DEVELOPMENT AND OPTIMIZATION OF NEW GENERATION START-UP INSTRUMENTATION SYSTEMS (SUI) FOR DOMESTIC CANDU REACTORS

by

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A Thesis Submitted in Partial Fulfillment

of the Requirements for the Degree of

Master of Applied Science

In

Nuclear Engineering

The Faculty of Energy Systems and Nuclear Science

University of Ontario Institute of Technology

December 2010

ABSTRACT

Due to the age and operating experience of Bruce Power units, equipment ageing and obsolescence has become one of the main challenges that need to be resolved for all systems, structures and components in order to ensure a safe and reliable production of energy.

The research objectives of this thesis will focus on methodology for modernization of Start-Up Instrumentation (SUI), both in-core and Control Room equipment, using a new generation of detectors and cables in order to manage obsolescence. The main objective of this thesis is to develop a new systematic approach to SUI installation/replacement procedure development and optimization. Although some additional features, such as real-time data monitoring and storage/archiving solutions for SUI systems are also examined to take full advantage of today's digital technology, the objective of this thesis does not include detailed parametrical studies of detector or system performance. Instead, a number of technological, operational and maintenance issues associated with Start-Up Instrumentation systems at Bruce Power will be identified in this project and a structured approach to developing a replacement/installation procedure that can be standardized and used across all of the domestic CANDU stations is proposed. Finally, benefits of Hierarchical Control Chart (HCC) methodology for all stages of plant life management, such as system design, development, operation and maintenance are demonstrated.

Keywords: Task Breakdown and Analysis methodology, installation/removal procedure development and optimization, risk-based analysis and optimization, Hierarchical Control Chart (HCC) methodology for system maintenance and troubleshooting, Start-Up Instrumentation (SUI), Ion Chambers, Fission Chambers, proportional counters, Shutdown System 1 (SDS1), Shutdown System 2 (SDS2).

ACKNOWLEDGEMENTS

First and foremost, I would like to thank my family for their love, support and encouragement. Without their help, this work would not be possible.

I would like to thank my advisor Professor Hossam A. Gabbar for encouraging me to work to the best of my abilities and for providing the environment that made this work possible.

I would also like to thank my Section Manager, Corinne Draesner (Bruce Power) for her support and encouragement in pursuing my academic goals.

My warm gratitude to my colleagues Moh Siddiqui, Navindra Persauld, and Colin Fisher for their help in gathering data and sharing their knowledge and experience.

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NOMENCLATURE

(F.P.)	Full Power, in [%] or [decade]
Nv	neutron flux, in [n/cm^2/s]
r_c	neutron count rate, in [count/sec]
l_{mod}	moderator level, in [cm]
V	signal amplitude in Volts [V] or milli-Volts [mV]
eV	electron-Volt
MeV	Mega electron-Volt
n	neutron
α	alpha particle
γ	gamma-ray
cm/Hg	centimetre of mercury
hr	hour
T	tritium, H-3
p	proton
E_p	energy of proton, equals to =0.573 MeV in the reaction $^3He + n \rightarrow T + p$
E_T	energy of Tritium, in =0.191 MeV in the reaction $^3He + n \rightarrow T + p$
°C	degrees Centigrade
°F	degrees Fahrenheit
cps/nv	neutron sensitivity in [counts per second / neutron flux]
DC	Direct Current, in [Amp]
AC	Alternating Current, in [Amp]
AMP	signal amplitude in [Amp]
^{10}B	Boron-10
^7Li	Lithium-7
$^7Li^*$	Lithium-7 in exited state
μm	[micro-meter]
mm	[milli-meter]
mg/cm^2	[milli-gramm per centimetre square]
S	detector sensitivity threshold Voltage in [c·s^{-1}/nv]
N	logarithmic scale detector signal, in [c·s^{-1}]
Φ	neutron flux, in [n·cm^{-2}·s^{-1}]
P	detector sensitivity curve slop, in [%]
Gy/h	gamma ray intensity in the air in [grays per hour]
a/nv	detector sensitivity to neutrons, in [amperes/neutron flux]
a/r/h	detector sensitivity to gamma, in amperes/roentgen/hour
R/hr	[Rem per hour]
mRem/hr	[milli-Rem/hour]
pF	capacitance, in [pico-Farad]
i and *j*	subscripts
R_i	risk
Δ	deviation
x_0, x_1, x_2	Pareto set solutions

F	Pareto feasibility region
$f_i(x_i)$	Pareto function
w_i	weighted ranking coefficient
cost	cost function $f_1(x)$
time	time function $f_2(x)$
$p(t)$	Gaussian function
σ	standard deviation
t	time
T_{50}	time interval point when 50% of the work is done
T_{90}	time interval point when 90% of the work is done
$\bar{t}$	mean
n	integer number
λ	rate, in [transactions/unit time]
r	job duration, in [min]
μ	service time, job processing time, in [min]
π	mathematical constant
c	contribution
a	amount of investment resource, in [$]
b	availability of investment resource
$\sum_{j=1}^{n} c_j x_j$	total investment, in [$]

LIST OF ACRONYMS AND ABBREVIATIONS

$M	Million $
.pdf	Portable Document Format
ACR	Advanced CANDU Reactor
AECL	Atomic Energy of Canada Limited
Al	Aluminum
ANSI	American National Standards Institute
Ar	Argon gas
AC	Alternating Current
AMP	Signal amplitude in Amp
B-10	Boron 10
BF_3	Boron Trifluoride
Bruce A	Bruce Power “A” Station
Bruce B	Bruce Power “B” Station
BWR	Boiling Water Reactor
CAD	Computer Aided Design
CANDU®	Canadian Deuterium Uranium
CatID	Catalogue ID
CEA	Commissariat à l'Énergie Atomique - Atomic Energy Commission, France
Ch	Channel
CNSC	Canadian Nuclear Safety Commission
CO_2	Carbon Dioxide gas
CoDeSys	Controller Development System
COG	CANDU Owners Group
CRL	Chalk River Laboratories
CV	Controlled Variable
DB	Database
DBA	Design Basis Accident
DBE	Design Basis Event
DC	Direct Current
DCC	Digital Control Computer
DIV	Division
EMI	Electro-Magnetic Interference
EP	Electrical Penetration (containment penetration)
EQ	Environmental Qualification
ERP	Equipment Reliability Program
eV	electron Volt
EV	Environmental Variable
F&P	Fuel and Physics (Department)
F.P.	Full Power
FBD	Function Block Diagram
FMEA	Failure Mode and Effect Analysis
G-2	Gentilly Unit 2 NGS
GUI	Graphical User Interface

HAZOP	Hazard and Operability Analysis
HCC	Hierarchical Control Chart
^{3}He	Helium-3
HV	Human Variable
I&C	Instrumentation and Components
I/C, I&C	Instrumentation/Control
IAEA	International Atomic Energy Agency
IC	Ion Chamber
ID	Identification
IDEF0	Integration Definition Language 0
IEC	International Electro-technical Commission
IEEE	The Institute of Electrical and Electronics Engineers
ISA	Instrumentation, Systems and Automation Society
IST	Imaging and Sensing Technologies
IT	Information Technology
IX	Ion Exchange
JB	Junction Box
LCD	Liquid Crystal Display
Li-7, Li	Lithium
LOCA	Loss of Coolant Accident
MasterCRIB	Electronic tool repository and database
MatLab	Matrix Laboratory (Mathworks, Inc.)
MCA	Multi-channel Analyzer
MCR	Main Control Room
MeV	Mega electron Volt
MOGA	Multi-objective Genetic Algorithm
Mtce	Maintenance
MV	Manipulated Variable
MW (e)	Megawatt (electrical)
NDA	Non Destructive Analysis
NFM	Neutron Flux Monitor
NFMS	Neutron Flux Monitoring System
NGS	Nuclear Generating Station
NimBin	Nuclear Instrumentation Bin
NMS	Nuclear Maintenance Services (Department)
NMS	Neutron Monitoring System
NOP	Neutron Overpower
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
O&M	Operation and Maintenance
OCR	Optical Character Recognition
OD	Outside Diameter
OP&P	Operating Policies and Procedures
OPEX	Operating Experience
OPGSS	Overpoisoned Guaranteed Shutdown State
OSHA	Occupational Safety & Health Administration

Outage	Planned or Forced Unit Shutdown
OV	Organizational Variable
Pu, Pu-239	Plutonium-239
P&ID	Piping and Instrumentation Diagram
PassPort	Ventyx PassPort Asset Suite
PBMR	Pebble Modular Reactor
PC	Personal Computer
PE	Procurement Engineering
PFD	Process Flow Diagram
PHWR	Pressurised Heavy Water Reactor
PI®	Plant Information System
Pickering A	Pickering NGS Station "A"
Pickering B	Pickering NGS Station "B"
PRA	Probabilistic Risk Assessment
Primavera	Oracle's Primavera Enterprise Project Portfolio Management Solution
PROMELA	Process Meta Language
PV	Process Variable
PWR	Pressurised Water Reactor
QA	Quality Acceptance
RDFL	Recipe Formal Definition Language
REM	Roentgen Equivalent Man
Restart	Bruce Power Unit 1 and Unit 2 Refurbishment Project
RF	Radio Frequency
RFI	Radio Frequency Interference
RHIS	Radiation Heath Information System
RM	Reactor Maintenance
RRS	Reactor Regulating System
SA	Shutoff Rod Assembly
SCA	Single Channel Analyzer
SCR	Station Condition Record
SDS	Shutdown System
SOE	Safe Operating Envelope
SPV	Single Point of Vulnerability
SSC	System, Structure, Component
SUI	Start-up Instrumentation
SV	State Variable
T	Tritium
TaskID	Task Identification (number)
Th	Thorium
TRTR	Test Research and Training Reactors
U1	Bruce Power Unit 1
U2	Bruce Power Unit 2
U-234	Uranium-234
U-235	Uranium-235
U-238	Uranium-238
U3	Bruce Power Unit 3

U4	Bruce Power Unit 4
U5	Bruce Power Unit 5
U6	Bruce Power Unit 6
U7	Bruce Power Unit 7
U8	Bruce Power Unit 8
UI	User Interface
UO_2	Uranium Dioxide
UP	Unit Procedure
VP	Viewing Port
WANO	World Association of Nuclear Operators
WWER	Water-Water Energetic Reactor

CHAPTER 1: INTRODUCTION

1.1 Introduction

In thermal nuclear reactors such as CANDU®, the process of splitting nuclear fuel and producing fission heat is induced by thermal neutrons, i.e. the neutrons that at room temperature have an average energy of about 0.025 eV [1]. Typically, neutron activity and neutron flux detectors designed for CANDU reactors are concerned with neutron detection and are not designed to measure neutron energies or distinguish between different neutron categories, such as fast or slow neutrons. Reactor regulation and power measurements rely on neutronic detectors for power adjustments, neutron overpower protection and safety system initiation. Detection and measurement of neutronic activities during all stages of reactor start-up, operation and shutdown is, therefore, of utmost importance.

Start-Up Instrumentation (SUI) is used at existing CANDU power plants following a prolonged shutdown and during a re-start phase at low and very low power levels. The main goal is to provide neutron overpower (NOP) protection via Shutdown System 1 (SDS1) trip circuits on high neutron count rate and to allow continuous monitoring of neutron flux level and rate. In general, SUI instrumentation is required to provide a reliable indication of the reactor power for a period of up to several weeks with the required range of sensitivity of the Start-Up Instrumentation is from 10^{-14} to 10^{-6} of reactor Full Power (F.P.) [2].

At Bruce A Nuclear Generating Station (NGS), which will be used as a case study for this paper, the SUI system is normally engaged once the reactor power level falls below -5 decades (10^{-5} F.P.) and provides a reliable bump-less transfer of core monitoring function from RRS Ion Chambers to SUI system at -6 decades. Once the power levels decay below 10^{-6} F.P., SUI detectors become the primary means of flux monitoring, with typical power levels between -7 and -8 decades during unit outages and shutdowns.

In this paper, challenges and limitations of the current SUI systems at Bruce Power units will be analyzed both in terms of the existing detector technologies and the overall current approach to system design and operation.

The two most common types of neutron detectors currently used in CANDU Start-Up Instrumentation (SUI) systems, Boron Trifluoride (BF_3) and Helium (^3He), will be examined along with the main disadvantages and system performance limitations. A new type of detector based on fission chamber technology will be proposed in order to overcome the current ageing and obsolescence issues affecting existing SUI detectors at Bruce Power (Bruce A) units 3 and 4.

Next, a new design and installation scheme will be proposed for existing CANDU reactors 3 and 4 at Bruce Power Nuclear Generating Station (NGS) in order to improve the overall SUI system functionality, and reduce dose rates and production losses that commonly result from the maintenance and operational challenges due to current system design limitations.

Next, a risk-based replacement and installation algorithm will be developed in order to formalize project phases and steps with the main objective of verification of the proposed work activity model. The case study of SUI instrumentation at Bruce Power Unit 3 and Unit 4 will be used in order to define and standardize the essential activities that can be used for all future SUI replacement/installation projects at domestic CANDU stations.

Next, risk analysis will be performed in order to identify potential hazards and risks associated with this project and the project activity model will be optimized in order to achieve maximum cost, time and radiation dose reduction benefits.

1.2 Motivation of Thesis

The main motivation for this thesis stemmed from the large number of technological, operational and maintenance issues associated with Start-Up Instrumentation (SUI) systems at Bruce Power and the need for a structured approach to developing a replacement/installation procedure that can be standardized and used across all domestic CANDU stations.

Given the age of the Bruce Power plant equipment, particularly in Units 3 and 4, detector ageing and obsolescence are perhaps the main issue affecting plant instrumentation and components. Currently SUI instrumentation at Bruce Power consists of older Centronic detectors combined with obsolete Tennelec and Canberra signal processing electronics. There are currently no spares or replacement parts available for amplifiers, portable panels and Multi-Channel Analyzer (MCA) computers. This situation is not unique to Bruce Power reactors. IAEA-TECDOC-1402 released in 2004 [3] highlighted the trend formed over the last two decades where many major manufacturers of components for Nuclear Power Plants (NPPs) have ceased to supply or support products with which the plants were built [3].

Another main motivation for this thesis is the fact that the existing Bruce Power SUI system design has a reputation for being both time and labour intensive to implement, and has been a causal factor to human errors, equipment disturbances and failures resulting in poor equipment performance.

Finally, radiological dose increases and outage extensions are among other undesirable outcomes associated with the existing SUI systems at Bruce Power units.

Presently, work packages for refurbishment projects at Bruce Power are usually developed based on previous experience or references to similar undertakings in other industries. No formal methodology is being used to identify potential hazards and risks and to optimize work packages in terms of time, cost and dose reduction. This, in its turn, results in many deficiencies and delays, where work packages have to be returned for re-assessment and re-work. With the current Bruce Unit 1 and 2 Restart project running significantly over time and over budget, process improvement and “Lean” approach have become some of the most important challenges that need to be overcome in order for time/cost savings and dose reduction to occur. An obvious demand for standard tools and practices has been identified, so that work packages and activity planning can be systematically examined from start to finish and optimized in order to eliminate their “wastes”, or steps that add no value, and enable a selection of the most optimal risk-based solution.

1.3 Objectives of Thesis

The main objective of this thesis is to analyze the main challenges with existing SUI technology at Bruce Power stations and to develop a risk-based optimization framework for replacement/installation of the next generation SUI systems. This will be achieved in the following steps:

1. Current operating challenges of SUI systems at Bruce Power units will be identified and analyzed.
2. Possible alternative solutions to conventional gas-filled detector technology will be analyzed and a new type and model of SUI detectors will be selected in order to address the technological limitations of the existing aged detectors. The new detector type and model will be selected based on the existing design specifications in order to demonstrate the proposed methodology for the SUI installation procedure development and optimization. This thesis will not include detailed parametrical studies or detector design validation analysis.
3. A new SUI system architecture and layout will be designed in order to address the existing operational and maintenance challenges at Bruce Power units. This enhanced SUI system architecture is shown in order to demonstrate the proposed framework for the SUI installation procedure development and optimization. This thesis will not include detailed physical system design, validation or performance optimization.
4. The installation procedure for SUI replacement will be developed and verified. Task analysis, hazard identification and barrier analysis as well as risk analysis will be performed and a method for procedure verification and optimization will be shown.

5. Hierarchical Control Chart (HCC) methodology will be proposed as a means for future use by system engineers, designers and maintainers to aid in system maintenance and fault troubleshooting.

This is shown in Figure 1.1 below in the Integration Definition Language 0 (IDEF0) [4] activity model format. Each process is shown as a box/arrow relationship. Inputs enter from the left side of the function box and are transformed by the function to produce outputs. Arrows leaving the right side are outputs. Controls entering on top specify the conditions required for the function to produce correct outputs. On the bottom of the function box are the mechanism and the call. The mechanisms are the tools used to generate outputs and the call enables the sharing of detail between models.

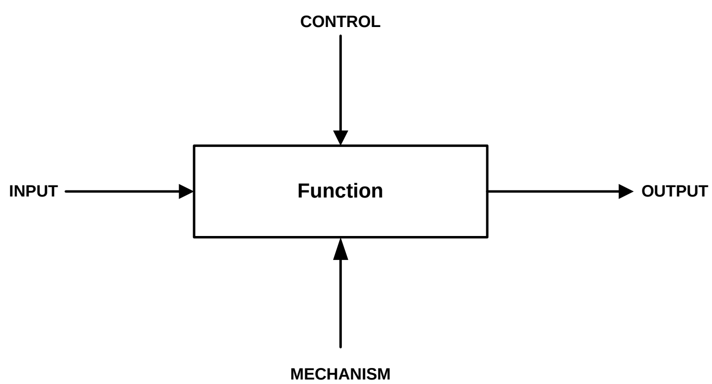

Figure 1. 1: IDEF0 activity model that will be used in this thesis. A sample process is shown with its associated input, output, controls and mechanisms.

Following this methodology, the framework for the proposed SUI design can presented in a series of steps, or processes, with feedback ties and decision points as shown in Figure 2.

The first process or activity can be described as “Analysis of Current SUI Challenges”. This is the initial step in this research and is being performed to identify and analyze the main challenges and limitations that the existing SUI systems present for the Bruce Power units. This analysis will involve both technological aspects of the aged Bruce Power SUI detectors, e.g. reduced sensitivity, as well as operational challenges such as outage extensions and high dose rates incurred by Maintenance personnel due to the current design and operating policies. This process is described in Chapter 2.

The second process corresponding to Chapter 3 can be described as "Analysis of alternative detector technologies" and will include the investigation of other detector types that are available on the market besides the conventional gas-filled proportional counters. The output of this process will conclude to selection of a new detector prototype that will be used for the proposed new SUI system design.

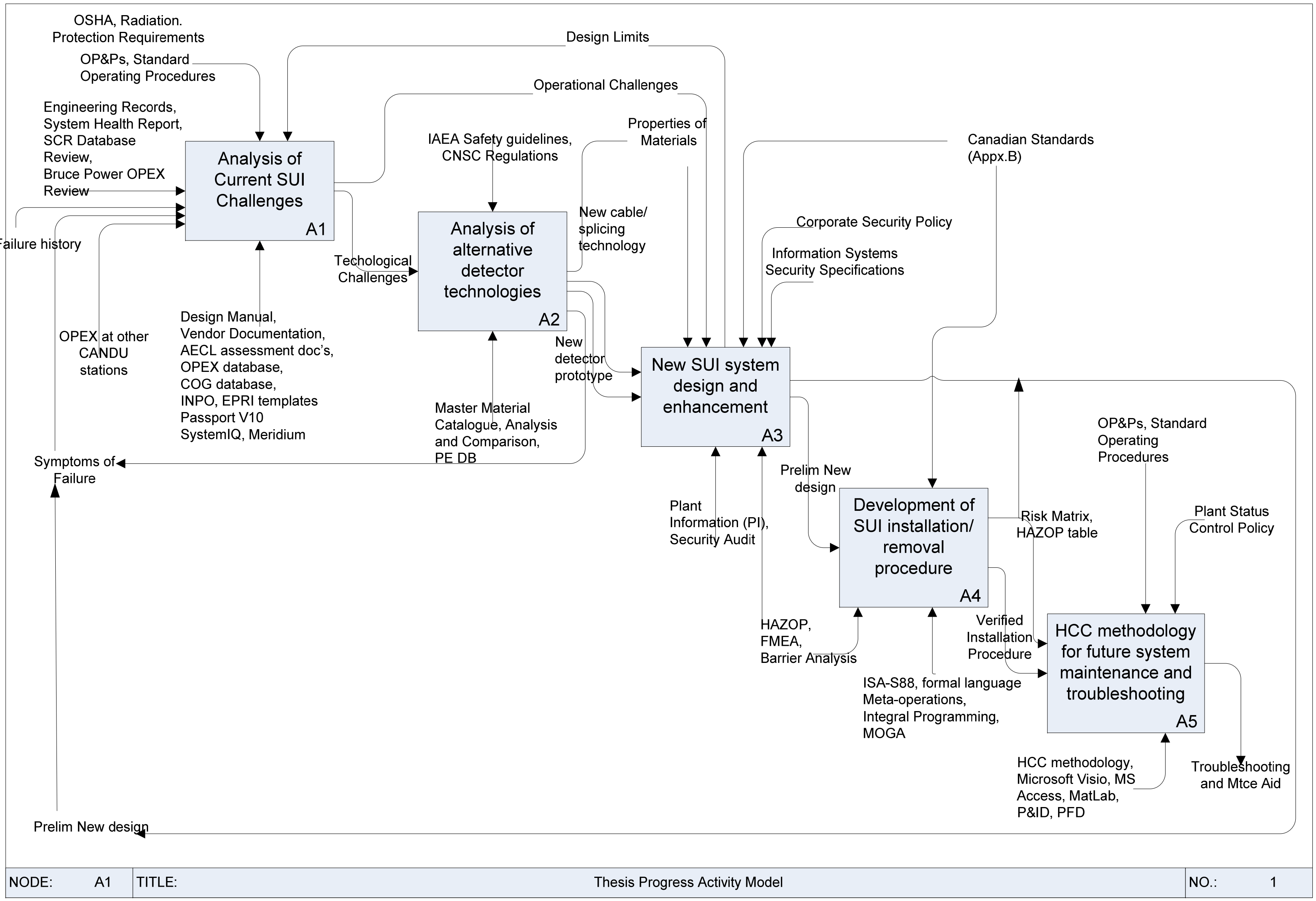

Figure 1. 2: Thesis Organization and Activity Model, where five main phases (or activities) are shown using IDEF0 activity model.

The third process shown in Figure 1.2 describes the development of a new SUI system architecture. It includes a conceptual design of the entire SUI system instrumentation loop, from the detectors in-core to the portable SUI equipment in the Main Control Room (MCR) and is described in Chapter 4. This chapter also includes a proposed new enhancement for the SUI systems, where networking and data storage capabilities are shown for the proposed design.

The fourth process shown in Figure 1.2 as “Development of SUI installation/removal procedure” is described in Chapter 5. In this Chapter, the need for a standardized procedure development approach will be examined, followed by detailed task analysis and procedure verification. A sample task in the proposed procedure will be used as an example to demonstrate a formal approach to procedure development and verification.

The last process shown in Figure 1.2 as “HCC methodology for future system maintenance and troubleshooting” will be described in Chapter 6. In this chapter, a “Hierarchical Control Chart” methodology will be proposed as a means to aid system engineers and operators, as well as helping designers with troubleshooting and routine verification tasks.

1.4 Organization of Thesis

In Chapter 1, an introduction to the research and background is given with the motivation and objectives of the thesis.

In Chapter 2, methodologies and techniques used in this thesis are given.

In Chapter 3, a detailed analysis of the existing technological and operational challenges associated with the existing SUI detectors and system architecture at Bruce Power units is given.

In Chapter 4, fission chamber based detectors are proposed as an alternative to the aged Bruce Power SUI detectors. A comparison between detector design, sensitivity ranges, signal discrimination capabilities, and other parameters is given for conventional gas-filled proportional counters and the new proposed detector technology. Three potential candidates for Bruce Power detector replacement are selected from the pool of immediately available detectors on the market based on the detector selection criteria. Rational for the selection of the Photonis CFUF-43 detector is given also. Next, a new SUI system architecture and layout addressing the current SUI challenges at Bruce Power units is proposed. Additional features, such as information storage and retrieval capabilities and real-time remote system monitoring features are given.

In Chapter 5, an installation/removal procedure for the proposed new SUI detectors and system architecture is developed and a methodology based on formal language-meta-operations is used to verify the procedure consistency and completeness. Risk-based optimization methodology for replacement/installation project is used to achieve the maximum cost/benefit ratio in terms of resource allocation, project timeline forecasting and budget estimation.

In Chapter 6, Hierarchical Control Chart (HCC) methodology is presented as a new tool for system designers, engineers and operators that can be used as an aid for system troubleshooting and verification tasks.

Finally, conclusions of the findings for this research and future work recommendations are given in Chapter 7.

CHAPTER 2: METHODOLOGY AND TECHNIQUES

The aim of this thesis is to develop a framework for systematic planning and optimizing installation/replacement procedures for Start-Up Instrumentation. By providing a formal structure and process for planning and executing the work, the proposed procedure for installation/replacement of SUI instrumentation can be modified to be used across various CANDU facilities as well as for further optimization and automation in the future.

The process that will be used in this thesis can be described in the following stages:

1. Analysis of Opportunity
2. Analysis of Current Challenges
3. Problem Definition
4. Proposal of a suitable solution
5. Solution Optimization
6. Performance Monitoring
7. Feedback and Modification for future implementation – Lessons Learned

This process is shown in Figure 2.1 below and further expanded in Figure 2.2

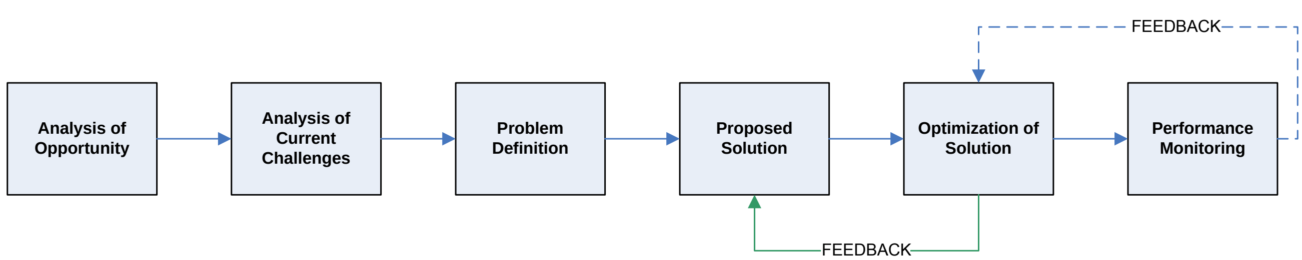

Figure 2. 1: Thesis development methodology.

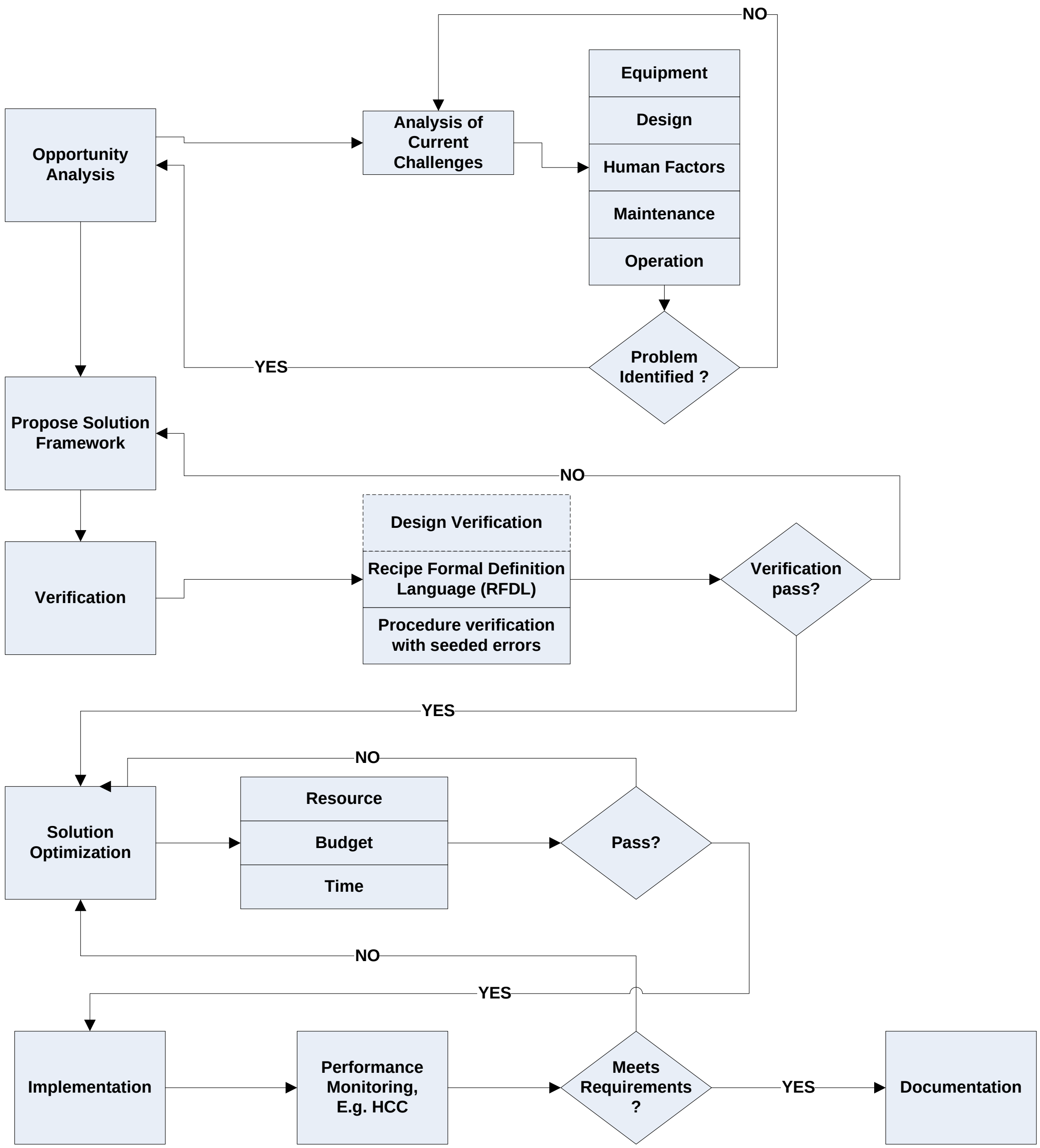

Figure 2. 2: Thesis framework development methodology where the stages of Opportunity Analysis, Analysis of Current Challenges, Proposed Solution, Solution Optimization and Implementation are shown.

2.1 Opportunity Analysis

In order to identify the existing gaps and limitations of current SUI systems at Bruce Power units, a thorough review of historical data, maintenance and Engineering records and Station Condition Records was conducted. Station Condition Record database (SCR) database was searched for the period between 2003 and present time. The following keywords were used: “SUI”, “Start-Up Instrumentation”, “Start-Up detectors”, “SUI detectors”, “Outage extensions”, “Human Performance issues”, “Equipment Reliability clock reset”, “ER clock reset” as well as by the unique system identification number used at Bruce Power. All entries that were found relevant were categorized by the nature/type of issue, e.g. equipment degradation, human error, outage extensions and delays, dose reduction areas for improvement, etc.

Next, Ventix Passport Suite used at Bruce Power was searched to investigate SUI system maintenance and equipment history. All work requests and work orders related to SUI equipment were reviewed and categorized in terms of detector failure, connector/cable failure, electronic modules/component failures and other types of faults that have occurred at Bruce Power stations between the year 2003 and present time.

Next, engineering records, such as System Health Reports and System Performance Monitoring Plans were reviewed to confirm the repetitive nature of failures and other issues with SUI equipment. It was determined that there is a large number of such issues that had been recorded and tracked in these documents.

Next, interviews with Engineers from Fuel & Physics department, Performance Engineering and Control Maintenance technicians were conducted to discuss the issues with the existing equipment.

Next, COG (CANDU Owners Group) records and IAEA technical documentation (TECDOC) were reviewed in order to determine the extent of condition throughout nuclear industry in Canada and worldwide. IAEA documentation is available in hard copy format through Bruce Power record management system as well as online through the Internet. COG records were reviewed online as well as during COG workshop meeting that took place in June 2010 where engineers representing various domestic and foreign CANDU power plants were interviewed to determine whether the SUI equipment concerns are common among other CANDU stations. The process of "Analysis of Opportunity" is shown in Figure 2.2 below.

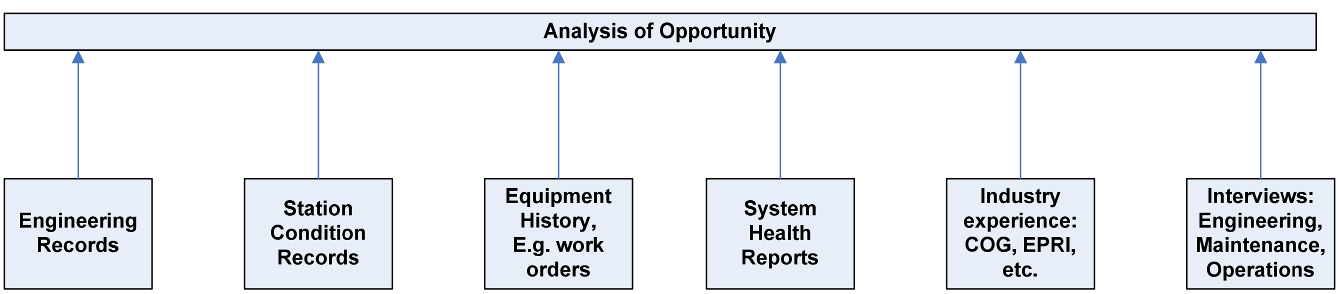

Figure 2. 3: Analysis of Opportunity model where various inputs, such as System Health Reports, interviews with Maintenance and Operations and others, are shown.

Finally, a research plan, shown in Chapter 1, was developed using Integration Definition Language 0 (IDEF0) methodology for the thesis activity model. IDEF0 standard was selected as the most suitable for developing a structured graphical representation for a complex process with multiple stages and interconnected inputs/outputs in order to provide a clear, consistent plan for thesis organization and objectives. Each chapter of this thesis is logically mapped to a process, e.g. “Analysis of Current SUI Challenges”, as shown in Figure 1.2 in Chapter 1. Following IDEF0 standard guidelines, each process is shown along with its tools, mechanisms, processes, operations information or objects, e.g. in Figure 1.2 the process of “Analysis of Current SUI Challenges” is shown in its inputs, such as Engineering Records, System Health Report, SCR Database Review, etc. that were used to determine and categorize the existing SUI system challenges as discussed earlier. Standard Operating Policies and Procedures, Radiation Protection Requirements and other standards and regulations are shown as controls that must be considered for the development of formulated technological challenges of this process, i.e. process output. Design Manuals, Vendor Documentation, COG database, SCR database, COG database and other tools used in this analysis are shown as “mechanisms” for this process.

Overall, the use of IDEF0 standard was found very successful for this task as it provided the required means for the analysis and development of a complex, multi-stage research framework as well as to incorporate both technical and business requirements in the same process analysis.

2.2 Analysis of Current Challenges and Problem Definition

Once the opportunity for improvement was identified, Analysis of Current Challenges was conducted and presented in Chapter 3 of this thesis. The existing deficiencies and limitations of Bruce Power SUI systems were analysed in greater depth. Station Condition Records were analysed in order to determine the main existing challenges due to obsolete SUI equipment as well as to categorize the root causes of the issue, such as technical specifications of the equipment, design, operation policies and procedures, human errors, etc. Detailed maintenance history, outage history, system notebooks and data ledgers were examined and data was compiled using MS Excel spreadsheets.

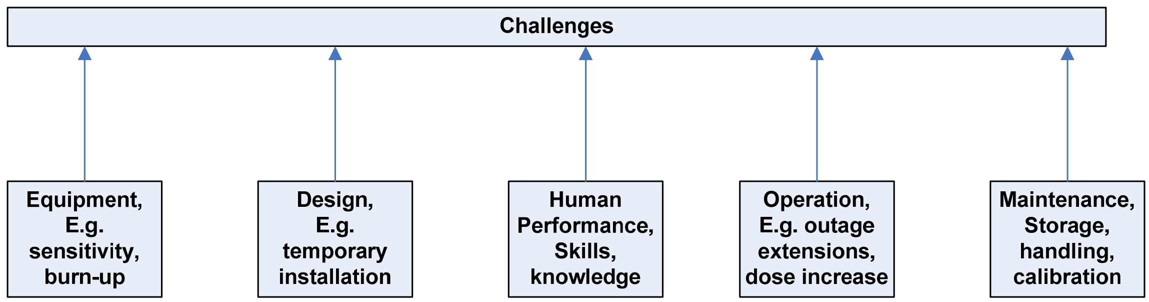

Figure 2. 4: Categorization of SUI related challenges at Bruce Power.

Next, COG meeting documentation was reviewed to find out whether other CANDU stations, such as G-2, Darlington, Pickering, etc. are affected by the same issues, e.g. Hydro-Quebec presentation by Denis Brissette at COG Workshop for "Start-Up Instrumentation & Physics Monitoring" that took place on 22-23 October 2003 where the current challenges of SUI instrumentation at Point Lepreau Nuclear Generating Stations was examined.

Next, existing literature, listed in the References section of this thesis, was reviewed to gather more background information regarding the technical specifications and design of conventional gas-filled detectors. Bruce Power SUI design manuals and specialized detector studies conducted by T. Qian, P. Tonner, N. Keller and other researchers at Chalk River Laboratories (available as restricted materials for Bruce Power internal use only) were reviewed as well.

Next, several physical plant walk-downs were conducted to assess condition of the equipment in the warehouse, Control Maintenance shop and the field locations, such as the Main Control Room portable SUI equipment rack and panel connections. Several photos that were taken during these walk-downs are used in this thesis for demonstration, e.g. Figure 3.11 where Bruce A Unit 2 SDS2 Horizontal Reactivity Management Deck is shown.

Next, vendor manuals for the existing Bruce Power SUI systems were reviewed to determine technical specifications for detectors, timers, counters, power supplies and other components of the system as well as to determine the applicable standards and regulations, shown in Appendix B.

Finally, additional interviews with subject matter experts, such as Jerry Cuttler (former AECL), Tom Tamagi (former AECL) and Navindra Persauld (Bruce Power Pressure Boundary Program) were conducted to finalize Chapter 3.

2.3 Proposed Solution

In Chapter 4, in order to address the identified gaps and limitations of the aged Bruce Power detectors, a comparative study of the alternative detector technologies and market trends were reviewed in order to propose a new detector model. Fission chamber characteristics were analysed using existing scientific literature, listed in the References section of this thesis, and compared to the requirements for Bruce power detectors identified in Chapter 3.

Next, a proposed enhancement to the design was developed based on the modern digital equipment and network capabilities. A physical walk-down was conducted along the proposed route of the new system installation. Location of the 6” Viewing Port on the Unit 3 Vertical Reactivity Deck, shown in Figure 4.11 in Chapter 4, was examined as well as the instrument racks and Main Control Room Modules and connection terminals.

The proposed design block diagram was developed using the Plant Instrumentation (PI) graphical module to demonstrate the proposed changes to the system architecture in order to develop a proposed framework for the installation procedure analysis and optimization. As discussed earlier, this thesis does not include detailed studies or optimization model for the new SUI architecture or parametric studies of specific components.

2.4 Solution Optimization

In Chapter 5, in order to develop SUI installation/replacement procedure, an activity model was created using IDEF0, as shown in Figure 5.1, and described in detail in sections 5.1 through 5.2. S-88.01 Standard guidelines were used to formally present the steps in the installation tasks in a sequential manner in order to produce a formal installation recipe. Following the process sequence identified in this model, all procedure tasks were broken down into smaller steps and assigned unique TaskID's using the guidelines of Recipe Formal Definition Language (RFDL), described in section 5.3, under the guidelines of ISA standards. RFDL standard methodology was selected in order to convert the procedure steps in to English-like statements so that the proposed approach could be standardized and used for other similar installations throughout the CANDU industry as well as to minimize the possibility of human error due to poor communication, elevated noise levels, personnel familiarity with descriptive English language, etc.

This was further used to produce Unit Procedure (UP) steps that were converted into a control recipe. Next, a proposal for the procedure verification using meta-operation language methodology was developed. In this proposal, an approach based on "seeded-error" technique is suggested for procedure verification and logic validation. In this approach error seeding is the process of purposefully adding known faults and logical hold points into a program in order to verify and eliminate unwanted logical loops and uncertainty conditions. In this thesis, this methodology is proposed for procedure task sequence verification and optimization. This is described in more detail in section 5.4.1 “Procedure Verification Methodology”.

Next, Hazard Operability Analysis was conducted for each procedure task, followed by Barrier Analysis in order to determine the potential hazards and develop the mitigating/control measurements. A risk matrix was developed and a proposal for further risk-based procedure optimization was made.

As a result of this analysis, several possible alternatives were developed, each presenting certain benefits in terms of time, budget or resource allocation. Thus, in order to select the best solution, a methodology for multi-objective optimization was needed, so that the best solution could be selected. In order to address this task, a Multi-Objective Genetic Algorithm (MOGA), given in section 5.5-5.8, based on Pareto optimal solution approach was proposed. Pareto-based approach was chosen in order to find the best possible choice out of a set of possible alternatives as described below.

2.4.1 Multi-Objective Optimization Methodology

As discussed earlier, during the risk-based optimization process it was determined that there are several possible solutions, each addressing certain parameters or constrains, e.g. time or budget, better than the others.

The need for a methodology for model optimization emerged during World War II where it was used to solve large-scale military logistics problems and has since been successfully used in many engineering, finance and other disciplines where the target is to make the most efficient use of resources such as time, money, time, staff, inventory and others.

To address this condition the optimization problem can be described using mathematical symbols and expressions, e.g. in a format of a function, so that it could be manipulated to achieve maximization, i.e. achieve the highest possible value or to be minimized, e.g. determine the least possible value where the function is still valid.

There are many various methods and approaches to model optimization, both mathematical and through the use of “off-the-shelf” computer applications such as LINDO optimization software available for Windows-based platforms that allows users to work with linear, non-linear, stochastic, integer and other types of models.

As discussed earlier, during the risk-based analysis portion of this thesis it was revealed that there are several alternative solutions, each with its associated benefits. Thus, the optimization methodology selected for this task will have to be able to address several equally important criteria, such as resource vs. time allocation or budget vs. time problem rather than a single objective one. Naturally, it is more desirable to achieve the target goal of installation/replacement of SUI equipment with the minimum cost, time and resource loading. This, however, may not always be possible. Quite often, it is necessary to spend additional funding in order to expedite execution of work or to assign more physical resources in order to meet the deadline. Thus, the problem of optimization comes to making a choice of which constrains will be given a priority, i.e. minimum cost, or which must be reduced, i.e. project time.

This condition can be represented by a mathematical model, where the alternatives are restricted by certain variables. Such mathematical model will, therefore, consist of an objective function *f(x)* and a set of constrains as shown in Equation 5.8-5.9 in Chapter 5. This approach of using mathematical modelling to solve complex problems is very typical and is often used in design, decision-making or financial portfolio development.

As mentioned earlier, there are many optimization methods and algorithms available, with over 4000 solution algorithms for different kinds of optimization problems: convex programs, separable programs, quadratic programs and the geometric programs. Some of these methods are better suited for certain problems that others. It is important to recognize the nature of a problem in order to choose the appropriate solution technique. Different optimization methods may vary in computational requirements, convergence properties, and so on. For example, Single-Objective Genetic Algorithms may not be the best approach to solving the problem with several competing priorities. Therefore, in this thesis Multi-Objective Genetic Algorithm (MOGA) based approach is selected for the proposed framework of SUI installation/replacement procedure so that several different objectives may be optimized at once.

Next, a set of feasible solutions will have to be determined, as shown in Equation 5.4 in Chapter 5. Any of the possible solutions x_i where all of the problem constraints are satisfied will be considered a feasible solution. Once the set of feasible solutions is developed, the algorithm can be further improved to find the solution with which the objective function has reached its maximum or minimum as shown in Equations 5.5-5.9. This result is called an optimal solution and will present the best cost-benefit answer to the procedure optimization. This is further explained in Chapter 5 for the selected case study of installation/replacement of SUI equipment for Bruce Power.

2.5 Performance Monitoring

In Chapter 6, Hierarchical Control Chart (HCC) methodology was developed and illustrated as a new interactive aid tool for system designers, developers and operators to be used for system troubleshooting and maintenance. The existing tools and aids available for control system designers, engineers and maintenance personnel at Bruce Power were analysed and the existing gaps and limitations were identified. For example, such tools as MatLab Simulink has been successfully used to design and simulate control circuits at AECL Chalk River Laboratories but by no means provide a good indication of the physical components of a system or their interconnections so that this information can be used for future system maintenance and troubleshooting.

Following this analysis, a set of requirements and functionalities for was formulated and Hierarchical Control Chart (HCC) methodology was developed to be used for nuclear power plant systems and components modeling in order to provide a single view of all elements and systems across a power plant. The objective of the new proposed automated HCC methodology is to aid system designers, operators and maintenance personnel with an automated tool for equipment, process lines and operations mapping, which offers a fast, intelligent and highly automated visual support for design as well as a troubleshooting and fault diagnostic tool.

Finally, several HCC functionalities were developed and implemented using Microsoft Visio and Visual Basic programming for graphical user interface set-up and a connection to MS Access database was developed to demonstrate the proposed interactive data capabilities.

2.6 Feedback and Modification for future implementation – Lessons Learned

This thesis will concentrate on the framework for installation/replacement of SUI equipment is will not include the installation process. In the future, should the proposal take place and the installation data become available, it can be used for feedback to fine tune the proposed framework in order to optimize the proposed solution and methodology.

CHAPTER 3: ANALYSIS OF OPERATIONAL CHALLENGES ASSOCIATED WITH THE EXISTING BRUCE POWER SUI SYSTEMS

3.1 Background – SUI System Purpose and Description

In a normal nuclear operations lifecycle, it is often necessary to shut down the reactor for maintenance, refurbishment, or due to a surplus of electrical power on the market. Following the shutdown, a reactor is normally placed in one of the shutdown states, such as Overpoisoned Guaranteed Shutdown State (OPGSS), or brought back to criticality. In either case, following a shutdown, activity of neutron sources in the reactor core decreases exponentially with time. During the normal start-up and operation at power, neutron flux is generated in the fuel pellets undergoing fission. This results in photo-neutron production that remains steady until the reactor is shutdown. Following continuous operation for a sufficient amount of time, fission product activity builds up to a level where it can be monitored by either in-core vertical flux detectors or out-of-core ion chambers. After a shutdown, the fission product activity decays away and might diminish below the levels where Shutdown System 1 and 2 (SDS 1&2) and Reactor Regulating System (RRS) ion chambers (IC) can accurately detect reactor neutron power and rate of change.

Thus, a means to monitor neutron activity at low and very low power levels is required once the SDS1 and SDS2 and RRS ion chambers go off-scale to ensure that the operators in the Main Control Room (MCR) can monitor neutron flux and obtain timely and accurate information on the status of the reactor in question. Table 3.1 below shows the nuclear instrumentation employed in CANDU reactors at various power levels.

Table 3. 1:Sensitivity ranges of CANDU Nuclear Instrumentation [2].

Power Range	Start-Up Instrumentation	RRS/SDS Ion Chambers	RRS/SDS In-Core Flux Detectors
Lower Limit	10^{-14} F.P.	10^{-7} F.P.	10^{-1} F.P. and above
High Limit	10^{-6} F.P.	1.5 F.P.	

Figure 3.1 below shows neutron count decrease as a function of number of shutdown days at Bruce A, where *nv (Φ)* or Neutron Flux (n/cm^2/s) is plotted versus the shutdown time. Ostensibly, the neutron activity decays quite noticeably, as can be seen below, and reduces to half of its original value after one week of outage time.

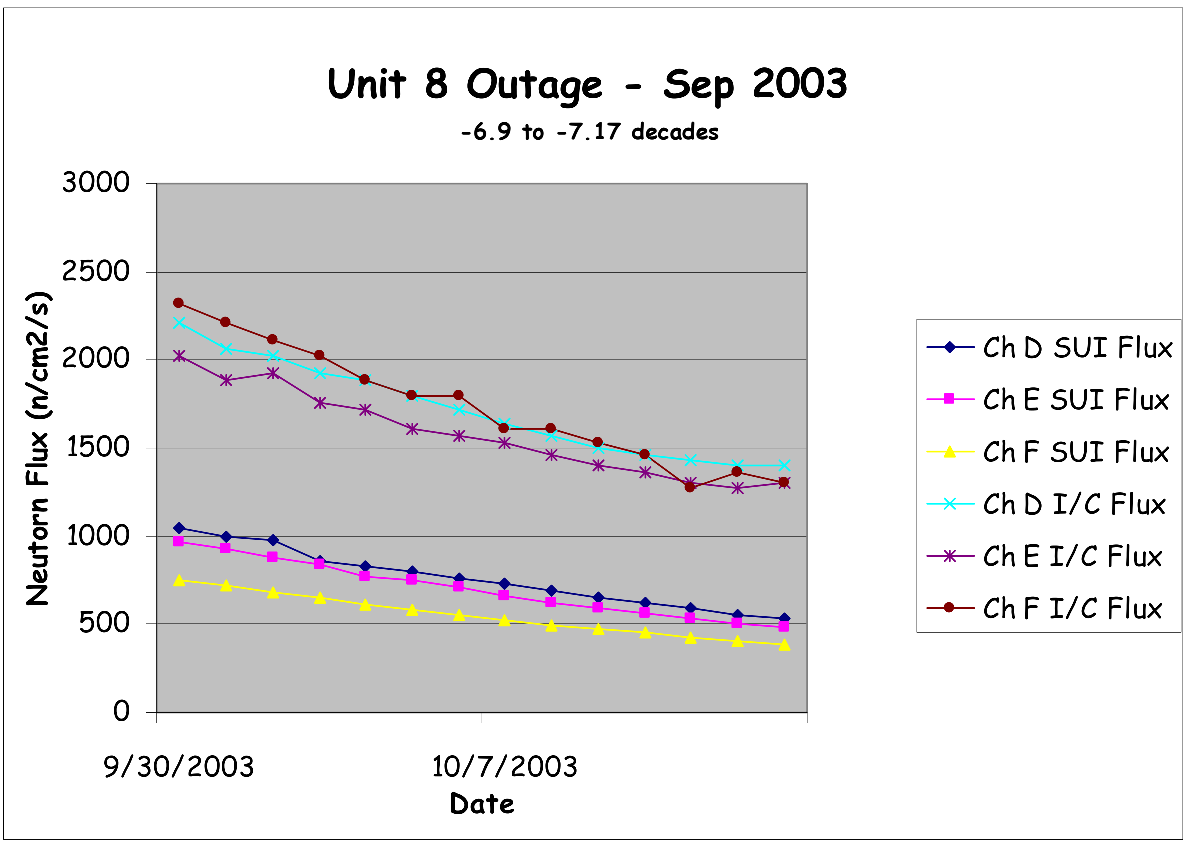

Figure 3. 1: Neutron Decay Curve as a Function of a Number of Shutdown Days, shown for a period of 1 week during Unit 8 Outage. Graph compiled using historical data manually recorded and stored in the system binder for "September 2003" outage.

For a specific case of prolonged outages when the moderator level is decreased or drained (Low Level Drained State or Drained Guaranteed Shutdown State), the SUI detectors will observe much higher counts than those shown in Figure 2.2 with the maximum read-out obtained when the moderator level is just below the location of the detectors. This happens mainly due to a number of thermal neutrons that escape from the over-poisoned moderator and form a cloud above the surface level. Figure 3.2 below shows a typical count rate r_c output as a function of moderator level l_{mod}.

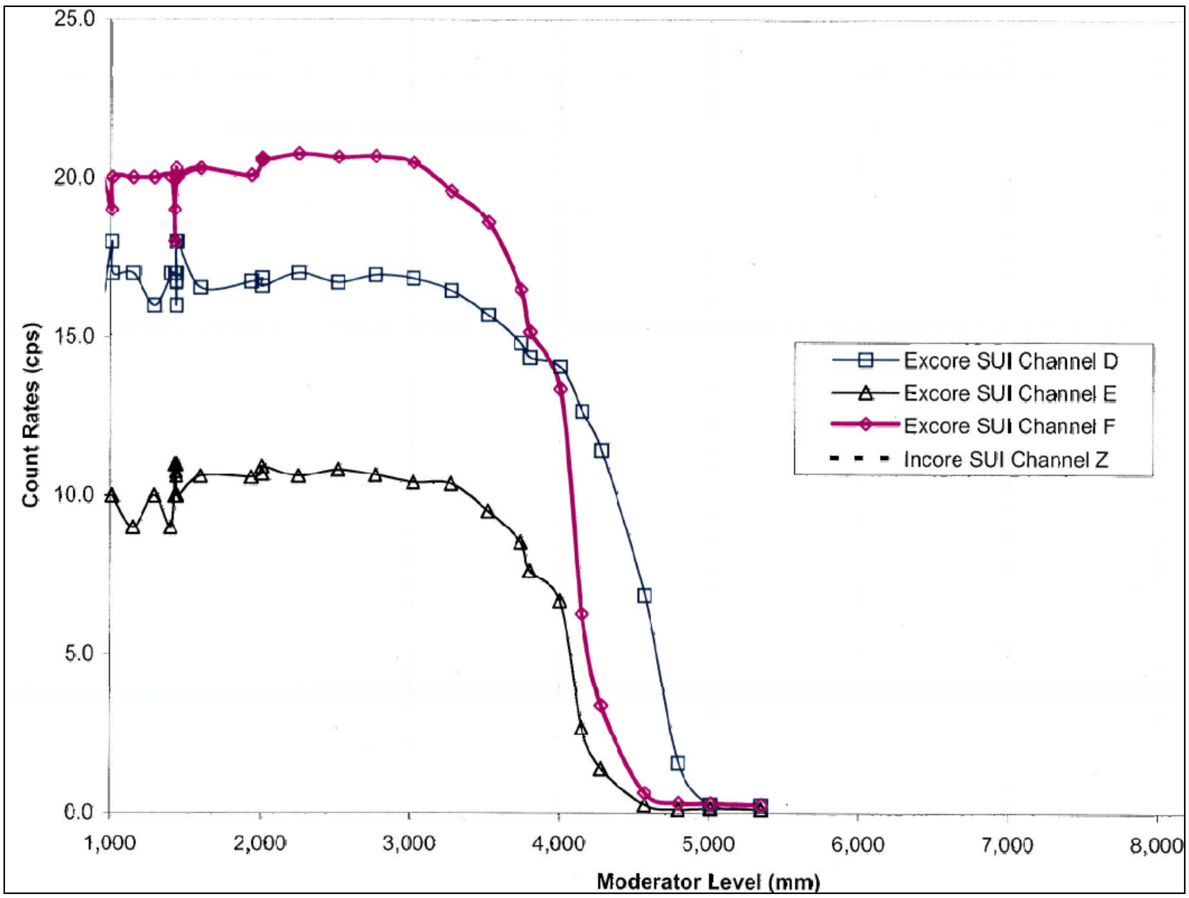

Figure 3. 2: Bruce Power Unit 4 Count Rates vs. Moderator Level (data sampled on 29-Mar-2003) [6]

Once the guaranteed reactor shutdown state is achieved and poison is being pulled from the moderator to enable approach to criticality, SUI continues to provide flux and rate monitoring. SDS1 trip capabilities are in place while RRS ion chambers provide an accurate and reliable reading and RRS can resume normal control of the reactor as shown in Figure 3.3 below.

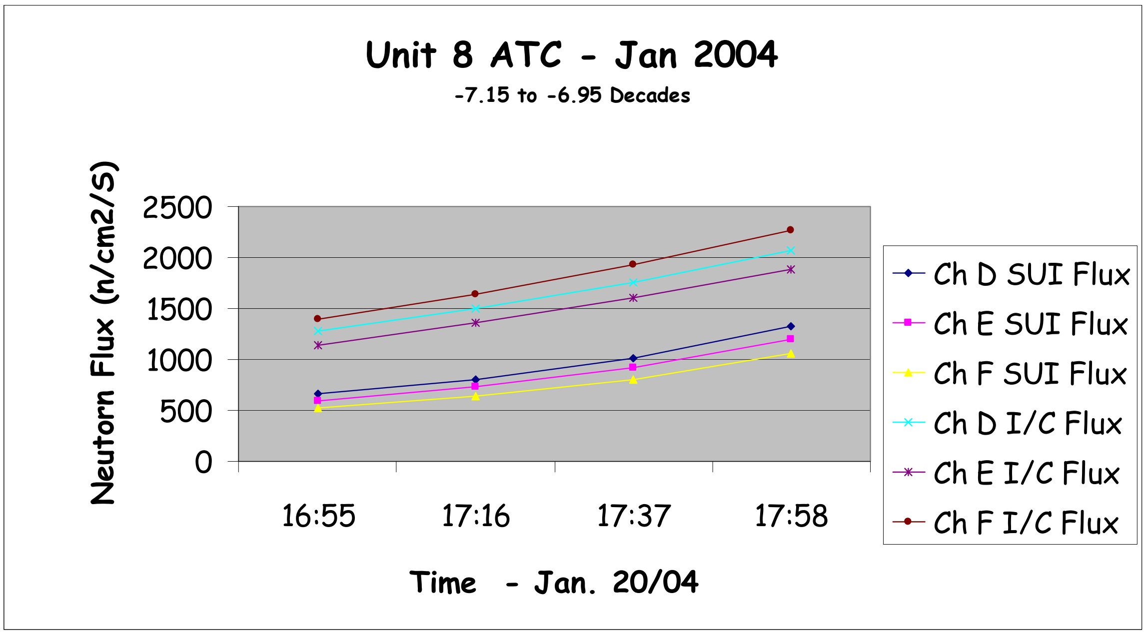

Figure 3. 3: Unit 8 Approach to Criticality Neutron Flux and SUI Counts, Unit 8, January 2004. Graph compiled using historical data manually recorded and stored in the system binder for "January 2004" outage.

Start-Up Instrumentation most commonly used today in CANDU-based power plants typically consists of thermal neutron detectors, electronic modules, cables and connectors. There is typically a dedicated computer terminal located in the main control room that is used for SUI data collection and processing. There are two major categories of Start-Up Instrumentation equipment used in CANDU power plants – in-core and out-of-core SUI modules.

In-core SUI Instrumentation can be positioned at various locations throughout the reactor core, both in a stationary and mobile manner. Stationary SUI detectors are permanently fixed at particular locations in the core to provide detailed information about the neutron flux distribution and magnitude. Mobile detectors could be of a "traveling type" where a detector assembly is outfitted with a mechanical drive unit and could be moved within the reactor core as needed. There are various types of in-core instrumentation available on the market, however they all share similar characteristics of size and shape restrictions imposed by the core geometry and available space between fuel channels. One of the main advantages of using in-core SUI instrumentation is that it requires lower neutron sensitivity. This, however, is somewhat offset by the stresses imposed on the system due to the extremely hostile environment inside the reactor core, which leads to faster component degradation and maintenance challenges due to physical inaccessibility.

Out-of-core SUI instrumentation is located outside the reactor core and is therefore better suited to provide bulk neutronic activity measurements. It is less affected by the reactor size and geometry factors as well as the high-temperature and pressure environment in the pressure vessels. Out-of-core SUI instrumentation is more easily accessible for maintenance and inspection; however it requires higher sensitivity and gamma discrimination capabilities than the in-core models.

Either type of SUI instrumentation is connected to the main control room indicators via a number of cables and connectors. A number of amplifiers, pre-amplifiers, signal analyzers, power supplies and other components may be used to comprise a complete SUI circuit. For reliability and redundancy reasons the SUI equipment is typically triplicated. Figure 3.4 below shows a typical arrangement for SUI equipment.

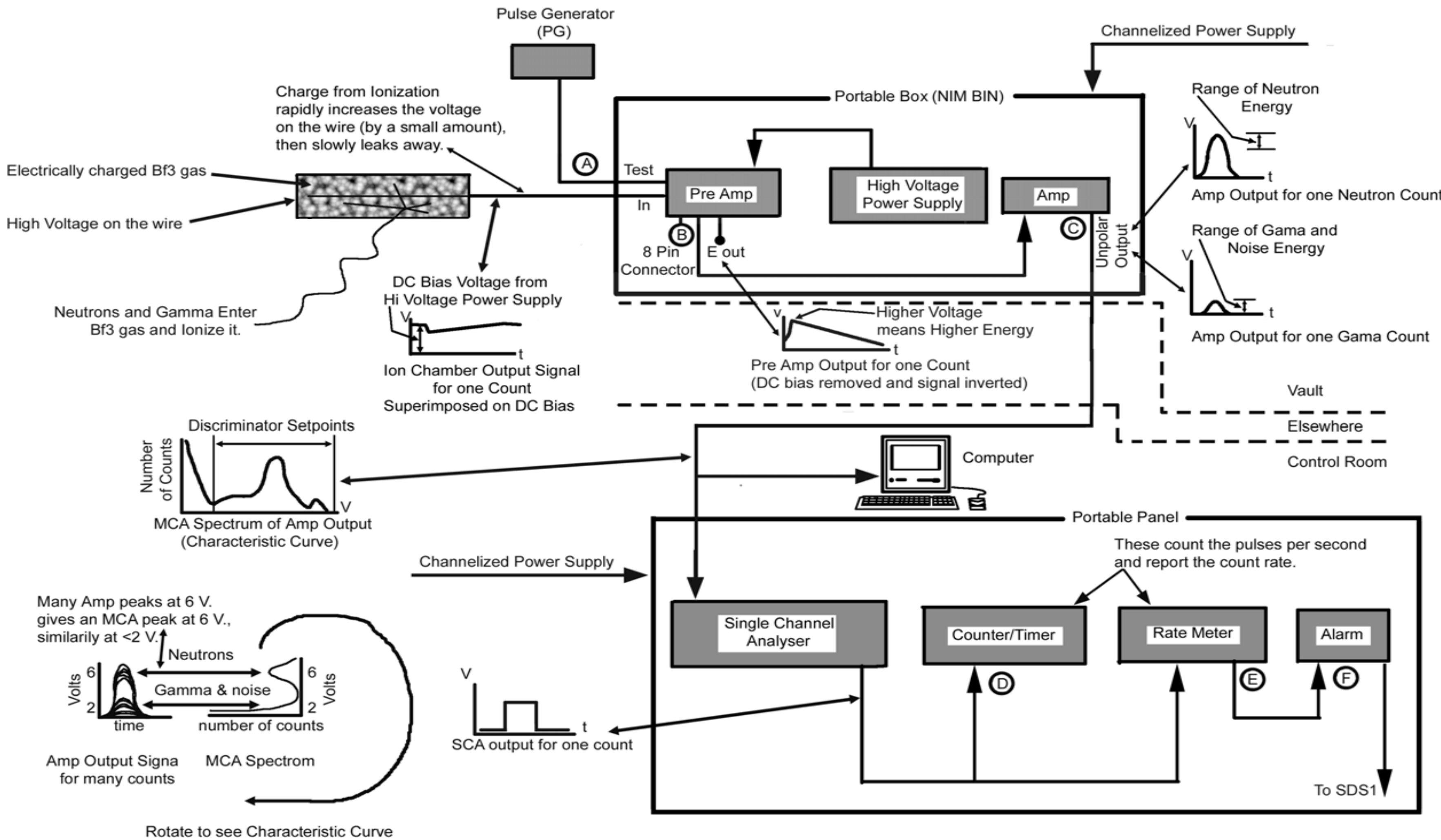

Figure 3. 4: Typical SUI Channel Set-Up [6].

Each of the three channels consists of a neutron detector, pre-amplifier, an amplifier, and a high-voltage power supply. Thermal neutron detector types vary depending on the plant location and application. However, all of them (irrespective of the technology used) serve the purpose of detecting neutron activity or so-called "neutron counts" and discriminating it from the associated gamma noise. The majority of neutron detectors used for reactor Start-Up Instrumentation are of the gas-filled type [1]. For CANDU power plants in Canada, the two most common detector types used are uncompensated ionization chambers with Boron Trifluoride (BF_3) and Helium (^3He) detectors. Both are of the thermal neutron-sensitive type. The type of detector chosen as well as the size and the gas filled pressure depend on the reactor power level and installation methods. A detailed description and comparison between the BF_3 and ^3He detectors will be provided in the following sections.

A signal received from a neutron interaction in the detector is sent through the pre-amplifier to the spectroscopy amplifier. The amplifier shapes the pulse proportionally to the charge deposited on the detector during the neutron interaction event and is consequently transmitted to the main control room through a series of cables penetrating the containment. It is important to note that the amplifier boosts both the detector signal and the noise picked up at the detector lead, connectors or along the cable. Therefore, cables and connectors used for the SUI instrumentation should be low-noise, environmentally qualified, as well as very high quality splicing and insulation materials should be used.

Once the signal is received in the main control room, it is connected by the single channel analyzer for information processing and interconnected to the SDS1 trip circuits for implementation of NOP protection. Single Channel Analyzer (SCA) is used to process the incoming signal to distinguish the true neutron activity response and the instrumentation and gamma noise. This signal is, in its turn, used by the counter and rate-meter components to determine the number of neutron counts per unit time, e.g. per second. This allows for translating the amplifier signal into a number of neutron counts representing the neutron source activity in the core as well as rate of change to determine whether it is increasing or decreasing over time. Figure 3.5 below shows a sample SUI pulse waveform as seen on oscilloscope.

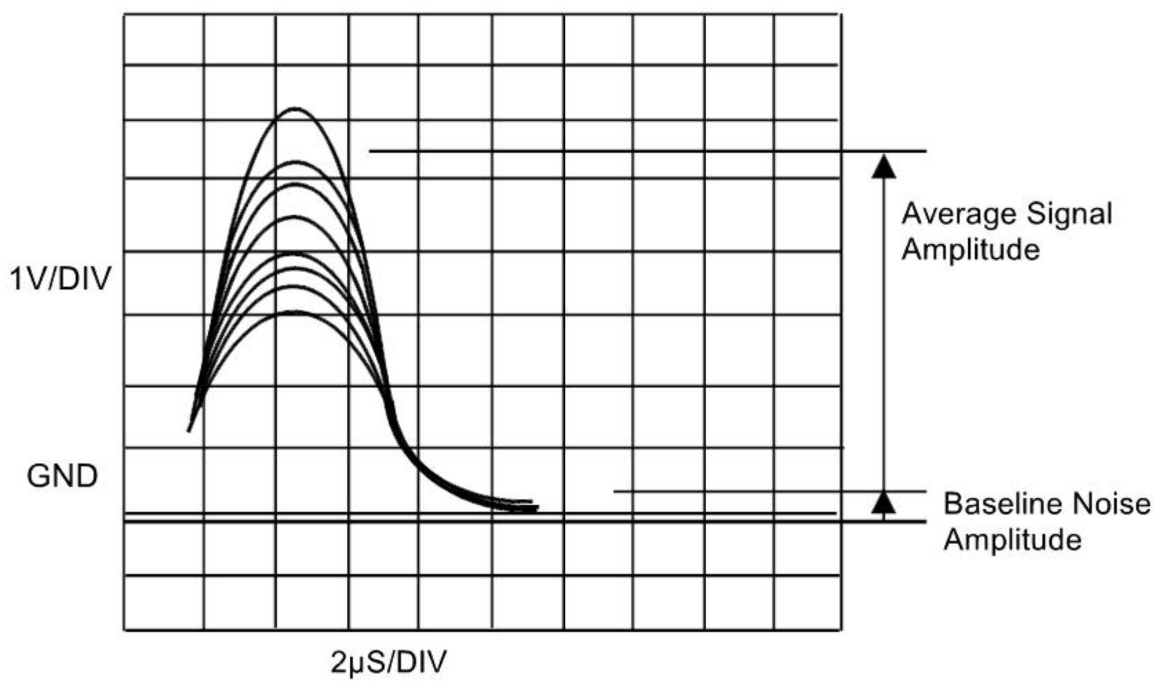

Figure 3. 5: A typical SUI Pulse Waveform As Seen on Oscilloscope [7].

Multi-channel Analyser (MCA) is normally connected to the SUI electronics, particularly during prolonged shutdown phases, to record the initial neutron count activities and count increase information. MCA creates a plot composed of the frequency with which specific pulse height voltages are recorded as function of the peak voltage via discretizing the voltage range into small portions or so-called channels. Each time the pulse height falls into the corresponding channel, its counter is incremented by “1”, and thus a neutron count is obtained even though the SUI detector does not directly interact with incident neutrons. Figure 3.6 below provides a sample output of SUI multi-channel analyzer (MCA) where a neutron peak can clearly be seen at 5.9 V versus low-energy (0.9-2.0 V) gamma noise.

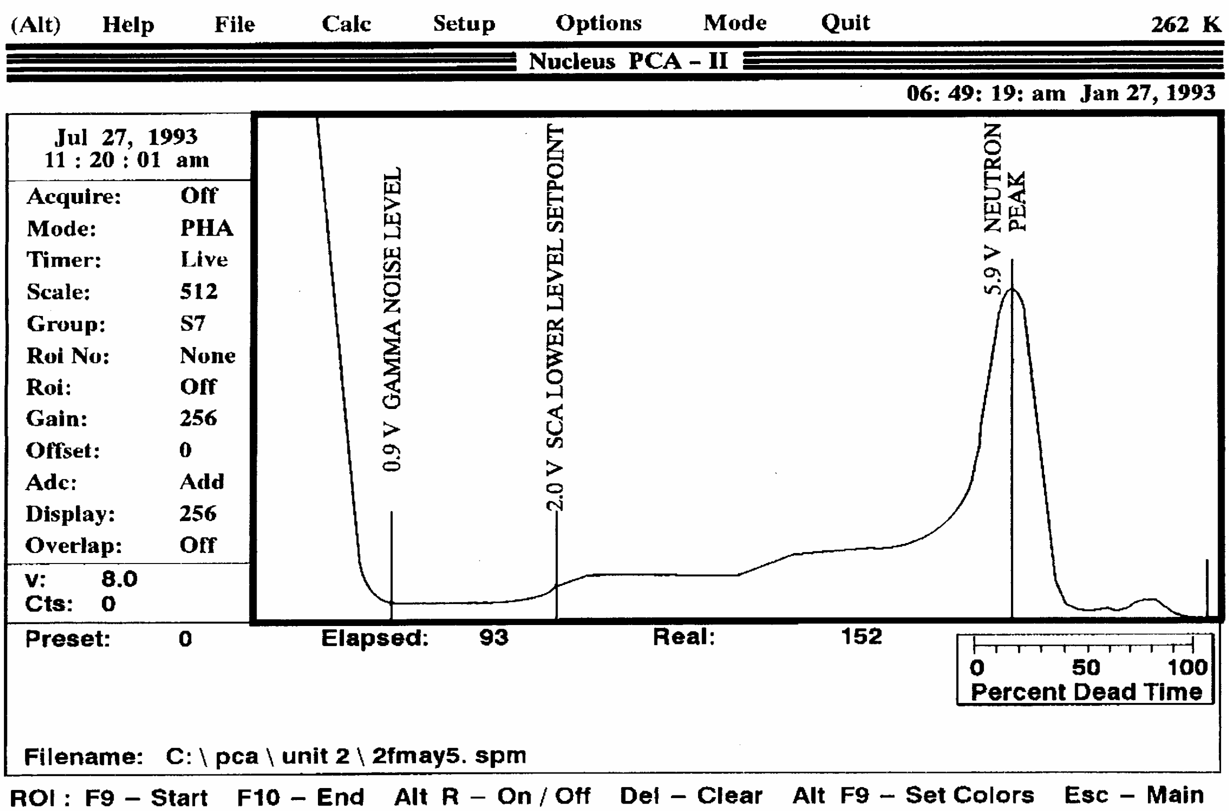

Figure 3. 6: SUI Multi-Channel Spectrum Analysis Sample with BF_3 detector taken on 27 July 1993. The 5.9V peak on the right-hand side corresponds to a neutron count, while the lover energy gammas can be seen on the left at 0.9 and 2.0V [6].

There is typically an alarm unit with high and low level alarm capabilities on the SUI main control room equipment. The high level alarm provides an indication that a potential loss of regulation may occur or the unit might be experiencing count rates higher than the design basis, which may lead to sensor damage or degradation. Low level alarms indicate to the control room personnel that the SUI detectors might be reading off-scale low or a loss of power for the SUI equipment has occurred.

To provide a good visual indication of the neutronic activity in the core there is typically a dedicated computer and printer installed in the main control room to be used with SUI equipment. This computer integrates the data obtained from the SUI channels and builds a graph that can be analyzed by the control room operators or station engineers and nuclear scientists to determine the accuracy of the SUI detector readings and to confirm the level of neutronic activity in the core.

3.2 Overview of Conventional SUI Detectors at Bruce Power

There is a wide variety of neutron detectors available on today's market. However, when it comes to nuclear power plant application, not all detectors perform well due to harsh environments inside the reactor containment, general inaccessibility for maintenance and the need to accurately discriminate between neutron and gamma activities.

In CANDU power plants in Canada the two of the most commonly used types of neutron detectors, Boron Trifluoride (BF_3) and Helium (^3He), belong to a family of gas-filled detectors. All gas-filled detectors share the ability to discriminate between neutron and gamma ray energies and are typically well suited to work over a wide dynamic range. Scintillation-based detectors are less suitable because of higher sensitivity to gammas and semiconductor detectors are significantly more susceptible to radiation damage.

Boron Trifluoride (BF_3) or Helium (^3He) detectors have also been selected as the technology of choice for SUI instrumentation in CANDU power plants, such as Bruce Power Nuclear Generating Station (NGS). It is often the case that high levels of gamma radiation are present in the reactor core during low power or start-up which makes BF_3 and ^3He detectors' ability to discriminate between a gamma-ray process and a neutronic reaction an important factor.

Both BF_3 and ^3He detectors are implemented based on the proportional counter principle. The detector gas tubes are constructed using a cylindrical outer cathode and a thin inner wire anode. Following exposure to neutron flux a number of charged particles, or so-called ion pairs, are created as a result of gas interaction with the incident neutrons. The charged particles recoil at high speed creating primary ion pairs and thus ionizing the gas inside the chamber resulting in a current flow between the cathode and anode. This current is subsequently detected and converted into voltage pulses, which are subsequently translated into neutron flux measurements. Table 3.2 below shows Bruce A Neutron Power Measurement and voltage conversion guidelines.

Table 3. 2: Bruce A Neutron Power Measurement and Voltage conversion [5].

Flux	Amps	Volts	% Power	Decimal	10^	Decade
$2.85x10^{14}$ n/s/cm^2	10^{-4}	4.0	100	1	10^0	0
			10	0.1	10^{-1}	-1
			1	0.01	10^{-2}	-2
			0.1	0.001	10^{-3}	-3
			0.001	0.0001	10^{-4}	-4
			0.0001	0.00001	10^{-5}	-5
			0.00001	0.000001	10^{-6}	-6
	10^{-11}	0.5	0.000001	0.0000001	10^{-7}	-7

One of the important features of all gas-filled detectors, including BF_3 and ^3He detectors is a so-called wall-effect. The wall effect occurs when charged particles, both the alphas and the other ions, e.g. lithium ion, to reach the detector wall before they had an opportunity to loose their excess kinetic energy in the gas. This will result in a pulse smaller than the one produced if the charged particle deposited their entire energy in the gas. The ions' energy ends up being deposited in the detector wall material and becomes neutralized there. The wall effect is typically present on all SUI spectrum readings and is an important factor in selection of SUI multi-channel resolution and discriminator bounds.

BF_3 and ^3He detectors that are currently used for Bruce Power SUI applications along with their associated limitations will be analysed in more detail further in this paper.

3.2.1 Bruce Power SUI Detectors – BF_3 Type

All BF_3 detectors are based on a well-known principle of indirect neutron detection. In BF_3 detectors, Boron Trifluoride serves both as a target for thermal neutrons and a proportional gas. The detector gas is highly enriched in ^{10}B, up to 96% [1] to provide the necessary efficiency. The thermal cross section for the ^{10}B (n, α) reaction is 3840 barns [1], which makes it extremely suitable for applications in thermal reactors. Another main advantage of BF_3 detectors is that Boron is a readily available element. Neutronic activity is derived from the ^{10}B interaction with an incident neutron, as shown below [8]:

$$^{10}B + n \rightarrow {}^{7}Li^{*} + \alpha + 0.48\ \text{MeV}\ \gamma\text{-ray} + 2.31\ \text{MeV}\ (94\%) \tag{3.1}$$

$$^{10}B + n \rightarrow {}^{7}Li + \alpha + 2.79 \text{ MeV } (6\%) \quad (3.2)$$

Typically, the resultant reaction statistically falls into two main categories – each with the reaction product ^{7}Li in a ground or in an exited state as shown above. It is well described in literature [1] [8] that about 94% of all reactions result in an exited state with the remaining 6% being in ground state. The produced alpha and lithium ions possess kinetic energies summing to b 2.4 MeV. If the entire kinetic energies are dissipated in the counter gas by ionization, approximately 80,000 primary ion-pairs are created. When a polarizing voltage of 1,800-2,000 Volts is applied, the avalanche results in 2x10 secondary ion-pairs. The conversion gain of the preamplifier is about -235 mV/(M ion-pair) so a positive pulse of 500 mV is obtained and passed on to the preamplifier. The pre-amplifier functions well up to 5V DC corresponding to the count rates up to 5×10^{5} counts per second, which is well within the system requirements for SUI equipment [5].

BF_3 detectors are commonly used in all Bruce A SUI instrumentation. Bruce A SUI detectors are tube-shaped Reuter-Stokes (Model No: RS-P1-0403-102) counters made of 1100 aluminium alloy with alumina ceramic detector insulation material. Their physical dimensions are 1.27 cm OD by 15.6 cm long with the fill gas being 96% enriched Boron-10 with BF_3-gas fill pressure of 55-60 cm/Hg. Table 3.3 below shows a summary of Bruce A BF_3 detector specifications.

Table 3. 3: Technical specifications summary for SUI BF_3 detectors at Bruce A [2].

Parameter	**Value**
Diameter	5/8" (16 mm)
Overall Length	5 ½ " (140 mm)
with MHV Connector	addl. 7/8"
Active Length	1" (24 mm)
Cathode	Brass (0.035")
Anode	1 mil tungsten
Connector	MHV-teflon
Insulators	Ceramic and glass
Gas filling	Enriched BF_3 (96% B^{10})
Gas Pressure	60 cm Hg
Temperature	-80 to +80°C
Neutron sensitivity	appx. 0.35 cps/nv
Neutron flux range	0.25 to $2.5x10^4$ nv

Bruce A BF_3 detectors typically possess a sensitivity of 0.35 neutrons per second per square centimetre (cps/nv). This is quite sufficient for all normal and long shutdowns and subsequent approach to criticality. The BF_3 detectors are deemed to perform well with the count rates as low as 5 counts per second. The RSN137 model currently used is designed to withstand a range of flux of 0.25 to 10^4 n/cm^2/s and a gamma dose rate of up to 100 Rad/h for 24 hours without significant deterioration [9].

Detection efficiency of a BF_3 detector for thermal energy neutrons (0.025) eV is approximately 91.5%, which makes it very suitable for start-up instrumentation applications. A sample print out of a SUI spectrum analysis using a typical BF_3 detector at Bruce A is shown in Figure 3.7 below.

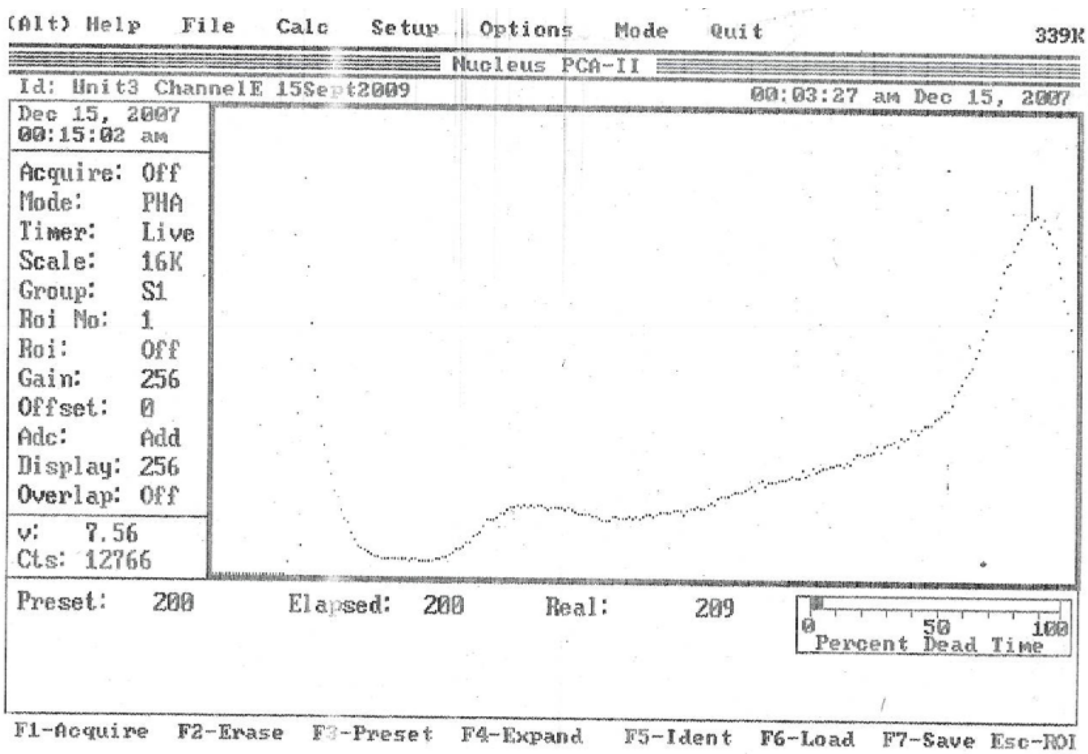

Figure 3. 7: BF_3 Detector Spectrum Analysis Sample at Bruce A, data taken Dec 15, 2007 for Unit 3, currently stored in a hard-copy format in the system binder for December 2007 outage.

Please note the poor quality of the printout due to obsolescence of printer components and unavailability of replacement ribbon. As discussed earlier, MCA plot is an instantaneous snapshot of data. The current SUI system MCR equipment has no data storage capability, so that the data cannot be reproduced later should there be a need for a better graph quality or further analysis. Also, the current system has no data transfer capabilities or connection ports for any external data processing software or hardware. As a result, the snapshot shown above cannot be reprinted or reproduced. This drawing is an ideal example why the system has to be upgraded in order to sustain a reliable performance and to ensure that data archiving and storage features are developed. This will be discussed in more detail in the proposed SUI System design enhancements.

3.2.2 *Bruce Power SUI Detectors – 3He Type*

A ^{3}He proportional counter is another type of detector widely used for SUI applications. Bruce B SUI instrumentation is typically implemented via ^{3}He detectors. ^{3}He gas possesses a cross-section for thermal neutrons of 5330 barns [1], which is noticeably higher than that of B10.

$$^{3}He + n \rightarrow T + p \tag{3.3}$$

$$E_p=0.573 \text{ MeV}, E_T=0.191 \text{ MeV} \tag{3.4}$$

Compared with BF_3 detectors, ^{3}He gas-filled tubes are of the same size and pressure. The traveling distances of the reaction products are much longer, which results in a more significantly wall effect [10]. This is largely due to the fact that ^{3}He has a very low atomic mass. Therefore, a small amount of a heavier gas, such as CO_2 or Ar is used in the fill-gas mixture to increase stopping power. Overall, ^{3}He filled tubes can be operated at much higher pressures and temperatures and can provide an acceptable performance output at temperatures as high as 200-250° degrees C. ^{3}He-based SUI detectors used at Bruce Power are GE Reuter-Stokes models. A summary of their technical specifications is provided in Table 3.4 below.

Table 3. 1: Technical Specifications Summary for SUI ^3He detectors at Bruce A [5].

Parameter	Value
Diameter	2.28" (57.9 mm)
Overall Length	15.69 " (398.5 mm)
Connector Type	HN
Body Material	304 S.S.
Connector Material	Brass, silver plated
Insulators	Alumina Ceramic
Gas filling	^3He
Gas Pressure	60 cm psig (0.41 MPa)
Temperature	-25 to +100° C
Plateau	200 V
Neutron sensitivity	110 cps/nv +/-10 %
Neutron flux range	1.004E-04 to 5.00E+02 nv

Another main difference between BF_3 and ^3He detectors is that the latter shows a much higher sensitivity and is capable of working with very low neutron counts. Typically, ^3He detectors can still produce an accurate indication of neutronic activities in the reactor core with neutron counts below 5 counts per second and are known to have been successfully used for power levels approaching the source level with the counts as low as 0.3 counts per second [11]. Figure 3.8 below shows a typical SUI ^3He detector spectrum analysis sample at Bruce A.

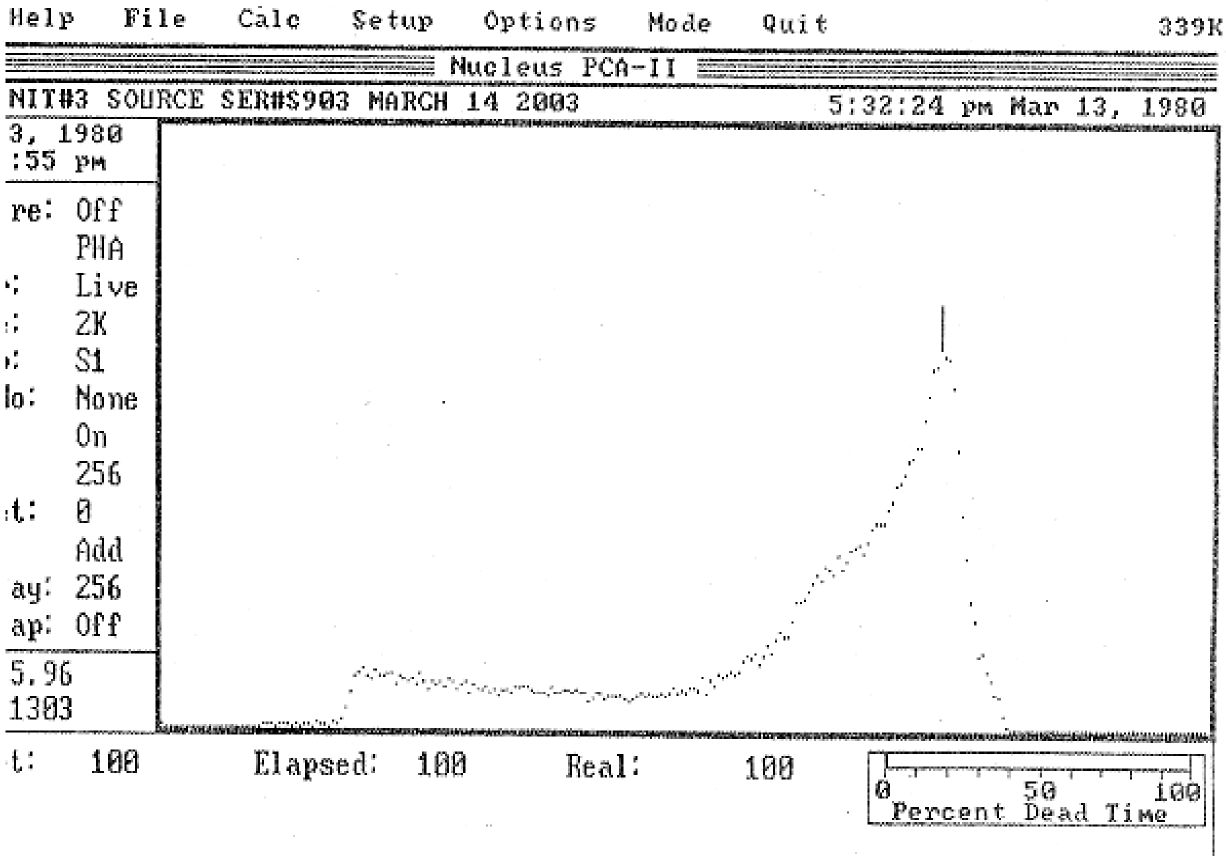

Figure 3. 8: ^{3}He Detector Spectrum Analysis Sample at Bruce A, data taken 13 March 1980 for U-3.

3.3 Analysis of Existing Challenges Associated with Bruce power SUI systems

3.3.1 Bruce Power BF_3 Detector Ageing and Performance Limitations

There are quite a few disadvantages and limitations associated with using existing BF_3 detectors as a means of neutron detection in CANDU reactors, particularly in SUI instrumentation at Bruce Power A and B stations.

There are several known disadvantages and limitation associated with BF_3 detectors.

First, BF_3 proportional gas detectors show degraded performance when operated at high pressure [8]. Therefore, the absolute pressure of gas inside the detector tube should be limited to 0.5-1.0 atm [1] and should be carefully monitored.

Also, because of high excitation voltage current, leakage through the detector insulators is common. This becomes particularly troublesome in high humidity environments, such as CANDU reactors.

Performance degradation due to ageing and radiation damage [12] is another main issue affecting BF_3 detectors. In 2003, degradation of Bruce Power SUI BF_3 detectors resulted in failure of five of these while in-service [13].

Contamination of the anode and cathode components by the disassociation products of avalanche reactions is another factor contributing to detector ageing and performance degradation. According to Dr. G. Knoll studies [1] this typically happens after 10^{10}-10^{11} counts.

Another limitation of BF_3 counters is susceptibility to vibration and shock [1]. Mechanical shocks against the HV detector cables or detectors themselves result in large noise pulses that get interpreted by the SUI electronics as large spurious increases in neutron counts. This becomes a significant limiting factor as the BF_3 SUI detectors at Bruce Power A and B are used in a mobile out-of-core configuration which requires frequent handling and positioning during unit outage periods. This will be discussed in more detail in the following sections where main operational challenges and concerns associated with current SUI instrumentation at Bruce Power are analyzed.

3.3.2 Bruce Power 3He Detector Ageing and Performance Limitations

Some of the main limitations of ^{3}He detectors include the fact that ^{3}He is a noble gas and must be used in its gaseous form to fill the detectors. Although ^{3}He gas is widely available for commercial applications [1], it could present higher costs associated with obtaining and maintaining the detectors. Performance degradation also results from the loss of containment on the detector, where gas escapes and air enters.

Although the high neutron sensitivity of ^{3}He detectors may be an important advantage in many applications, such as radiation detection in the biomedical industry or research, it becomes a hindering factor for SUI instrumentation in commercial CANDU reactors such as Bruce Power NGS. Neutron sensitivity as high as 110 cps/nv [8] might result in slow response of the detector electronics and the maximum count it can handle before burn-out damage occurs to the internal components.

^{3}He detectors are also prone to build-up of electronegative poisons in the gas with time. Bruce Power operational history shows that ^{3}He detectors are more susceptible to problems with electronic equipment saturating at high powers than the BF_3 tubes. It is a regular operational concern that a rapid increase in reactor power at start-up might result in neutron fields being too high for the detectors' electronics to handle.

Another main limitation of ^{3}He detectors is their low neutron-gamma discrimination capability. They are significantly more sensitive to both gammas and neutrons with much lower ability to discriminate between the two, often times counting both as one simultaneous count. As a result of this, BF_3 detectors have to be shielded in lead even when positioned in-core.

3.3.3 *Bruce Power Challenges With Current SUI System Operation and Maintenance*

As per current Bruce Power operational practices, SUI detectors are installed in spare horizontal SDS2 Ion Chambers. Detectors are installed 7-10 days following reactor shutdown and stay in service during approach to criticality and until reactor power is raised up to 10^{-5} decades when RRS Ion Chambers can reliably indicate current neutron flux in the core.This practice involves a number of steps before the SUI equipment can be placed in service.

First, SUI detectors have to be shop calibrated. Next, the detectors, scaffolding materials and tools have to be transported to the accessible area. Once the scaffolding is installed, the shield plugs are removed to allow the insertion of detectors in to the spare SDS2 Ion Chamber (IC) housing.

Next, the SUI neutron detectors are installed in each of the three SDS2 Ion Chamber housings, in the tube that normally hosts the spare ion chamber. To accommodate SUI detector the spare IC is removed and replaced by a SUI detector as shown schematically in Figure 3.9 below.

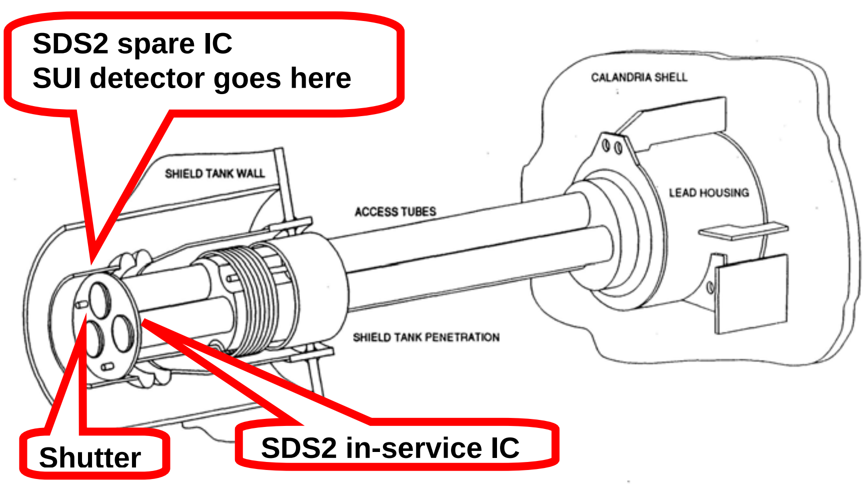

Figure 3. 9: Schematical Arrangement for SUI Detectors in SDS2 Ion Chamber Housing [2].

This is further illustrated in Figure 3.10, where the photo shot of an actual SDS2 Ion Chamber is shown with the spare Ion Chamber shield plug removed and SUI detector installed in service.

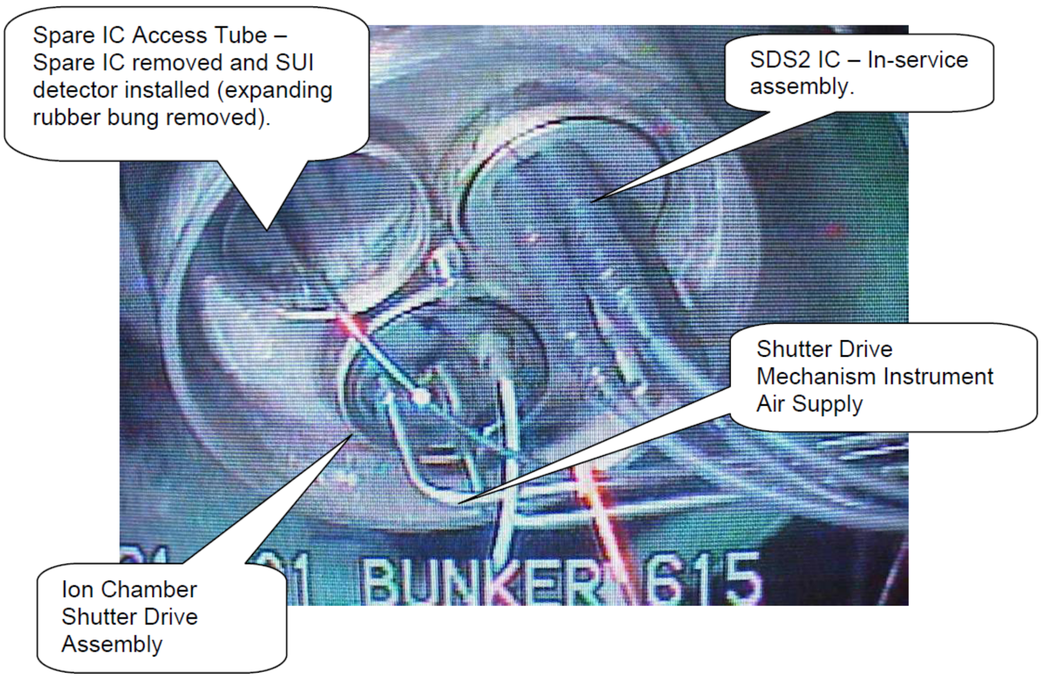

Figure 3. 10: SDS2 Ion Chamber Housing with Spare Ion chamber Removed and SUI Detector Installed in Service. Photo taken during SUI detector installation performed in September 2009 outage and stored in the online system folder for engineering records.

Next, the detector is wired to the portable intrumentation rack positioned on the north side of the vault. The rack contains a pre-amplifier and a high voltage power supply unit.

Once the equipment is installed in the vault, it is connected to the main control room (MCR) panels via a number of cables so that output signal from the amplifier is sent to the MCR via a fixed-penetration junction box. Both vault and MCR equipment is powered up from the same 120 V Class II bus. Next, the alarm unit output from the MRC panel is connected to SDS1 Channel D, E, and F trip logic.

Following a prolonged shutdown, it is expected to encounter core conditions where the gamma flux from the fission product buildup may be significantly higher than the neutron flux. It may become necessary to vary the detector's sensitivity to the magnitude of neutron flux. This is achieved by adjusting the depth of detector insertion in the tube. Initially, following a reactor shutdown, the detectors are inserted at various depths depending on the recommendations given by SUI engineers and Fuel & Physics specialists. Detectors are moved inside the access tube through the shield tank following the neutron decay pattern in the reactor core. During extended outages with very low power levels BF_3 detectors can be further inserted into the IC housing lead shielding to maintain sensitivity.

To reduce sensitivity of detectors for subsequent start-up and approach to criticality this process is repeated in reverse, the detectors are pulled back away from the calandria wall, out of the lead shielding and into the outer shield tank wall area. BF_3 detectors are typically repositioned several times until the maximum distance of 24 inches is reached. Neutron count rates half with every 4 to 5 inches of withdrawal, so that at 24-inches of distance the detector's sensitivity can be reduced by a factor of 100, or 2 decades in count rates.

Another typical scenario that routinely occurs during extensively long outages and shutdowns is when BF_3 detectors need to be swapped for ^3He detectors due to a need for higher sensitivity. Detector change-out routines result in additional delays, dose rates and error-likely situations as the SUI BF_3 detectors have to be removed from service and ^3He

detectors have to be obtained from the storage facility. Once obtained, they are then calibrated and placed in service following the steps described earlier. Once the unit is on a ramp up to power and moderator purification columns are valved in, the ^{3}He detectors can no longer be used due to their high susceptibility to saturation at high power. Thus, the detectors have to be swapped again for BF_3 counters, which in their turn will have to be re-positioned several times during approach to criticality.

3.3.4 Challenges Due To Mechanical Disturbance and RF/EMI Effects

At very low power, any work performed on the SDS2 platform in the vault has the potential to trip SDS2, particularly if IC cables are jarred or even touched. It is therefore extremely important to make sure that only one IC is being approach at a time, so that the chance for a rated trip on more than one channel is minimized. There have been a number of trips and plant upsets resulting from arc-welding jobs taking place in the close vicinity of instrument rooms as well as trips resulting from personnel walking near SUI cables [14]. One of the most significant events occurred at Bruce A in 2003 while work was being done by U4 outage crews in a congested area near the SUI location. Channels E and F tripped multiple times with count rates increasing dangerously close to an alarm level. It is believed that this may have been the result of inadvertent contact with SUI cabling during scaffolding modification work. Figure 3.11 below shows a snapshot of U2 SDS2 platform where SUI installation and repositioning tasks are taking place.

Figure 3. 11: Bruce A Unit 2 SDS2 Horizontal Reactivity Management Deck. Photo taken during SUI detector installation performed in September 2009 outage and stored in the online system folder for engineering records.

Following this event, operating policies and procedures were modified to ensure that access to SUI instrumentation areas is strictly controlled. However, as can be seen from the photo above, this policy cannot completely eliminate the challenges associated with movement of personnel or equipment in the tight congested areas of the vault.

Another famous event occurred on the 8th of March, 2007, when some SUI instrumentation was discovered missing from the Bruce Power Unit 6 vault [15]. The following investigation revealed that SUI equipment was unintentionally removed to another area. It was also discovered that there was garbage and other items left in the dedicated SUI area that could potentially cause contamination and mechanical damage to SUI units.

EMI/RF noise induced SUI trips have been another long standing issue for Bruce A. SUI detectors have been known to pick up noise from portable radios in and around the instrument rooms as well as that produced by motors, pumps and other rotating equipment. SUI connectors and cables that are presently in service at Bruce A are showing significant visible signs of wear and tear, as well as ageing related degradation. Control maintenance personnel routinely find that constant positioning and repositioning of cables results in accelerated wear and tear, which in turn results in additional tasks and work orders that have to be completed prior to SUI installation as well as throughout the unit shutdown. Even though best efforts are made to improve signal cable connections and shielding in order to eliminate or minimize noise affecting SUI channels, it is still a strong contributor to the number of spurious trips that occur every outage.

3.3.5 Outage Extensions and Delays to Critical Path Challenges

Delays in SUI installation have been a long standing issue at Bruce Power due to the detector assemblies' degradation problems, installation challenges and SUI equipment reliability. Since SUI instrumentation must be placed in service before the reactor power decays below -7 decades, any delays resulting from unavailability of SUI equipment presents very significant challenges to the operation, engineering and maintenance staff and results in elevated work load, dose rates and delays to the critical path.

For example, on 5 April 2007, Unit 4 was in a shutdown state with all three channels of SDS1 ion chamber readings of -7 decades. As per current operating policies and procedures, Start-up Instrumentation had to be installed before any SDS1 Ion chamber signal decays to -7.0 decades (as shown below) as there was no other means available for Engineering to monitor Bruce A SDS1 or SDS2 ion chamber signals at this power level.

April 5 2007 Log Neutron Power indication for SDS1:

63725-RIA2D – 1.4x10-7
63725-RIA3D – 1x10-7
63725-RIA2E – 1.4x10-7
63725-RIA3E – 1x10-7
63725-RIA2F – 1.4x10-7
63725-RIA3F – 1x10-7

The readings shown above were obtained with SDS1 Shutters already jumpered open on April the 3rd in order to increase the ion chamber signals. The actual readings, therefore, would have to be 0.1-0.2 decades lower that the ones shown above if the shutters were in the closed position. There also were several moderator level changes that affected the indicated power. SUI instrumentation, however, was not installed until April 8 with several delays in the process resulting in impairment to the neutron monitoring system as well as elevated stress on the operators, engineering and control room personnel during those uncertain times.

Similarly, for the Unit 8 outage there was approximately 13 hours delay with the installation. For unit 6 the System Engineer spent 16 hours on site for SUI installation. There were also multiple issues and delays for the installation of SUI during the A731 outage.

Overall, it is estimated that positioning of SUI detectors takes at least 4-8 hours of critical time. In the year 2009 alone Bruce Power lost approximately 70 reactor-days of generation because of extensions to planned outages, partially due to SUI equipment installation and condition issues.

3.3.6 Bruce Power SUI Challenges Due to Human Performance Errors

It can be appreciated that the described earlier process of detector selection and installation/repositioning in service is quite a complex one, where input from multiple disciplines, e.g. Fuel and Physics, Engineering, Maintenance, Instrumentation &Components (I&C) department, etc., is often required several times throughout the unit outage work. Inadequate understanding of differences and specifics of Start-up Instrumentation detectors and neutron-gamma activities in the shutdown reactor core has resulted in multiple incidents and events. In 2002, SUI instrumentation was found impaired during U7 outage due to a wrong type of detector, namely ^{3}He, being selected and placed into service [16]. The following investigation identified several causes contributing to the event:

- Premature and incorrect installation of BF_3 detectors
- Inadequate knowledge of SUI in Engineering
- Inadequate knowledge of SUI in Fuel & Physics
- Inadequate understanding of procedure content
- Inadequate operating manuals and procedures
- Inadequate training on SUI

Failure of SUI instrumentation to properly respond to the removal of moderator poison on approach to criticality resulted in an emergency shutdown and return to overpoisoned state (OPGSS). This resulted in additional delays in the outage schedule, stress to the plant equipment and dose rates to the maintenance crews.

Furthermore, impairment to SUI equipment and intent function could have had a direct impact on nuclear safety, as it is believed that the SUI detectors provided an invalid indication of neutron flux from November 30 until December 24, 2002 [16].

3.3.7 Radiological Hazards of SUI Systems at Bruce Power

Installation, re-positioning and removal of SUI equipment as per current mobile scheme involve significant radiological hazards. Control maintenance personnel working in the vault have to be aware of high gamma fields present at SDS2 platform.

Another source significantly contributing to radioactive exposure is the SDS2 piping located near the Shield Tank. High gamma fields in the range of Rem/hr exist at the IC access tube openings and around the neutron flux monitors (NFM) as well, but none is as high as the gamma and neutron beams emitting from empty SDS2 ion chambers access tubes.

This was also reflected in the annual CNSC Staff Report [17], where the final collective annual doses to workers at all stations in Canada are consistently higher than the projected dose targets, particularly at Bruce A and Bruce B plants. This is attributed mainly to human factors, increase in outage scopes or durations and equipment problems. With that in mind, it can be appreciated that installation of SUI instrumentation alone contributing a dose of 1 REM per person per outage [18] at Bruce A is a significant area for improvement.

3.3.8 Conventional Hazards of SUI Equipment at Bruce Power

Some of the main conventional hazards associated with detector installation and re-positioning involve electrical hazards and chemical toxic substances.

Installation of SUI equipment in service and frequent adjustment and re-positioning of cables requires control maintenance personnel to work in tight confined spaces in the vault. This results in elevated risks of inadvertently coming into contact with terminals of high voltage power supplies in the portable equipment racks.

Boron Trifluoride (BF_3) fill-gas in the current Bruce A SUI detectors is a known severe lung, eye and skin irritant. Detectors have to be checked and the area monitored to ensure no gas leaks are present during the detector calibration phase in the shop.

CHAPTER 4: PROPOSED SOLUTION FOR BRUCE POWER SUI SYSTEM REFURBISHMENT

4.1 New SUI Detector Technology and Prototype

4.1.1 Past Experience with Refurbishment of Aged Gas-filled Detectors

Attempts to resolve the degradation and short life expectancy of neutron detectors has been an on-going effort since the outset of commercial power generation industry throughout the world. In 1966 Stokes proposed to use activated charcoal as fluorine absorbent [29], however this technique was not investigated further or used by Reuter Stokes. Mitsubishi Electric has also been working on resolving the limitations of BF_3 detectors [20, 21] in an attempt to create a detector with much higher resistance to degradation at exposures 1000 times larger than in the previous models by using activated charcoal [21].

In 1995 CANDU Owners Group (COG) SUI team made a recommendation to investigate the possibility and potential advantages of using modified BF_3 detector types that may be left in CANDU stations [22], following which various other types and models of detectors have been proposed and tested over the years as an alternative to the traditional BF_3 detectors. Imaging and Sensing Technologies (IST) BF_3, modified BF_3 detector from Reuter Stokes and N. Wood detectors were thoroughly examined at CRL laboratories to determine the extent and nature of degradation for each detector type and model [23].

To date, these attempts have been unsuccessful mainly due to the lack of a practical solution to address degradation mechanisms affecting both BF_3 and ^3He detectors resulting from detector irradiation and burn out. Short useful life time and undesirable degradation/recovery characteristics remain the main obstacles for implementation of permanent SUI equipment installation in CANDU power plants.

Given that no significant progress has been achieved to-date that allows improving life cycle characteristics of the existing BF_3 and ^3He detectors currently used at Bruce Power and age-related performance degradation, it is clear that a need for a newer approach to implementing detector technology has to be found. In the following sections a proposal will be made to implement SUI detectors via miniaturized Fission Chamber technology and exploration of this approach to overhaul Start-up Instrumentation system at Bruce Power Stations will be taken.

4.1.2 Alternative Detector Technology - Fission Chambers

In addition to the conventional gas-filled proportional counters, such as BF_3 or ^3He detectors, an alternative family of “fission-chambers” or ‘fission-counters” based detectors are currently widely used throughout the nuclear industry. Fission based counters as a method for neutron detection is not a new development and has been known and used in the nuclear industry for quite some time. Large amounts of kinetic energy from fission fragments, 168 MeV [24], released in a fission reaction inside the fission chamber can be easily detected and distinguished from gamma-noise effects and other activities. This makes it an extremely attractive option for Start-up Instrumentation detectors, as this detection mechanism is very effective even with very low count rates.

4.1.3 *Fission Counters Design and Principle of Operation*

Typically fission detectors are implemented similar to ionization chamber principles with the inner surface coated with a fissile material that produces fissions in response to incident neutrons. The products of fission reactions inside the chamber will ionize the fill gas in the manner similar to the proportional counter principle. The ion current from the fission chamber is then used to detect and measure neutronic activities in the reactor. A typical design for a fission chamber detector used in Boiling Water reactors (BWR) is shown in Figure 4.1 below. For the body of a typical fission chamber, similar to the one shown below, stainless steel is used for the walls and aluminium for the electrodes with an operating voltage in the range of 50V to 300V [1]. The operating voltage is an important factor in the selection of a fission chamber model as an increased voltage is required at higher count rates to achieve ion current saturation and prevent recombination [1].

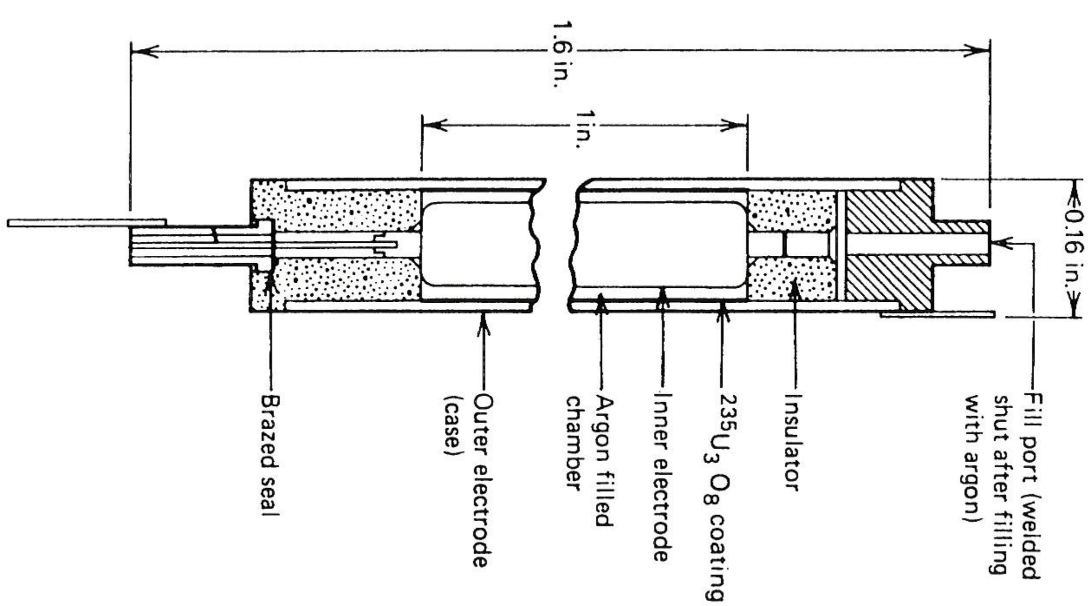

Figure 4. 1: Typical In-core Fission Chamber Used in BWR NFMS [1].

Highly enriched uranium is a common choice of fissile material. Fissile material selection initially presented some challenges as the neutron sensitive material, such as U-235, will gradually burn up, thus causing a decrease in the detector's sensitivity. An exposure to a neutron flux of about 1.7×10^{21} n/cm^2 will result in a 50% detector sensitivity decrease [25]. This limitation was subsequently solved by combining fertile and fissile material in the lining. While the original fissile material is being used in the burn up process, it will be replaced by fertile to fissile isotope conversion. Several mixture compositions have been studied over the years and the results show that no more than +/-5% change over 4.8×10^{21} n/cm^2 flux has been achieved [26] using a mixture of U-234 and U-235 as well as U-238 and Pu-239.

For typical fission chambers with a very thin fissile layer it is common to obtain the energy spectrum where light and heavy fission products are distributed according to their fission product yield curve shown in Figure 4.2 below.

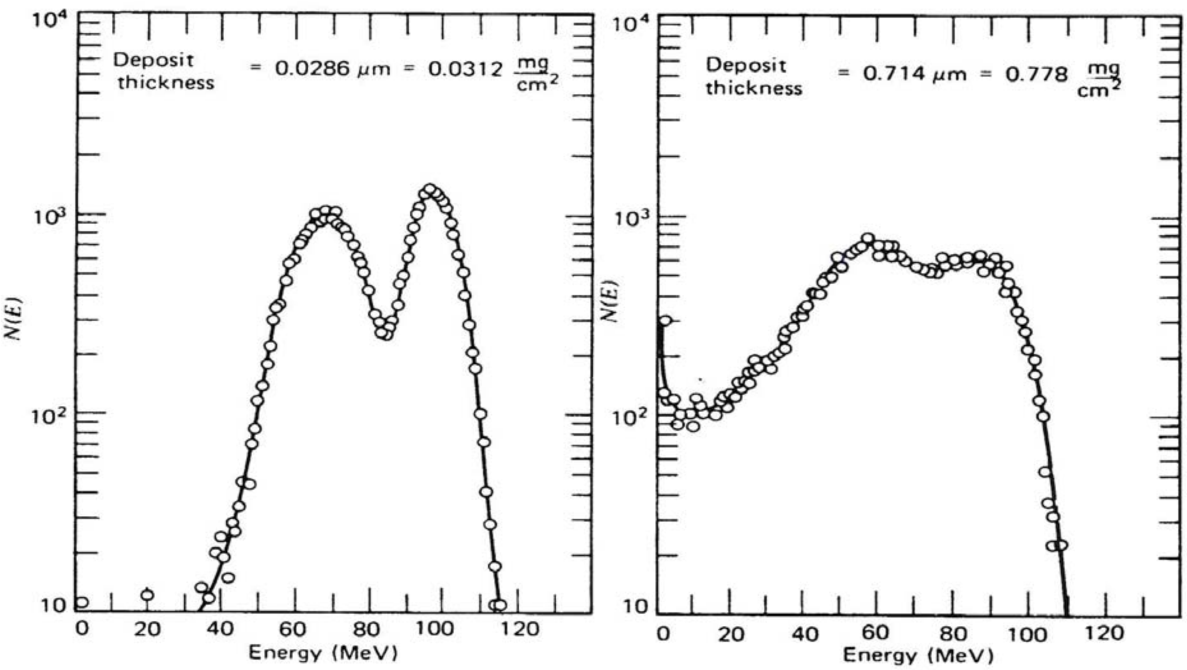

Figure 4. 2: Energy Spectra of Fission Fragments emerging From Flat UO_2 Deposits of Two Different Thicknesses [27].

As shown in Figure 4.2, changes in the thickness of UO_2 deposits produces changes in the spectrum shape. Thus, material thickness and number of layers of detector lining can be manipulated to achieve the desired detector characteristics depending on the practical application they are used for.

The two fission fragments produced in the fission reaction induced by thermal neutrons are born with kinetic energies and large positive charges, and conservation of momentum means that they move in opposite directions. As the initial charge of fragments is in the order of 15 to 20 electronic charges [1], the energy losses at the beginning of their trajectory are most significant. As the energy loss continues and the fragments slow down, more and more additional charges are picked up on the way. This may result in only one fragment reaching the active volume of the chamber. To counteract this phenomenon, various designs for fission chamber linings and backing materials have been developed, but typically a common choice for fill gas for all detector models has been Argon or Nitrogen. The detector is filled up to several atmospheres to ensure that the range of travel of fission fragments is within the dimensions of the detector.

One of the main challenges initially encountered with the fission chamber counters is the fact that the fissionable material used is typically an alpha emitter. This results in a consistent alpha particle production that cannot be eliminated or minimized. This feature, however, can be overcome quite easily since the alpha particle and the fission fragments have very distinctive energy signatures. Average alpha particle fragment typically possesses 5 MeV of energy while a typical fission fragment's energy is 10 times larger [1]. No gas multiplication is required and fission chambers can be used in pulse-counting mode. Discrimination between alphas and fission fragments is based on the pulse amplitude with subsequent arrangement of signal processing components to suppress alpha pulse pile-ups [28].

4.1.4 Industry Experience with Fission Chamber Detectors

Fission chambers are widely used in Pressurised Water Reactors (PWR), Boiling Water Reactors (BWR) at various commercial nuclear power plants throughout the world. The French Atomic Energy Commission (CEA) selected fission chamber technology for in-core , start-up, intermediate and wide power range monitoring. Today, each Areva (formerly FRAMATOME) power plant uses 5 or 6 of CFUF43 fission chambers [29].

A special wide range of CFUE32 fission chamber operating in pulse mode has been developed for Studsvik Instrument Sweden to be used during start-up mode for six ABB-ATOM BWR plants in Sweden and two in Finland. [29].

Beznau Nuclear Power Plant, Switzerland, employs LOCA and post-LOCA qualified CFUG08 wide-range fission chambers for the safety instrumented channels. To date, four detectors have been installed in Westinghouse PWR blocks to provide wide range neutron flux indications over a range of 11 decades [29].

WWER-based REKON Bohunice V1 plant in Slovakia has adapted CFUL08 fission chambers for wide-range safety instrumentation and 12 fission chambers have been installed in the safety instrumentation channels of both blocks [29].

French COGEMA fuel reprocessing plants in Marcoul and La Hague are using fission chambers as well. Fission chambers are also used for non-destructive analysis (NDA) measurements for the DUPIC fuel cycle. Safeguard verification of mixed oxide (MOX) and spent-fuel assemblies are performed by a combination of a pair of U235 fission chambers and an ion chamber in a fork-shaped holder to verify the declared burn-up [30]. A complete list of nuclear facilities where Photonis fission chamber technology is used is provided in Appendix B.

Fission chambers are also routinely used in Test Research and Training Reactors (TRTR) and medical isotope reactors such as MDS Nordion reactor at AECL Chalk River Laboratories in Canada.

CANDU 6 technical summary [31] released in June 2005 mentions two sets of triplicated fission chambers that are used to cover a very low flux range of 10^{-14} to 10^{-10} F.P. and 10^{-11} to 10^{-6} F.P. without explicitly naming the facility where this takes place. It is, however, believed to be based on Pickering A's reactor example, where signals from fission chambers are used for reactor trip function and for monitoring the reactor power on a continuous basis [32].

Another CANDU neutron detection system based on fission chambers mounted in housings on the calandria shell is proposed for the Advanced CANDU Reactor ACR-700 [33] as supplementing in-core flux detectors; however no practical implementation of this technology on the domestic CANDU market has been achieved yet.

4.2 New Detector Model Selection and Validation

There are many different types of neutron detectors widely available on the market. However, in order to be useful for Start-Up Instrumentation purposes they have to possess certain specific characteristics such as: technical specifications, size, cost efficiency and environmental qualifications.

In order to satisfy the existing technical specifications, new SUI detectors must respond to relatively low neutron counts common during routine and prolonged shutdowns and have good discriminatory capabilities to distinguish between neutron and gamma activities.

Secondly, as detectors are required for use in existing reactor cores, detector size and shape becomes an important factor. New detectors must possess small dimensions and specific geometry in order to become a viable option for existing CANDU reactors.

Cost efficiency is another important factor affecting new detector selection. Complex research-grade systems are expensive, both to obtain and maintain, particularly in harsh environments that exist in large commercial Pressurised Heavy Water (PHWR) reactors.

New SUI detectors must possess certain environmental qualifications in order to be considered as a feasible alternative for refurbishment of existing SUI systems. Prolonged detector exposure to high temperature, pressure and radiation must not impede their ability to provide stable, reliable output signals.

4.2.1 Physical Design and Geometry

Both ion chambers and fission chambers belong to a family of gas-filled chambers where the same concept of proportional counter is used. Similar to BF_3 and ^3He detectors described earlier in this paper, fission chambers are composed of coaxial electrodes where the inner electrode is the anode. Once the excitation DC supply voltage is applied, a charged particle or a gamma-ray entering the inter-electrode space ionizes the filling gas. The electrons created are collected by the anode, and the ions are attracted to the cathode, thus resulting in a current pulse proportional to the energy of incident particles. In a fission chamber the incident neutrons are captured by the converter material, e.g. U-235, which then produces ionizing particles as a result of a nuclear reaction.

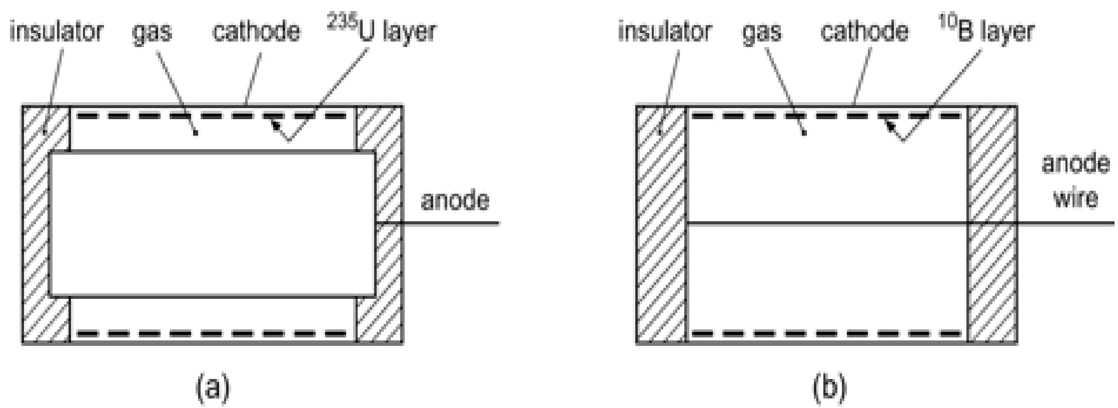

Figure 4. 3: Fission Chamber Design (a) and Boron-lined Detector Design (b) [26]

As shown in Figure 4.3, both conventional gas filled and fission chamber based detectors are designed with the same conceptual principle in mind. Modern fission chambers typically follow the same detector geometry and sizing conventions as the older BF_3 or ^3He detectors.

4.2.2 Detector Sensitivity Ranges

One of the main requirements for the proposed Bruce Power Start-Up Instrumentation is that the detector should provide accurate and reliable readings at low and very low power levels. That is another reason why U-235 lined fission chambers are proposed as an alternative solution to the replacement of ageing BF_3 and ^3He detectors that are currently in use. U-235 fission chambers possess sensitivity ranges down to -10 decades F.P. Figure 4.4 below shows reactor power ranges for thermal neutron detection in CANDU reactors, including fission chamber detector range, to illustrate how detectors based on fission chambers technology can be utilized in SUI applications in CANDU reactors.

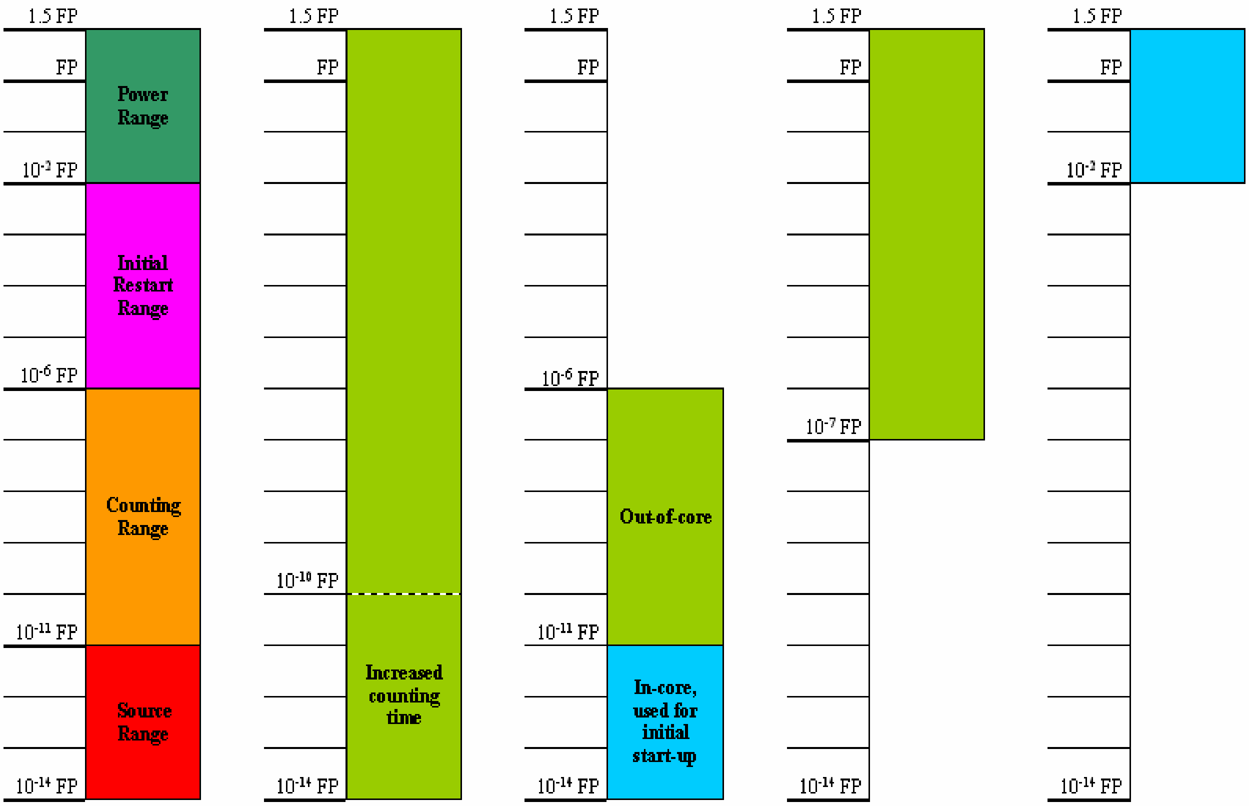

Figure 4. 4: Reactor Power Ranges for Thermal Neutron Detection in CANDU Reactors [32].

As can be noted from Figure 4.4, the expected power ranges in CANDU reactors reach as low as 10^{-14} [32] F.P during initial start-up (shown in blue). As discussed earlier, at 10^{-6} F.P. the reactor power monitoring function can be transferred to the RRS ion chambers. Thus, in order to successfully satisfy the requirements for continuous monitoring of neutron power while the RRS Ion Chambers remain off-scale, the proposed SUI detectors must provide accurate readings for the power levels ranging from 10^{-14} F.P. to 10^{-6} F.P. for both routine outage shutdowns as well as prolonged shutdowns required for unit refurbishments when fresh fuel is used for the start-up.

4.2.3 *Signal Discrimination Characteristics*

A traditional method of assessing performance of neutron detectors is through analyzing their signal discrimination curve to see how the counting rate changes in response to the discriminator threshold voltage applied to a detector in question with all other parameters being constant. The discrimination curve is also the most significant indicator that the detector is functioning correctly. A typical discrimination curve for a conventional proportional ion chamber is shown in Figure 4.5 below.

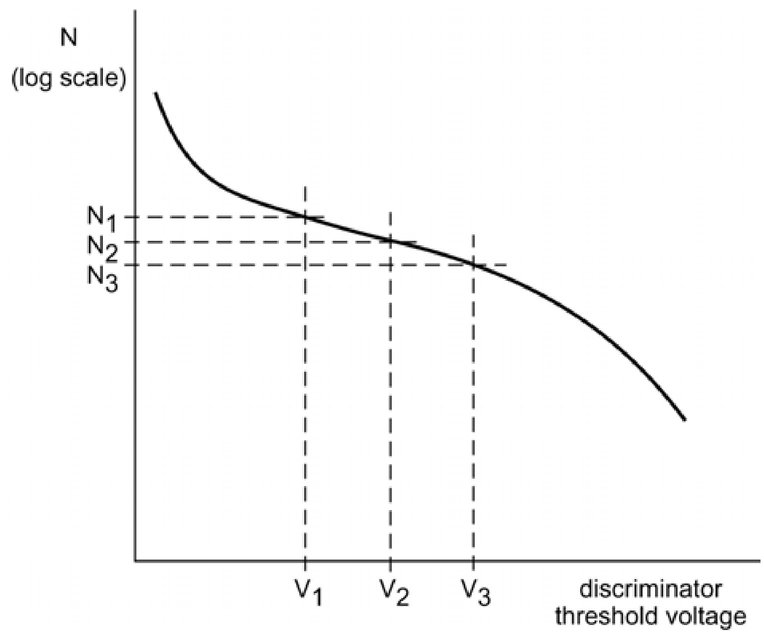

Figure 4. 5: Ion Chamber Discrimination Curve [32].

To produce this curve correctly for a conventional gas-filled ion chamber, the neutron flux should be stable and the background gamma radiation should be as low as possible. Also, the counting rate should be high enough to obtain good statistical accuracy but not exceeding the detector's dead time. From the discriminator curve, the detector sensitivity, S, can be calculated for a given threshold voltage as:

$$S\ [c\cdot s^{-1}/nv] = N\ [c\cdot s^{-1}]/\Phi[nv] \tag{4.1}$$

where nv stands for $[n\cdot cm^{-2}\cdot s^{-1}]$ [32].

The slope, P, of the curve around a specific operating threshold voltage can be described as [32]:

$$P[\%] = (N1 - N3)/N2 \times 100 \tag{4.2}$$

Monitoring changes in the curve slope will provide an indication of the fluence seen by the counter and can indicate with good accuracy the remaining life of the detector.

For fission chambers the discrimination curve can be used to determine the detector sensitivity S, for a certain threshold voltage of a determined α count. First, the "α" count rate is measured without any neutron flux present and plotted as an α curve. This represents the count rate due to the "α" activity of the uranium layer itself and the electronic background noise at the smaller threshold voltages. Next, the (α + n) counts are measured with a neutron source present and (α + n) curve is plotted as shown in Figure 4.6 below.

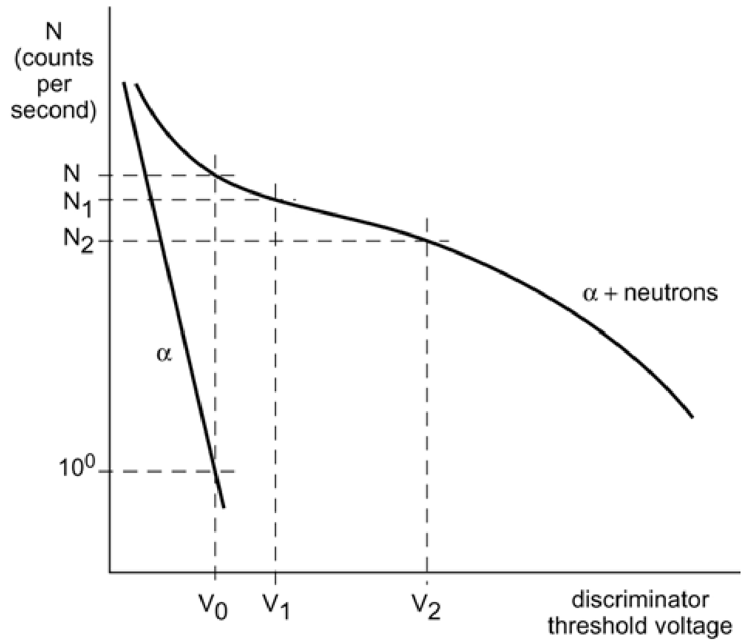

Figure 4. 6: Fission Chamber Discrimination Curve [32].

Once the discrimination curve is plotted, the detector sensitivity S, for a certain threshold voltage of a determined α count rate can be obtained as shown below:

$$S\ [c\cdot s^{-1}/nv] = N\ [c\cdot s^{-1}]\ /\Phi\ [nv] \tag{4.3}$$

where nv stands for $[n\cdot cm^{-2}\text{-}\cdot s^{-1}]$ [32]. Depending on the detector type, it is typically 1, 0.1 or 0.01 $c\cdot s^{-1}$.

4.2.4 Resistance to Gamma Radiation

For low and intermediate gamma ray intensity, ion chambers are proven to provide an accurate and consistent reading. Since gamma rays are less ionizing than the "α" and Li ions from the B-10 fission process, they result in lower pulse amplitudes easy to discriminate from pulses caused by neutrons. As the gamma dose rate increases, a 'pile-up' effect deforms the discrimination curve, resulting in less accurate count rates.

Furthermore, with the increase in total ionization ion chamber space charge phenomenon occurs resulting in reduction of the electric field and decrease of the multiplication factor. This, in its turn, leads to increase in the slope of the discrimination curve.

For a typical fission chamber, at low threshold voltages the high counting rate is mainly due to "α" activity plus some electronic background as well as some small-amplitude pulses due to gamma radiation. Once the operating voltage has reached saturation, ionisation multiplication no longer occurs. This results in the pulse amplitude being constant even if the electric field between electrodes increases.

Also, since there is no ionization due to gamma radiation, it has no effect on the amplitude of the neutron-induced pulses and the count rate. Although at extremely high gamma dose rates, the 'pile-up' effect disturbs a correct response of the chamber, it has been proved to properly operate in gamma flux of up to 104 Gy/h [32].

4.2.5 Summary of Detector Comparison Discussion

To summarize the comparison between the conventional ion chambers and fission chambers presented in the sections above, a side-to-side comparison of main key characteristics is complied and presented in Table 4.1 below.

Table 4. 1: Comparison of Ion Chamber and Fission Chamber based Flux Monitoring Systems, based 28th Annual CNS Conference materials on Fission Chambers [32].

Characteristics	Ion chamber based flux monitoring system	Fission chamber based flux monitoring system
Principle of operation	(n,α) reaction with B^{10}	Neutron induced fission of enriched U^{235}
Signal sensitivity	Signal is sensitive to thermal neutrons and gamma rays.	Designed to be sensitive to thermal neutrons and operate in high gamma radiation fields.
Sensitivity lifespan	Small decrease of sensitivity over time. Have operated satisfactorily for up to 25 years in more than 30 CANDU® reactors with no significant impact on performance.	Maintains specified performance and very small decrease in sensitivity over 40 years, through proper selection of the coating material, which is made up of fissile plus fertile material.
Post-accident application	Gamma radiation at low power levels complicates monitoring.	Designed to meet post-accident conditions.
Start-Up application	Can only be used above 10^{-7} F.P.	Provides monitoring for: 1) Start-up from fresh fuel – initial core condition. 2) Start-up after fuel channels replacement / refurbishment.
Inherent discrimination feature	Needs a lead shield to shield against gamma radiation.	1) In the counting mode, pulse amplitude discrimination is used. 2) In MSV mode, provides large amount of inherent discrimination against alpha and gamma pulses.

4.2.6 Proposed Fission Chamber Detector Model Selection

Several fission-chamber detector models were considered as an alternative to the ageing Bruce A gas-filled detectors. Three models, IST WX-33073 made by Mirion Technologies/IST, 300i Neutron Flux Monitoring System (NFMS) by Thermo Scientific, and Photonis Fission Chambers were selected as suitable candidates for replacement. Their features and characteristics will be described and compared below to determine the best model to satisfy Bruce Power Start-Up Instrumentation requirements.

4.2.6.1 *Alternative 1 - IST WX-33073 by Mirion Technologies/IST*

The first detector that was considered for this project was IST WX-33073 made by Mirion Technologies/IST. WX-33073 detectors belong to a family of miniature fission chambers designed for monitoring thermal neutron flux levels. Detector chamber (shown below in Figure 4.7) is made of stainless steel with entire inorganic high purity aluminum oxide insulation to be used in-core in any position and is able to withstand temperature ranges up to 300°C (572°F) in a wet or dry environment. The type of fissionable isotope can be selected from a variety of U-235, U-238 and Th-232 inventory depending on the application range. The WX-33073 provides neutron sensitivity of 0.7×10^{-18} a/nv to 1.7×10^{-18} a/nv and a maximum gamma sensitivity of 7.0×10^{-15} amperes/roentgen/hour [33].

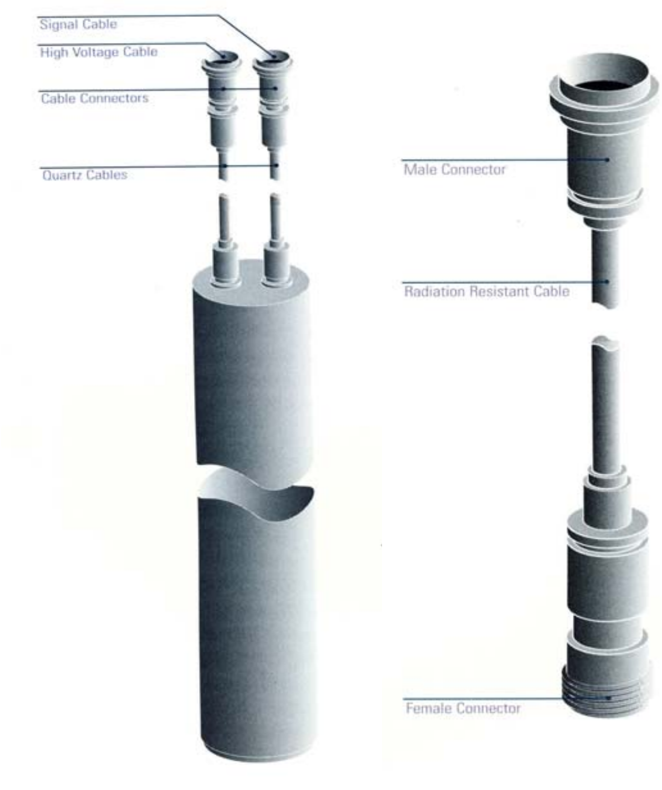

Figure 4. 7: IST Miniaturized Fission Chamber and Assembly [33].

Detector’s chamber coaxial connector cable is equipped with Inconel-600 jacket and aluminum oxide insulation [33]. The coaxial cable can be adjusted from a nominal of 29 meters (95 feet) to any length per application requirement up to 2,000 feet of continuous cable run. All cables, connectors and assemblies are made of purely inorganic materials, including insulators to provide highest degree of resistance to high temperature and high radiation. Also, ceramic to metal seals are used for cable terminations to prevent contamination of the high purity mineral insulation required to produce the assembly’s electrical resistivity.

IST’s fission chamber, connectors and assemblies have been qualified as Class 1E post-accident neutron flux monitors meeting the requirements of Regulatory Guide 1.97 [34] and IEEE 323 Standard for both LOCA (Loss of Coolant Accident) and seismic conditions [35]. The IST WX-33073 Fission Chamber Assembly can be used with specially designed pre-amplifiers, amplifiers, compensators, panel alarm meters and test circuits to implement a complete signal processing system. The main characteristics of IST WX-33073 Fission Chamber Assembly are summarized below.

Table 4. 2. Summary of main characteristics of IST WX-33073 Fission Chamber Assembly.

Parameter	Measurement/Type
Mechanical	
Chamber Diameter (max/min)	3.00 ±0.05 / 0.118 ±0.002 (mm/inch)
Cable Diameter (max/min)	1.570.05 / 0.062 ±0.002 (mm/inch)
Length: Chamber (maximum)	36 / 1.4 (mm/inch)
Sensitive Length	13 / 0.5 (mm/inch)
Connector	Standard Male BNC
Materials	
Cable Sheath	Inconel 600
Reduces Induced Radioactivity material	0.3% Mn and 0.1% Co by weight
Detector Outer Case	304 Stainless Steel
Inner Electrode	304 Stainless Stee
Detector Insulation	Al_2O_3
Cable Insulation	Al_2O_3
Neutron Sensitive Material	U_3O_8 Uranium enrichment ≥ 90% U235
Gas Fill	Argon
Ratings	
Temperature, excluding internal heating (max)	300°C / 572°F
Pressure (max)	3.5 / 50 (kg/cm2 / psig)
Thermal Neutron Flux (max)	4.0 x 1014 (nv)
Total Integrated Neutron Flux at 300°C (572°F)	3.0 x 109 (R/Hr) for 10% loss in sensitivity (min)
Typical Operation	
Thermal Neutron Flux Range	$7 \times 10^{10} \rightarrow 4.0 \times 10^{14}$ (nv)
Thermal Neutron Sensitivity (TRIGA Test Reactor)	$0.7 \rightarrow 1.7 \times 10^{-18}$ (A/nv)
Maximum Gamma Sensitivity	7.0×10^{-15} (A/R/Hr)

4.2.6.2 Alternative 2 - Thermo Scientific 300i Source Range/Intermediate Range Neutron Flux Monitoring System (NFMS)

The Thermo Scientific 300i Neutron Flux Monitoring System (NFMS) is designed with a single, 40-year-life, fission chamber-based system. 300i NFMS is a system designed to provide neutron activity monitoring from cold reactor shutdown to 200% F.P. or, in other words, for ranges more than 11 decades [36]. The 300i Neutron Flux Monitoring system is designed to operate under normal service conditions and to operate through a design basis event (DBE). It is qualified for Safety Grade Class 1E and US NRC RG 1.97 Post-Accident Monitoring applications [35]. A summary of technical specifications for 300i NMFS system is provided below.

Table 4. 3. Summary of technical specifications for Thermo Scientific 300i NMFS system.

Source Range	
Sensitivity	.20 cps/nv (thermal)
Flux Range	10^{-2} nv to 10^{4} nv
Output Range	1.0 cps to 10^{6} cps
Mechanical Specifications	
Detector Housing	152 cm (60 in) x 14.3 cm (5.625 in) O.D.
Amplifier	61 cm (24 in) x 51 cm (20 in) x 25 cm (10 in)
Temperature Specifications	
Detector Normal	.0°C to +93°C (+32°F to +200°F)

Thermo Scientific 300i NFMS system is well built with modern electronic components to provide high reliability and immunity to electromagnetic interference and noise. Its modular design allows easy component swapping and replacing for low maintenance.

4.2.6.3 *Alternative 3 - Photonis Fission Chambers Detection Instrumentation*

Photonis fission chambers are another famous example used for wide-range neutron flux control in a complete operational range of WWER-1000 and WWER-440 reactor installations. More than sixty AREVA reactors use PHOTONIS neutron detectors in their safety instrumentation channels [26]. Photonis fission chambers function in the range from 1 to 1×10^6 $n \cdot cm^{-2}/s$ for start-up applications and are designed for the detection of thermal neutrons in a state of high flux. The fissile material used in the chamber is enriched U-235 (>90%) with the typical layer thickness of 0.06 - 2 mg/cm2. A typical detector design is shown in Figure 4.8 below.

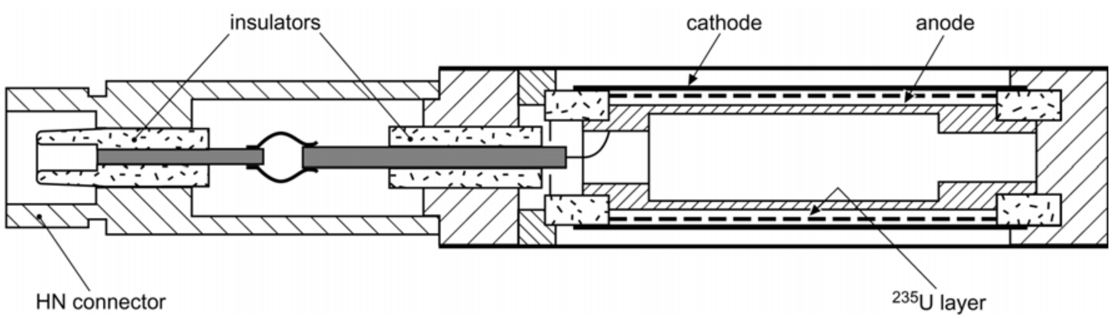

Figure 4. 8: Photonis Fission Chamber [26]

Photonis fission chambers are made of two to six high-purity aluminum concentric electrodes. For in-core detectors that are designed to operate in 250 to 600°C environment stainless steel or Inconel are used. Some types have an inner sealed subassembly protected by an external shell/housing of aluminum, stainless steel or Inconel to strengthen the construction and to provide post-LOCA protection.

Pure Argon is typically used as fill gas for ‘slow-response’ chambers used in high-temperatures up to 600°C. The composition of the fill gas can be modified by addition of 4% of Nitrogen to allow for faster response. This however, comes at the cost of reduced operating temperature of 400°C. The gas pressure varies in the range of 100-800 kPa depending on the mode of operation.

Only high purity Al_2O_3 material is used for the detector's insulation to provide the required resistance to radiation. The detector assembly comes with an integrated long non-organic mineral-insulated cable.

As can be seen from the discussion above, these detectors are well adapted for in-core measurements under very severe environmental conditions such as high temperature, high humidity and high gamma flux. As declared by the manufacturer, the chamber source life expectancy is at least 30 years [26]. A summary of Photonis detection specifications are shown in Figure 4.9 below.

Type		Unit	CFUE 32[1)]	CFUE 43	CFUF 34	CFUF 43[2)]
Neutron sensitivity	Curent mode	$A/n.cm^{-2}.s^{-1}$	10^{-16}	7×10^{-17}	-	10^{-17}
	Fluctuation mode	$A^2.Hz^{-1}/n.cm^{-2}.s^{-1}$	4×10^{-29}	3×10^{-31}	4×10^{-29}	3×10^{-31}
	Pulse mode	$c.s^{-1}/n.cm^{-2}.s^{-1}$	10^{-3}	-	10^{-3}	-
Pulse operating range		$n.cm^{-2}.s^{-1}$	10^3 - 10^8	-	10^3 - 10^9	-
Max operating T°		°C	600	500	400	350
Nominal operating voltage		V	200	150	250	150
Detector capacitance		pF	-	-	-	-
Charge collection time		ns	150	-	20	-
Structure			stainless steel	Inconel	stainless steel	stainless steel
Filling gas			Ar	Ar	$Ar+4\%N_2$	Ar
Nominal diameter		mm	7	7	4.7	4.7
Detector length	nominal	mm	150	85.5	81	86
	sensitive	mm	56	15	27	27
Integral cable			6 mm coaxial	3 mm Inconel	4 mm coaxial	1 mm coaxial
Connector			HN (female)	BNC (male)	HN (female)	BNC (male)

Figure 4. 9: Photonis Fission Chambers Detector Summary [26].

4.2.7 *Detector Selection Discussion and Results*

Based on the discussion above, it is believed that the Photonis fission chamber detector family will be best suited for implementation in upgrading and refurbishing of older Start-Up Instrumentation modules at Bruce Power. It should also be noted that the detector length and geometry was a key factor influencing the selection, since the detector will have to be installed into the existing Bruce Power reactor units. As such, CFUF-43 detector was selected as possessing high sensitivity while operating in temperature ranges up to 350° C.

4.3 Proposed New SUI System Design – Permanent In-core Installation

Potential benefits of placing SUI detectors in service once and leaving them in the reactor core include: eliminating a complicated SUI installation procedure minimizing wear and tear on SUI instrumentation and perhaps, most importantly, significantly reducing downtime and outage extensions due to SUI installation challenges and spurious trips. This in turn will greatly reduce the additional dose rates incurred by the control maintenance personnel involved in SUI installation and removal phases as well as Operation and Maintenance (O&M) costs. Outage planning, scheduling, and coordination would be greatly simplified once the necessity to install SUI instrumentation is eliminated. The control maintenance personnel involved in SUI installation and placement in service has to possess highly specialized skills and knowledge as well as field experience. These qualifications are in great demand due to so few personnel having the required training, particularly during busy shutdown schedule. With permanently installed SUI equipment, outage crews will be able to proceed to pertinent tasks and jobs without delays due to SUI detector installation and removal.

Several attempts and initiative have been made in the past to determine the feasibility of setting up SUI instrumentation permanently in reactor. These will be discussed in more detail below.

4.3.1 Past Experiences and Results

Originally Bruce Power stations were designed with the intent for SUI instrumentation based on N. Wood BF_3 detectors to be permanently installed in the core. However, degradation of SUI detectors over time became a significant factor in the decision to remove SUI in-core instrumentation and use it temporarily on as-needed basis. Laboratory degradation tests at Chalk River Laboratories (CRL) [23] showed that the standard BF_3 detectors originally used at Bruce A, as well as at most domestic CANDU stations, displayed undesirable degradation issues which could not be resolved at the time.

As a result, Centronic BF_3 detectors were chosen as the model showing the least amount of degradation [23] and subsequent changes were made to operating procedures and work management practices to incorporate installation and removal of SUI instrumentation into planned outage scopes.

By 1984, original SUI detector guide tubes and assemblies were removed from viewing ports and SDS2 spare ion chamber housing was adapted to accommodate temporary SUI detector installation in both Bruce A and Bruce B units.

Following this, a comprehensive suite of procedures, guidelines and training materials were developed for installation and removal of SUI detectors using spare SDS2 ion chamber housings at Bruce A. Control maintenance procedures and operating manuals had to be updated and modified to incorporate the specific requirements for the installation and change over procedures for SUI detectors. Although this helped to improve detector useful life time and to slow down the degradation rates, significant “side effects”, such as additional dose rates, spurious trips and outage extensions were produced.

With the new fission chambers-based detector technology, a proposal to implement permanently installed SUI detectors has become a viable alternative to the existing SUI system design at Bruce Power units.

4.3.2 Proposed New SUI System Conceptual Design

With the CFUF-43 detectors permanently installed in the reactor core, a complete new SUI system will consist of the following components:

- Source range detector assembly CFUF-43 (with integral cable)
- Junction box
- In-containment cable assembly (from junction box to inside penetration)
- Amplifier cable assembly (from outside penetration to amplifier)
- Amplifier assembly
- Optical isolator assembly
- Signal processor cables
- Source range signal processor

- Shut down margin monitor assembly
- Wall mount signal processor assembly

A conceptual block diagram for the proposed system outlining the installation in the wall, containment penetrations and connections to the portable NimBins and the MCR panels are shown in Figure 4.10 below. The physical location of the 6"Viewing Port on the Vertical Reactivity Deck is shown in Figure 4.11. This location and the Port's geometry are typical and are identical across all Bruce Power units and are ideally suited to house the new SUI detectors as they are easily accessible during unit shutdowns. It is also accessible at power, although with higher gamma and neutron dose rates than during shutdown periods. However, this is a substantial improvement to using the SDS2 horizontal platform that is not accessible at power at all. The Viewing Port plugs will be removed and the new fission counters will be lowered down the guide tube. Once the detector position and orientation in relation to fuel channels are confirmed, the port will be sealed again using compression bungs and the modified shield plugs.

The SUI instrumentation loop starts when the signal is obtained from the detector in core and is transmitted via a junction box to the amplifier assembly located in the existing Instrumentation Rooms outside the containment.

Next, the signals from the wide range amplifier are received by the rack mount signal processors located in the main control room (MCR). This panel will be dedicated to the SUI equipment and will be permanently installed in the MCR. The new Photonis neutron monitoring systems (NMS) are designed with the consideration that they will be used to replace existing Start-Up Instrumentation at older nuclear plants. Therefore, a request to Photonis will be made to ensure that the new signal processors look similar in appearance to existing equipment to minimize training of plant personnel and changes to plant operating procedures. As a result, very little interruption to the normal operating policies and procedures are expected as a result of this addition.

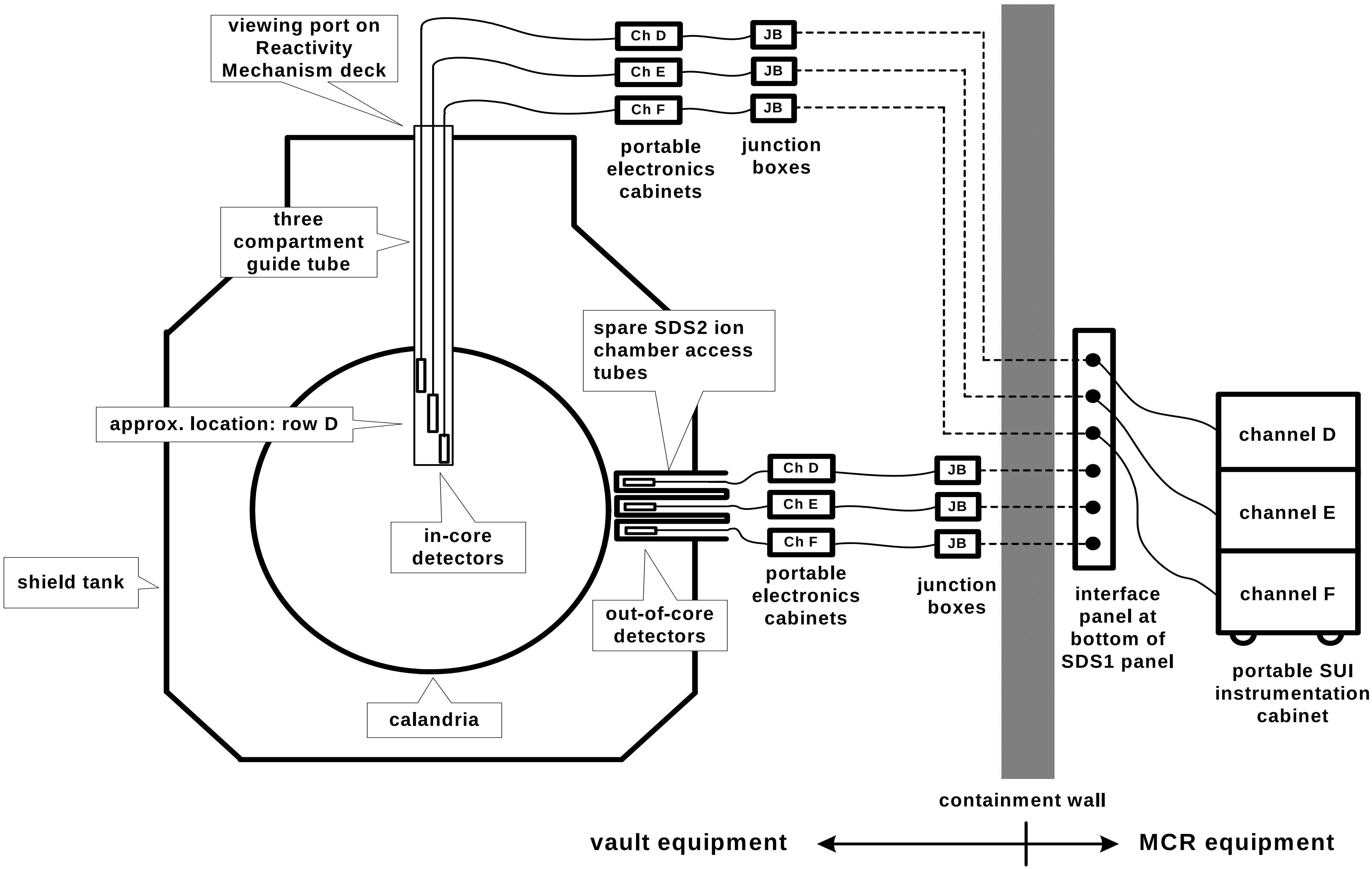

Figure 4. 10: SUI Detector Arrangement in the Vault and Connection to the Portable SUI Bin [38].

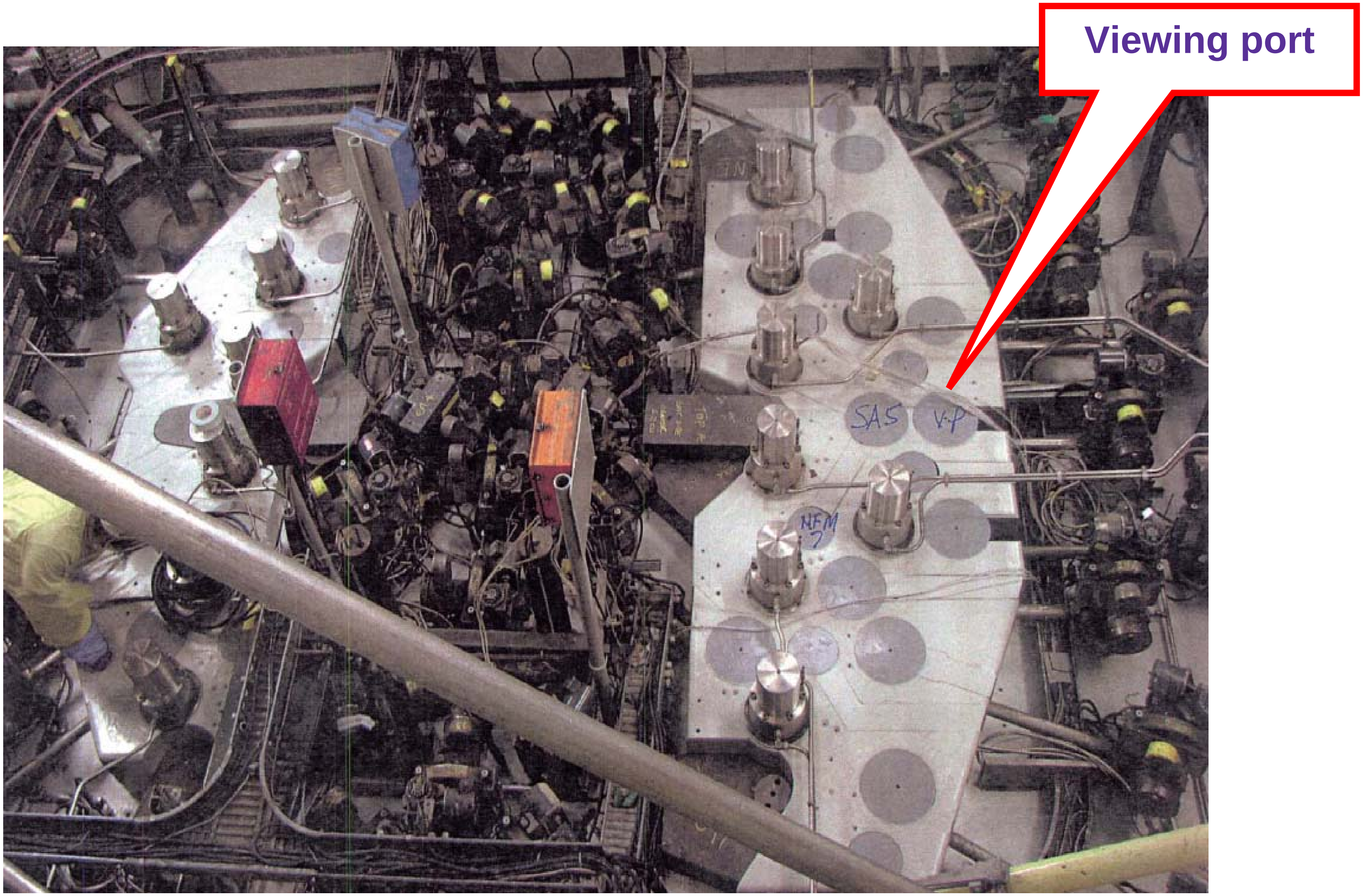

Figure 4. 11: Proposed New Location for the Permanently Installed SUI Detectors - 6" Viewing Port, Reactivity Deck. Photo taken during SDS1 walk-down for February 2010 outage and stored in the online system folder for engineering records.

Next, the MCR monitors convert the signals from the amplifiers into signals that represent the source range count rate and the rate-of-change of reactor power level. There are also alarm functions provided to alert the MCR operators when neutron activity in the core exceeds the set points. The Source Range signal processors used for Start-Up applications are designed in standard 19 inch rack mounted enclosures with LCD-digital screens and bar graph meters.

The output of the SUI MCR panel will be connected to the SDS1 trip logic on the MCR panel in the same way it is done today. The connection to the SDS trip circuits will be normally jumpered out during routine operations and will be activated only as required during outage periods.

4.3.3 Proposed SUI System Design Enhancements - Data Logging and Archiving Solution

Currently Start-Up Instrumentation at Bruce A has no dedicated tags in the Plant Information, or so-called PI system [39] to enable real time data logging. Therefore, neither SUI engineers nor control room operators have the means to store SUI data for future trending and analysis electronically. SUI counts have to be printed from the MCR via a printer and are processed and logged manually in paper format.

Should it be desired to have the SUI connected to the PI interfaces, it is done at a custom request from the engineering personnel via spare unassociated tag circuits. This requires specialized knowledge of the station DCCs (Digital Control Computers) and PI application architecture, so an experienced professional from the Computer Design Group has to be closely involved in the set up of this arrangement. It can be appreciated that this arrangement is highly undesirable and is used very often due to complexity and time constrains.

To address this limitation, particularly in the view of upcoming prolonged reactor outages for Unit 3 and 4 refurbishment projects, it is proposed to establish connections between SUI channels and the PI servers as shown in Figure 4.12 below with an additional archiving server dedicated to viewing, documenting and copying of the SUI archive data.

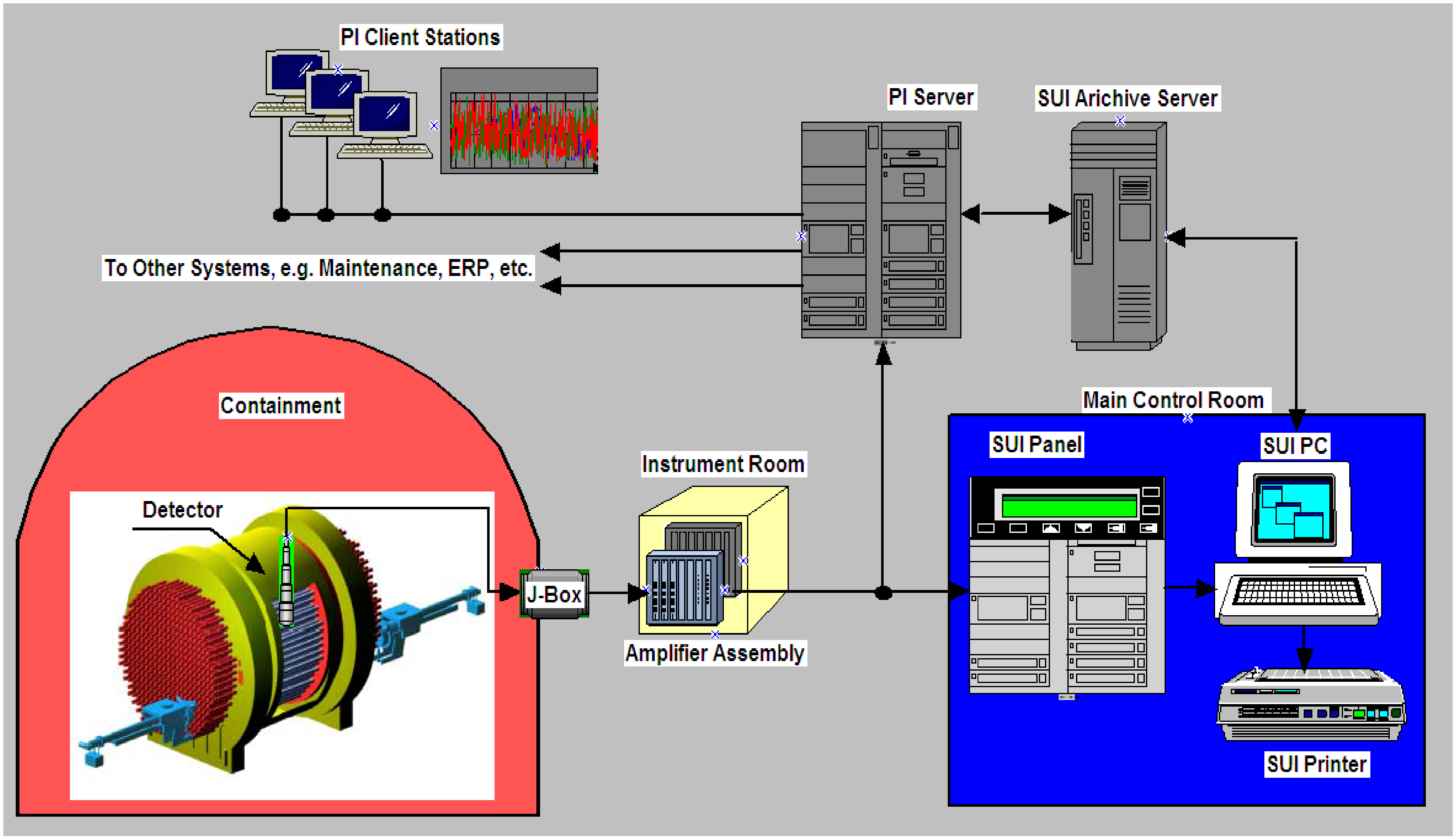

Figure 4. 12: New SUI System Including PI Server and Archive Server Connections Conceptual Block Diagram created using PI graphical module to demonstrate the proposed system architecture and enhancements.

CHAPTER 5: INSTALLATION PROCEDURE DEVELOPMENT AND OPTIMIZATION

In this chapter a risk-based installation methodology for the proposed SUI system will be developed. First, the task installation sequence will be produced and analyzed in order to identify main steps and to create mitigation/control barriers. Next, these control and mitigation barriers will be incorporated into the installation procedure in order to verify the step-by-step installation sequence against all known hazards and adverse results. Next, the installation procedure will be verified using a proposed methodology of using formal language definitions and meta-operation language. Finally, the procedure will be analysed in further detail in order to achieve maximum risk, time and cost optimization.

5.1 Activity Model for Procedure Development and Verification

In order to follow this approach, the analysis will be conducted in the following sequence:

1. Develop a step-by-step task procedure for transportation and installation steps in order to identify probable hazards.
2. Verify the procedure.
3. Develop Task Hazard Identification and Risk Assessment matrix along with the list of Preventative measures/controls.

Finally, in the last section, an attempt will be made to show that how the proposed SUI installation/replacement procedure can be optimized.

These phases are described below using IDEF0 (Integration Definition Language 0 [4] in order to model those activities formally, using a structured graphical representation of the steps rather than an informal flow-chart style models. The main stages of the framework development/verification are composed of hierarchical layers of process diagrams that gradually display increasing levels of detail describing functions and their interfaces within the system. Figure 5.1 below shows the algorithm development and optimization process with four main stages shown with their expected outcomes.

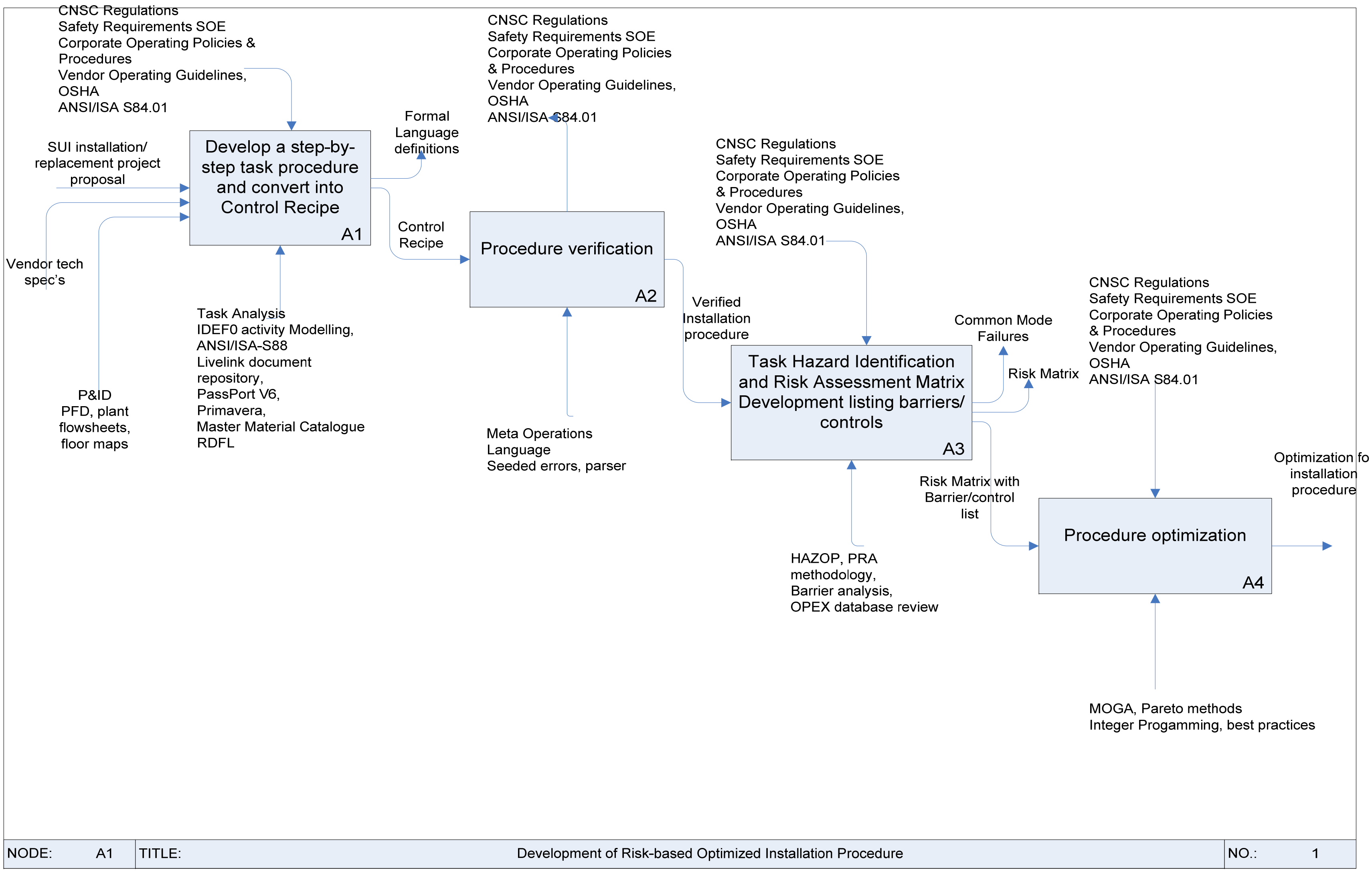

Figure 5. 1: Activity Model for Procedure Development and Verification developed using IDEF0 standard.

The first process is in the IDEF0 diagram above is “Develop a step-by-step task procedure and convert into Control recipe”. The initial step in this task is to perform the task analysis and breakdown into smaller actions/steps that can be easily represented by a sequential flow chart diagram, so that the task sequence can be converted into a formal language and the proposed procedure can be written in a control recipe format.

Next, this control recipe is sent to the second process known as “Procedure Verification”. The second process analyses the control recipe and verifies that all conditions and constraints have been satisfied as well as ensures that the procedure is structurally and logically sound and free of unknown references or conditions.

In the next step, the procedure is analyzed task by task in order to determine the potential hazards and their consequences. At this step, a Risk Matrix is created and the identified hazards are evaluated against the criteria set in the matrix. A list of barriers or control measures is also developed at this stage in order to address or minimize the adverse effects of the potential hazards.

The last process, labeled “Risk-based optimization”, uses the data produced at the previous stage in order to determine whether the proposed SUI installation/replacement procedure can be optimized to reduce the risks associated with each task. Each Process will be discussed in more detail in the following sections of the report.

5.2 Procedure Task Analysis and Breakdown

In this chapter, a portion of the proposed Bruce SUI systems replacement project will be used as a case study in order to develop a structured approach to replacement/installation procedure and automate it for future applications across any existing CANDU units.

Presently, at most nuclear power plants in Canada installation procedures and work packages with detailed task instructions are produced using the so-called "tribal knowledge". Procedures are typically put together by the most experienced person that has performed a similar job in the past. Tasks are analyzed using a traditional "barrier-analysis" method in order to identify "what can go wrong" scenarios and come up with potential mitigation solutions. Next, a procedure is written using native language in a descriptive manner, e.g. "Obtain jumper tool No.141-06-00. When using a jumper, use caution to ensure no ground faults are introduced on the circuit which could cause a loss of the 90Vdc". One can appreciate that this approach is not desirable (in general) as it largely depends on the individual's level of experience, knowledge and situational awareness. Also, for infrequently performed tasks and evolutions this approach presents a highly likely possibility of human or organizational errors since none or very limited previous personal experience exists. In addition to the challenges of following such procedures during the execution stages, the manual approach for procedure development requires significant time, effort and expertise from plant personnel which contributes to outage extensions and delays.

The other main challenges encountered in any task analysis typically fall under one of the following categories:

- The need to handle stochastic processes: e.g. work duration is best expressed by a probability distribution
- The need to allow iteration
- The need to analyze activities or processes of arbitrary complexity

In order to address the challenges described above, the installation procedure using a formal language will be developed in order to provide a structured approach that can be used for similar projects at Bruce Power or any other CANDU based stations. ANSI/ISA-S88 standards were selected to develop a hierarchical definition of task sequence procedure and to map them to plant structure. ISA standards were also chosen as a means to standardize procedure tasks and sequence across the nuclear industry worldwide. This will allow for procedure development and verification that can be easily adapted to any similar process of installation/replacement of SUI systems at Bruce Power, OPG and other CANDU stations, as well as any other type of reactor technology.

First, the overall project was divided into several distinctive stages, namely "Development", "Implementation", "Commissioning/QA", and "Plant Operation" as shown in Figure 5.2 below.

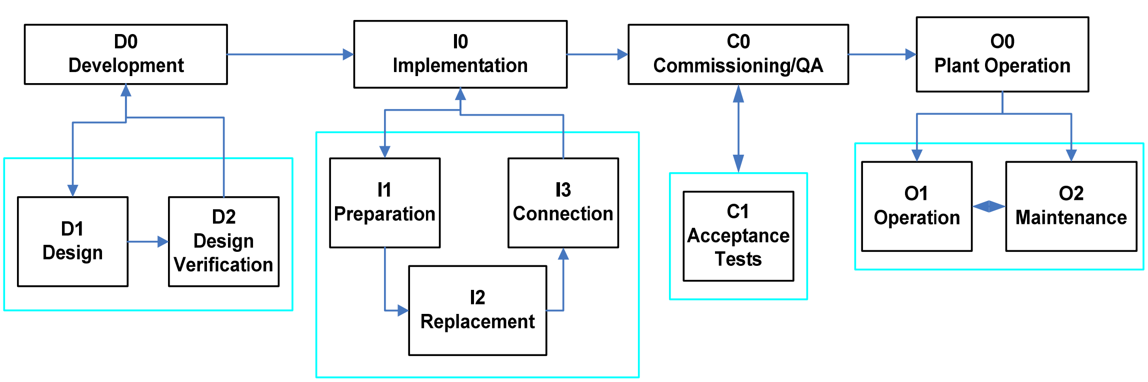

Figure 5. 2: SUI Replacement/Installation Project Phases developed to demonstrate the breakdown for procedure stages such as "D0" – Development, "I0" – Implementation, etc.

In this thesis, the focus will be made on the “Implementation” phase which involves the actual replacement/installation activities. “Development”, “Commissioning/QA”, and “Plant Operation” will not be included in this analysis.

The analysis of “Implementation” phase of the project was subsequently broken down into three functional stages – “Preparation” where materials and tools are manufactured, inspected and transported to the Reactivity Management Deck, “Replacement” where the shield plug of the Viewing Port is removed and the new SUI detectors are placed and secured in the guide tube, and “Connection” phase, where the signal cables are connected to the junction boxes and signal/data processing instrumentation.

Next, each phase was broken into sub-tasks, e.g. “Preparation” phase was divided into “In-house” and “External” based on whether the tasks are to be performed at Bruce Power facility or, alternatively, by Bruce Power personnel or by an external third party. Next, the tasks were further subdivided based on the nature of what is to be done, e.g. mechanical, electrical, I&C, transportation, etc. as shown in Figures 5.3. Each step was given a formal description using key words to describe of what is occurring in each unit procedure step in formal language and assigned a Task ID, e.g. M1.1.1 for the task of performing checks for the scaffold material. A formal language approach was chosen in order to make the task descriptions sound less complex [40] in order for the tasks to make sense to any user, whether familiar or unfamiliar, with the proposed procedure and eliminate any ambiguity or doubt as to what each individual task is supposed to be. Formal language implies absolutely accurate and precise definitions of the underling system, which can be used to validate the system/process [41].

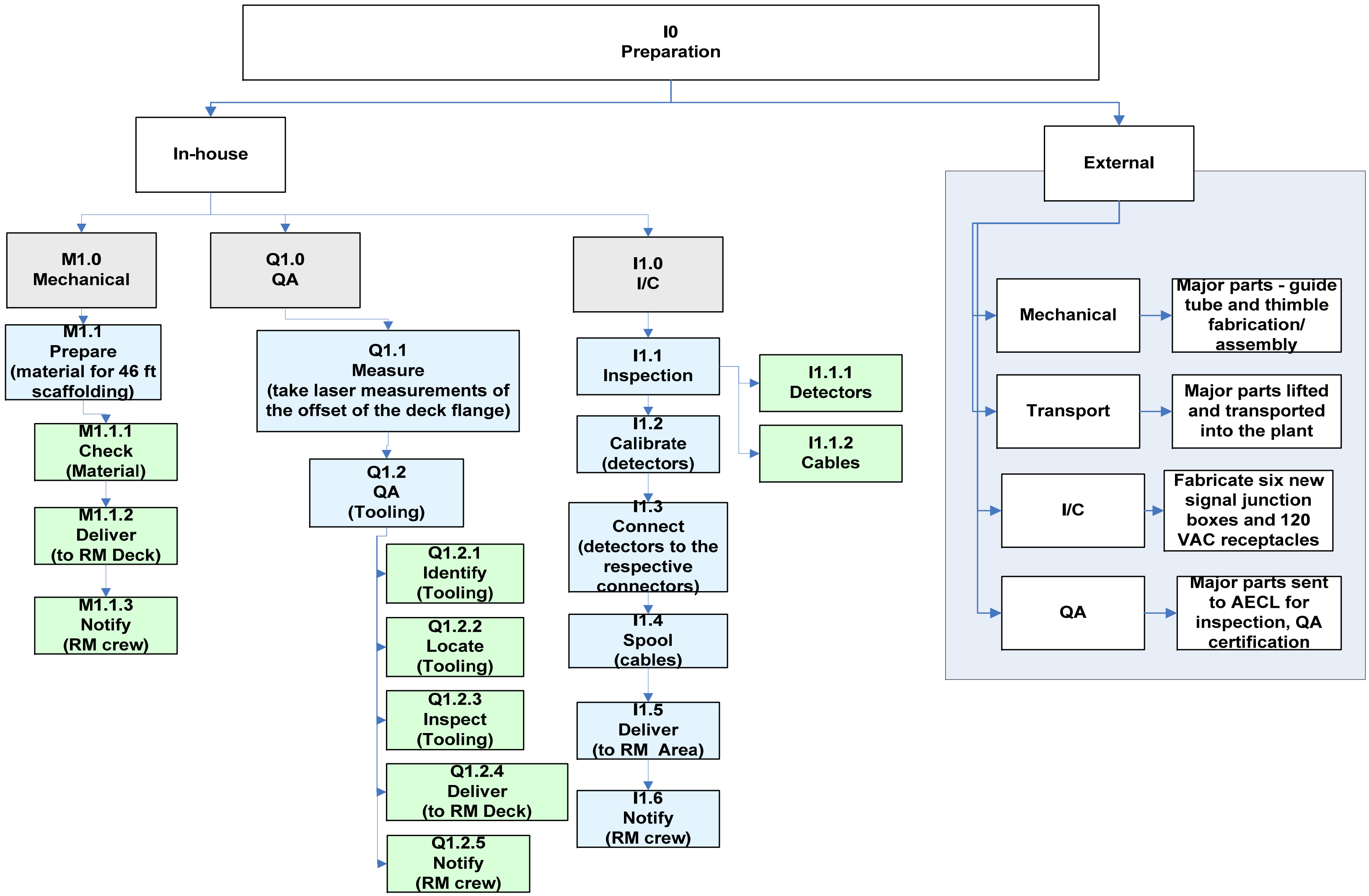

Figure 5. 3: Procedure Task Breakdown Diagram, where each task is defined using a formal key word and assigned a unique TaskID.

Next, M1.1 "Prepare" tasks were broken down into further detail and each assigned a unique identifier and definition. e.g. M1.1.2.1 "Load" for the task of loading the scaffold parts onto the delivery vehicle.

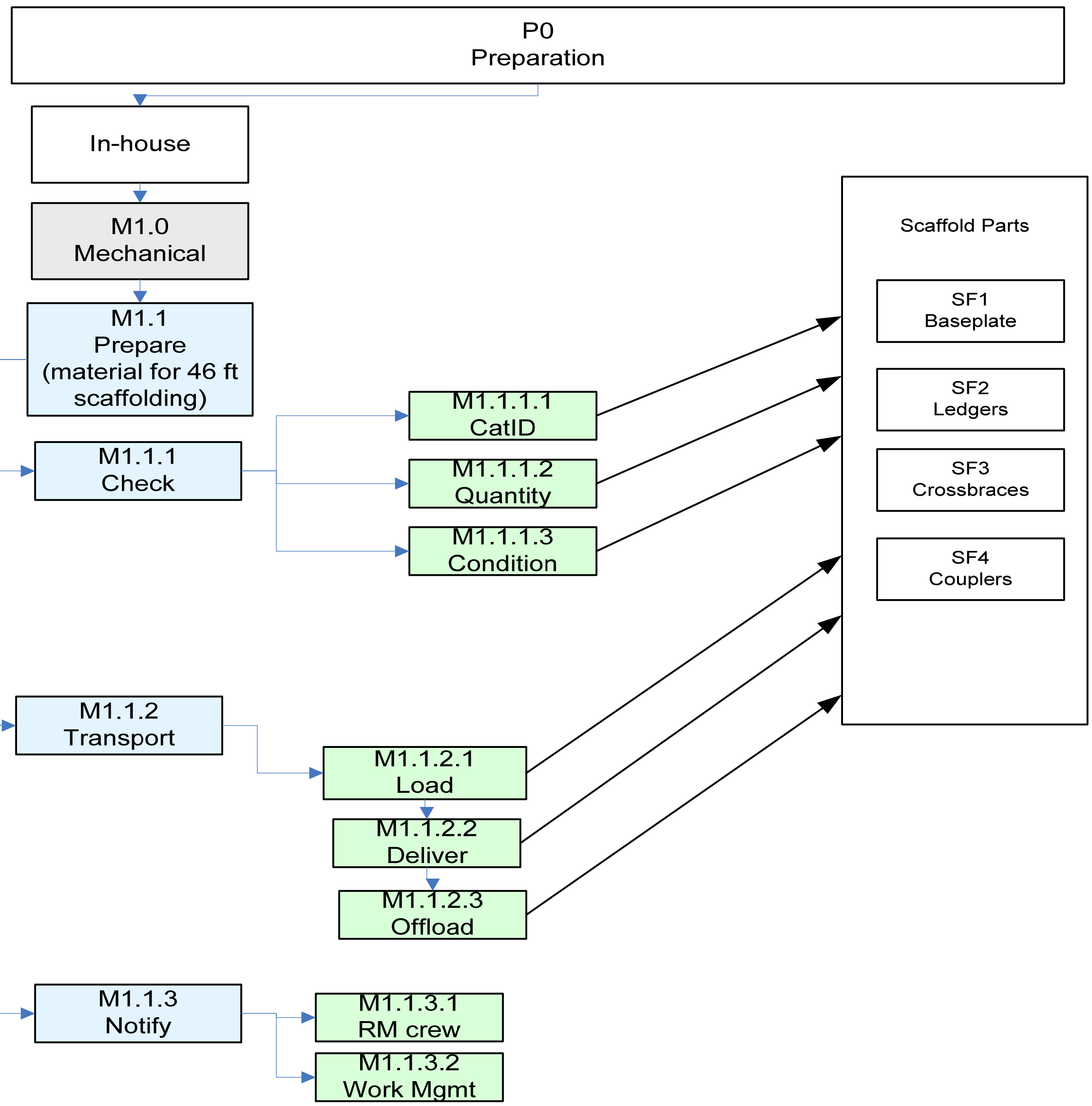

Figure 5. 4: Three levels of task analysis breakdown for In-house Mechanical task of prepare scaffold material shown in three different colors to distinguish task hierarchy.

5.3 Control Recipe Development for the Procedure Task Sequence

According to S-88.01, a recipe is defined as the necessary set of information that uniquely identifies the production requirements for a specific product [42]. For the purpose of the installation procedure, these guidelines can be adapted to formally present the steps in the installation task in a sequential manner in order to produce a formal installation recipe. The tasks have to occur in a specific sequence with the preceding tasks being successfully completed in order to proceed on to the next task.

Following this approach, a sample task of delivering scaffolding parts to the designated staging location could be described by the following sequence of task IDs:

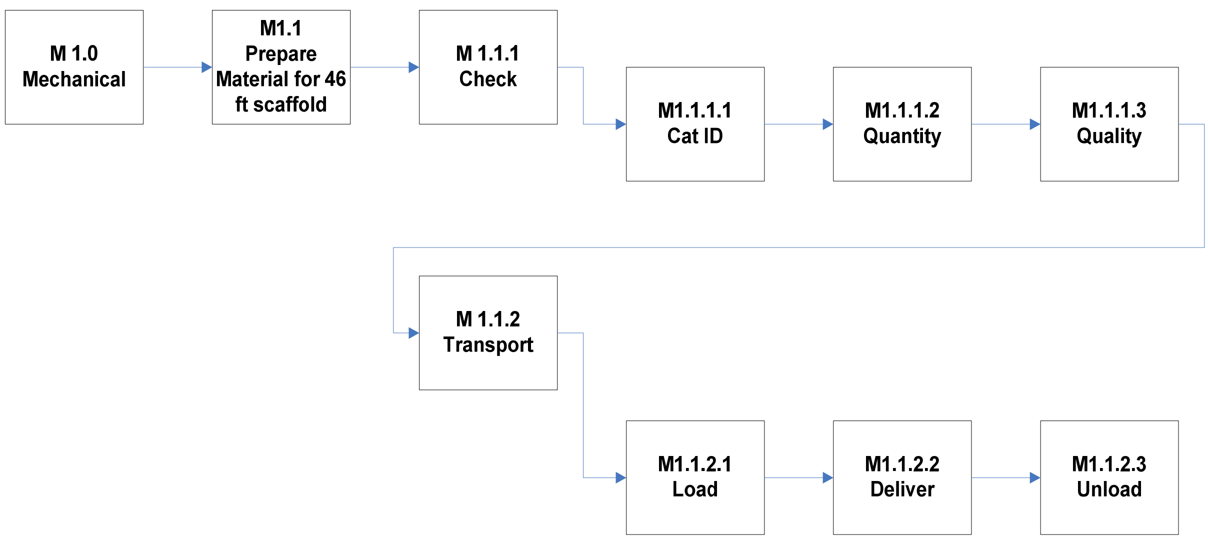

Figure 5. 5: Task sequence presentation via a number of subsequent TaskID's.

Next, this sequence of tasks can be converted into a control recipe-style list of instructions. A control recipe gives the step-by-step sequence for execution of the tasks outlined in the general procedure in a formal language rather than a descriptive manner or flow chart diagrams. The control recipe for the tasks in Figure 5.5 is created using Recipe formal definition language (RFDL) [41] under the guidelines of ISA standards in order to use a common language so that miscommunication is prevented between various operators and facilities across the nuclear industry worldwide. Therefore, the proposed RFDL formal language is written in the form of English-like statements, where each statement is composed of a sequence of keywords called tokens. This allows producing a description of what is occurring in each Unit Procedure step (UP) using clear, concise statements in a language that can be easily understood by the Engineering and Maintenance community worldwide. The functions required at each step of the procedure can be further described in terms of a control recipe language (i.e. Procedure_Action, Structure_Action, Control_Action, etc.) in order to clarify the nature of the action as well as to list the initiating and terminating conditions for each step as shown below:

- `Procedure_Action` refers to starting, restarting, or ending a function.
- `Structure_Action` refers to a specific physical task that the user has to execute, e.g. physically load scaffolding components on the transport track.
- `Control_Action` refers to the specific parameters that need to be controlled within a specified range or value, such as maximu weight that can be lifted at a time during "Load" task or the maximum speed the transporter can travel through the plan during "Deliver" task phase.
- `Init_trigger` means that step is initiated if the condition specified inside the bracket is met.
- `Term_trigger` is the termination trigger meaning the step ends if the condition in the bracket is satisfied.

- *`The token "|"`* is used to show that either the tokens before or after can be used, e.g. stop delivery if the light or horn on the transport vehicle is not functioning.

This methodology was used to develop the control recipe for the example of M1.1.2 "Transport" portion of the procedure in the following way. First, the assumption was made that any procedure will contain a description, initiation trigger, termination trigger, and action that need to be executed. Each task in will be initiated by the initiation trigger, e.g. successful completion of the previous task, and will be terminated by the termination trigger, e.g. an obstruction on the designated delivery route will result in the termination of the Delivery task. An example of RFDL – based control recipe is for M1.1.2 Transport phase and its subtasks – "Load", "Deliver", "Unload" is shown below.

Objective: Load, transport and off-load the material for 46 ft scaffold to the designated area at the Vertical reactivity Deck.

Procedure:

<u>`UP001.`</u> This step occurs first. The material received from the Stores has to be loaded on the transport vehicle according to the lifting/rigging procedures and restrictions. All components of the scaffold have to be loaded and accounted for in order to start the delivery. This step will be terminated if a wrong number of parts is identified.

```
Init_trigger(COMPLETE_RECEIVING)
Procedure_Action(START_LOAD)
Structure_Action[LOAD_MATERIAL]
Structure_Action [COUNT_MATERIAL)
Structure_Action [CONFIRM_DESTINATION)
Term_trigger (COUNT IS FALSE)
```

<u>`UP002.`</u> This step occurs after and if the LOAD action is complete and all material has been counted and loaded on the transport vehicle with no missing parts or components identified. The delivery has to be performed safely with warning lights/horn functioning to warn the by-passers of the potential danger. This step will terminate if scaffold components fall off the vehicle, or the warning system is not functioning, or there is an obstruction on the route.

```
Init_trigger (LOAD IS TRUE)
Procedure_Action (START_DELIVERY)
Structure_Action [START_ENGINE]
```

```
Structure_Action [ON_LIGHT|HORN]
Term_Trigger (ENGINE|COUNT|DESTINATION LIGHT|HORN IS FALSE)
Term_Trigger (OBSTRUCTION IS TRUE)
```

UP003. This step occurs after and if the DELIVERY action is complete and all material is delivered to the correct location. The material has to be counted to confirm that all required components have been delivered. The step will terminate if a wrong number of components have been delivered, or the delivery location was identified to be wrong.

```
Init_trigger (DELIVERY_TRUE)
Procedure_Action (START_UNLOAD_PROCEDURE)
Structure_Action [CONFIRM_LOCATION]
Structure_Action [UNLOAD_MATERIAL]
Structure_Action [COUNT_MATERIAL]
Term_Trigger (COUNT|DESTINATION IS FALSE)
```

Figure 5. 6: Sample Control Recipe for the task of transporting scaffold material developed using Recipe formal definition language (RFDL) [41] under the guidelines of ISA standards in order prevent miscommunication between various operators and facilities across the nuclear industry worldwide.

Once the control recipe for the installation procedure is established, it can be verified as shown in the next section.

5.4 Procedure Verification

As mentioned earlier, the final step in the task analysis and procedure development is verification. The accident report for Three Mile Island [43] describes "a series of events - compounded by equipment failures, inappropriate procedures, and human errors and ignorance – that escalated into the worst crisis yet experienced by the nation's nuclear power industry". Inappropriate or inadequate procedures or work coordination plans are often among the top causes contributing to failures and accidents in nuclear industry.

Although the importance of using validated, comprehensive procedures is currently well understood, at Bruce Power stations the phase of independent procedure verification is conducted through a subject matter expert review typically done by System Engineers or Maintenance Assessors. Procedure documentation is usually written in a natural language, as discussed earlier, and is submitted at various stages of completion, e.g. draft or pre-approved stage in order to identify any deficiencies or risks. The reviewer analyzes the proposed documentation for technical integrity, regulatory compliance and conventional/radiological safety aspects. The aspects of clarity and consistency as well as comprehensiveness are typically open to interpretation. For those well-familiar with the job, an additional level of detail appears unnecessary and distractive as it is increases the complexity of the document. For those not familiar or barely familiar with the proposed project, a more detailed description and a comprehensive list of precautions is deemed to be a necessary and mandatory part of the procedure. Similar to manual procedure development methodology, this greatly relies on the person's experience, skills and knowledge of the subject and often results in time delays and cost increase even at this early stage of the proposed project. Also, since it greatly relies on human factors, this approach is difficult to capture and quantify, so that it could be repeated for a similar procedure or work coordination plan again at a different facility. This is particularly important since the proposed task analysis and installation/replacement for obsolete SUI detectors is meant to be used for any subsequent projects at other facilities. Therefore, a structured approach to procedure verification, as well as task analysis, is needed to ensure consistency and ease of transparency for future installations.

5.4.1 Procedure Verification Methodology

The ultimate goal of procedure verification is to confirm that following the proposed procedure the intended tasks or objectives are achieved as expected. Another main goal of procedure verification is to eliminate inconsistencies or structural errors when the steps of the proposed procedure are written out of order or no hold/back out conditions are provided for all possible scenarios.

For the selected case study of the proposed SUI installation/replacement procedure and its associated tasks, a meta-operation language similar to “PROMELA” process meta-language developed by the formal methods and verification group at Bell Laboratories [44] will be used in this thesis. The main advantage of the Meta-operations language method is that it is well suited to describe actions and sequences of actions in a chronological manner, as well as conditions that allow or forbid advancing to the next step, until a clearly stated hold point or back-out condition is specified. This will be demonstrated in more detail below, however it is important to point out that this approach is proposed as a general methodology and will not be explored further into the script and parser development.

First, the proposed installation procedure or work coordination package can be converted into Meta-operation language programs using the same formal-language task ID’s developed earlier. For example, for tasks in M1.1.2 “Transport”, actions such as ‘Load” or “Deliver” can be easily converted into a processes object, e.g. LOAD. In Meta-operation language processes are global objects that represent the concurrent entities of the system with the behavior of a process defined by a “*proctype”* declaration. For example, the following declares a process type "LOAD" with a Boolean variable stating "True" that corresponds to a successful completion of the task:

```
proctype LOAD()
{
bool state;
state = True;
}
```

Since the scaffolding material for 46 ft scaffold construction will have to be loaded in sequence, one component after another, further sub-tasks in "LOAD" process can be represented through a loop structure with the number of repetitions set to the number of the components to be loaded:

```
do
:: count = count + 1
:: scaf = scaf +1
:: (count == 0) -> goto done
od
done:
skip;
```

Next, we can write the code to represent a "proceed" condition, i.e. once the task of "LOAD" is completed and is "true", the user can proceed to the next task of "DELIVER":

```
if
:: (LOAD == true) -> DELIVER;
:: else ->; fallthrough_option;
If
```

Following this logic, the entire work coordination package or procedure could be converted into a meta-language in order to verify process interactions. Depending of the level of verification that is required, the model can be expanded into the required number of levels and verified with different type of assumptions or risks, e.g. for the task ID M1.1.3.1 “Notify”, used for notification of RM crew members that the scaffold material has been delivered to the correct location, it could be verified against the risks that the message will get lost. Parsing the model against this assumption, or the so-called seeded error, will ensure that a correct barrier is thought of and build into the proposed procedure, in this case ensuring that there is a message delivery confirmation mechanism specified in the procedure. Since condition can only be executed (successfully passed) when it holds, it will be blocked and paused until an alternative step is provided. For the purpose of the procedure verification, this translates into a hold point or back-out actions clearly stated for the future user, outlining the steps they need to take in case this condition cannot be met, e.g. place the “LOAD” task on hold and contact supervisor if the number of scaffold parts received from store is less than stated in the procedure. This way, personnel that will be executing the procedure in the future won’t have to deal with the uncertainty of what direction to take if faced with the situation for which the procedure does not provide clear instructions.

Once the correctness of a sample procedure or work coordination package has been verified, the proposed installation procedure can be analyzed in terms of the potential hazards and their associated consequences in an effort to develop and implement a list of control measures or barriers to the known and anticipated hazards and, finally, to optimize the installation procedure.

5.5 Risk Matrix Development and Criteria

In order to perform the proposed design modifications and installation of new SUI detectors and electronics at Bruce Power, a task analysis of the major project steps, outlined in the previous section, was performed with the main intent of identifying potential hazards and risks associated with each project phase.

The following definitions were used in order to identify and assess the risk associated with the installation procedure:

- Hazard – something that has the potential to cause harm
- Probability – the chance of occurrence of an unwanted event
- Consequence – the extent of harm or severity of event
- Risk – probability of occurrence multiplied by consequence of occurrence.

In order to quantify the risk associated with the proposed installation procedure, certain ranking has to be assigned to probability of occurrence as well as severity of consequences. Although it is fairly easy to estimate the severity of consequence, the probability of occurrence of unwanted events or conditions has to be roughly estimated based on previous knowledge and the overall complexity of the project, rather than real data as no current procedures exist and the proposed installation project is the first of its kind.

Shown below is the ranking table for magnitude of consequences with the highest ranking assigned to the most significant outcome – “Catastrophic” event resulting in multiple human fatalities.

Table 5. 1: Ranking for Magnitude of Consequences shown, where the highest ranking of 10 is assigned to the most significant consequence.

Consequences	Ranking
Catastrophic (multiple fatalities, complete failure of the project)	10
Disaster (a few fatalities, significant challenges to the project).	8
Accident (one fatality, lost time/production challenges).	6
Event (serious injury, serious damage to equipment, outage extensions).	4
Incident (no injuries, damage to plant, serious schedule interruptions)	2
Near miss (no consequence, lessons learned)	1

It is important to note that other impacts, both real and potential, have to be considered in this analysis in addition to human casualties (conventional safety). For example, impact to nuclear safety, environmental safety, radiation safety, reputation, regulatory compliance, production, cost, schedule, outage, etc. resulting from unwanted or adverse impact of the installation procedure may cause a serious financial or regulatory damage to the generating station and have an adverse negative impact for the rest of the nuclear industry. However, for the purpose of this assessment and following a conservative decision making approach generally adopted for the nuclear industry in Canada, the magnitude of consequences are ranked based primarily on their effect on conventional safety, i.e. human casualties or injuries.

Similarly, the probability of occurrence of an adverse condition or failure during the installation procedure can be ranked as shown in Table 5.2 below, with the highest ranking of 10 assigned to the most probable occurrence.

Table 5. 2: Ranking Probability of occurrence is shown, where the highest ranking of 10 is assigned to the events that are most likely or "expected" to happen.

Probability	Ranking
Most likely and expected	10
Quite possible (50/50).	8
Unusual but possible.	6
Remotely possible (has happened somewhere).	4
Unlikely	2
Practically impossible	1

5.6 Barrier Analysis and Development of Mitigating Measures/Controls

Following the ranking criteria in the previous section, the project step M 1.1.2 "Prepare material for 46 ft scaffold"- "Transport" was analyzed to determine barriers/control actions required at each step in order to mitigate the known and anticipated hazards. This is shown as an iterative process in Figure 5.7, where the output of the Task Hazard Analysis process is used for Barrier Analysis. This will allow identifying an appropriate barrier, whether procedural or physical, for each task during at the Barrier Analysis stage. Next, the output result of barrier analysis will be used as a feedback-input for the Task Analysis process in order to use this information for the task analysis stage. This iteration enables a maximum procedure optimization where mitigating actions or controls are identified and included during the procedure development stages for maximum risk optimization.

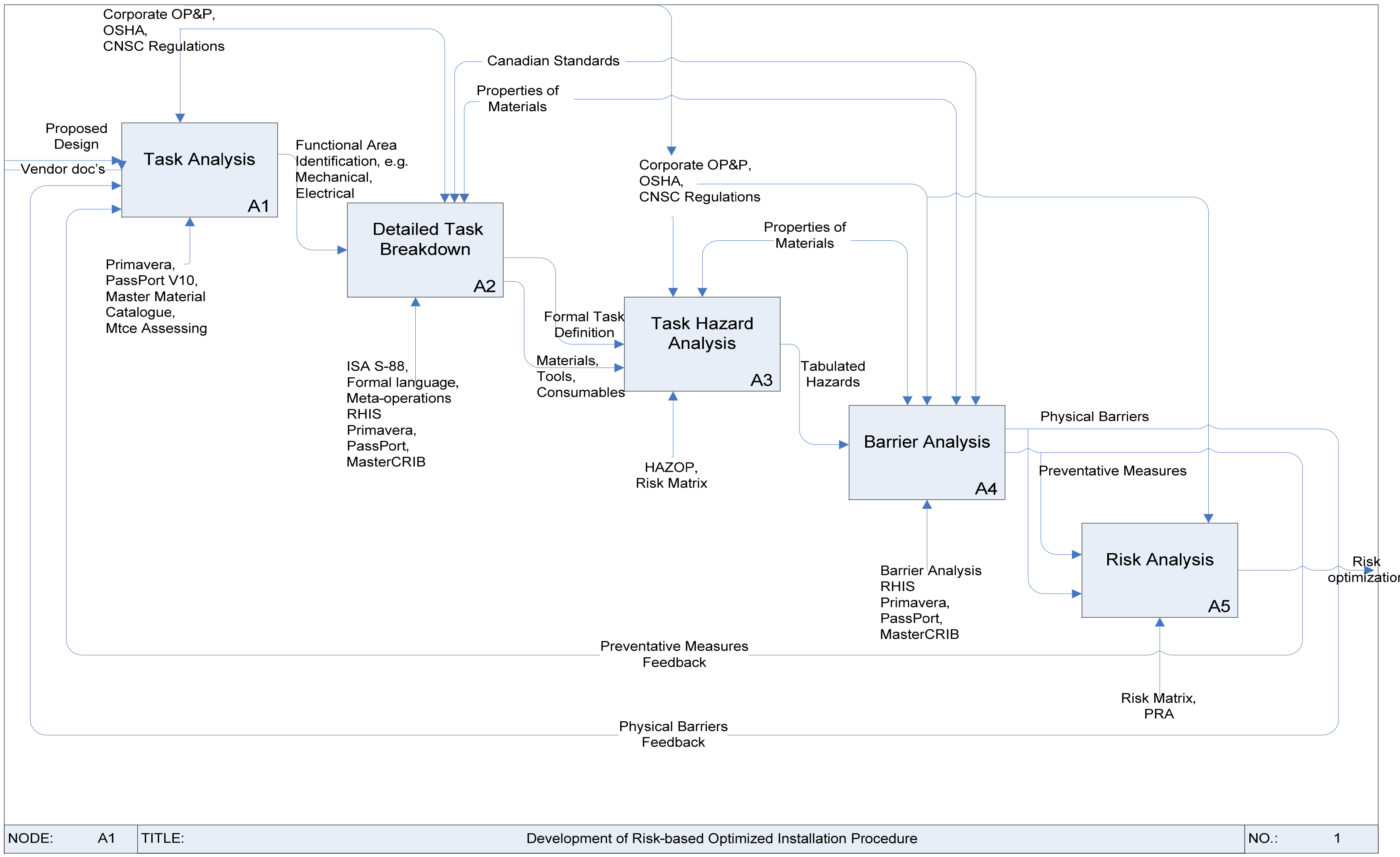

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Figure 5. 7: Procedure optimization using feedback from Barrier Analysis and Risk Analysis process as input to the Task Analysis stage, developed using IDEF0 standard.

For example, for the task of loading of scaffolding material onto the transport vehicle, one of the main expected hazards is bodily damage to the crews due to scaffold components falling off the transport or lifting mechanism. The main preventative measure procedural barrier for this hazard is strict adherence to the lifting/rigging procedure written under the guidelines of OHSA. The main preventative physical barrier is the use of Personal Protective Equipment, such as safety footwear, by the loading personnel and the use of brackets/restrains to keep the parts properly locked in place.

5.7 Tabulated Risk Assessment Matrix and Analysis Results

As discussed in section 4.6, the results of hazard analysis will be used in the next step of the process in order to develop a task risk matrix. The product of probability of risk and magnitude of possible consequences will be used to quantify the projected risk of each task. For this particular example, a sample risk analysis was conducted for the task of transporting scaffold material from the load bay to the Reactivity Deck where it will be assembled. The results of risk assessment for the task of transporting, including the estimated probability of occurrence as well as the magnitude of consequence of the adverse conditions produced by each hazard are shown in Table 5.3 below. During this analysis the highest score in terms of risk was determined to be due to a high probability that the scaffold components may cause bodily damage during the loading/unloading operation with the consequences reaching as high as 4 in ranking, i.e. a chance of causing a serious injury or serious damage to equipment. Similarly to Barrier Analysis process, the results of risk assessment stage will be used as feedback-input into the procedure development process where the identified risks can be minimized for maximum procedure optimization.

Table 5. 3: A sample table showing hazards, their estimated probabilities, consequences, and risks are shown for the task of loading the scaffold material M1.1.2.1.

Process ID	Process Desc	Activity	Hazard	Cons	Prob	Risk	Controls
M 1.0	Mechanical						
M 1.1	Prepare material for 46 ft scaffold						
M 1.1.2		Failure to deliver correct scaffold material to RM deck (Qty/Qlty)					
			Damage to components due to falling/improper handling	2	8	16	Use correct procedure for lifting/rigging/stacking. Ensure personnel receive correct training.
			Damage to transport vehicle	2	6	12	Use correct procedure for lifting/rigging/stacking. Ensure personnel receive correct training.
			Bodily injuries due to weight of the components	4	10	40	Wear protective clothing - gloves, hardhats, safety footware, googles. Use correct procedure for lifting/rigging. Conduct training for proper lifting/rigging techniques
			Transport not available	2	8	16	Schedule verification and confirmation
			Personnel not available or wrong number of people	2	8	16	Verification for resource planning and loading. Arrangement to be made to have addt'l on-call personnel available
			Lifting/rigging equipment not available	2	6	12	Equipment to be obtained from Stores and staged prior to execution
			Procedure missing	2	4	8	Job Assessment and walk-down prior to execution
M 1.1.2.1		Load	Personnel not trained for the job	4	6	24	Personnel qualifications reviewed prior to assignment

5.8 Installation/Replacement Procedure Optimization

In this section, the proposed SUI installation/replacement procedure will be further analyzed to determine whether it can be optimized for even better results. In general, optimization is the process of selecting the best available alternative or modifying the proposed solution in order to take maximum advantage of the resources available, minimize the costs and reduce the anticipated risk. The analysis of these three factors will be given in the following sections with the recommendations as to the most efficient solution to adopt.

5.8.1 Risk-based Procedure Optimization

Once the formal work coordination plan or “procedure’ has been developed and verified and hazards and risks associated with each step have been determined, it is possible to optimize the algorithm in order to minimize or eliminate those risks. This will be described in this chapter.

First, let’s consider the “Preparation” tasks, where the material required for 46ft scaffold has to be prepared. The fist step in this process would be the selection of the scaffold parts as shown below:

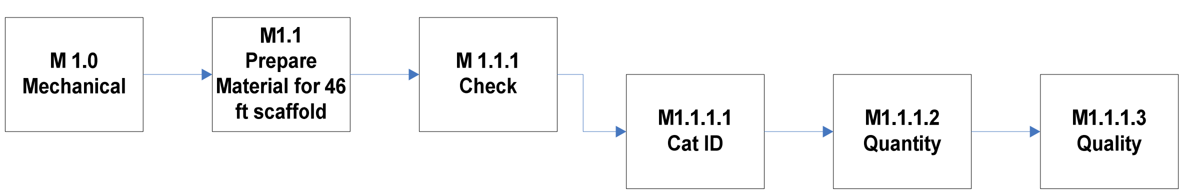

Figure 5. 8: Breakdown of the M1.1 task for scaffold material preparation, broken down in to further sub-tasks of M1.1.1 “Check”, M1.1.1.1 “CatID”, etc.

This process can be presented in a Cause-Failure Mode-Consequence format, as shown below:

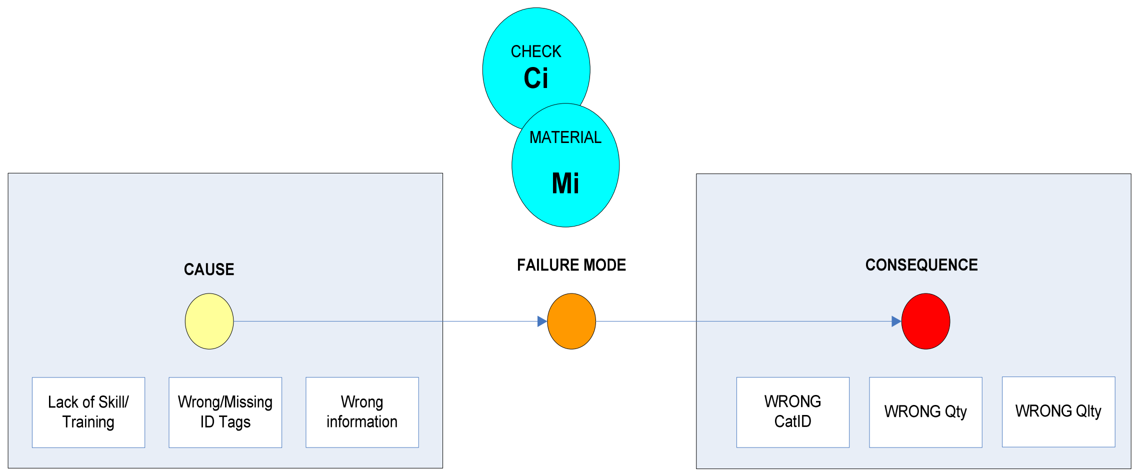

Figure 5. 9: Cause-Failure Analysis for M1.1 Task of preparation of scaffold material, showing the relation between potential causes and consequences, e.g. Lack of Skills/Training may lead to selection of wrong quantity, quality or material or wrong material type (i.e. wrong CatID).

Here, for the task of checking the material three main causes are shown. For example, for lack of skill/training of the personnel retrieving the material from the stores would result in either wrong CatID, or wrong quantity being selected. It also could result in the wrong quality of material being selected, e.g. if the correct CatID is picked up from a scrap location where the discarded scaffold parts are store until the time they could be shipped off the site. Thus an expression:

$$Ci*Mi \tag{5.1}$$

can be used to represent the product of causes (for *Check* and *Material* respectively) contributing to the task failures, while

$$Cj*Mj \tag{5.2}$$

can be used to present the product of consequences to check the correct material.

Next a set of related State Variables (SV) can be developed as shown below:

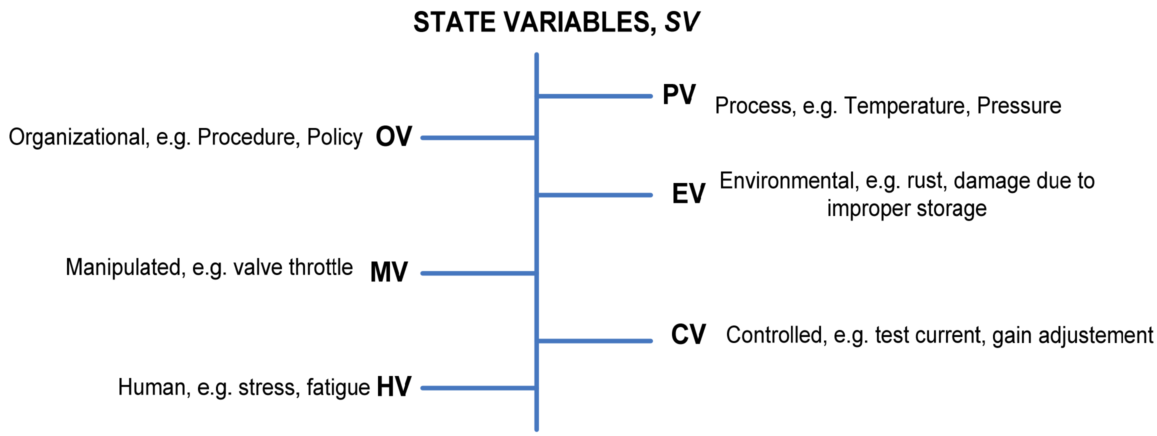

Figure 5. 10: State Variable Tree, where the possible State Variables are shown, e.g. Process Variable (PV) may include Temperature, Pressure, etc. Similarly, Human Variable (HV) may include such factors as physiological or psychological stress, fatigue, etc.

Next, it is possible to define the list of *SV's* related to the selected task "Check-Material" and to determine the possible deviations, or Δ_i, which will result in possible Risk or R_i. This is shown below:

$$SV_i + \Delta_i \rightarrow R_i + \Delta R_i \qquad (5.3)$$

where Δ_i coefficient varies depending on the task scenario.

Thus, for the same task of checking the material depending on how the task is performed the deviation Δ_i could be quite different. To illustrate that, three different alternatives for this task are described below.

First alternative consists of designated Stores (plant wear-house) personnel performing the material retrieval and verification. The Stores personnel are trained and qualified for this task, which reduces the possibility that they would select wrong CatID or quantity of the material. The main constraint for this alternative is that although the

Stores are open 24x7, the Stores personnel are only present during the day shifts and only on Mon-Fri schedule. Therefore, should the task of checking the material be scheduled for after-hours, this job will have to be performed by the scaffolding crews who may not be qualified or experienced to do so. This option, however, carries the least amount of additional cost as the existing Stores personnel are already trained and qualified for the job and possess the most experience with the Master Material Catalog and storage location.

The second alternative is to assign the task of checking the material to the scaffold personnel and to provide a pre-job training so that they would have the formal qualification and knowledge to perform this task. This would eliminate the constraints of the first alternative, namely the need to perform this task during Mon-Fri business hours. This alternative, however, would have the associated risk of assigning an important step to the personnel who has no experience with the task and received formal training only. Since the scaffold crews are not normally admitted into the Stores warehouse area, this may result in a significant time delay to the task and increases the chance that the wrong material or not enough material will be picked up. Also, this option carries the additional costs associated with the training material, instructor's fees and facility booking as well as time required to complete the training.

The third alternative is to outsource this task to the trade's union and have them deliver the scaffold parts to the loading bay. This alternative will eliminate the need to train Bruce Power personnel or to manipulate the project schedule to coordinate between the two groups. This, however, will have its own risk of not having the control over the quality of the scaffolding parts, e.g. amount of stress, rust, cracking, life expectancy and storage conditions. Also, this alternative will introduce the element of risk in terms of delivery time and possible delays due to road conditions, labor strikes or adherence to schedule by non-Bruce Power employees. This option carries the highest costs as the union and contractors' fees will likely need to be added to the project budget.

This is shown in Table 5.7 below, where the highest ranking of 3 is assigned to the factor that has the most significant effect on the risk.

Table 5. 4: Definition of SV deviations defined for three alternatives to task M1.1

Alt	**Description**	**Hazard**	**Cause**	**Consequence**	**Dev cost** Δ_1	**Dev time** Δ_2
1	Task performed by the Stores Personnel	task scheduled during after-hours	Lack of coordination	material not available OR wrong Qty/Qlty is selected	1	2
2	Task performed by the scaffold crew	lack of experience/familiarity with the storage area	Lack of training/experience	material not available OR wrong Qty/Qlty is selected	2	2
3	Task performed by an external contractor	risk to schedule adherence, quality	no control or oversight over the external contractor	material not available OR wrong Qty/Qlty is selected OR time delay	3	3

As can be seen from the table above, several Δ_i's have been estimated to have the same value, thus this approach faces the challenge of deciding between conflicting criteria. The next step of this analysis is, therefore, developing an optimization algorithm where both deviations, in terms of time and cost, are considered. Since it is desirable to keep both to the minimum in order to minimize the risks of the project, the next step is to determine the alternative with the minimum cost and time. This will be addressed by using an algorithm similar to the MOGA – Multi Objective Genetic Algorithm method, typically used for the cases where the mathematical description of performance criteria are used for cases when they are usually in conflict which each other [45].

The approach used for this analysis begins with defying the Pareto set solutions x_i [46] for a feasible region *F:*

$$x_0, x_1, x_2 \in F \tag{5.4}$$

The Pareto optimal solution in the minimization problem will be x_0 with the following criteria satisfied:

if $f(x_1)$ is said to be greater than $f(x_2)$, i.e. $f_i(x_1) \geq f_i(x_2), \forall\ i = 1, 2, ..., n$ (5.5)

and

$$f_i(x_1) > f_i(x_2), \exists\ i = 1, 2, ..., n\ , \tag{5.6}$$

then

x_1 is said to be dominated by x_2

therefore, if there is no $x \in F$ such that x dominates x_0, then x_0 is the Pareto optimal solution. Next, a weighted sum can be used to combine multiple objectives into single objective [16] as such:

$$f(x) = w_1 f_1(x) + w_2 f_2(x) + \cdots + w_n f_n(x) \tag{5.7}$$

For this case study, the weights are the ranking that is assigned to each option's Δ_i, e.g. the ranking of 3 is the highest and corresponds to the worst, i.e. having the most impact, on the risk of the project. Also, numerically for this case study there are two objective functions that are being considered, namely minimal cost and minimal time:

$$f(x) = w_1 \cdot f_1(x) + w_2 \cdot f_2(x) \tag{5.8}$$

or

$$f(x) = \Delta_1 \cdot \cos t + \Delta_2 \cdot time \tag{5.9}$$

where:

$f_1(x) = \text{cost}$, $f_2(x) = \text{time}$

The goal of the solution is to achieve the condition when *f(x)* is minimal. The optimal solution will, in this case, occur when both Δ_i are minimal, thus contributing minimal deviations to the estimated project risks. It is also the most desirable in terms of project planning and execution.

To illustrate that, a case where the lest-costly alternative is achieved at the expense of a significant additional time is not desirable as it involves other costs, e.g. facility use costs or additional costs of storage. Similarly, the alternative option 3, where the time is minimized to the least possible value but at the cost of additional budget that is required, as it is typical in a case of contracting the tasks out to an external organization, the deviation to the project risk is significantly increased due to the need for additional budget. This decision, regarding the tradeoff between cost and time deviations becomes significantly more complicated once the number of tasks goes up as well as other constraints, e.g. investor confidence and delays to project schedule, are taken into consideration.

This methodology could be further explored with MOGA-Multivariable Object Genetic Algorithm [47] for analysis of the entire project with multiple stages and options for each alternative.

5.8.2 Project Time Allocation and Optimization

One of the main challenge for resource allocation and optimization that was not discussed in the previous section is due to the fact that a very few work activities are performed in a constant or predictable amount of time. The risk minimization strategy above assumed that a certain fixed amount of time is required to perform a specific task, e.g. a task of loading the scaffold component onto the transport will *always* take 10 minutes.

In reality, the same task can vary in duration every time it is performed, even by the same individual. Currently, the time allocation for task execution, particularly for a new procedure, is done subjectively, based on the assessor's personal experience or level of knowledge of what is required in order to complete the task. It also highly depends on the overall work schedule and personnel availability, e.g. during planned outages it is commonly anticipated that more resources are available because of the increase in the number of external contractors, trades-people and other personnel. This common practice routinely results in situations where a significantly lower number of work hours are allocated to a particular project. The tasks of pre-job briefing, checking the drawings, verifying the correct tools and consumables are routinely omitted from the estimate. As a result, the time allocated to the project tasks is typically assigned arbitrarily and inconsistently.

Therefore, it is required that this gap be addressed to show that there might be more benefit to using a consistent methodology for time allocation for project tasks. A method of statistical analysis of the previous operating experience data was used on the case of the sample task of loading the components of the 46ft scaffold onto the transport truck, which is one of important steps in the proposed procedure.

The analysis of the previous operating experience showed that the same task of lifting and rigging a similar component was performed with the duration ranging from 0.3 min to 13.8 min as shown below:

Table 5. 5: Historical data for the duration of loading the scaffold components ranging from 0.3 to 13.8 minutes.

Task Duration (min)	Frequency
0.30	1.00
1.00	3.00
2.20	2.50
3.50	7.00
4.80	12.00
6.10	18.00
7.40	20.00
8.60	18.00
9.90	6.00
11.20	6.00
12.50	5.00
13.80	2.50
15.00	0.00

Plotting this numbers shows that this closely follows a normal distribution pattern:

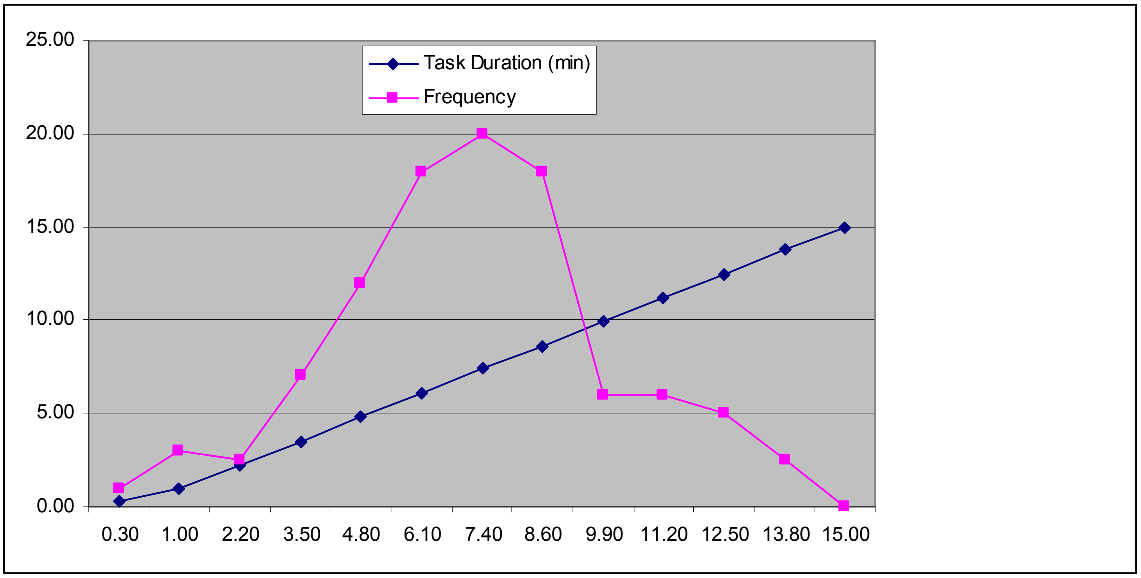

Figure 5. 11: Plotting statistical data shows that the task duration follows a normal distribution.

Therefore, the shape of this curve can be approximated by the "Gaussian" probability distribution.

$$p(t) = \frac{1}{\sigma\sqrt{2\pi}} e^{-\left(t-\bar{t}\right)^2 / 2\sigma^2} \tag{5.10}$$

Next, σ and t can be determined from the table of the data collected during past operating experience review:

$$t = \frac{\sum_{t=1}^{n} t_i}{n} \qquad \text{and} \qquad \sigma = \sqrt{\frac{\sum_{t=1}^{n} \left(t - \bar{t}\right)^2}{n-1}} \tag{5.11}$$

Further defining as T_{50}: the time interval point when 50% of the work is done and T_{90} as the time interval point when 90% of the work is done would reveal the following:

$$\bar{t} = T_{50} \qquad \sigma = 1.188,415...\left(T_{90} - T_{50}\right) \tag{5.12}$$

Therefore, a more realistic estimate of the time required to complete this task for the proposed procedure can be derived based on the previous statistic, rather than arbitrary "guesstimating" this value.

For the selected task of loading the scaffold material onto the truck, the following assumptions were made. One crew member, labeled as Joe, loading a part of a scaffold material is considered for this example. The scaffold component is handed to the crew member at a rate of λ = 1 part /10 minutes = 0.1/min. When Joe receives the part, he has to examine the part to determine its physical condition. Joe also has to check the serial number on the part and cross-reference it with the list of components that he has been given during the pre-job brief. Once the component passes the two acceptance criteria above, Joe loads it onto the truck. This process of inspection and verification takes on average 9 min (as shown earlier, it can range from 0.3 to 13.8 min based on the previous experience). This can be represented as "service time" and expressed as μ = 1 permit/9 minutes = 0.111…/min

The average job duration from the queuing theory is:

$$r = 1/(\mu - \lambda) = 1/(0.111 - 0.1) = 90 \text{ min} \tag{5.13}$$

Running more simulation data we get the following:

Table 5. 6: Average job duration versus number of components.

Number of Parts	Avg. Job Duration (min)	Error (%)
10	27.2	69.8
100	85.6	4.9
1000	88.5	4.7

This analysis can be further fine-tuned until the optimum forecasted job duration can be determined with an acceptable error and the time allocation for execution of the procedure tasks can be optimized for maximum efficiency so that that over-allocation of time, resulting in project extensions and cost increase, can be prevented.

Another main advantage of applying a formal approach to the time allocation process is that automated solutions for time allocation can be developed based on one of many formal methods, similar to the automated resource planning solutions used in the airline industry. In addition to reducing the human factors, such as the assessor's familiarity with the job or level of understanding of the complete task sequence, it can also reduce the time during the project planning stage as well as minimize the room for error.

5.8.3 Resource Allocation and Optimization

As discussed earlier, the proposed SUI installation/replacement procedure can be described as a dynamic, real-time process or system since the personnel executing the task sequence will be operating in a continuously changing environment. As shown during the task hazard analysis stage, there is a significant probability of workflow interruptions due to various reasons, both equipment and human-factor related. An example of this could be that a wrong number of personnel are assigned to the job, or there may be enough physical bodies to perform the job, but the operators/maintainers may not have the right training or skill-set to perform the job in a timely fashion if at all.

With all of this in consideration, the projected workload during this stage will fluctuate, thus challenging the resources allocated for the job, which in its turn results in an additional risk factor challenging the successful execution of the procedure. Therefore, the resource allocation for this project should be robust enough in order to mitigate those potential adverse conditions but at the same time should attempt to keep the allocated resources to a reasonable minimum in order to comply with the allocated budget and availability.

Some of the questions in resource allocation process are how the resource should be assigned. For example, should the resource be assigned based on one of the following:

- First available
- Lowest cost
- Best fit (skill-set and training)

Currently, the practice of resource allocation for projects at Bruce Power is conducted in a conventional way, where the need for resource loading is "guesstimated" by the task assessors based on their level of familiarity or understanding of the job in question. In order to optimize this process, similar to the project time allocation, a formal method can be adopted in order to ensure that the optimal balance between the job duration and the resource loading is achieved. To illustrate that, the same example of loading the components of the scaffold onto the transport track can be used. The proposed methodology for this case study is loosely based on the Integer Programming model used for resource allocation. It is commonly used for sales resource allocation [49], or transit fleet resource management. The Integer Programming approach is well-suited for these tasks as it allows trying all the alternatives in terms of adding resources, reordering or combined tasks, or using different resources which is routinely done due to personnel unavailability. It also takes into the account that often the procedure tasks get reassigned to others, equally trained and capable resources, if the originally assigned personnel is needed to work on another, higher priority job.

Previously, as described in the previous chapter, there was one technician “Joe” loading the scaffold components. For the case when one more resource is “loaded”, i.e. assigned to the same tasks, the following can be derived from queuing theory:

average job duration $r = 9.866{,}824$ min.

Running a simulation will reveal:

Table 5. 7 Average job duration versus the number of parts analysis for the case when an additional person is assigned to perform the same task.

Number of Parts	Avg. Job Duration (min)	Error (%)
10	9.17	7.1
100	9.52	3.5
1000	9.72	1.5

This exercise can be repeated until the optimum number of crew members can be selected in order to optimize the task duration with the required degree of accuracy. Similar to time allocation solutions, an automated resource loading solution can be developed in order to assign optimal personnel numbers while running the optimization algorithms for the accuracy of the qualifications, experience, industrial safety regulations, collective agreements and government regulations.

5.8.4 *Project Budget Allocation and Optimization*

Refurbishment of older CANDU nuclear plants requires billions of dollars in capital with no immediate "visible" return on investment. Over the course of years since Bruce Power Unit 1 and 2 Restart project, there has been a consistent tendency for project cost and budget overruns. An important lesson that was learned from these challenges is that for any future refurbishment project an accurate and effective budget allocation and optimization of cash flow is a key to the project success.

The selected case study of installation/refurbishment of the SUI instrumentation for Bruce Power units is estimated to be in the range of $1.5 M. Although this number may appear high, it is worth noticing that the budget for such a project includes not only the costs of purchasing the new hardware, i.e. detectors, cables, servers, etc, but also the cost of engineering work, such as design and verification, installation, commissioning, QA assurance, as well as other costs associated with human and facility resources, maintenance costs, tools and equipment needed for the project, etc. Currently, as is the case with the Unit 1 and 2 Restart project, the project budget has to be drawn from the company's investors and partners resources. Based on the history of the Restart activities to date, it does not appear that the financial resources are forecasted and allocated in a systematic manner. Therefore, as part of the project preparation stage, a better budget analysis and optimization approach may prove to be more effective.

With a number of large, medium and small scale projects happening simultaneously, the investors have a wide variety of options to choose from. Assuming that c_j is the contribution resulting from the j- th investment and that a_{ij} is the amount of resource i , such as cash or manpower, used for the j th investment, the problem can be defined as the need to maximize

$$\sum_{j=1}^{n} c_j x_j \tag{5.14}$$

that is subject to:

$$\sum_{j=1}^{n} a_{ij} x_j \leq b_i \tag{5.15}$$

where (i=1, 2,…m) $x_j = 0$ or 1 (j=1, 2,…m)

The objective is to maximize total contribution from all investments without exceeding the limited availability b_i of any resource. This, however, is a much bigger job and is typically performed by Projects or Investor Relations department.

Another main challenge during a project of this magnitude stems from the multi-phase nature of the tasks and their sequencing. It typically involves a lot of coordination between various work groups in order to align the resource loading, equipment and facility use and plant status conditions. This results in extended time lines for the project completion. Historically, this resulted in a situation where the project budget was released in portions, either based on a fixed schedule of completion progress. This can be mathematically described as

$$\sum_{j=1}^{n} a_{ij} x_i \leq b_i \tag{5.16}$$

to show the incremental balance in each time period. Here the coefficients a_{ij} represent the net flow of finances from investor j in time period i. If the project requires additional budget in period i , then $a_{ij} > 0$. It is important to note that a_{ij} is always going to be greater than zero, i.e. $a_{ij} < 0$, since there is no expectation that the investment will generate immediate profit. Coefficients b_i in this equation represents the incremental exogenous cash flows. If additional budget is provided during time period i , then $b_i > 0$. Similar to a_{ij} coefficient, it is reasonable to assume that the project budget will not be recalled or reduced, thus bi can never be less than 0, i.e. $b_i < 0$. Further analysis can be conducted detailing down to the task level and automated for a more thorough investigation or for better planning. This, however, will not be discussed in this paper as it represents a very specialized area of finance and accounting operations.

What is important is that there are tools and techniques available on the market that would allow determining and optimizing the project budget to be within 10% of the final value. This is done through a formal task analysis approach with procedure standardization, verification, optimization of resources and effective time allocation as discussed earlier.

CHAPTER 6: HIERARCHICAL CONTROL CHART METHODOLOGY FOR FUTURE SUI SYSTEM MAINTENANCE AND TROUBLESHOOTING

6.1 The Need for a New Methodology

In today's industry, in order to successfully design and operate plant and systems, engineers and designers are required to have right tools for analysis of plant layout, physical interconnections, functions and control hierarchy, which requires knowledge of all measured and control variables as well as determination of all of the components, processes and their relation.

Typically, in order to describe the way a system operates, system engineer's designers use text documents and process flow diagrams [49]. This might work for designing a small brand new plant with no existing process or control systems in service. For a case of refurbishing or design modifications to the existing ageing or obsolete nuclear reactor units, such as Bruce Power SUI systems, this presents significant challenges. The existing paper-based or microfiche drawings are routinely outdated, with many plant modifications that take place over the years not being reflected. There are components and parts found in the plant that have incorrect specifications, that cannot be found on the associated drawings or simply have no documentation at all.

Another challenge system engineers have to deal with on a daily basis is the fact that the existing drawings are of poor quality, often bearing hand-made comments or remarks and exhibit soiled, faded or worn-out ink. Although efforts are being made to digitize the Bruce Power drawing library, there is still a large number of drawings that exist only on microfiche cards. Engineers have to rely on older microfiche viewers and printers to print out a paper copy, which often is too blurry or too dark or otherwise barely readable. There are also a very limited number of microfiche stations. Confounded by a long warm up time, unreliable performance, missing or jammed cards, and a long manual search through the card drawers makes this system quite an obstacle during troubleshooting of faults. This results in additional delays to critical path and increases the levels of frustration and stress during troubleshooting process.

On the plant drawings that do exist electronically or in paper format, individual system components are shown with equipment labels, e.g. 3-63743-LT2D-MV2D, which provides only basic indication of what the component specifications are. Although it is possible to deduct that MV2D stands for a motorized valve, no further information regarding the make/model or technical specifications, i.e. flow range or temperature cannot be determined from the drawing along. Information on component availability, if required for replacement, cannot be determined either and requires an additional manual search in the plant master material catalogue.

Another major limitation of the existing paper-based design tools is that the existing old system drawings are very specific where the system under consideration is often shown on its own, with no identified interconnections or process flow lines to other systems. In order to verify system interconnections, a design engineer has to physically trace the line on the microfiche reader screen or paper drawing through a series of valves, pumps and tanks carefully paying attention to their functions and states and the direction of process flow.

Operational flow sheets, on the other hand, show the process connections but provide no detail information on the system components. Neither type of sheet has any indication of the type of information exchange, e.g. sensor data sent or control command issued, and is meant to show only material or energy exchange. This may work well for process systems but becomes a major limitation for control or I/C systems, where data processing and information exchange functions are just as important as terminal locations or line bus numbers.

6.2 Existing Troubleshooting Tools and Their Limitations

There are a few products currently available on the market that are designed to aid designers, engineers and system maintainers in getting access to the required data electronically. Although all of them have their own advantages, there is currently no tool that can provide the capabilities of the proposed Hierarchical Control Chart (HCC).

6.2.1 Open Text ECM Suite Tools

Open Text ECM Suite [50] has developed a search technology that incorporates full-text indexing and string-search technology through their Livelink application. A user is able to search for a specific item of interest, e.g. JB902 (junction box 902) in alpha-numeric format. The Livelink search engine is designed so that any record, e.g. purchase order, bill of material, design or operating manual referencing JB902 will be pulled from the document repository and made available for the end user. This is achieved through Open Text Federated Query Server which provides a single access point to query multiple repositories to obtain a unified set of results, enabling instant access to information from across the enterprise [50].

The main limitation of this method is that a user must search for a component by manually entering its name into the search engine. A vast list of all associated documentation will be produced in textual .pdf format. Next, the user has to manually search for the required information by opening and searching through each of these documents until the required data is found. This method is ideal for augmenting paper-based design process but is very time and labor consuming for system troubleshooting and diagnostics.

6.2.2 ECM Documentum Tools

ECM Documentum introduced its Electronic Document Management System (EDMS), which is a client-server product for electronic document management, in 1993 [51]. This product managed access to unstructured information stored within a shared repository, running on a central server. It included an integrated full-text search engine for retrieving documents from the repository. Although Documentum has proved to be a very successful solution for document management, the process of information search and retrieval is similar to the one described for Livelink. Drawings and blueprints can be retrieved both in .pdf and various CAD formats for authorized users, but no interactive information retrieval function is available.

6.2.3 Ventyx PassPort V10 Asset Suite

Ventyx PassPort V10 Asset Suite [52] is another popular solution for information management at CANDU power plants. Asset Suite enables users to gain access to stored electronic documents and drawing as well as check the existing parts inventory and equipment maintenance history. This application, however, still does not address the need for a fast instantaneous access to the information on a specific component or connector. In an example of obtaining temperature ratings on a specific connector for the proposed SUI system, in order to find this data, a user has to find the number of the system wiring diagram first. This number, e.g. NK21-EGAN- 63176, for the proposed system will be searched for in the PassPort application in order to determine that: a) such a drawing exists, and b) the revision number for the latest master copy. Next the latest approved master copy of the drawing is selected and retrieved from LiveLink and viewed in Brava! Server. The retrieved drawing is a scanned .pdf version of an old manually paper-based drawing produced by hand in 1973. It can be appreciated that in addition to poor readability, this drawing does not provide much information about the connector under investigation other than an equipment tag name.

Next, in order to access its specifications it is necessary to go access a different module of passport and do Equipment/Component header search. Next, the required component is selected and searched to determine its Catalog ID (CatID). This, however, will only provide system engineers with a unique numerical tag for this connector with no technical specifications, such as temperature rating or restrictions. In order to obtain that data, system engineers have to contact personnel in the Procurement Engineering department. Based on the CatID provided, they will be able to search for the manufacturer's specifications document for that particular component through their own electronic material catalogues or paper-based records in the records vault. Once this information is received, a system engineer will be able to determine whether the connector under investigation is subject to high temperature effects or is environmentally qualified for the environment it is currently installed, therefore temperature fluctuations need not to be included in the FMEA for this component.

6.2.4 Anark Interactive PMI Pro Tools

Anark Interactive PMI Pro is an interactive CAD modeling solution used in manufacturing industry to produce 3D PDF documents that contain geometric dimensions, manufacturing tolerances, and other 3D annotations [53]. By adding advanced model interrogation capabilities to new and existing 3D PDFs, Anark Interactive PMI Pro enables clear communication of original design intent between internal and external stakeholders across the extended manufacturing enterprise. Once the 3D PDF is processed and enhanced, the PMI data may be interrogated on any version of Acrobat Pro or Acrobat Pro Extended with or without the Anark plug-in [53].

Although the Anark solution has shown to be quite successful in the manufacturing industry, it lacks the functionality of showing a component's connections and hierarchy and provides no indication of where in the overall plant design this component is located. Specific data that is reflected on the 3D drawing, such as geometrical dimensions, tolerances, etc. is quite useful for a mechanical designer or component engineer for mechanical systems, but offers no help for system engineers or troubleshooters in cases of I&C systems or complex systems and their interconnections.

6.2.5 Attachmate Online Wiring Tool

Attachmate Corporation Extra! Online Wiring solution [54] is another tool widely used for system wiring and interconnection design and troubleshooting. It provides an excellent tool to I&C engineers and technicians, but offers only a tabular list of data associated with each component or connection. Although this is sufficient for I&C designers or electrical maintenance personnel, it provides no overview of component hierarchy across the plant domain or possess interactive features for instantaneous information retrieval. The process of obtaining more detailed data, e.g. for the example below the EQ specifications of Panel-2651, is again performed manually via a number of other means, such as PassPort Searches, LiveLink or manual document retrievals.

6.2.6 MatLab/Simulink

Existing control system design tools, such as MatLab and Simulink commonly used by AECL designers for CANDU reactors, is another conventional tool used to implement a high fidelity real-time control models. Although they have proved to be an excellent method for time and frequency response analysis or system logic stability tests, MatLab models by no means provide a good indication of system physical components and their interconnections. Systems, Structures and Components (SSC) and their parts are presented as mathematical models with no data on physical aspects of the elements, i.e. make, model, EQ, etc., plant layout or maintenance strategy.

This is further illustrated in Figure 6.1 and 6.2 below. Figure 6.1 shows a process flow diagram of Maple reactor flux control. Figure 6.2 gives a MatLab model of the same process.

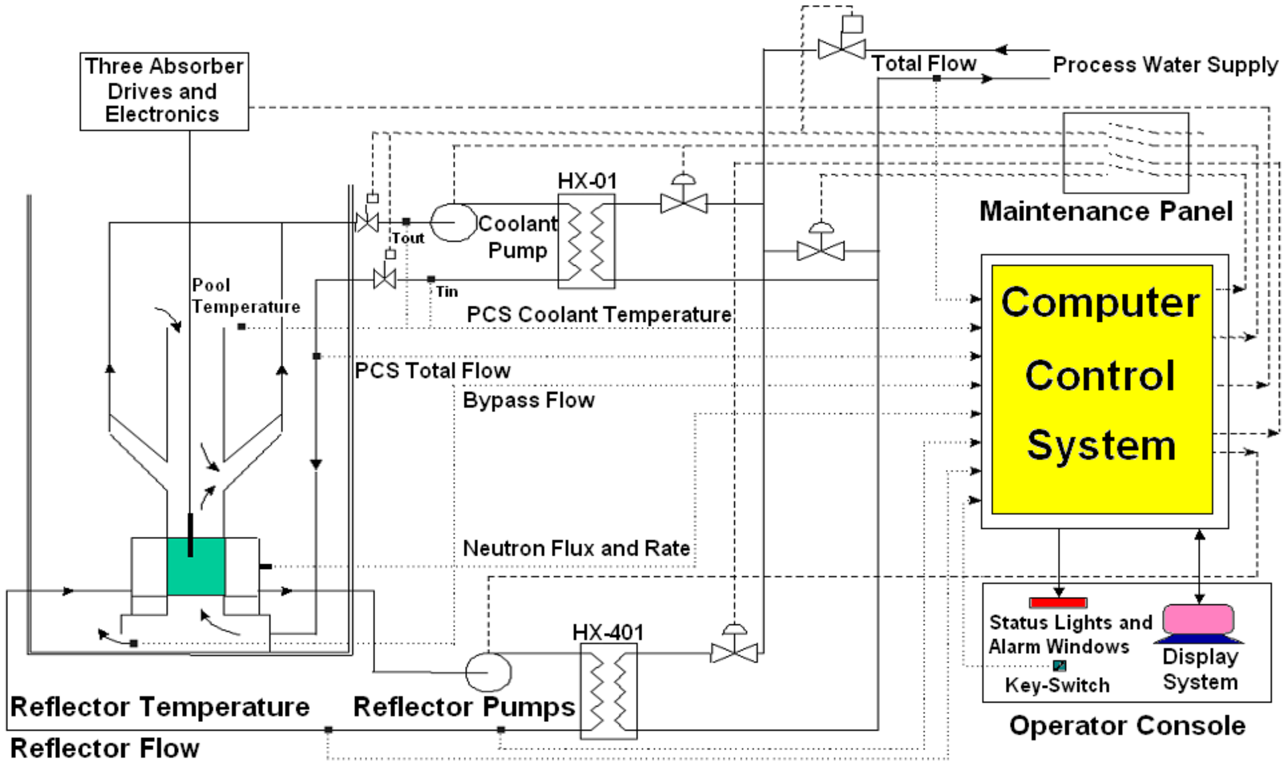

Figure 6. 1: Maple Reactor Flux Control model Process Flow Diagram (PFD) [55].

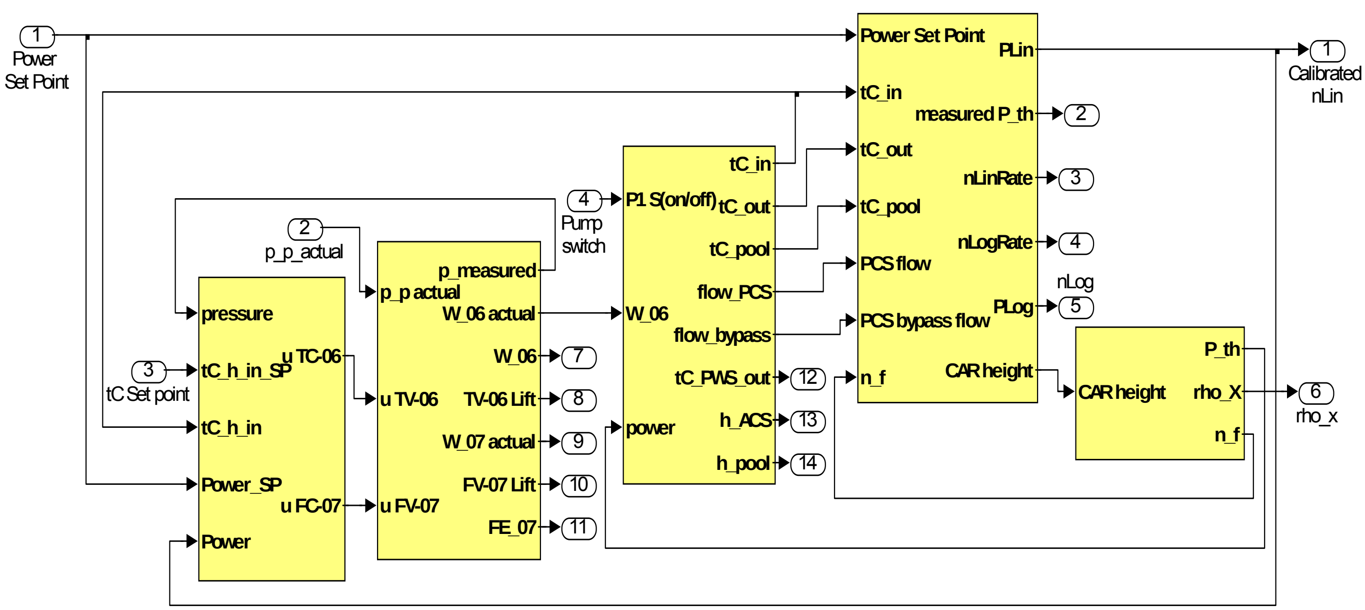

Figure 6. 2: Maple Reactor Simulink Model [55].

Performing a basic visual comparison between the two models of the same system shown above illustrates the concept that the Plant model and a Control model are rather difficult to reference to each other.

6.3 Summary of Current Limitations and Requirements for Additional Functionalities

Currently Bruce Power engineers are using a combination of the tools described earlier, the most popular being PassPort, Online Wiring and Livelink, while AECL system designers are typically using MatLab Simulink models. With all the limitations of each tool described earlier, the existing process of data mining and cross-referencing that needs to be done during system troubleshooting becomes time and labor consuming and results in multiple iterations and switching between various applications.

6.3.1 Design Stage Limitations

This becomes particularly important during design validation and verification stages and may present certain challenges during implementation stages. The Plant model based on PFD or P&ID tools has no fault forecasting or behaviour analysis capabilities while mathematical models developed in Simulink provide no means for effective equipment troubleshooting or parts specifications. Procurement and supply chain personnel will have to find a component that is available on the market that could fit into its required specifications and serve its intended function. As can be seen from Figure 6.1 above, a logical model of a reactor offers no indication of which manufacturer would be able to fabricate and procure the required components. The models that are used will have to be rendered and cross-referenced into a PFD and P&ID designs, after which each component will have to be looked up manually in the existing material catalogues or custom-fabricated to suit its design basis.

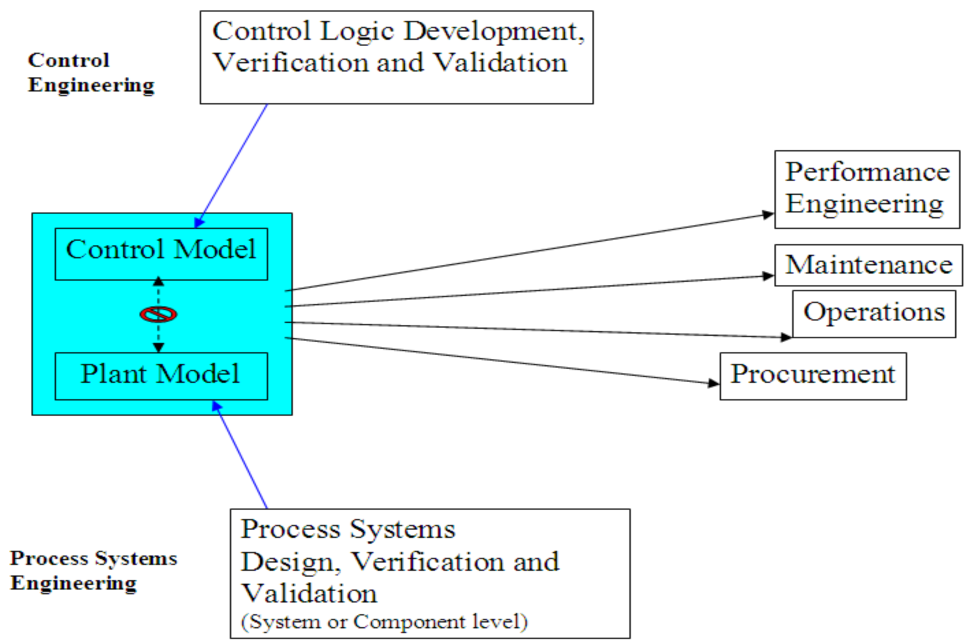

Figure 6. 3: Plant Process and Control Models developed via traditional methods.

A clear disconnect between Plant Process and Control model might result in significant changes and modifications to the system design during implementation stages when the current design limitations are discovered. This may result in a need to design re-works and re-validation or plant modifications done in-situ at later stages, after which system design basis will have to be evaluated and re-assessed in order to prove that the required safety margin is not compromised. With most of nuclear refurbishment projects running over time and over budget, this additional extensions and time losses present significant challenges to project deliverables that have to be met within timelines and budget constraints. Investor and public confidence in the industry's ability to deliver both new installations and refurbishing old units can be greatly affected by unanticipated delays and modification backlog.

6.3.2 *Online Troubleshooting Limitations*

For the systems that are already installed and are in production, quite often the troubleshooting process spans over several components or parts of the system and the exercise of looking up the relevant data and establishing process connections and information exchange paths has to be repeated many times. Even for redundant components, e.g. serial fuses, it is erroneous to assume that the same technical characteristics are applicable to all items with the same functionality. Due to the age of the plant, many components and parts have been replaced in the past resulting in a situation when several different models produced by various manufacturers are installed in the same system. Conservative thinking approach to engineering decision making, particularly in a case of special safety systems and nuclear installations, requires that a system engineer analyses every part and component of the system with due diligence.

6.3.3 *Summary of Required Functionalities and Requirements*

Following the discussion above, it is obvious that a better tool is needed to aid system engineers, designers and trouble-shooters. Current set-up where plant information is stored in various many databases is valid and well grounded. It provides for a better information ownership where each group has a better control of how their data is handled, modified and protected. It also reduces a number of Single Points of Vulnerability (SPV) where malfunction or failure of one data storage will not affect information quality and integrity of others. An example of this could be a failure of a firewall on a Livelink archive server where documental information, such as design manuals or drawings, is stored. Since no direct bi-directional connection exists between LiveLink document servers and MatLab, control system models in Simulink will not be compromised.

This however, results in a gap where a system engineer has to manually conduct searches using various tools on different existing applications and gather the data needed for a basic analysis and evaluations through engaging personnel in other departments, spending time and effort tracing system components across mechanical, electrical and logical models and search engines. Thus, there is a clear need for a new tool where immediate, fast and accurate information can be obtained for any component or connection in the system:

- Conceptual design of the plant system process.
- Preliminary design of the plant system process and preliminary design of system functions.
- Detailed design of the plant functions, preliminary design of operation state.
- Design verification and validation
- Risk Analysis
- Fault forecasting, barrier analysis and implementation
- Troubleshooting and Root Cause Analysis
- Maintenance Planning and Optimization

The main objective is, therefore, to create a highly modular solution that allows easy corrections, extensions, migration and data import/export on as needed basis for an easy integration and collaboration as well as to provide convenience of smooth and easy customization capabilities.

6.4 Hierarchical Control Chart Methodology

Hierarchical Control Chart (HCC) is a proposed new methodology that was developed to be used for nuclear power plant systems and components modeling in order to provide a single view of all elements and systems across a power plant. The proposed Hierarchical Control Chart (HCC) technology enables automation of information exchange between system designers, operators and maintenance personnel on the basis of ISA S-95/S-88 [56, 57]. This also allows standardizing the terminology and object models at various stages of system design, development, operation and troubleshooting.

HCC is integrated with interactive data access and information retrieval capabilities that enables a fast, automated access to the information about processes and parameters across the power plant domain. The information stored in the database captures data at different levels of process and control hierarchy, as well as specifies how each piece of knowledge in the system is interconnected with the others. The proposed knowledge base contains trace information for each piece of knowledge defined which makes it easy to reuse, extend, and translate the contents in the future. The entire existing knowledge bases could be reused or incorporated into the new systems in their entirety or only in the relevant portions.

The objective of the new proposed automated HCC methodology is to aid system designers, operators and maintenance personnel with an automated tool for equipment, process lines and operations mapping, which offers a fast, intelligent and highly automated visual support for design as well as a troubleshooting and fault diagnostic tool. In this thesis it will be used as a modeling technique for risk-based analysis of new generation SUI systems as well as to aid in developing a safety control recipe.

6.4.1 HCC Main Functions and Features

The proposed new HCC system will serve for all, but not limited to, the following purposes:

- Promotes visual recognition of equipment location and process connections across the power plant domain;
- Improves operator awareness of the overall structure of the power plant, which is particularly useful for new employees and operators in training;
- Allows to access required initial information in a fast, consistent and error-free manner;
- Helps to visually map process alarms and faults to the corresponding physical location of the equipment throughout the plant, thus promoting communication with the maintenance department and assisting in troubleshooting activities;
- Make more efficient use of available control room resources through support for visualization of operation and improved standardized naming conventions, hence reducing operating errors, time and workload, and improving operation efficiency and communication between various departments and individual employees.

6.4.2 *HCC Design and Implementation Process*

Figure 40 below shows the processes and tools involved in the design and implementation of the proposed HCC.

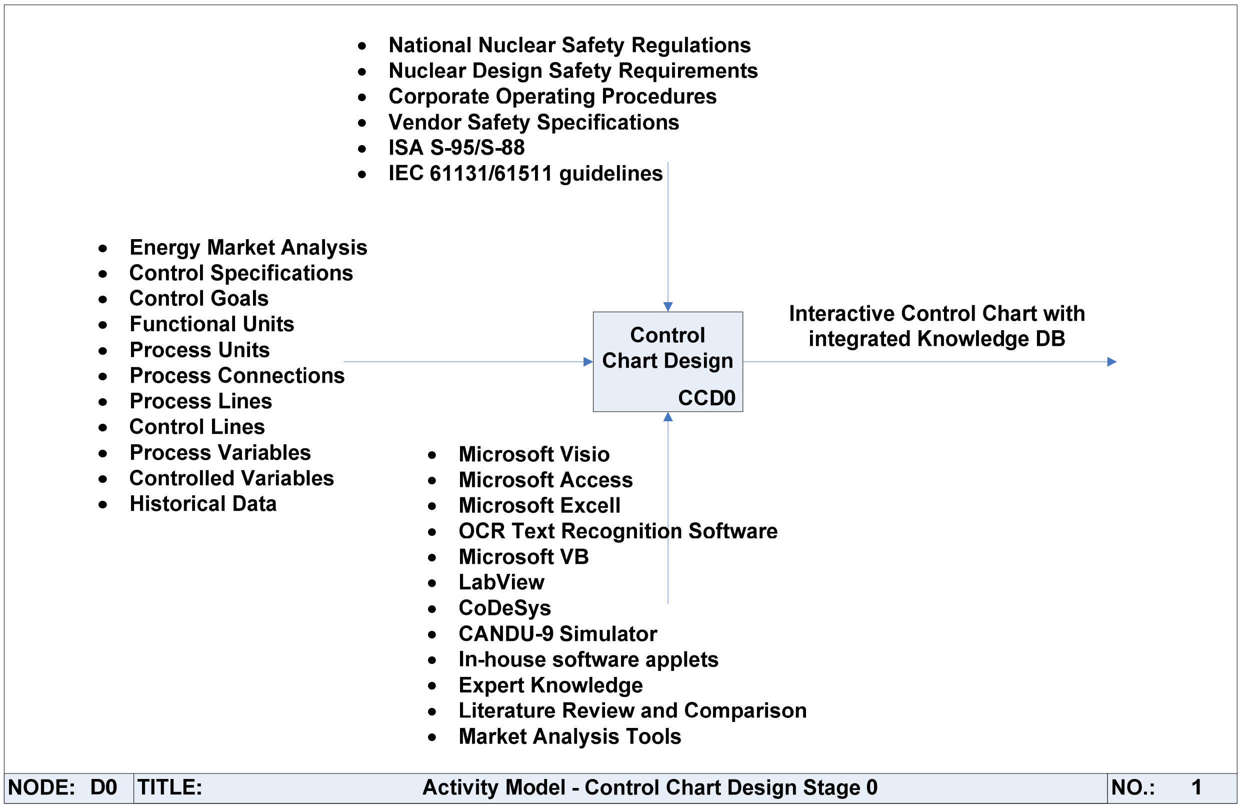

Figure 6. 4: Overall View of the HCC Design Process and Inputs, developed with IDEF0 standard.

As shown in Figure 6.4, the design of the proposed HCC solution was based on the existing ISA S-95/S-88 and IEC 61131/61511 standards and follows the guidelines and regulations of Canadian Nuclear Safety Commission (CNSC) as well as corporate operating policies and procedures. A number of existing tools and computer applications, such as Microsoft VB. LabView and CoDeSys were used along with widely available Microsoft Office tools in order to create HCC User Interface (UI). The step-by-step HCC development process is shown below in Figure 6.5.

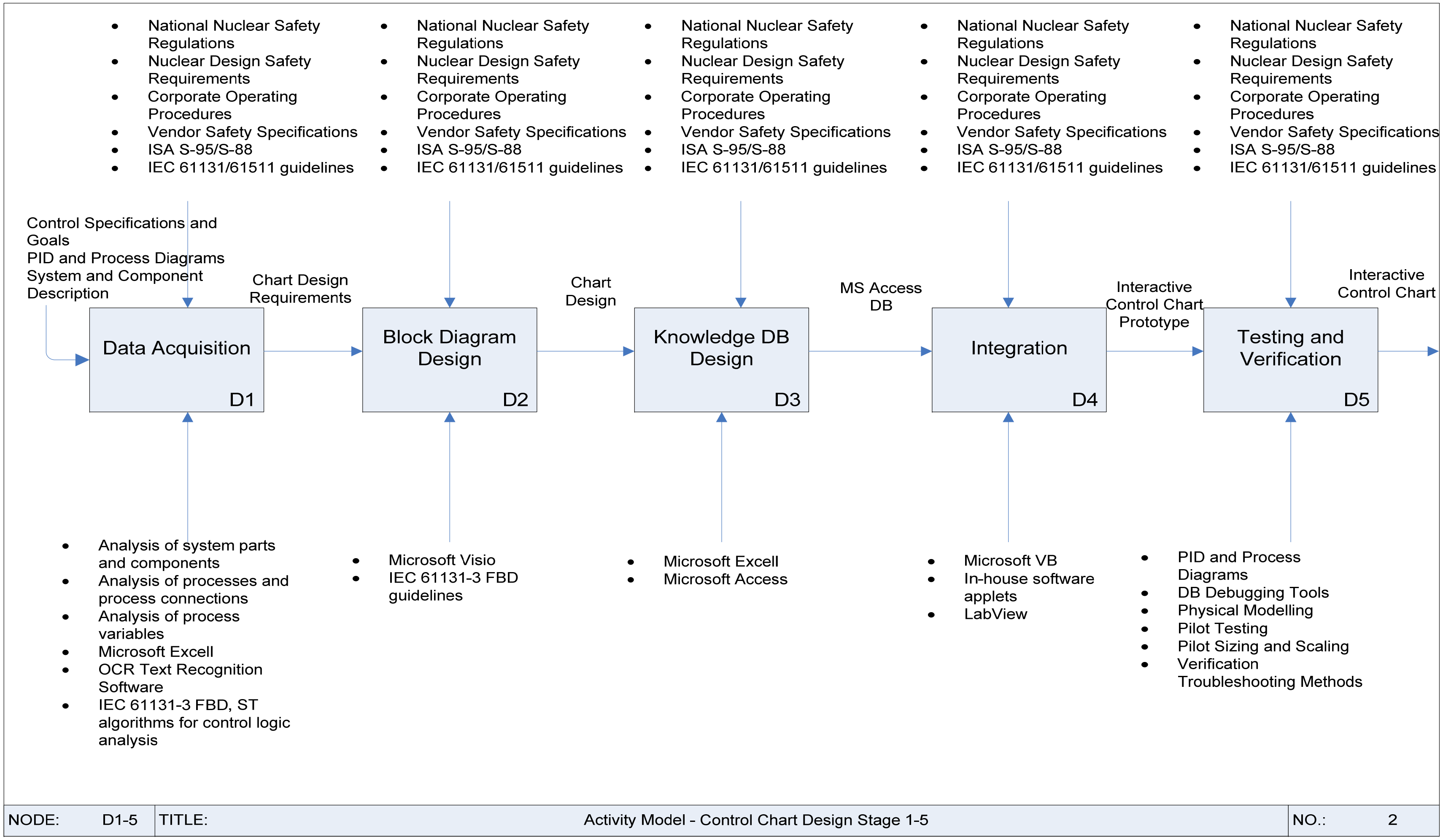

Figure 6. 5: Activity model for proposed HCC solution design process showing the data acquisition, block diagram design, database design, and integration and testing & verification stages. IDEF0 standard is used for model development.

In HCC, the plant hierarchy is described in Layers, e.g. all process systems reside on Process Layer, all control systems reside on Control Layers, sensors and data loggers reside on Data Acquisition & Processing Layer, etc. Next, all Systems, Structures or Components (SSC) in HCC are categorized as Process Blocks, Control Blocks or Data Storage Blocks and their respective connections are classified as either Process (material or energy), Control (regulation) or Data (information) exchange lines. Therefore, the first step in the development process was to identify and categorize all SUI components and connections for the proposed new design.

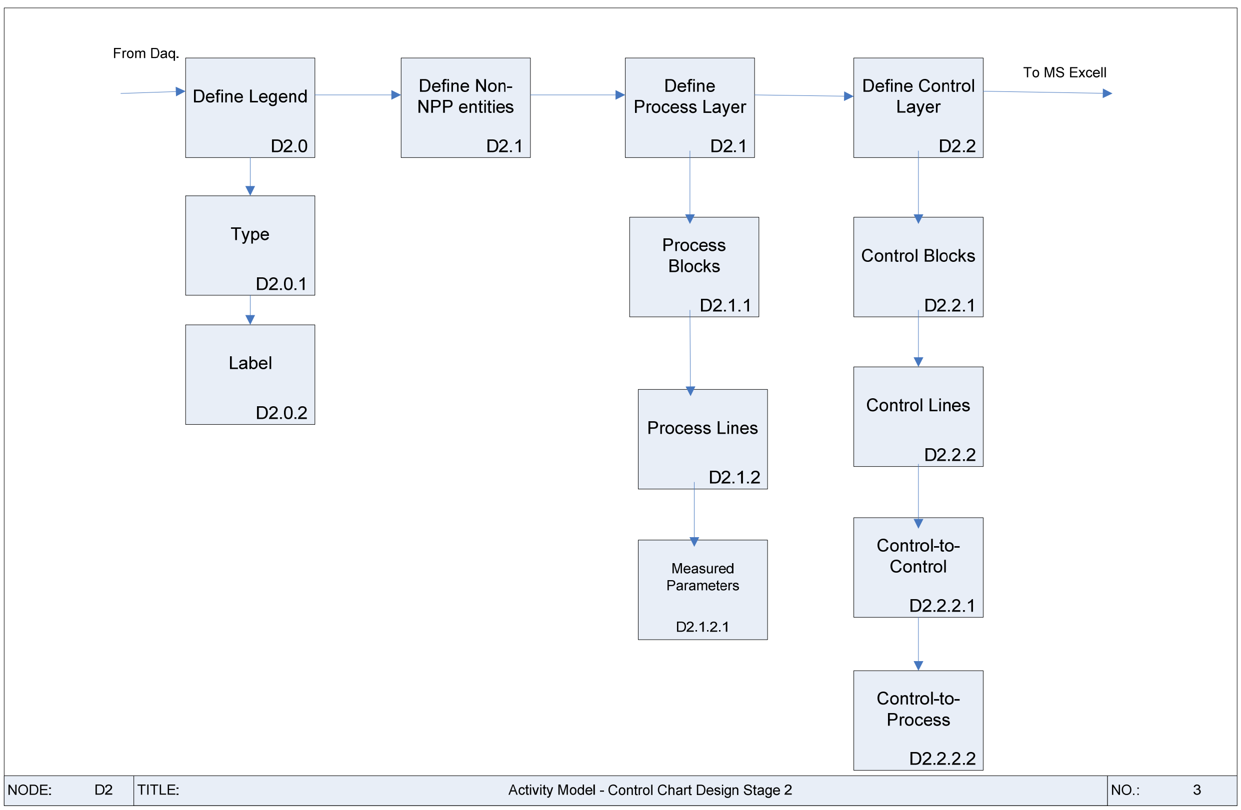

Figure 6. 6: Analysis of SUI System Components.

Next, a stencil is developed for each object or connecting line in order to represent these components in GUI. HCC stencil represents a standardized library of plant components is identified along with the legend describing the blocks and lines naming and graphical conventions used throughout the HCC. A sample stencil is shown below in Figure 6.7:

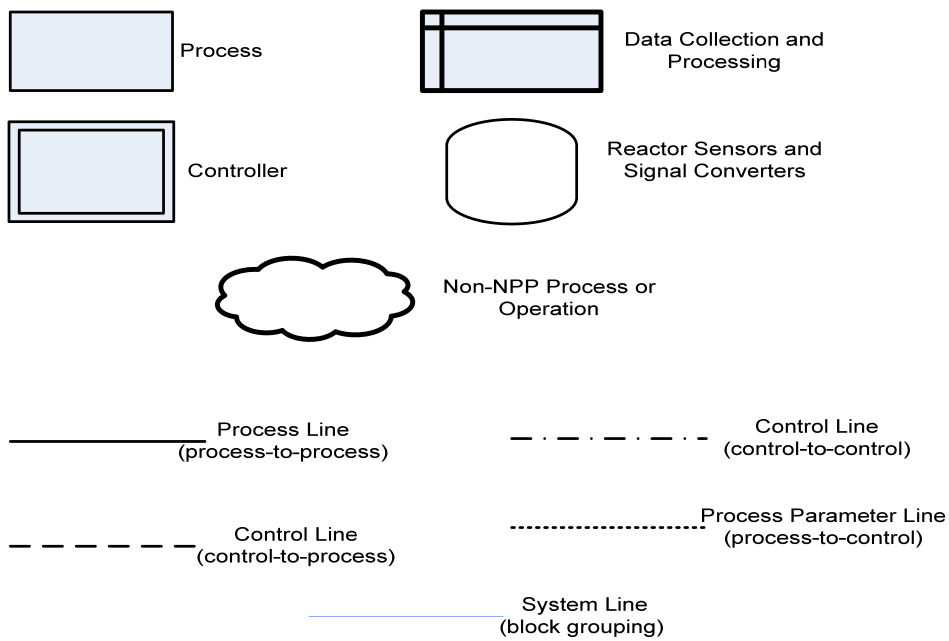

Figure 6. 7: HCC Legend describing the graphical and naming conventions and objects available in the stencils template.

Next, the plant hierarchy can be developed at any level, starting from the most general and can be further expanded to a component level. Through the integrated Knowledge DB, HCC will retain all information about the objects along with their connections and termination data and type of information exchange, e.g. material, sensor data, control instruction, etc.

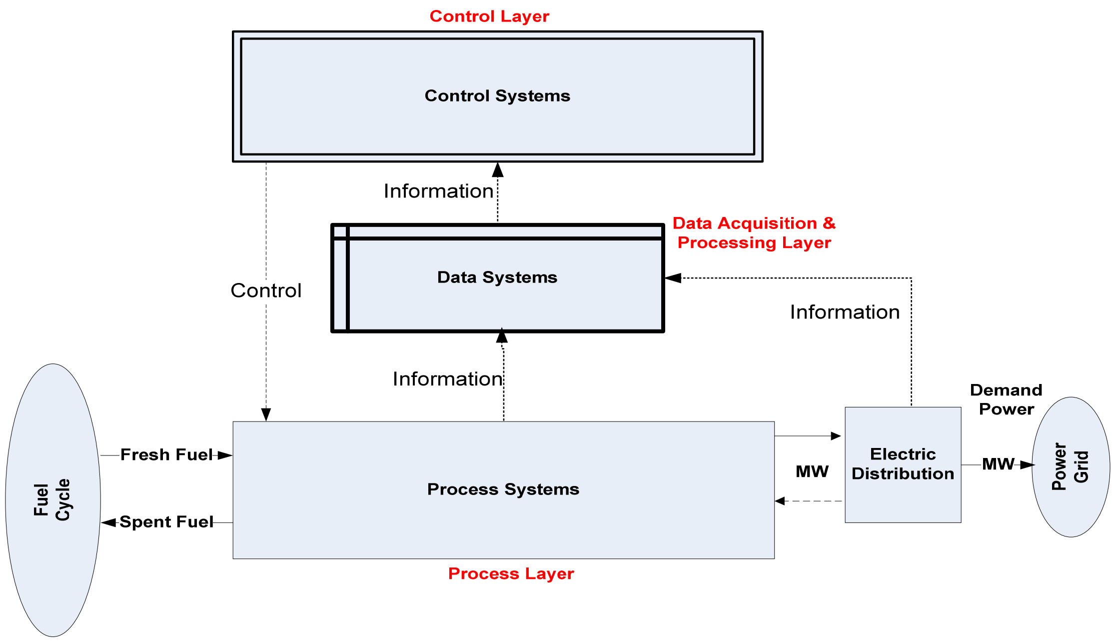

Figure 6. 8: HCC Plant Overview Where All Plant Systems Are Categorized as Objects in Either Process or Control Layer.

This general view can be further expanded to the desired level of detail, as shown in Figure 6.9, where reactivity control block is shown in HCC. While the plant hierarchy is being defined and saved via GUI, all the information associated with HCC objects, e.g. name, model, process variables, etc. is automatically recorded along with the direction, termination points and type of information exchanged between the objects, e.g. flow, steam, control instruction or sensor data.

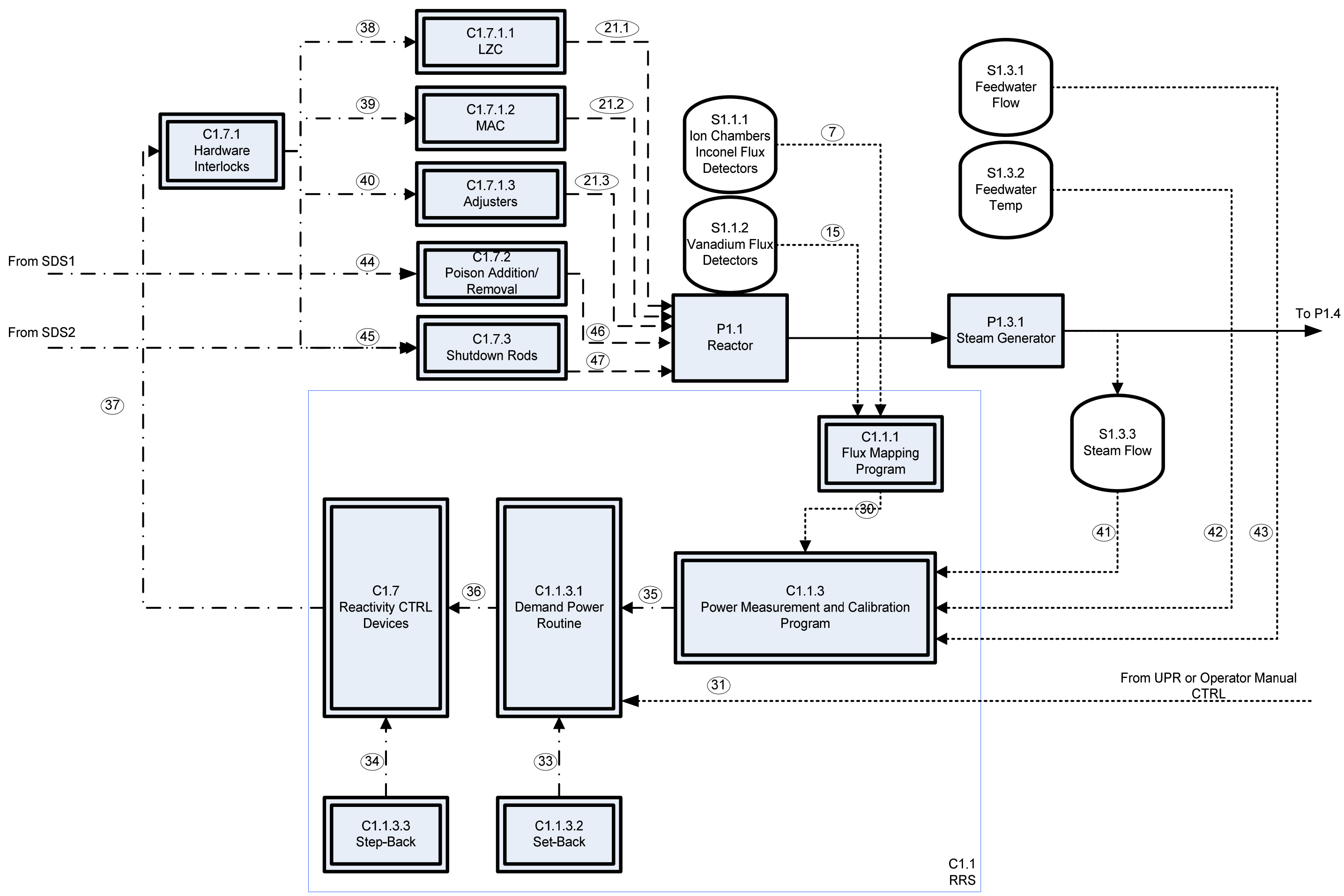

Figure 6. 9: HCC user interface – reactivity control block, where SDS1 trip logic and components are shown along with power monitoring equipment and RRS software ties.

This view can be further expanded to incorporate SUI components for the proposed new SUI system design. The main function of the SUI system is to provide continuous accurate neutron monitoring capabilities during prolonged outages or when RRS Ion Chamber (IC) detectors are off-scale low. Thus, SUI system has no process or control function and will be identified at Data Acquisition and Processing Layer as shown below.

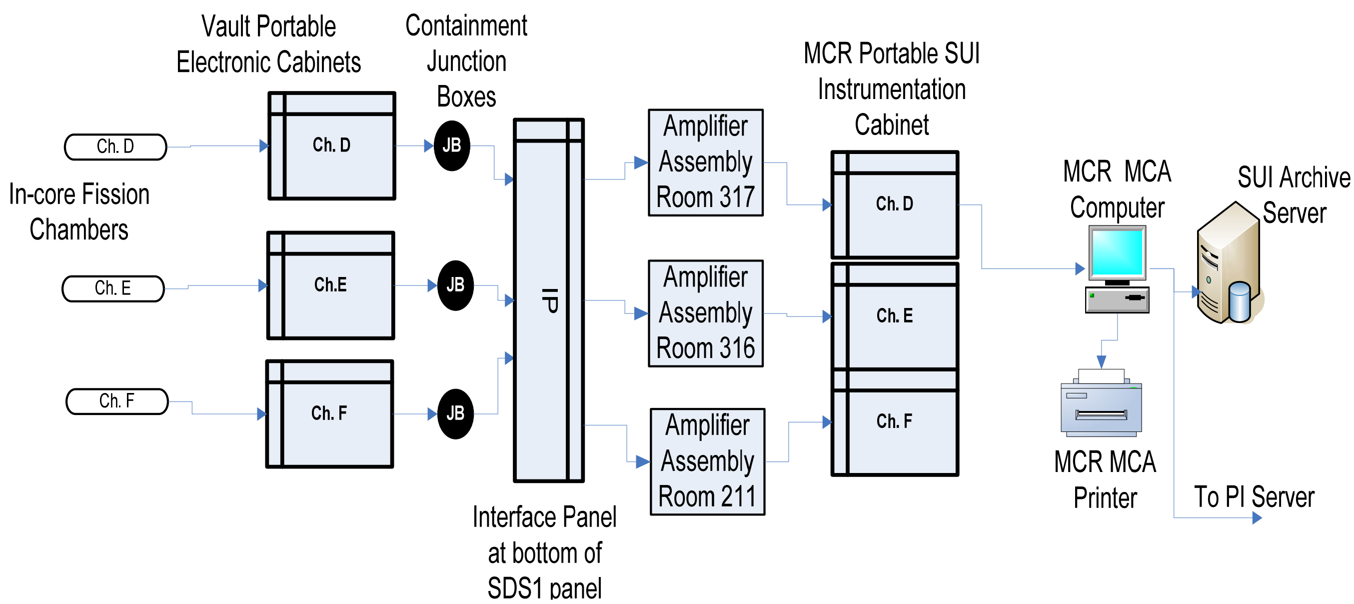

Figure 6. 10: HCC preliminary model for SUI Components.

It is important to note that HCC Graphical User Interface is integrated with the HCC Knowledge Database, where information for various components and connections is entered when a new graphical object is created and automatically stored. Process connections and termination points are also automatically recorded and can be interactively retrieved whenever required. A snapshot of HCC user interface and its capabilities is shown below.

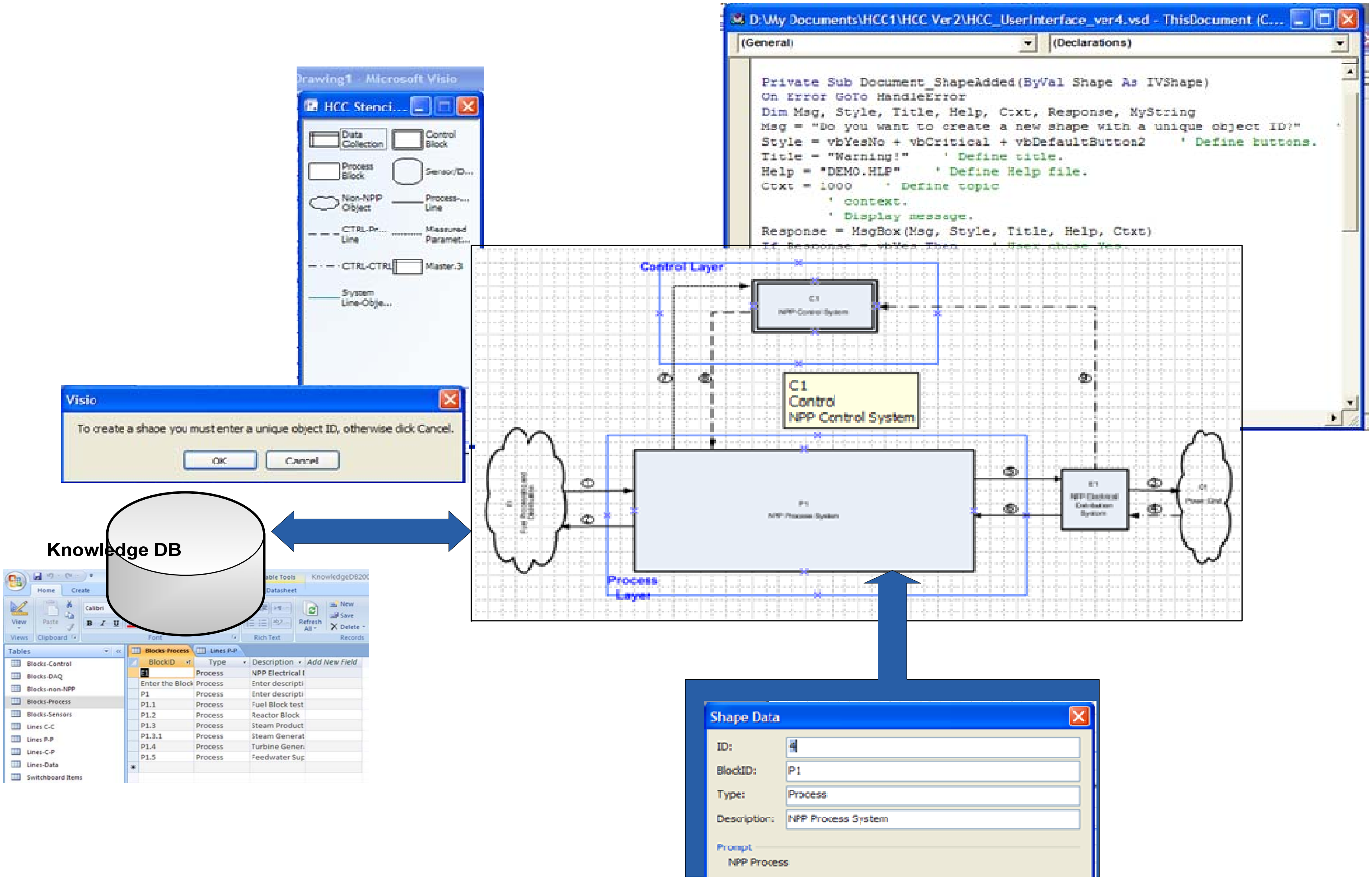

Figure 6. 11: HCC user interface with some of the available capabilities.

Once a plant hierarchy is established and all system and component related information is entered in the Knowledge DB, it can be converted in to FBD-style diagram, as shown in Figure 6.12 and the proposed activities could be mapped into control instructions using control recipes on the basis of IEC-61131-3 standard.

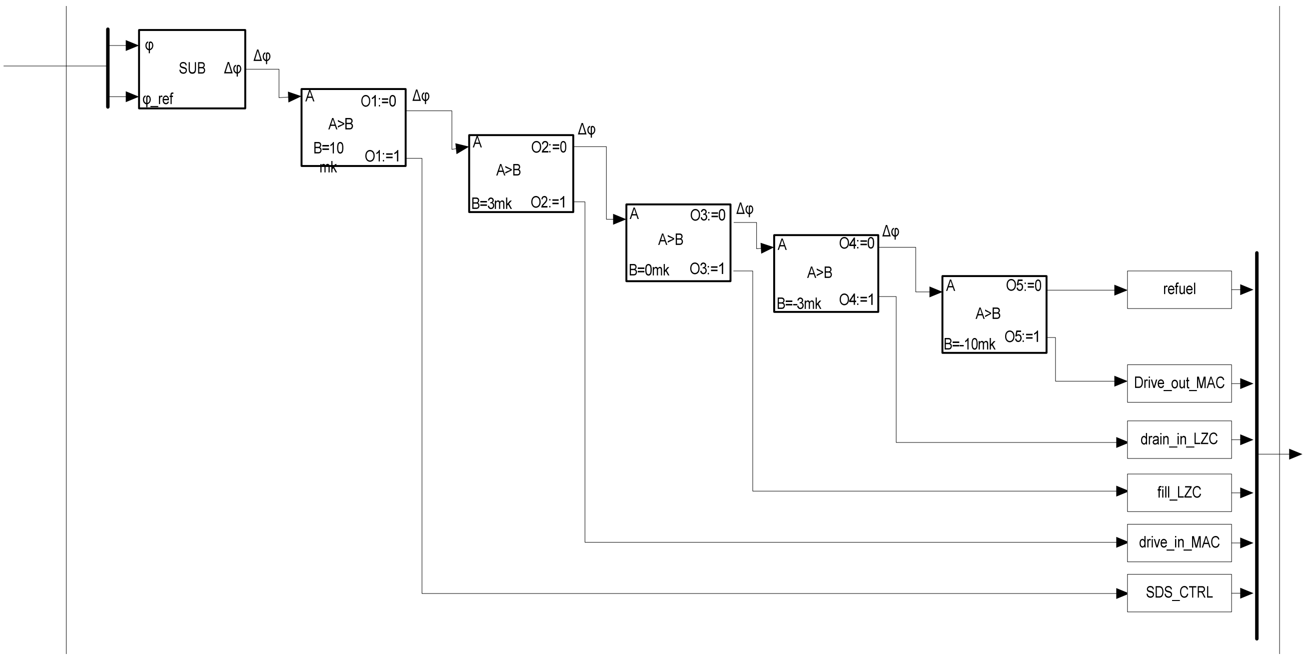

Figure 6. 12 IEC 61131-3 reactivity device selection FBD diagram.

The proposed HCC approach is highly modular and allows easy corrections, extensions, migration and data import/export on as needed basis for an easy integration and collaboration with the existing control systems already in use in Canada and worldwide, as well as providing the convenience of smooth and easy customization for proprietary software and applications.

6.4.3 Hierarchical Control Chart (HCC) Methodology Application for Nuclear Power Plants

The proposed Hierarchical Control Chart solution (HCC) is a new tool that can prove to be a useful aid at all stages of the NPP control system design, development and operation. Its interactive user interface combined with the integrated knowledge database provides control system designers, testers, troubleshooters or users with a single view of all elements and systems across a power plant with an accurate and fast information retrieval capabilities reducing the time spent on proofing and verification of the NPP control system paper-based or still CAD drawings, hence increasing the process efficiency and eliminating unnecessary grounds for human error.

The objective of the new proposed automated control chart and operations mapping solution is to aid the control room operators with an automated tool for equipment, process lines and operations mapping, which is offering a fast, intelligent and highly automated visual support for daily routine operations as well as a troubleshooting and fault diagnostic tool. Its standardized ISA S-95/S-88 [56] [57] interface enables interoperability of control systems in CANDU-based plants in Canada as well as with other members of international nuclear community at all levels of involvement of control system design, implementation, operation and troubleshooting.

6.4.3.1 HCC Application for Control System Designers

The importance of well-designed operator interfaces for reliable human performance and nuclear safety is widely acknowledged [58] [59]. The design of modern control systems starts with the analysis of control goals and control hierarchy, which requires knowledge of all measured and control variables as well as determination of all of the components, processes and their relation. Should a control system designer chose to use the existing paper-based process and connection diagrams they will be faced with time-consuming engineering materials that cannot be easily translated into CAD software, often bear hand-made comments or remarks and exhibit soiled, faded or worn-out ink. Should they chose to use the existing plant simulators or graphical interfaces for the plant processes, they will be faced with multiple objects with bright contrasting colors as well as excessive labelling contributing to clatter on the user's screen and providing highly specialized highly detailed level of information with the lack of distinction between process and control lines, making it hard for the user to trace the line's origin and function [60]. In order to address these and other limitations, the proposed HCC solution could be employed among other tools to aid the control system designers.

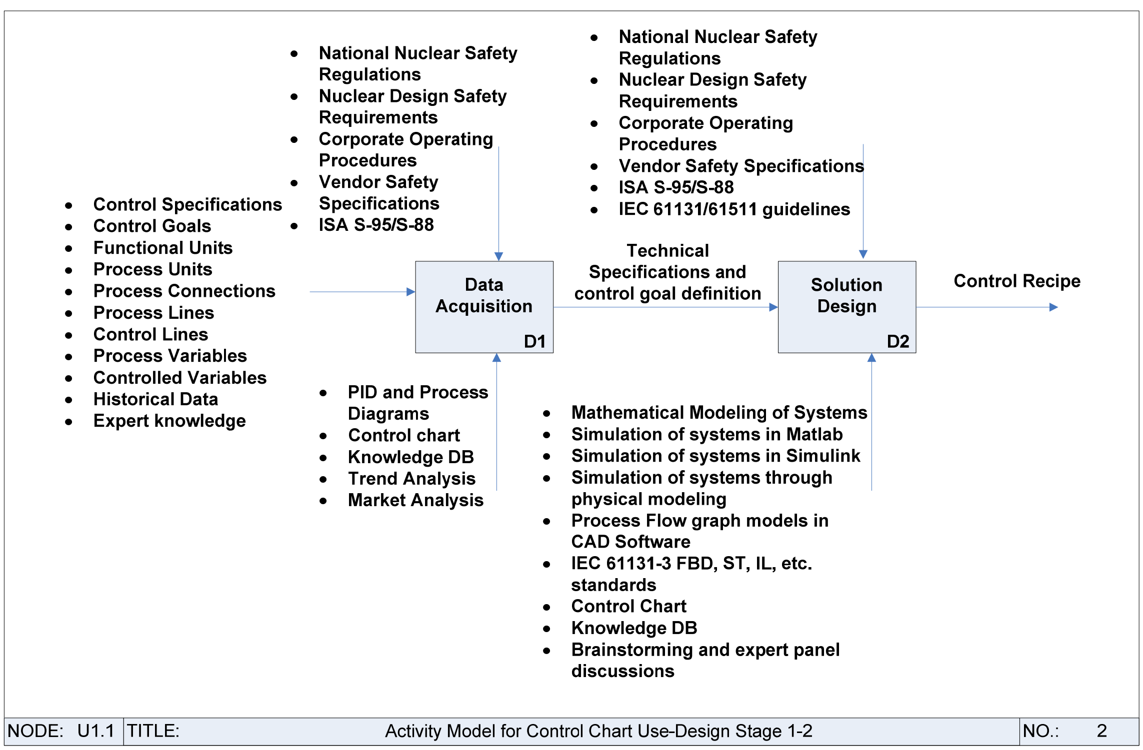

Figure 6. 13: A break-down of Control System Design process with HCC utilization at the data acquisition and solution design stages.

As shown in Figure 6.13, the proposed HCC solution could be used both at the data gathering stage as well as during the control system conceptual design period.

6.4.3.2 HCC Application for Control System Developers

Once the system has been designed and transferred to programming stage for implementation, to verify the system performance a control system programmer will often need to physically trace the line on the screen or paper diagram through a series of valves, pumps and tanks carefully paying attention to their status and functions. In modern nuclear power plants, software developers create logic ranging from something fairly simple, e.g. simulating current device conditions, to actuating safety alarms and reactor trip functions, which are vital to the plant safety and productivity. A minor mistake in a pipe or valve connection could be considered quite excusable for a control system programmer, most of whom are highly specialized IT experts or electric/electronic engineers, but could lead to absolutely devastating consequences for the power plant operation and public safety. This leads to enormous amount of time spent on verification and proofing of new control system at all stages of development prior to installation. The proposed HCC solution could be used to aid the control system developers and programmers during the implementation stage as well as for testing and verification purposes.

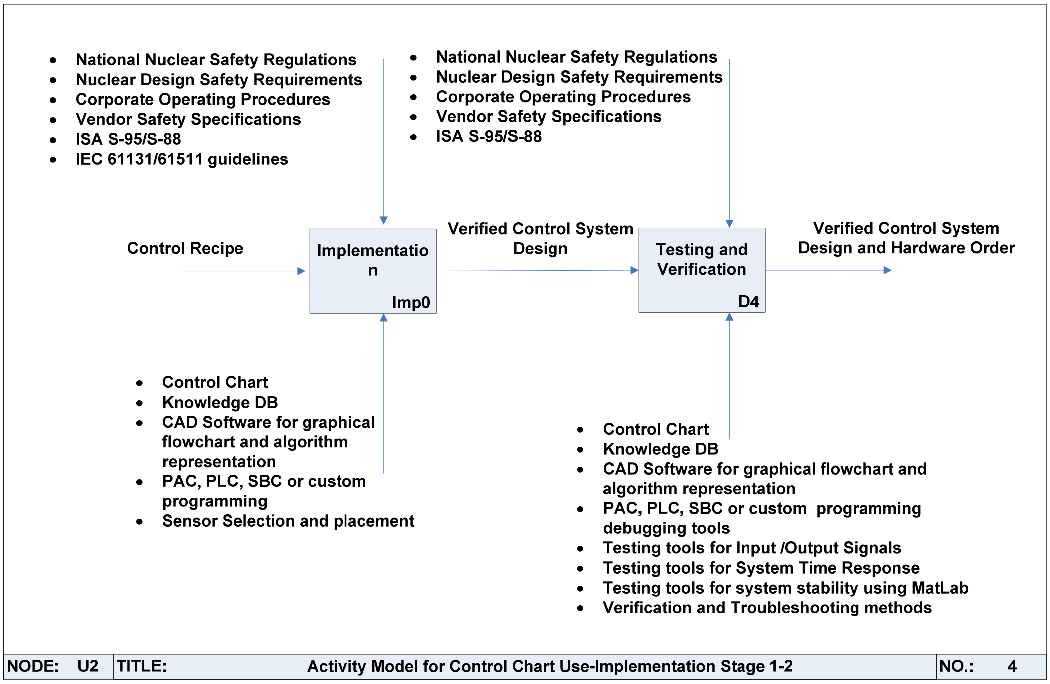

Figure 6. 14: Implementation stage of the control system development showing that the proposed HCC can be used to benefit the control system programmers as well as testers and troubleshooters.

6.4.3.3 HCC Application for Control System Operators

Once the control system is complete and installed at a power plant, it is a common scenario, well described in literature [61], that quite often a process operator is faced with an emergency situation where the operator is overwhelmed by a huge amount of information that has to be processed at a very high speed in a limited time, which increases the operator workload and making it almost impossible to deal with the emergency [61].

This is even more crucial for the nuclear power plants, where the control room operators supervise and monitor all major plant systems and equipment, often mentally translating and interpreting information from multiple sources. Typically, control room personnel comprise five or six operating crews working around the clock. Each crew spends part of their four, five or six week cycle rotating on each of the shifts - days, afternoons, and nights lasting up to 12 hours. This system has proven itself to be very successful in CANDU-based power plants, however, it is highly dependent on the knowledge and expertise of the control room operators who have to be carefully selected and spend approximately eight years in training. Even though increased advancements in evolution of digital technology and process automation have lead to the replacement of operating panels with dedicated instruments by general-purpose workstations, time required for analysis of operating events with potential fault propagation scenarios and consequences across the power plant domain still remains the leading cause of the loss of plant production and operational availability. Long working hours as well as the effects of concentrated mental work (including prolonged visual work) can cause the excessive fatigue and reduce the operators' ability to maintain awareness of equipment location and the process relations across the plant, leading to the obvious implications for workplace and public safety. Field studies [59] showed that in a scenario where the plant procedures are written on paper, the operators have to browse the paper procedures volumes continually, going from general diagnostic flow charts to the detailed manual actions and procedure check-lists and vice-versa, which consumes valuable time during emergency situations, as well as routine fluctuations. Considering the complexity of modern nuclear power plants, the amount of time spent on retrieving and filtering paper-based information by plant operators increased up to a point where it became a factor in hindering operators' performance and creating additional grounds for operation errors. Operational experience worldwide has demonstrated that the accuracy, completeness and efficiency with which work is performed in the plant control centre is a key enabling factor in attaining effective plant operation. [62]. The proposed automated HCC control chart solution is a highly automated, visually-enhanced system that provides nuclear power plant operators with a single view of all elements and systems across the power plant domain.

CHAPTER 7: SUMMARY AND CONCLUSIONS

7.1 Summary and Conclusions

Gas-filled detectors, such as BF_3 and ^3He, for Start-Up Instrumentation in domestic CANDU reactors have been the primary means of monitoring neutron activity at low power to provide sufficient neutron overpower protection during unit shutdowns for routine maintenance.

However, several inherent disadvantages of gas-filled detectors have been well-known since the early day of CANDU program and several efforts have been made to provide a solution to address this issue. Originally SUI systems in domestic CANDU power plants, such as Bruce A were designed to be permanently installed in the vault to provide immediate core monitoring activities following a shutdown and subsequent start-ups. Since no progress was made to overcome rapid detector degradation in high gamma fields, SUI implementation was re-visited and a portable, temporary solution was adopted. Original SUI guide tubes and view ports were removed and a complex, multi-step process of positioning, re-positioning and removal of detectors from service was developed and refined over the years. Presently, following prolonged unit outages it is often necessary to use two different types of detectors, namely BF_3 and ^3He at various stages of reactor power decay and inversely during moderator poison pull and approach to critical.

Although this method helped prolong the detectors' useful life, it resulted in significant additional time and labour costs, with a solid history of plant upsets, spurious trips, equipment misplacement, noisy signals, losses of critical path time and additional REM-dose to the plant personnel.

In the view of upcoming Bruce A Unit 3 and Unit 4 refurbishment project, it is quite likely that the neutron flux in the reactors in question might decay to low and very low levels, potentially down to 10^{-14} decades. It is expected that monitoring reactor powers after such an extensive shutdown with potential de-fuelling of some or all channels might present significant challenges for the ageing BF_3 and ^3He detectors and worn out SUI cables and connectors. Once the reactors are fuelled with fresh un-irradiated fuel the approach to criticality process will require especially accurate instrumentation readings and timely response.

Features and characteristics of several neutron detectors based on fission chambers technology that are available on the market, such as Mirion Technologies/IST, Thermo Scientific and Photonis, were analyzed in addition to the conventional BF_3 and ^3He detectors presently used at Bruce Power. Their main key features have been analyzed and compared and a selection was made in favour of Photonis CFUF-43 fission chambers as the most suitable alternative to the ageing and obsolescent SUI gas-filled detectors at Bruce A.

Also, a proposal for permanently installed SUI instrumentation system based on the new fission-chamber technology was made, as opposed to the existing temporary SUI installation approach, in an attempt to eliminate the challenges and hurdles incurred by the Bruce A maintenance personnel during every outage due to time, effort and radiation dose involved in temporary SUI installations. Additionally, new SUI data logging and archiving capabilities were proposed to provide SUI engineers and control room personnel with the options of computerized data analysis and trending as well as electronic SUI data storage facility.

Also, a clear need for a formal approach to installation/replacement procedure and project optimization was identified and several suggestions were made in order to address this limitations, e.g. using formal language and Meta-operations approach for procedure verification and risk-based analysis for optimization and minimization of risk, cost and time factors.

Finally, a new Hierarchical Control Chart Methodology (HCC) was shown as a new interactive tool that can be used for system designers, engineers and operators for future system troubleshooting and maintenance tasks.

7.2 Recommendations

The framework for SUI procedure development and optimization presented in this thesis can be further expanded to include development of Post-Accident Monitoring capabilities for Bruce Power units using the proposed framework for new generation SUI system as currently the stations do not posses this capability.

Additionally, the framework for installation of fission chambers shown in this thesis could be further developed to provide a solution for Prompt Fraction Test Requirements for Bruce Power Neutron Overpower (NOP) system. Currently several NOP detectors at both Bruce A and Bruce B stations are trending down and showing accelerated degradation and burnout. With new fission chambers technology in place a new methodology can be developed to address this limitation using a systematic structured approach as discussed in this research.

7.3 Future Work

Upgrading the existing Bruce Power SUI instrumentation with fission chamber detectors permanently placed in service will be able to provide neutron over-power coverage throughout the units' refurbishment project well into the future of the station, as well as a long-term reliability and significant economical benefits.

However, given the complexity of the project and the important nature of the special safety systems further detailed technical feasibility studies involving reactor component designers, nuclear safety assessment experts, DCC computer design group and manufacturer's product line specialists will be required to finalize this proposal in terms of financial investment, time and labour costs that will be incurred. As this work expands and matures, it is expected to develop into a sustainable approach for ongoing improvement to all critical and safety system refurbishment projects.

APPENDIX A: MODERN NUCLEAR POWER PLANTS WHERE PHOTONIS FISSION CHAMBERS ARE USED

AREVA (PWR) 900 MW(e) EDF reactors, France
Detectors: 2 CPNB44P and 5 CFUF43P per block

• Bugey 4 blocks
• Fessenheim 2 blocks
• Dampierre 4 blocks
• Gravelines 6 blocks
• St Laurent 2 blocks
• Chinon 4 blocks
• Le Blayais 4 blocks
• Cruas 4 blocks
• Tricastin 4 blocks

AREVA (PWR) 1300 MW(e) EDF reactors, France
Detectors: 4 CPNB44P and 6 CFUF43P per block

• Cattenom 4 blocks
• Nogent 2 blocks
• Penly 2 blocks
• Paluel 4 blocks
• Flamanville 2 blocks
• Belleville 2 blocks
• St Alban 2 blocks
• Golfech 2 blocks

AREVA (PWR) 1450 MW(e) EDF reactors, France
Detectors: 4 CPNB44P and 6 CFUF43P per block

• Chooz 2 blocks
• Civaux 2 blocks

AREVA (PWR) 900 MW(e), Belgium
Detectors: 2 CPNB44P and 5 CFUF43P per block

• Doel 1 block
• Tihange 2 blocks

AREVA (PWR) 900 MW(e), Republic of China
Detectors: 2 CPNB44P and 5 CFUF43P per block

• Daya Bay 2 blocks
• Ling Ao 2 blocks
• Quinshan 3 blocks
• Tianwan 2 blocks

AREVA (PWR) 900 MW(e), South Africa
Detectors: 2 CPNB44P and 5 CFUF43P per block

• Koeberg 2 blocks

AREVA (PWR) 900 MW(e), South Korea
Detectors: 2 CPNB44P and 5 CFUF43P per block

• Ulchin 2 blocks

WESTINGHOUSE (PWR) 360 MW(e), Switzerland
Start-up and intermediate range Detectors: CFUG08

• Beznau 1 4 detectors
• Beznau 2 4 detectors

ABB-Atom (BWR)
Start-up and intermediate range Detectors: CFUE32S

• Forsmark 3 Sweden 8 detectors
• Oskarshamn 3 Sweden 8 detectors
• Barsebäck 1 Sweden 4 detectors
• Oskarshamn 1 Sweden 4 detectors
• Barsebäck 2 Sweden 4 detectors
• Oskarshamn 2 Sweden 4 detectors
• Olkiluoto 1 Finland 8 detectors
• Olkiluoto 2 Finland 8 detectors

VVER (PWR) 410 MW(e), Czech Republic
Start-up and intermediate range Detectors: CPNB44K

• Dukovany 6 detectors

VVER (PWR) 440 MW(e)
Start-up and intermediate range Detectors: CFUL08, CPNB44

- Bohunice 1,2,3,4 Slovakia 26 detectors
- Paks 1 Hungary 3 detectors

VVER (PWR) 440 MW(e), Bulgaria
Start-up range - safety instrumentation Detectors: CPNB44K

- Kozloduy 1 6 detectors
- Kozloduy 2 6 detectors

VVER (PWR) 950 MW(e), Ukraine
Start-up and intermediate range Detectors: CFUL08

- Khmelnitski 8 detectors
- Rovno 8 detectors
- Zaporozhe 24 detectors

BREEDERS, France
Detectors: Numerous R&D fission chambers

- Marcoule PHENIX
- Greys-Malville SUPERPHENIX

AREVA FUEL REPROCESSING PLANTS
Detectors: Numerous fission chambers

- Marcoule France
- La Hague France
- Rokkasho Japan

APPENDIX B: LIST OF CANADIAN STANDARDS AND COMPLIANCE REGULATIONS FOR ELECTRICAL, ELECTRONIC AND I&C EQUIPMENT

Canadian Standards Association; Quality Program - Category 3, CSA Z299.3.

Canadian Standards Association; CSA 22.2 NO142 - Process Control Equipment Industrial Products-General Instruction No 1-5.

Institute for Interconnecting and Packing Electrical Circuits; IPC-A-600 - Acceptability of Printed Circuit Boards.

Institute for Interconnecting and Packing Electrical Circuits; IPC-A-610 - Acceptability of Printed Board Assemblies.

Institute for Interconnecting and Packing Electrical Circuits; IPC-A-770 - PC Component Mounting.

Institute for Interconnecting and Packing Electrical Circuits; J-STD-001 - Requirements for Soldering Electrical and Electronic Assemblies.

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IEC-61000-4-4, Electromagnetic compatibility (EMC) - Part 4-4: Testing and measurement techniques - Electrical fast transient/burst immunity test-Second Edition.

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