transportation of the nuclear material in the public domain, the proliferation resistance is greatly enhanced as the risk of theft of the nuclear material is reduced. The fuel cycle facilities can be located in a number of combinations, and the combinations can determine the **PR** value of a plant. Table - 6.9 shows the variation in **PR** utility values for various combinations of co-location.

6.2.14 Optimisation of Nuclear Material Flow in the Fabrication Lines

Section 3.4.18 gives the merit in optimisation of material flow in fabrication lines of fuel fabrication facilities. Optimisation of nuclear material flow can help in reducing the risk of theft by restricting the amount of the nuclear material present in process lines to the minimum. This can greatly improve proliferation resistance. Table - 6.10 shows the variation in the **PR** utility values where such optimisation has been done.

6.2.15 Optimisation of Manpower

Section 3.4.16 describes the merit of optimisation of manpower in fuel fabrication facility. Total number of personnel in a plant can have a bearing on the proliferation resistance. Neither a high number, nor a low number for equivalent capacity of production, is desirable. Fig. 6.5 shows the variation of the **PR** utility value with manpower as per the experts' opinion.

6.2.16 Implementation of Safeguards-by-Design (SBD)

Details of SBD for enhancing **PR** are described in sections 3.4.1 and 3.5. SBD enhances the safeguardability of the facility by the incorporation of various safeguards measures right at the stage of design and has gained importance in recent times. In any facility, SBD can be incorporated at various stages. Table - 6.11 shows the **PR** utility values when safeguards measures are incorporated at different stages of the design and construction.

6.2.17 Improvements in Equipment Design

The advantage of better design of equipment is described in section 3.4.14. Table - 6.12 shows the effect on the **PR** utility value when such measures are provided.

6.3 Relative Importance of Various Measures

Twenty measures for enhancing proliferation resistance in nuclear fuel fabrication facilities for thorium based fuels have been considered above. **PR** utility values for these measures have been derived after discussion with the experts. When a measure is not present, the utility value is close to zero. Similarly, when the measure is present in full, the utility value is close to one. The weighting factors are also assigned values by the experts between zero and one while maintaining the sum of the weighting factors for the twenty measures as one. The variation in **PR** utility values for each of the twenty safeguards measures has already been discussed in detail. For comparison and assessment of overall proliferation resistance by MAUA, two facilities are considered. In one facility, it is assumed that none of the measures are implemented, while in the other, all twenty measures are incorporated in full.

As explained earlier in eq. 3, the general form of the additive multi-attribute utility function used is

$$u(x) = \sum_{i=1}^{n} k_i \, u_i \, (x_i). \qquad \text{.........(4)}$$

where u_i are the utility functions for the individual attributes, normalized between zero and one and k_i are the weighting factors for each attribute indicating an attribute's importance relative to the others. Table - 6.12 shows the variation in utility values and weighting factors for the safeguards measures. It also shows the results of the calculation of the value of proliferation resistance based on MAUA.

The hypothetical facility where no measures are implemented has a proliferation resistance value of 0.1499, while for the facility where all the measures are assumed to be implemented, the proliferation resistance value is 0.7557. The values for **PR** are dependent on the extent of implementation of different measures. To study the contribution of various measures, a sensitivity analysis was carried out. When none of the 20 measures is present, the **PR** value is 0.1499 and when a particular measure is implemented, the **PR** value improves. The extent of improvement will depend upon the extent of implementation of the safeguard measure. Let us take the case of implementing automation. As automation is increased from 0% to 100%, utility value changes from 0.10 to 0.85 and the **PR** value changes from 0.1499 to 0.217. This is plotted in Fig. 6.6 with **PR** value as ordinate and utility value as abscissa. This is studied for all 20 measures. It may be seen that the impact of implementing NRTM on the **PR** value is the greatest.

A reverse exercise was also carried out. When all 20 safeguards measures are fully implemented, the **PR** value is estimated as 0.7557. When automation is removed, it drops to 0.688. **PR** value is plotted against utility value in Fig. 6.7. The effect of removing other measures is similarly evaluated and plotted.

Based on this sensitivity analysis, one can define an importance factor of every measure as the ratio of the overall **PR** value with and without the measure. This can help rank various measures. This can be done by two different methods: one based on the increase in the **PR** value when a measure is added and the other based on the decrease in the **PR** value when a measure is removed. Table - 6.14 lists the safeguards measures in decreasing order of the importance factor for a facility when the importance factor is calculated by addition of a measure and Table - 6.15 when it is calculated based on removal of a measure.

Within the different categories, the importance of each measures in decreasing order are given below;

Conceptual

1) Safeguards-By-Design
2) Co-location of facilities
3) Provision of nuclear material storage during physical inventory verification
4) Isolation of services

Design Related

1) Systems for plant imaging
2) Measurement of nuclear material inventory at every box / cell
3) Nuclear material tracking systems using RFID and bar codes
4) Overall reduction in the total number of items of process equipment

5) Footprint reduction of the plant

6) Reduction in ventilation ducting length

Engineering Related

1) Implementation of automation in the plant

2) Incorporation of efficient process powder recovery systems

3) Integration of process equipment

4) Integration of QC equipment with main process equipment

5) Improvements in equipment design

Operational

1) Implementation of near real time monitoring

2) Implementation of dynamic nuclear material accounting

3) Optimization of overall manpower deployment

4) Computerized tracking of nuclear material in the plant

5) Optimization of nuclear material flow in fabrication lines

6.4 Analysis of Expert Opinion by JAEA Methodology

For the twenty safeguards measures, the feedback from the experts was also obtained to evaluate **PR** using the JAEA method. In the JAEA method the proliferation resistance range is from 0 to 5, where 0 indicates no proliferation resistance and 5 indicates maximum proliferation resistance. Another difference is that the sum of the weightage factors for all the attributes need not be 1. This means that the overall **PR** value calculated can be more than 1. Table – 6.16 shows the calculation of **PR** for the two plants using JAEA methodology. For the plant not having any of the studied safeguards measures, the overall **PR** is 15.605. Similarly for a plant having all the measures for safeguards incorporated, the **PR** value is 43.246. It is also noticed that the sum of all the weightages for all the 20 measures is 10.79.

It is difficult to compare these numbers from those arrived at by using MAUA. Therefore, the JAEA methodology was modified. This is done by normalizing the weightages in a way that the sum of all the weightages is 1.0. Table -6.17 shows the calculated values for the two plants by what can be called as modified JAEA methodology. The values for the plant not having any measures is 0.2893 and the **PR** value for plant having all the measures in full is 0.8019. If these values are compared to those arrived at by MAUA, viz. 0.1499 and 0.7557, they reflect a similar pattern of **PR** values.

6.5 Highlights of Comparison of MAUA and JAEA Methodologies

The proliferation resistance has been evaluated using two methods; MAUA and the modified JAEA method. Two plants have been compared. In one plant none of the measures are present while in the other all the measures have been implemented. For both the methods, expert opinion has been used to analyse and arrive at the **PR** values. In the **PR** evaluation by MAUA, sensitivity analysis was also carried out to study the contribution of measures to improvement in the **PR** value. An importance factor was defined and was calculated to rank the 20 safeguards measures. It is found that the following measures are very important for improving safeguardability of a fuel cycle facility (Table – 6.15).

1. Near Real Time Monitoring – (O)
2. Automation (E)
3. Safeguards-By-Design (C)
4. Dynamic Nuclear Material Accounting (O)
5. Plant Imaging (D)

The evaluation by the **PR** by JAEA method has shown that the most important measures are as below (Table - 6.16 & Table - 6.17);

1. Automation – (E)

2. Near Real Time Monitoring (O)
3. Powder Reduction (E)
4. Dynamic Nuclear Material Accounting (O)
5. Safeguards-By-Design

On comparison of results of **PR** by MAUA and modified JAEA, the similarities in the overall values is evident, viz. for the plant having none of the measures the MAUA yields **PR** value of 0.1499 and modified JAEA yields 0.2813. Similarly for the plant having all the 20 measures MAUA yields the **PR** value 0.7557 and modified JAEA gives 0.816. On calculation of ratio of the two values for the two plants, in case of modified JAEA

$(PR_{20measures}) / (PR_{0measures}) = 0.816 / 0.2813 = 2.9$

and for MAUA

$(PR_{20measures}) / (PR_{0measures}) = 0.7557 / 0.1499 = 5.04$

In terms of the measures that are important for affecting the overall **PR** value, the two methods have 4 measures which are common;

1. Near Real Time Monitoring – (O)
2. Automation (E)
3. Safeguards-By-Design (C)
4. Dynamic Nuclear Material Accounting (O)

Only in case of evaluation of **PR** by JAEA, plant imaging is also one of the top 5 measures and in case of evaluation of **PR** by MAUA it is powder reduction.

Though the safeguards measures described in the study have been evaluated to enhance the proliferation resistance for thorium based fuel fabrication plants, the measures are general in nature and are applicable equally to facilities handling fuels other than thorium, to the extent those design features are present in the other fuel cycles.

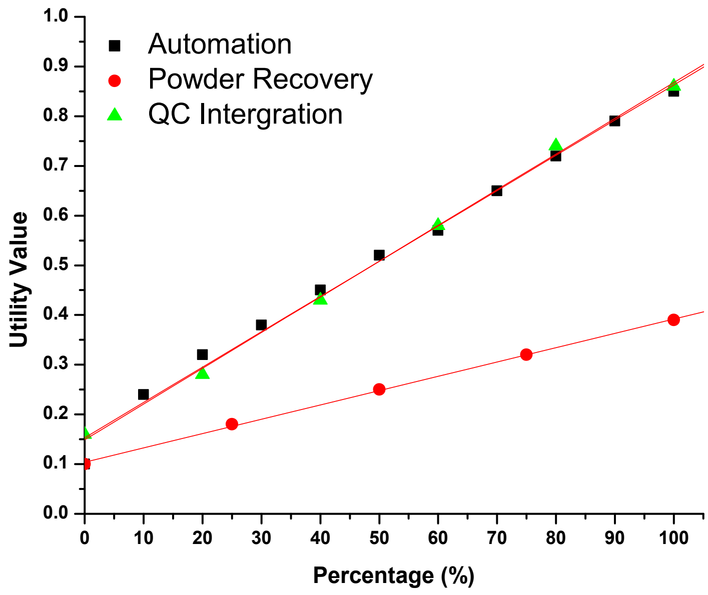


Figure – 6.1 : Effect of Automation, QC Equipment Integration and Process Powder Recovery on PR Utility Value

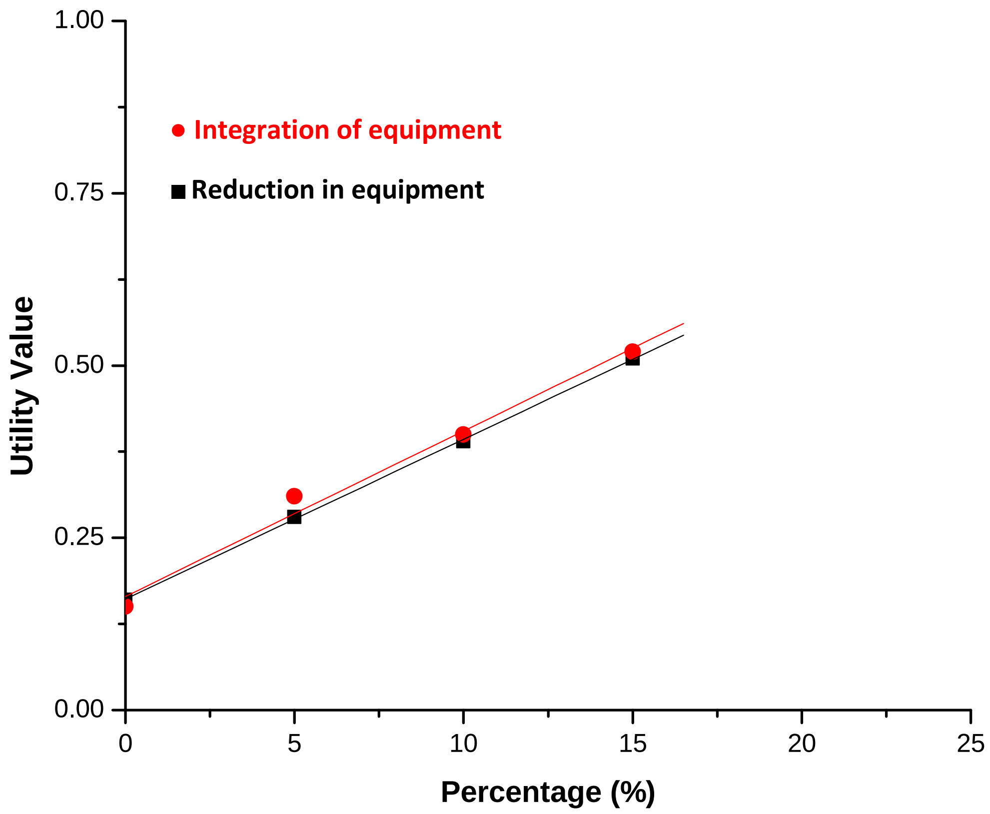


Fig. 6.2 : Effect of Integration of Process Equipment and Reduction in Equipment on PR Utility Value

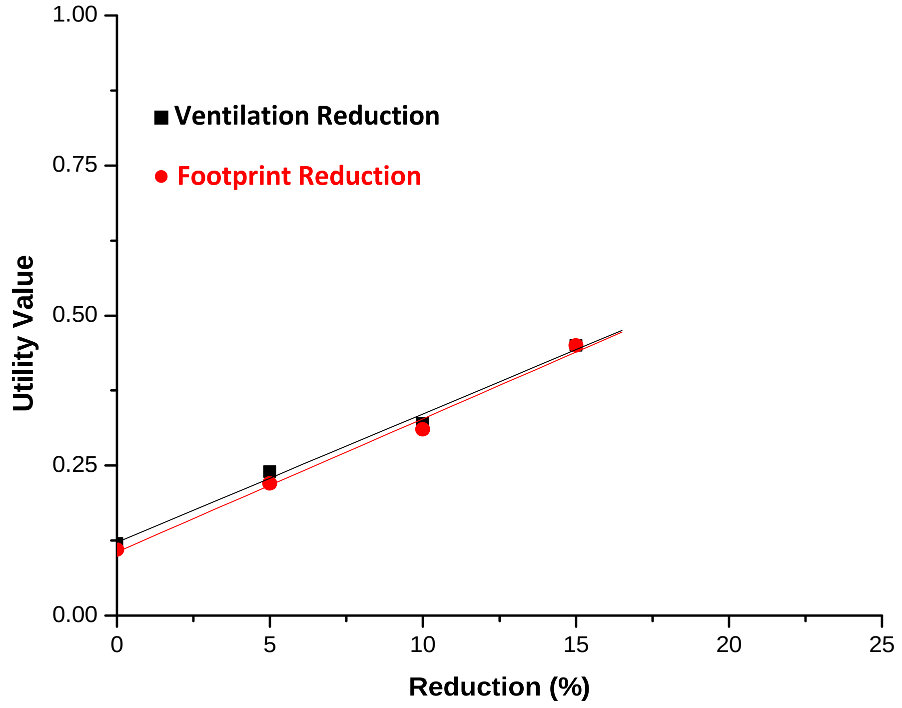


Fig. 6.3 : Effect of Reduction in Footprint and Reduction in Length of Ventilation Ducting on PR Utility Value

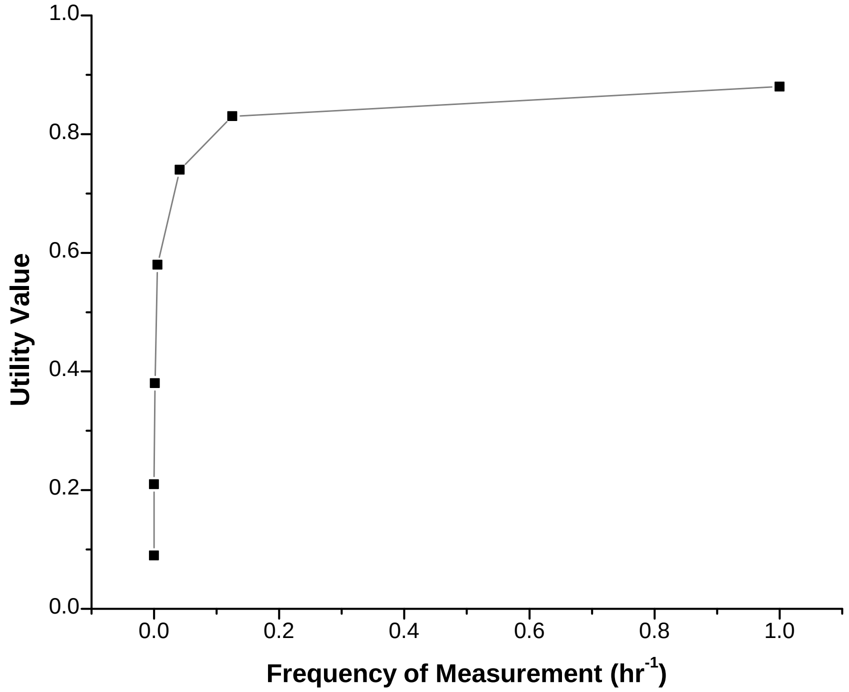


Fig. 6.4 : Effect of Frequency of Measurement on PR Utility Value

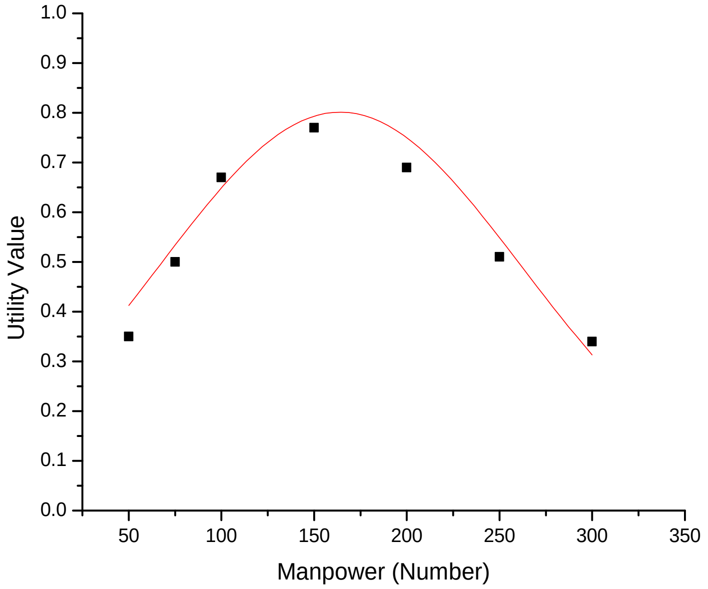


Fig. 6.5 : Effect of Manpower on PR Utility Value

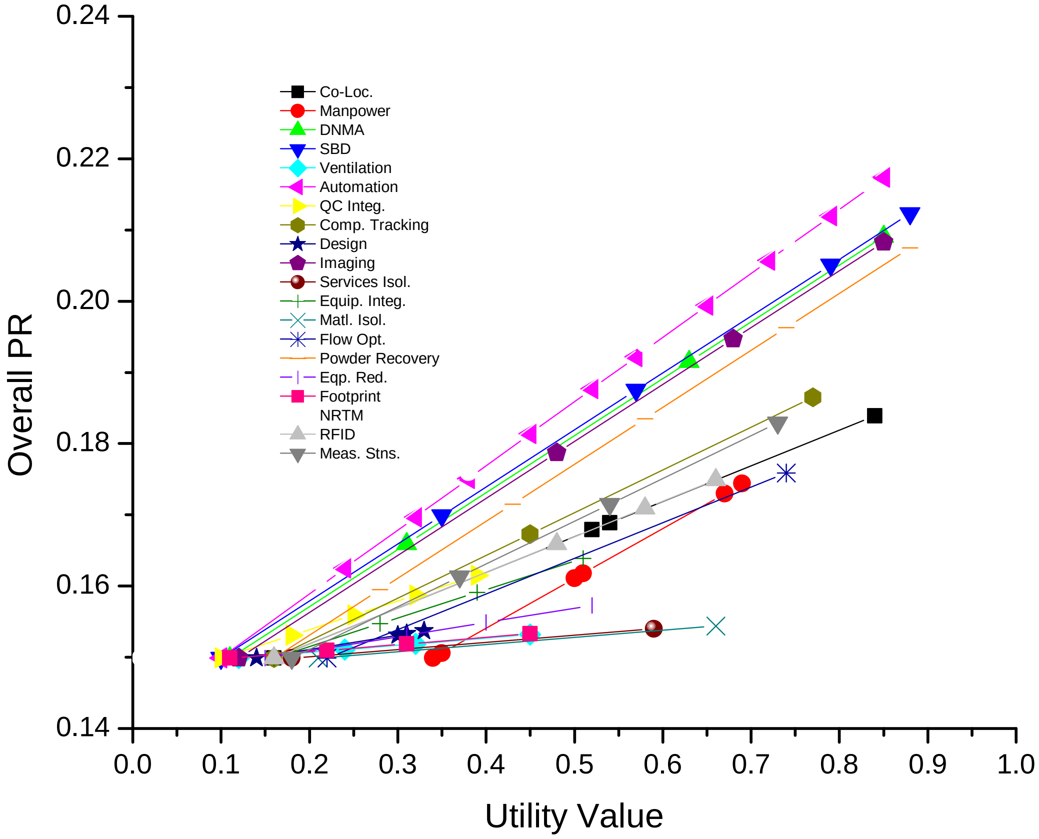


Fig. 6.6 : Effect of Individual Measures on Increase in Overall PR

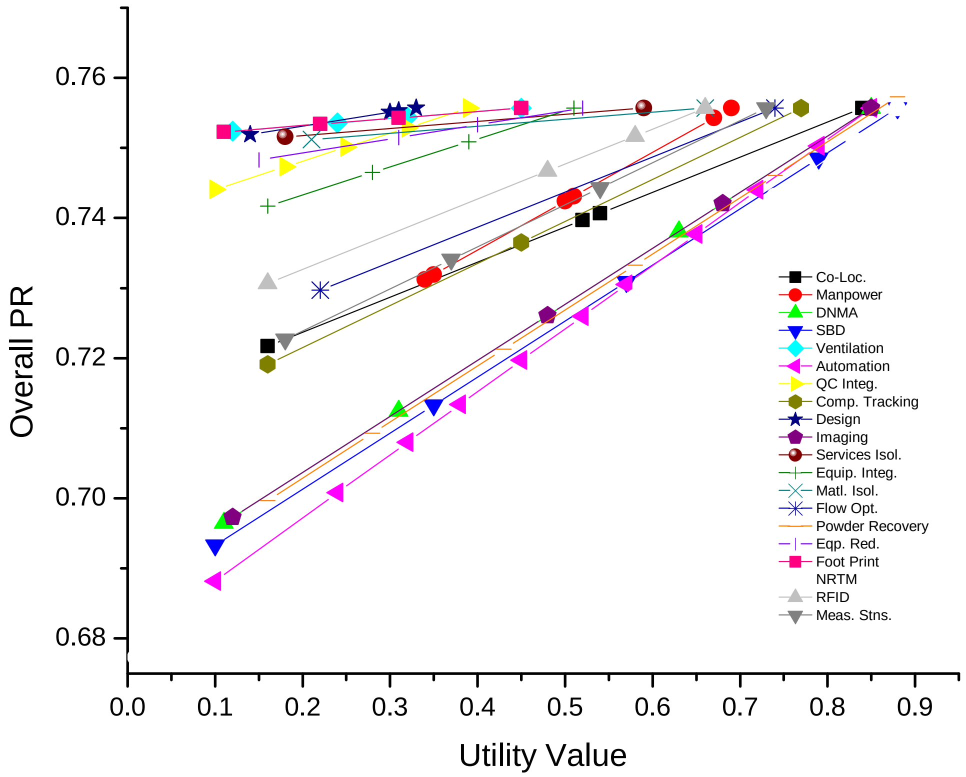


Fig. 6.7 : Effect of Individual Measures on Decrease in Overall PR

Table - 6.1 – Variation in PR Utility Value due to Progressive Implementation of DNMA

DNMA	Utility Value		
	Minimum	Maximum	Average
Not available	0.01	0.2	0.11
All Entry and Exit Points	0.02	0.6	0.31
Every Processing Area	0.25	0.9	0.63
Every Box / Cell	0.4	0.97	0.85

Table – 6.2 – Variation in PR Utility Value for Different Frequencies of NRTM

NRTM	Utility Value		
	Minimum	Maximum	Average
Not Available	0.01	0.5	0.09
Yearly	0.09	0.5	0.21
Monthly	0.10	0.6	0.38
Weekly	0.16	0.75	0.58
Daily	0.2	0.9	0.74
Every Shift	0.4	0.90	0.83
Hourly	0.7	0.95	0.88

Table - 6.3 – Variation in PR Utility Value due to Provision of Isolation of Services

Isolation of Services	Utility Value		
	Minimum	Maximum	Average
Not provided	0.02	0.4	0.18
Provision made	0.4	0.8	0.59

Table - 6.4 – Variation in PR Utility Value due to Provision of Measurement in Different Areas of Plant

Measurement in Areas	Utility Value		
	Minimum	Maximum	Average
No Provision	0.05	0.3	0.18
Every Processing Area	0.2	0.5	0.37
Every Processing Line	0.3	0.8	0.54
Every Processing Box / Cell	0.35	0.90	0.73

Table - 6.5 – Variation in PR Utility Value due to Incorporation of Laser Engraving / RFID

Laser Engraving / RFID	Utility Value		
	Minimum	Maximum	Average
No Provision	0.05	0.45	0.16
RFID on Containers	0.3	0.6	0.48
Individual Pin Engraving	0.4	0.7	0.58
Engraving on Fuel Assembly	0.4	0.8	0.66

Table - 6.6 – Variation in PR Utility Value due to Provision of Computerised Material Tracking

Computerised Material Tracking	Utility Value		
	Minimum	Maximum	Average
No Provision	0.05	0.4	0.16
Only in Process Areas	0.2	0.6	0.45
Provision in Individual Box / Cell	0.3	0.9	0.77

Table - 6.7 – Variation in PR Utility Value due to Provision of Plant Imaging

Plant Imaging	Utility Value		
	Minimum	Maximum	Average
No Provision	0.05	0.3	0.12
General Plant Coverage	0.2	0.6	0.48
Processing Area Coverage	0.3	0.8	0.68
Individual Box / Cell Coverage	0.5	0.95	0.85

Table - 6.8 – Variation in PR Utility Value due to Provision of Isolation of Nuclear Material during PIV

Provision for Isolation of Nuclear Material during PIV	Utility Value		
	Minimum	Maximum	Average
No Provision	0.05	0.4	0.21
Provisions made at various places in the fabrication line	0.4	0.8	0.66

Table - 6.9 – Variation in PR Utility Value due to Co-Location of Plants

Co-Location	Utility Value		
	Minimum	Maximum	Average
No Provision	0.01	0.3	0.16
Fabrication and Reactor	0.2	0.7	0.52
Fabrication and Reprocessing	0.2	0.8	0.54
Fabrication, Reactor and Reprocessing	0.4	0.95	0.84

Table - 6.10 – Variation in PR Utility Value due to Optimization of Material Flow

Material Flow	Utility Value		
	Minimum	Maximum	Average
Not optimised	0.05	0.5	0.22
Optimised by Computer Analysis	0.6	0.9	0.74

Table - 6.11 – Variation in PR Utility Value due to Incorporation of Safeguards at Different Stages of Design and Construction

Incorporation of safeguards	Utility Value		
	Minimum	Maximum	Average
No provision	0.01	0.2	0.10
Post Construction	0.1	0.5	0.35
Construction Stage	0.2	0.7	0.57
Design Stage	0.5	0.85	0.79
Concept Stage	0.7	0.95	0.88

Table - 6.12 – Variation in PR Utility Value due to Design Improvement

Design Improvement	Utility Value		
	Minimum	**Maximum**	**Average**
No Provision	0.02	0.25	0.14
Smooth Surface Finish	0.2	0.5	0.31
No Blind Spots	0.2	0.5	0.30
No Sharp Corners	0.2	0.6	0.33

Table - 6.13 – Weighting Factors and Change in PR Utility Value due to addition of a measure and Overall PR

No.	Measure	Weighting factors (k_i)			Utility Value (u_i)					
					Old Plant			New Plant		
		Min.	Max.	Avg.	Min.	Max.	Avg.	Min.	Max.	Avg.
1.	NRTM	0.02	0.10	0.09	0.01	0.70	0.09	0.09	0.88	0.88
2.	DNMA	0.02	0.10	0.08	0.01	0.40	0.11	0.20	0.97	0.85
3.	SBD	0.01	0.10	0.08	0.01	0.70	0.10	0.20	0.95	0.88
4.	Co-Location*	0.01	0.10	0.05	0.01	0.40	0.16	0.30	0.95	0.84
5.	Plant Imaging*	0.03	0.10	0.08	0.05	0.50	0.12	0.30	0.95	0.85
6.	Equipment Reduction*	0.01	0.05	0.02	0.05	0.40	0.15	0.30	0.80	0.52
7.	Footprint Reduction*	0.005	0.05	0.01	0.05	0.40	0.11	0.25	0.70	0.45
8.	Ventilation Reduction*	0.005	0.03	0.01	0.05	0.50	0.12	0.20	0.70	0.45
9.	Automation*	0.01	0.10	0.09	0.01	0.30	0.10	0.40	0.95	0.85
10.	Manpower Reduction*	0.01	0.08	0.07	0.01	0.15	0.34	0.30	0.95	0.69
11.	Equipment Integration*	0.02	0.05	0.04	0.01	0.40	0.16	0.30	0.80	0.51
12.	Isolation of Material for PIV*	0.005	0.05	0.01	0.05	0.40	0.21	0.40	0.80	0.66
13.	Material Flow Optimisation*	0.01	0.07	0.05	0.05	0.60	0.22	0.50	0.90	0.74
14.	QC Integration*	0.01	0.07	0.04	0.01	0.20	0.10	0.20	0.60	0.39
15.	RFID Tagging*	0.01	0.06	0.05	0.05	0.40	0.16	0.45	0.80	0.66
16	Computer Tracking*	0.01	0.07	0.06	0.05	0.30	0.16	0.40	0.90	0.77
17	Powder Reduction*	0.05	0.10	0.08	0.05	0.30	0.16	0.50	0.95	0.86
18	Measurement at every station*	0.01	0.08	0.06	0.05	0.35	0.18	0.30	0.90	0.73
19	Design Improvement*	0.01	0.05	0.02	0.02	0.20	0.14	0.25	0.60	0.33
20	Isolation of Services*	0.005	0.05	0.01	0.02	0.40	0.18	0.40	0.80	0.59

* Indicate measures proposed for the first time

$u(x) = \sum_{i=1}^{20} k_i u_i (x_i)$	Old Plant	New Plant
	0.1499	0.7557

Table - 6.14 – Importance Factors of Safeguard Measures Calculated by Addition of Measures

Rank	Safeguard Measures	Importance Factor	Type
1	Near Real Time Monitoring	1.103	Operational
2	Automation	1.098	Engineering
3	Dynamic Nuclear Material Accounting	1.084	Operational
4	Plant Imaging	1.083	Design
5	Safeguards –By-Design	1.080	Conceptual
6	Powder Reduction	1.080	Engineering
7	Computer Tracking	1.050	Operational
8	Co-Location	1.047	Conceptual
9	Measurement at Every Station	1.045	Design
10	Material Flow Optimisation	1.035	Operational
11	RFID Tagging	1.034	Design
12	Manpower Reduction	1.033	Operational
13	Isolation of Material	1.019	Conceptual
14	QC Integration	1.015	Engineering
15	Equipment Reduction	1.009	Design
16	Equipment Integration	1.005	Engineering
17	Design Improvement	1.005	Engineering
18	Isolation of Services	1.005	Conceptual
19	Footprint Reduction	1.004	Design
20	Ventilation Reduction	1.004	Design

Table - 6.15 – Importance Factors of Safeguard Measures Calculated by Removal of Measures

Rank	Safeguard Measures	Importance Factor	Type
1	Near Real Time Monitoring	1.474	Operational
2	Automation	1.450	Engineering
3	Safeguards –By-Design	1.416	Conceptual
4	Dynamic Nuclear Material Accounting	1.394	Operational
5	Plant Imaging	1.389	Design
6	Powder Reduction	1.373	Engineering
7	Computer Tracking	1.244	Operational
8	Co-Location	1.227	Conceptual
9	Measurement at Every Station	1.220	Design
10	Material Flow Optimisation	1.173	Operational
11	RFID Tagging	1.167	Design
12	Manpower Reduction	1.163	Operational
13	Equipment Integration	1.093	Engineering
14	QC Integration	1.077	Engineering
15	Footprint Reduction	1.057	Design
16	Equipment Reduction	1.049	Design
17	Isolation of Material	1.030	Conceptual
18	Isolation of Services	1.027	Conceptual
19	Design Improvement	1.025	Engineering
20	Ventilation Reduction	1.015	Design

Table - 6.16 – Analysis Using the Conventional JAEA Methodology

Process Step	**Weightage**	**Old**	**New**	**Old Wt**	**New Wt**
Near Real Time Monitoring	0.87	1.21	4.28	1.0527	3.7236
Dynamic NMA	0.79	1.96	4.05	1.5484	3.1995
Safeguards by Design	0.78	1.34	4.08	1.0452	3.1824
Co- Location	0.61	1.61	4.14	0.9821	2.5254
Surveillance Imagery	0.75	1.33	4.2	0.9975	3.15
Equipment Reduction	0.3	2.38	3.62	0.714	1.086
Footprint Reduction	0.15	2.09	3.58	0.3135	0.537
Ventilation System Reduction	0.16	1.93	3.28	0.3088	0.5248
Automation	0.88	1.18	4.47	1.0384	3.9336
Manpower Reduction	0.67	1.52	4.07	1.0184	2.7269
Integration of Equipment	0.45	1.57	3.5	0.7065	1.575
Ease of Isolation	0.29	1.48	2.86	0.4292	0.8294
Material Flow Optimisation	0.6	1.8	4.05	1.08	2.43
QC Integration	0.45	1.42	3.24	0.639	1.458
RFID / Laser Tagging	0.5	1.35	3.94	0.675	1.97
Computerised Tracking	0.55	1.27	3.99	0.6985	2.1945
Powder Reduction	0.81	1.1	4.35	0.891	3.5235
Measurement at Each Station	0.67	1.28	4.27	0.8576	2.8609
Design Improvement	0.25	1.19	3.62	0.2975	0.905
Isolation of Services	0.26	1.2	3.5	0.312	0.91
	10.79			15.605	43.246

Table - 6.17 – Analysis Using the Modified JAEA Methodology

Process Step	Wt.	Wt Normalised	Old	New	Old Normalised	New Normalised	Old Wt	New Wt
Near Real Time Monitoring	0.87	0.0806302	1.21	4.28	0.242	0.856	0.0195	0.069
Dynamic NMA	0.79	0.0732159	1.96	4.05	0.392	0.81	0.0287	0.0593
S-B-D	0.78	0.0722892	1.34	4.08	0.268	0.816	0.0194	0.059
Co- Location	0.61	0.0565338	1.61	4.14	0.322	0.828	0.0182	0.0468
Imagery	0.75	0.0695088	1.33	4.2	0.266	0.84	0.0185	0.0584
Equipment Reduction	0.3	0.0278035	2.38	3.62	0.476	0.724	0.0132	0.0201
Footprint Reduction	0.15	0.0139018	2.09	3.58	0.418	0.716	0.0058	0.01
Ventilation System Reduction	0.16	0.0148285	1.93	3.28	0.386	0.656	0.0057	0.0097
Automation	0.88	0.081557	1.18	4.47	0.236	0.894	0.0192	0.0729
Manpower Reduction	0.67	0.0620945	1.52	4.07	0.304	0.814	0.0189	0.0505
Integration of Equipment	0.45	0.0417053	1.57	3.5	0.314	0.7	0.0131	0.0292
Ease of Isolation	0.29	0.0268767	1.48	2.86	0.296	0.572	0.008	0.0154
Material Flow Optimisation	0.6	0.055607	1.8	4.05	0.36	0.81	0.02	0.045
QC Integration	0.45	0.0417053	1.42	3.24	0.284	0.648	0.0118	0.027
RFID / Laser Tagging	0.5	0.0463392	1.35	3.94	0.27	0.788	0.0125	0.0365
Computerised Tracking	0.55	0.0509731	1.27	3.99	0.254	0.798	0.0129	0.0407
Powder Reduction	0.81	0.0750695	1.1	4.35	0.22	0.87	0.0165	0.0653
Measurement at Each Station	0.67	0.0620945	1.28	4.27	0.256	0.854	0.0159	0.053
Design Improvement	0.25	0.0231696	1.19	3.62	0.238	0.724	0.0055	0.0168
Isolation of Services	0.26	0.0240964	1.2	3.5	0.24	0.7	0.0058	0.0169
	10.79	1					0.2893	0.8016

Chapter 7

Conclusions and Directions for Future Work

Thorium fuel cycle is a key to India's nuclear programme. It is attractive from the viewpoint of proliferation resistance as well. The closed fuel cycle facilities comprise of the nuclear reactors, fuel fabrication facilities and the reprocessing plants. While safeguards are easier to implement in case of nuclear reactors, they pose challenges in the fuel fabrication and reprocessing facilities. Conventionally, the nuclear fuel is fabricated using the powder pellet route. In power pellet type of fabrication facilities, the nuclear material is present in the form of powder, green pellets, sintered pellets and fuel pins. This makes safeguards implementation in such facilities more demanding, but in the opinion of author more achievable. As the concerns of nuclear proliferation grow, there have been advancements in the field of safeguards implementation. Such measures are legal, administrative and technical. In the present study, various safeguards measures have been proposed to enhance safeguardability of the nuclear fuel fabrication facility. Additionally, measures have been identified that can be implemented as SBD in thorium based powder-pellet fabrication plants. A comparison based on safeguards measures has been carried out to study two configurations of Thorium Fuel Cycle Facilities. An assessment on the impact of safeguards measures on the overall **PR** was also carried out by seeking expert opinion and analysing opinions using MAUA and JAEA methodology. Based on the work carried out in the present study, the following conclusions have been arrived at:

1) This study has proposed a design for hybrid layout for fuel fabrication line for Thorium Fuel Fabrication facility. This design has found to have enhanced safeguardability features as compared to linear layout.

2) This study has presented 20 measures which can be implemented in fuel fabrication facilities for thorium fuels. Three of these proposed measures are improvements on the existing methods, but 17 measures are original ideas which are presented in this thesis. These 20 measures have been classified under four categories, viz. a) Conceptual, b) Design, c) Engineering and d) Operational.

3) The effect of these 20 measures on the **PR** of a fuel fabrication plant has been studied in detail using MAUA and JAEA methodologies. Sensitivity analysis has also been carried out to study the contribution of measures to improvement in the **PR** value. An importance factor was also defined and was calculated to rank the 20 safeguards measures. Although all the measures were found to be helpful in enhancing the overall PR, the following measures were found to have major impact.

    a. Near Real Time Monitoring
    b. Automation
    c. Safeguards-By-Design
    d. Dynamic Nuclear Material Accounting
    e. Plant Imaging
    f. Powder reduction

4) The present study has identified 11 safeguards measures for effective implementation as SBD in fuel fabrication plants. This study has shown merits of incorporation of safeguards measures in such plants at the design stage itself.

5) A detailed comparison of two configurations of thorium fuel cycle facilities has been carried out on the basis of safeguards implementation. The study showed the merits and limitations of hub-and-spoke and self contained configurations.

Though the safeguards measures described in the study have been evaluated to enhance the proliferation resistance for thorium based fuel fabrication plants, the measures are general in nature and are applicable equally to facilities handling fuels other than thorium, to the extent those design features are present in the other fuel cycles.

Directions for Future Work

The work that flows as an extension of the current study can be of various types;

a) Study for uranium and plutonium bearing fuels: The thorium bearing fuels are processed in alpha tight shielded hot cells. Uranium fuels can be fabricated in natural atmosphere and plutonium bearing fuels need alpha tight glove boxes. The three type of facilities have different challenges in terms of safety, security and safeguards. Detailed and separate studies can be carried out for such facilities.

b) Novel safeguards measures for spent fuel reprocessing facilities: The current study has focus on powder pellet type of fuel fabrication facilities. In fuel cycle, the reprocessing plants contain nuclear material in bulk form as sintered pellets, solutions and powders. Safeguards measures for implementation in such facilities can be studied.

c) **PR** evaluation by additional methods: The current study has used MAUA and method based on JAEA for evaluation of **PR.** Other methods for evaluation of **PR** can be attempted and the results of various methodologies can be compared.

APPENDIX- 1

India's Nuclear Power Programme

In India, nuclear energy development began with the objectives of improving the quality of life of the people and self-reliance in meeting their energy needs as compared to the other advanced countries where nuclear power came about as an out come from the development of the military programme. Number of research reactors were progressively designed and constructed during different stages of the nuclear programme in India. In 1956, a swimming pool type of reactor (APSARA) was built and it was the first Asian reactor to go critical. Subsequently in 1960, a 40 MWth reactor CIRUS was constructed for research applications. In early 1961, a zero energy critical facility named ZERLINA was built, for studying lattice parameters with heavy water moderation and natural uranium as fuel. Later on, need for the higher neutron flux and larger irradiation volume led to construction of 100 MWth research reactor DHRUVA. In addition, the first reprocessing plant was set up in 1965 to separate plutonium from irradiated uranium fuel of the research reactors. Another test reactor named PURNIMA was built for studying the behaviour of plutonium fuel in a Pulsed Fast Reactor (PFR). Following this, a critical facility called PURNIMA-2 was also designed, with a solution containing 400 gms of uranyl nitrate serving as the fuel for this facility. Apart from thermal power reactors, a Fast Breeder Test Reactor (FBTR) was commissioned in 1985, with indigenous uranium-plutonium mixed carbide fuel, providing valuable design and operational experience. In addition, as a part of studies with U^{233} fuel, a 30 kW pool type research reactor, KAMINI, was designed and built in 1996. This reactor is being extensively used as a neutron source for research applications such as neutron radiography of irradiated nuclear fuel. The development of most of these reactor systems is an integrated result of detailed understanding of science and technology involved and provided useful experience for power reactor technology. Capability in all aspects of R&D, reactor physics design, engineering design and safety analysis was demonstrated in the country by DAE and other R&D organisations. In short, self-reliance through R&D has been the hallmark of the Indian nuclear power programme right from the inception of Department of Atomic Energy (DAE). India has become one of the few countries in the world that has acquired expertise in the entire range of nuclear fuel cycle activities.

DAE is responsible for implementation of nuclear programme in India. It has under it a number of research centres where research is carried out in the fields of basic sciences, astronomy, astrophysics, cancer research & healthcare, nuclear agriculture and desalination of water, etc. Advanced technologies like accelerators, lasers and supercomputers are key areas of development. In addition, work is being carried out on applications of radiation in processing of food, industrial and medical products. The Department is also involved in strengthening of education and research in nuclear sciences and technologies and allied disciplines. All aspects of nuclear programme, prospecting, mining and processing of uranium, fuel fabrication, fabrication of structures and tubular materials of different nuclear materials for reactor use, industrial scale heavy water production for the pressurized heavy water reactors (PHWRs), boron carbide production for control material for reactors and manufacture of nuclear electronic instrumentation are also being carried out by the Department. The nuclear power programme is being integrated and executed by two organizations namely Nuclear Power Corporation of India Ltd. (NPCIL) for thermal reactors of the first stage programme of DAE and Bharatiya Nabhikiya Vidyut Nigam Ltd. (BHAVINI) for Fast Breeder Reactors of the second stage programme of DAE. As part of human resource development, a regular Training School started functioning in 1957 and the programme has evolved as a primary resource for providing training in nuclear science and technology to engineers and scientists from various disciplines. India has a large pool of highly qualified and trained nuclear engineers and scientists.

Three Stage Nuclear Programme

Figure-1 shows schematic of India's three-stage nuclear power programme. The three-stage power programme of DAE has been planned based on a closed fuel cycle concept, requiring reprocessing of spent fuel from every reactor. The objective is to judiciously utilize mined uranium and thorium resources of the country to a maximum extent. Uranium has just 0.7% fissionable U^{235} isotope while U^{238}, the balance dominant isotope of uranium is a fertile material. It needs to be converted to fissile Pu^{239} for further use. The process of this conversion takes place in a nuclear reactor where uranium fuel is used. Uranium's fissile isotope, U^{235} present as a very small fraction in uranium, produces excess neutrons over and above those required for maintaining a steady fission chain reaction. Some of these excess neutrons invariably get absorbed in the major isotope U^{238} and result in production of Pu^{239}. By suitable reactor physics design of a nuclear reactor the production of Pu can be optimized. Thus, the spent fuel from thermal reactors contains a small quantity of Pu^{239}, along with residual uranium (predominantly U^{238}). The spent fuel can be reprocessed chemically to separate plutonium, residual uranium and the fission products, etc. Pu^{239} is fissile material and when used in fast reactors, where neutron energies are kept high (by not slowing them through use of a moderator as in thermal reactors), is more efficient in producing excess neutrons during fission chain reaction. These excess neutrons are used by proper reactor physics design to convert U^{238} into additional plutonium. By suitable choice of fuel type and reactor configurations such fast reactors can produce a little more Pu than they consume, hence breed more fuel from spent uranium obtained from reprocessing, thus the name 'Breeder' Reactors. Similarly, thorium is a fertile material and has to be converted to a fissile material, viz. U^{233} to be used for power production.

A closed fuel cycle approach as mentioned above, involving reprocessing of spent fuel to separate the useful fissile and fertile isotopes from spent fuel and reusing them in nuclear reactors has been adopted as a guiding principle for our nuclear energy programme to ensure long term energy security for the country. The second stage, comprising of Fast Breeder Reactors (FBRs) are fuelled by fuels based on plutonium mixed with reprocessed uranium recovered by reprocessing of the first stage spent fuel. In FBRs, Pu^{239} undergoes fission producing energy, and at the same time, producing Pu^{239} by transmutation of U^{238}. Over a period of time, growing plutonium inventory can multiply the number of FBRs. The process of increasing the nuclear power capacity can thus be achieved to a desired level in the country through plutonium based FBRs. Th^{232} is not fissile and has to be converted to U^{233} by transmutation in a reactor for use as a fissile material. In the second stage, once sufficient nuclear power capacity is built through plutonium-based FBRs, Th^{232} will be introduced as a blanket material to be converted to U^{233}. The third stage of the programme will be using a Th^{232}–U^{233} fuel cycle in the reactors. Direct use of Th^{232} as a fuel will thus be in the third stage reactors. Th^{232}–U^{233} fuel cycle does not permit attractive breeding characteristics like that of Pu–U cycle but would facilitate the nuclear power capacity built during second stage of the programme to be sustained for as long as thorium, which is quite large in the country, is available.

The technology for dealing with low and intermediate level radioactive wastes from nuclear power plants has been well-established and the processes for treatment and disposal are in practice for the past several decades. India has developed the technologies for waste management of high level liquid waste using immobilization in inert glass matrix by a process of vitrification. After intermediate storage ultimate disposal of immobilized waste will be in a repository. In addition to conditioning as well as disposal, including deep geological repository development in partitioning and transmutation of waste are also being pursued.

First stage

First stage of Indian Nuclear Power Programme comprises of thermal reactors fuelled by uranium. Two units of boiling water reactors (BWR) with US collaboration were set up at Tarapur in 1964. Subsequently, pressurized heavy water (PHWRs) were selected to be set up in different parts of the country as these reactors produce plutonium efficiently which is required to fuel the second stage on Indian nuclear programme. The reactors were set up at Rajasthan, Madras, Narora, Kakrapar and Kaiga. These PHWRs were designed for generating 220 MW electricity. Successful commissioning and subsequent operation of these stations have demonstrated that India has mastered the technology involved and is fully capable of utilizing the same in the commercial domain. To further optimize the power generation capacity of the indigenous nuclear reactors, 540 MWe PHWRs were designed and constructed. Subsequently, these PHWRs have been optimized further to a design capacity of 700 MWe. Currently, four units of such PHWRS are under various stages of construction. Additionally, two units of 1000 MWe VVER type of reactors in collaboration with Russian Federation were planned. The first unit is operational and the second unit is under commissioning.

Nuclear safety is an important aspect of India's nuclear power programme. Defense-in-depth concept has been adopted for ensuring safe operation of Indian nuclear reactors. Post Fukushima accident, a review of Indian nuclear reactors have been carried out and necessary upgrades have been incorporated.

Second stage

The second stage of the programme began with the construction of a 40 MWe fast breeder test reactor (FBTR) in 1985. This indigenous reactor is fueled with mixed uranium plutonium carbide. A 500 MWe Prototype Fast Breeder Reactor (PFBR) is at final stages of construction. This reactor will be a stepping stone for future fast reactors of India which will breed additional plutonium and uranium-233 from thorium. Uranium-233 and plutonium generated from these fast reactors will be driver fuels for the reactors of the third stage.

Third stage

Thorium is at the centre stage of the third stage of the programme. The third stage technology for utilization of thorium has been demonstrated in small measures. For example, the KAMINI reactor, in IGCAR, the only currently operating reactor which uses U^{233} as fuel. This fuel was bred, processed and fabricated indigenously. Efforts are currently on to enlarge that experience to a bigger scale.

As part of thorium utilization an Advanced Heavy Water Reactor (AHWR) is being designed. The AHWR is an innovative concept, which is a bridge between the first and third stage systems. It uses light water as coolant and heavy water as moderator. It can be fuelled to suit a variety of fuel cycles like a mixture of enriched uranium and thorium-based fuel or plutonium and thorium, with a sizeable amount of power coming from Thorium. This reactor also has been designed to use a number of advanced passive safety features making it the next generation safe plant.

India is also working on high temperature reactor-based power packs for high temperature process heat and hydrogen fuel production, accelerator driven systems and molten salt reactors including breeders for generation of electrical power, high temperature heat and transmutation of minor actinides. India is also a partner in the international experimental initiative on harnessing fusion power for the future, the International Thermonuclear Experimental Reactor (ITER) project. India is supplying several components for the experimental reactor.

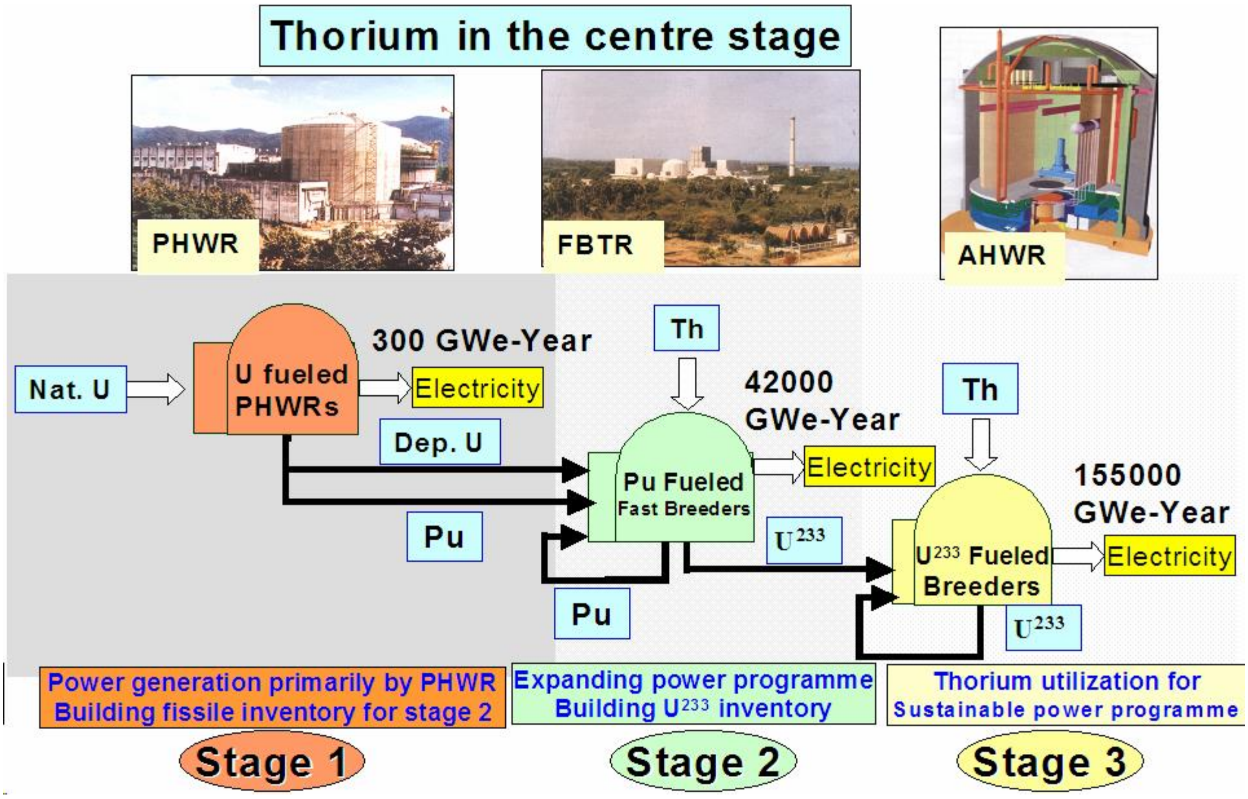


Fig. A-1 India's Three Stage Nuclear Power Programme

REFERENCES

1. Acheson-Lilienthal, Report on the International Control of Atomic Energy, http://www.learnworld.com/ZNW/LWText.Acheson-Lilienthal.html accessed on 20th October, (2013).

2. Artisyuk, V., Saito, M., and Ezoubtchenko, A., "Development of Methodology to Assess Proliferation Resistance of Nuclear Heavy Metals", Progress in Nuclear Energy **50**, 647-653 (2008).

3. Bari, R., Peterson, P., Roglans, J., and Mladineo, S., Report on "Proliferation Resistance Modeling", ESARDA/INMM Workshop; Como, Italy, **39** (2004).

4. Bari, R., Nishimura, R., Petersons, P., Roglans, J., Bjonard, T., Bley, D., Cazalet, J., Cojazzi, C.J.M., Delaune, P., Golay, M., Renda, G., Rochau, G., Senzaki, M., Therios, I. and Zentner, M., Report on "Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems: An Overview", BNL-75546-2006-CP, May (2006).

5. Bari, R.A., Johnson, S.J., Hockert, J., Wigeland, R., Wonder, E.F., Zentner, M.D., Report on "Overview of the Facility Safeguardability Analysis (FSA) Process", PNNL-20829, October (2011).

6. Bathke, C.G., Ebbinghaus, B.B., Collins, B.A., Sleaford, B.W., Hase, K.R., Robel, M., Wallace, R.K., Bradley, K.S., Ireland, J.R., Jarvinen, G.D., Johnson, M.W., Prichard, A.W. and Smith, B.W., "The Attractiveness of Materials in Advanced Nuclear Fuel Cycles for Various Proliferation and Theft Scenarios", Nuclear Technology **179**, 5-30 (2012).

7. Bean, R., Bjornard, T., and Larson, T., Report on "Simulation Enabled Safeguards Assessment Methodology", INL-CON-07-12943, September (2007).

8. Beckers, J., Jaunet, P., Boella, M., Koehne, W., Burrows, B., Koutsoyannapoulos, C., Colin, A., Pattern, J., Crousilles, M., Pujol, E., Haas, E., Stanley, W.T., Ingels, R., and Xerri, C., "Control of Nuclear Material Hold-up; the Key Factors for Design and operation of MOX Fuel Fabrication Plants in Europe", IAEA-SM-367/8/04 (2004).

9. Bjornard, T., Bari, R., Hebditch, D., Peterson, P. and Schanfein, M., Report on "Improving the Safeguardability of Nuclear Facilities", INL/CON-09-15834, (2009).

10. Bjonard, T., Bean, R., Durst, P.C., Hockart, J. And Morgan, J., Report on "Implementing Safeguards-by-Design" INL/EXT-09-17085 (2010).

11. Blair, D.S., Rexroth, P.E., Rochau, G.E., Sype, T.T. and Wyss, G.D., Report on "A Risk-Based Methodology for Nuclear Proliferation Decisions," SAND2002-1579C, Sandia National Laboratory (2002).

12. CEA 2014-15, Load Generation Balance Report, available at http://www.cea.nic.in/reports/yearly/lgbr_report.pdf accessed on 14th January (2015).

13. Chakraborty, S., Danny, K.M., Saraswat, A., Gangotra, S. and Kumar, A., "Simulation of Material Flow in the Process Layout for Production of Mixed Oxide Fuel for PFBR", Proc. of Conf. on Quality Control of Nuclear Fuels –CQCNF, (2009).

14. Charlton, W.S., LeBouf, R.F., Gariazzo, C., Ford, D.G., Beard, C., Landsberger, S. and Whitaker, M., "Proliferation Resistance Assessment Methodology for Nuclear Fuel Cycles", Nuclear Technology, **157**, 143-156 (2007).

15. Chirayath, S., Metcalf, R., Ragusa, J., and Nelson, P., "Assessment of Proliferation Resistance Requirements for Fast Reactor Fuel Cycle Facilities", 8th International

Conference on Facility Operations – Safeguards Interface, Portland, Washington, March 30 – April 4, (2008).

16. Chirayath, S., Charlton, W., Stafford, A., Meyers, C., Goddard, B., Alfred, J., Carroll, M., Sternat, M., and Rauch, E., Report on "Risk-Informed Safeguards Approaches for Fast Reactor Fuel Cycles Utilizing MAUA based Proliferation Resistance Assessments," NSSPI-10-002, Nuclear Security Science and Policy Institute report, Texas A&M University (2010).

17. Chirayath, S., "Multi-Attribute Utility Analysis for Proliferation Resistance Assessment", Presentation at the INMM Workshop on Proliferation Assessment, Texas A&M University, (2010).

18. Chirayath, S., Elmore, Royal, Hollenbeck, Gordon, Chandregowda, Nandan G., Charlton, William S., Metcalf, Richard, and Ragusa, Jean C., "Proliferation Resistance Analysis and Evaluation Tool for Observed Risk (PRAETOR)—Methodology Development", Journal of Nuclear Materials Management, Vol.XLIII,No.2, 22-37 (2015).

19. Cleary, V.D., Rexroth, P.E., Rochau, G.E., Saltiel, D.H., Charlton, W.S,, Ford, D.G. and Giannangeli, D., Report on "Strengthening the Foundations of Proliferation Assessment Tools", SAND2007-6158, Sandia National Laboratories (2007).

20. CLND-2010, available at http://dae.nic.in/sites/default/files/civilnucliab.pdf, accessed on 14th May (2011).

21. CLND Rules-2011, available at http://dae.nic.in/writereaddata/liab_rules.pdf, accessed on 19th June (2014).

22. Cojazzi, G.G.M., Renda, G. and Sevini, F., "Proliferation Resistance Characteristics of Advanced Nuclear Energy Systems: a Safeguardability Point of View", ESARDA Bulletin No. **39**, 31-40 (2008).

23. Coles, G. A., Brothers, A. J., Gastelum, Z .N., and Thompson, S. E., Report on "Utility of Social Modeling for Proliferation Assessment", PNNL- 18438, (2009).

24. Danny, K.M., Chakraborty, S., Saraswat, A., Gangotra, S. and Kumar, A., "Optimisation of Material Flow in Fuel Fabrication Process Layout for PFBR.", International Conference on "Energy related Materials: Processing, Performance and Phenomena", Proc. of National Metallurgists Day- Annual Technical Meeting, Indian Institute of Metals, NMD-ATM (2007).

25. DeMuth, S., Budlong-Sylvester, K. and Lockwood, D., 2010, Report on "Next Generation Safeguards Initiative (NGSI) Program Plan for Safeguards by Design", LA-UR-10-01933 (2010).

26. Findlay, T., "Nuclear Energy and Global Governance – Ensuring Safety, Security and Non-Proliferation", Routledge Global Security Studies, Pub: Routledge, ISBN: 9780415532488 (2012).

27. Fischer, D., "History of the International Atomic Energy Agency- the First Forty Years", IAEA, Pub: International Atomic Energy Agency, ISBN: 9201023979 (1997).

28. Gangotra, S., Grover, R. B., Ramakumar, K. L., Kamath, H. S., and Panakkal, J. P., "Safeguards-by-Design (SBD) Concepts for Thorium-Based Fuel Fabrication Facilities" , Journal of Nucl. Mat. Management, **41 (1)**, 43-51 (2012).

29. Gangotra, S., Grover, R. B. and Ramakumar, K. L., "Comparison for thorium fuel cycle facilities of two different capacities for implementation of safeguards", Nuclear Engineering and Design, **262**, 535-543 (2013).

30. Gangotra, S., Grover, R. B., and Ramakumar, K. L., "Analysis of Measures to Enhance Safeguards, and Proliferation Resistance in Thorium Based Fuel Fabrication Plants", Progress in Nuclear Energy, **77**, 20-31 (2014).

31. Giannangeli, Donald D.J., "Development of the Fundamental Attributes and Inputs for Proliferation Resistance Assessments of Nuclear Fuel", M.Sc. Thesis, Texas A&M University, (2007).

32. GIF/PRPPWG/2011/003, "Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems", Revision 6, prepared by The Proliferation Resistance and Physical Protection Evaluation Methodology Working Group of the Generation IV International Forum (2011).

33. Grenèche, D., Rouyer, J. L. and Yazidjian, J. C., Report on "SAPRA: A Simplified Approach for the Proliferation Reistance Assessment of Nuclear Systems", AREVA, Inc. (2006).

34. Greneche, D., J. Cazalet, J., and Delaune, P., Report on "Proliferation Resistance Assessment: An Illustration Through the French Fuel Cycle", IAEA-CN-114/D-4 (2004).

35. Grover, R. B., and Chandra, S., "Scenario for Growth of Electrical Energy In India", Energy Policy, **34**, 2834-2847 (2006).

36. Grover[1], R.B., "National Framework for Governance of Nuclear Power", India's Nuclear Energy Programme, Future Plans, Prospects and Concerns, Ed. R. Rajaraman, ISBN: 9789332700307, Pub: Academic Foundation, (2013).

37. Grover[2], R.B., “Green Growth and Role of Nuclear Power: A Perspective from India”, Energy Strategy Review, (2013).

38. Grover, R.B., “Nuclear Power for India’s Energy Security: External and Internal Challenges”, India’s National Security Analysis Review 2013, Ed. Satish Kumar, ISBN:9781138796386, Pub: Routledge (2014).

39. Guidance Document, “Guidance on International Safeguards and Nuclear Material Accountancy at Nuclear sites in the UK, 2010 Edition, Rev.1”, UK Safeguards Office, (2010).

40. IAEA Safeguards Glossary, 2001 Edition.

41. IAEA-STR-332, Report on “Proliferation Resistance Fundamentals for Future Nuclear Energy Systems” (2002).

42. IAEA-415, Technical Report on “Status and Advances in MOX Fuel Technology”, Series No. **415,** Pub: International Atomic Energy Agency, ISBN:9201031033 (2003).

43. IAEA-TECDOC-1434, “Methodology for the Assessment of Innovative Nuclear Reactors and Fuel Cycles,” Report of Phase 1B (first part) of the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO), International Atomic Energy Agency, ISBN: 9201163045 (2004).

44. IAEA-TECDOC-1450, Thorium Fuel Cycle – Potential Benefits and Challenges, ISBN: 9201034059 (2005).

45. Nuclear Security Recommendations on Physical Protection of Nuclear Material and Nuclear Facilities (INFCIRC/225/Revision 5), IAEA Nuclear Security Series No. 13, 2011.

46. IAEA-1279, available at http://www-pub.iaea.org/MTCD/publications/PDF/Pub1279_web.pdf, accessed on 14th October (2013).

47. IEA-2014, available at http://www.iea.org/publications/freepublications/publication/keyworld2014.pdf accessed on 16th February (2015).

48. Inoue, N., Hori, M., Hori, K. and Takeda, H., “Methodologies of Nuclear Proliferation Resistance Assessment for Nuclear Fuel Cycle Options,” Proc. 44th Annual Meeting of the Institute of Nuclear Materials Management, Phoenix, Arizona (2003).

49. Inoue, N., Kurakami, J. and Takeda, H. “Review of JNC’s Study on Assessment Methodology of Nuclear Proliferation Resistance,” Proc. 45th Annual Meeting of the Institute of Nuclear Materials Management, Orlando, FL (2004).

50. JNMM, Special Issue: The Next Steps in International Safeguards, Journal of Nuclear Materials Management, **37(4)**, Ed. Mangan, D., Pub: Institute of Nuclear Materials Management Inc., ISBN: 9780750686730 (2009).

51. Kakodkar, A., International Atomic Energy Agency, 48th General Conference, Vienna 22nd Sep 2004, Statement by India, available at http://www.iaea.orjg/About/Policy/GC/GC48/Statements/india.pdf accessed on 11th March 2013.

52. Kaplan, S. & Garrick, J. B., “On the Quantitative Definition of Risk”, Risk Analysis, **1(1),** 11-17 (1981).

53. Kang, J., “Analysis of Nuclear Proliferation Resistance”, Progress in Nuclear Energy, **47(1-4)**, 672-684 (2005).

54. Kimura, Y., Saito, M. and Sagara, H., "Evaluation of Proliferation Resistance of Plutonium Based on Decay Heat", Journal of Nuclear Science and Technology, **48(5)** 715–723 (2011).

55. Kiriyama, E. and Pickett, S., "Non-Proliferation Criteria for Nuclear Fuel Cycle Options", Progress in Nuclear Energy, **37(1-4)**, 71-76 (2000).

56. Kuno, Y., Inoue, N., and Senzaki, M., "Nuclear Proliferation-Resistance and Safeguards for Future Nuclear Fuel Cycle", Journal of Nuclear Materials, **385,** 153–156 (2009).

57. Larry, A. L., Eller, P. G., and Stanbro, W. D., Report on "Complexities in Gauging Time-dependency of Proliferation Resistance", LA-UR-04-4300, (2004).

58. Liu, Y., Y., Shuler, J., Report on "Radio Frequency Identification (RFID) Tracking and Monitoring System", APT No. 64438, CTMA Workshop, Melbourne, Florida (2009).

59. Manohar, S., Patel, V.P, Dani, U., Venugopal, M.R., and Wattal, P.K., "Engineering Scale Demonstration Facility for Actinide Partitioning of High Level Waste", BARC Newsletter, **332** (2013).

60. Metcalf, R., Report on "Determination of Relative Importance of Non-proliferation Factors", Institute of Nuclear Materials Management, 50th Annual Meeting, INC/CON-09-15994, (2009).

61. Metcalf, R, "New Tool for Proliferation Resistance Evaluation Applied to Uranium and Thorium Fueled Fast Reactor Fuel Cycles" M.Sc. Thesis, Texas A&M University, (2009).

62. Mourogov, V., Fukuda, K., and Kagaramanian, V., 'The Need for Innovative Nuclear Reactor and Fuel Cycle Systems; Strategy for Development and Future Prospects", Progress in Nuclear Energy, **40(3-4)**, 285-299 (2002).

63. Niemeyer, I., "Safeguards Information from Satellite Imagery", Journal of Nuclear Materials Management, **37(4)**, 41-48 (2009).

64. Ninagawa, J., Nagatani, T., Asano, T. and Fujiwara, S., Report on "Experiences and Achievement on Safeguards by Design for the Plutonium Fuel Production Facility (PFPF)", IAEA-CN-184/66 (2010).

65. Planning Commission, Integrated Energy Policy: Report of the Expert Committee, New Delhi, Planning Commission, Government of India, (2006).

66. Sinha, R. K. and Kakodkar, A., "Design and Development of AHWR- the Indian Thorium Fueled Innovative Nuclear Reactor", Nuclear Engineering and Design, **236(7-8)**, 683-700 (2006).

67. Sinha R K. Sinha, R.K., Kushwaha, H.S., Agarwal, R.G., Saha, D., Dhawan, M.L., Vyas, H.P., and Rupani, B.B.,, "Design and Development of AHWR – the India Thorium Fuelled Innovative Nuclear Reactor". Power from Thorium – Status. Strategies and Directions INSAC, Proc. on Ann. Conf. Mumbai, 2000, Indian Nuclear Society, **1**, 81-106 (2000).

68. Skutnik, S. E., "A Methodology for Enhancing Nuclear Fuel Cycle Proliferation Resistance Assessment" Doctoral Thesis, North Carolina State University, (2011).

69. Sleaford, B. W., Collins, B. A., Ebbinghaus, B. B., Bathke, C. G., Prichard, A. W., Wallace, R. K., Smith, B. W., Hase, K. R., Bradley, K. S., Robel, M., Jarvinen, G.D., Ireland, J. R., and Johnson, M. W., "Nuclear Material Attractiveness: An Assessment of Material from PHWR's in a Closed Thorium Fuel Cycle", Proc. of European Nuclear Conference, Barcelona, LLNL-CONF-429545, April (2010).

70. TOPS, Report on "Technological Opportunities to Increase the Proliferation Resistance of Global Civilian Nuclear Power Systems (TOPS)," TOPS Task Force on the Nuclear Energy Research Advisory Committee, US Department of Energy; available at http://www.nuclear.gov/nerac/FinalTOPSRpt.pdf (2001).

71. Wallace, R., "Safeguards Information from Open Sources", Journal of Nuclear Materials Management, **37(4)**, 30-40, (2009).

72. WNA, World Nuclear Power Reactors and Uranium Requirements, available at http://www.world-nuclear.org/info/Facts-and-Figures/World-Nuclear-Power-Reactors-and-Uranium-Requirements/, accessed on 14th May (2015).

73. Yue, M., Cheng, L. and Bari, R., Report on "Applications of Probabilistic Methods of Proliferation Resistance: Misuse, Diversion and Abrogation Scenarios," Brookhaven National Laboratory (2005).

74. Yue, M., Cheng, L.Y., and Bari, R., Report on "Methodology for Proliferation Resistance for Advanced Nuclear Energy Systems", BNL-75428-2006-CP (2006).

75. Zentner, M., "Nuclear Proliferation, Nuclear Power Deployment, Operation and Sustainability, Ed: Dr. Pavel Tsvetkov, ISBN: 9789533074740", Pub: InTech Publication, http://www.intechopen.com/books/nuclear-power-deployment-operation-and sustainability/nuclear-proliferation (2011).