SAFE OPERATION OF THE NRU RESEARCH REACTOR NOW AND BEYOND 2021

S. MISTRY
Safety Engineering and Assurance,
Atomic Energy of Canada Limited –
Nuclear Laboratories, K0J 1J0 Chalk River, – Canada

ABSTRACT

This paper will describe the approach that has been taken by Atomic Energy of Canada Limited (AECL) to ensure that the National Research Universal (NRU) reactor designed in the 1940's continues to remain safe and reliable to operate now and for the near future (2021 and beyond). This paper focuses on two major projects, the NRU Upgrades Project undertaken in the 1990's and the Integrated Safety Review (ISR) resulting in the Integrated Implementation Plan (IIP) that is currently underway.
Through the NRU Upgrades Project, AECL was able to identify areas for safety improvement and implement changes in the field. Following the NRU Upgrades Project, AECL was able to demonstrate that for design basis accidents that the reactor was able to meet the four basic safety requirements namely:-

- It shall be possible to shut down the reactor and maintain it in that state indefinitely;
- The capability of removing decay heat from the fuel during this shut down period shall be maintained;
- The confinement structure shall continue to be capable of limiting radioactivity release; and
- Continuous monitoring of reactor safety functions shall remain available.

The NRU Upgrades Project enabled AECL to continue to operate the NRU reactor beyond the year 2000 but it was recognised in 2008 that if operations were to continue up to and beyond 2021 then another assessment was warranted. This assessment resulted in the ISR project. The ISR project consisted of reviewing the NRU design against current codes and standards and, where applicable, addressing gaps identified. This project identified not only gaps in the analysis basis for NRU, it also identified the need to replace ageing equipment that was reaching the end of its design life. The findings of the ISR project have been captured in the IIP; IIP has enabled AECL to prioritise equipment replacement to enable continued safe and reliable operation of the NRU reactor beyond 2021.
The paper demonstrates that, in order to safely extend reactor life, it is important to address both design issues (NRU Upgrades Project) and ageing management issues (ISR Project/ IIP).

1 Introduction

This paper will provide an overview on some of the activities undertaken by Atomic Energy of Canada Limited (AECL) to ensure that the National Research Universal (NRU) research reactor operates safely now, and into the future. The paper focuses on two significant projects: the NRU Upgrades Project and the Integrated Safety Review (ISR) Project resulting in the Integrated Implementation Plan (IIP). An overview of both projects and the plan will be presented together with a description of changes that were, and continue to be, implemented in the field to support continued safe operation. The paper will show that extending safe operation is complex and that many factors need to be considered. It will also show that, though complex, these factors can be managed, as demonstrated by AECL's continued safe operation of the NRU reactor.

1.1 NRU History

The NRU reactor is located at AECL's Chalk River Laboratories in Canada. It is a heavy water moderated and cooled 135 MWt reactor that first achieved criticality in 1957. Originally designed to use natural Uranium, it has been converted to use low enriched Uranium (LEU, 20% U-235); conversion being completed in the early 1990's.
The original role of the NRU reactor as a nuclear research tool in the 1950's was later augmented with a second role; that of a supplier of medical isotopes such as Molybdenum–99 (Mo-99). The need to safely support nuclear research and medical isotope production resulted in AECL reviewing NRU operation in the 1990's and taking steps to maintain this capability past the year 2000.
In 2008, as in the 1990's, AECL recognised the need to conduct a review of the NRU reactor to ensure both nuclear research and medical isotope production could continue to be safely supported. The review was focused on enabling the NRU reactor to safely operate up to, and, beyond 2021.

2 The NRU Upgrades Project

In 1991 following a detailed systematic review of the NRU reactor, AECL concluded that the reactor was in good condition and being operated safely. The assessment also identified areas where safety could be improved, leading to the creation of the NRU Upgrades Project. An overview of the major steps of this project will be provided, followed by a brief description of the changes that were implemented by the project and the resultant improvement to safety following these changes.
The NRU Upgrades Project was broken into a number of key steps as shown in Figure 1.

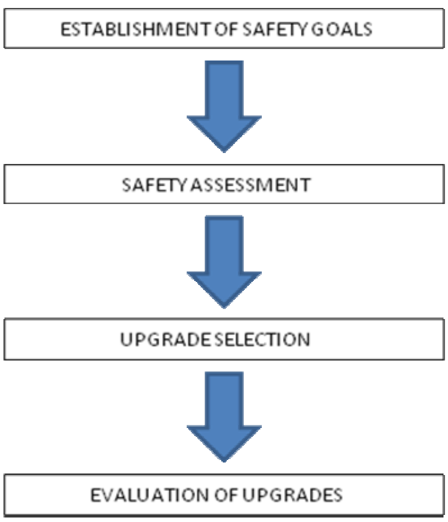

Fig 1. NRU Upgrade Project Steps

2.1 Establishment of Safety Goals

As shown in Figure 1, the first step in the project was to establish the safety goals that would be used to determine the effectiveness of the project. The NRU Upgrades Project developed a set of safety goals for the NRU reactor based on both Canadian and International guidance documents, [1] [2]; these goals were then used to help determine the nature of the NRU Upgrades and the resulting safety improvement. The safety goals chosen were:

1) Release Limit
2) Fuel and Core Damage Frequencies

The release limits defined the acceptable dose to off-site personnel i.e. members of the public based on a consultative document released by the Canadian Regulator [1]. These requirements stated that:

1) Any single event or event sequence giving rise to an off-site dose greater than 0.25 Sv shall have a frequency not greater than 10^{-5} per annum; and
2) Any single event or event sequence giving rise to releases greater than 10^{-5} per annum shall be limited to dose levels given in Figure 2.

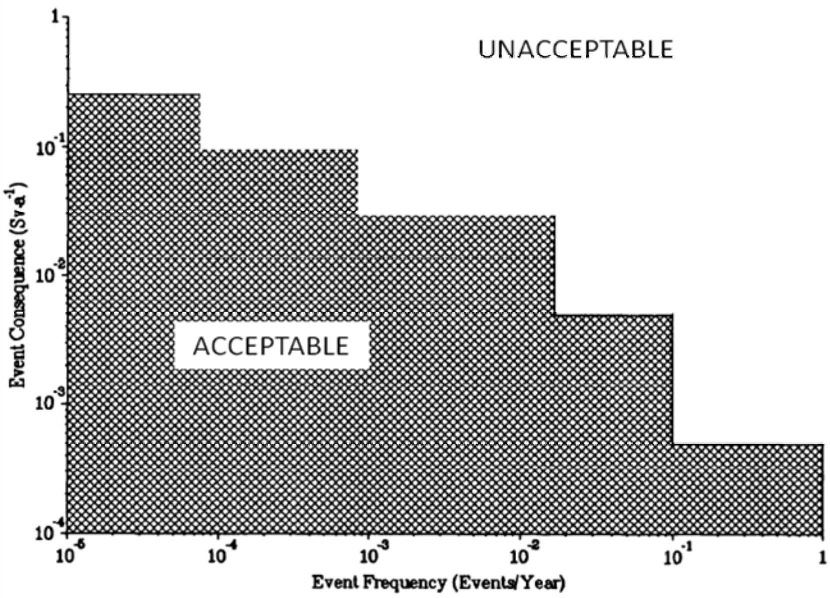

Fig 2. Proposed NRU Safety Upgrade Release Goals

The fuel failure frequency was defined such that, significant fuel failures shall be prevented for any single event, or event sequence that has a frequency greater than 10-3 per annum. The project used the IAEA's INSAG-3 document for guidance [2] in determining the core failure frequency requirement. The core failure frequency requirement stated that, any single event, or sequence of events, shall have a frequency no more than 10^{-4} per annum.

The safety goals can be graphically represented as shown in Figure 3.[1] Figure 3 highlights the relationships between fuel failure, core damage and dose limits clearly showing that fuel failure does not always result in core damage, and that core damage does not always result in a dose that exceeds 0.25 Sv.

[1] The off-site dose acceptability for event frequencies greater than 10^{-5} has already been provided in Figure 2.

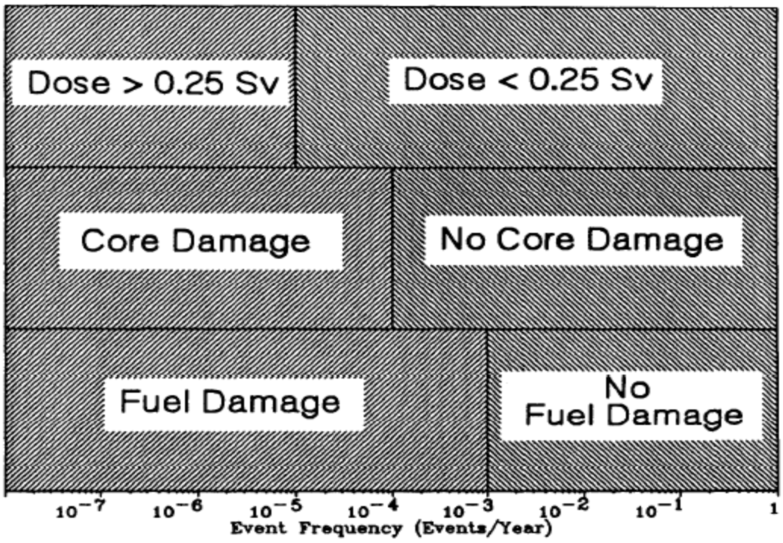

Fig 3. NRU Safety Upgrade Goals

2.2 Safety Assessment

The establishment of the safety goals provided the NRU Upgrades project with a measure with which to conduct the safety assessment. The next step in the project was to conduct a safety assessment to determine if there was an opportunity for a safety improvement. The approach taken during the assessment is shown in Figure 4.

Fig 4. Safety Assessment Process

The determination of the initiating events employed a variety of tools and techniques, including

1) Review of the existing safety analysis report for NRU;
2) Review of internal operating experience;
3) Review of external operating experience (CANDU – 6);
4) Review of external operating experience from international research reactors; and
5) Systematic review of the NRU plant design.

The output from all of the above techniques was a list of initiating events that was used to perform the safety assessment.

As with the determination of the initiating events, the analysis of the events employed a variety of tools to determine the impact of each initiating event in terms of Off-Site Dose, Fuel Failure Frequency and Core Damage Frequency [3]. The processes used were:
1) Deterministic Analysis;
2) Probabilistic Analysis;
3) Accident Analysis, - Including accident progression / intervention opportunities;
4) Consequence Analysis; and
5) Common Cause Effect Analysis.

The output from this stage of the project provided the consequences for each initiating event as well as a frequency. For any event sequence that did not meet the safety goals a number of options were considered by the team, they were:
1) Changes to Operating Procedures;
2) Changes to Emergency Procedures;
3) Implement Design Changes; and
4) Any Combination of the Above.

The safety assessment process is summarized in Figure 5.

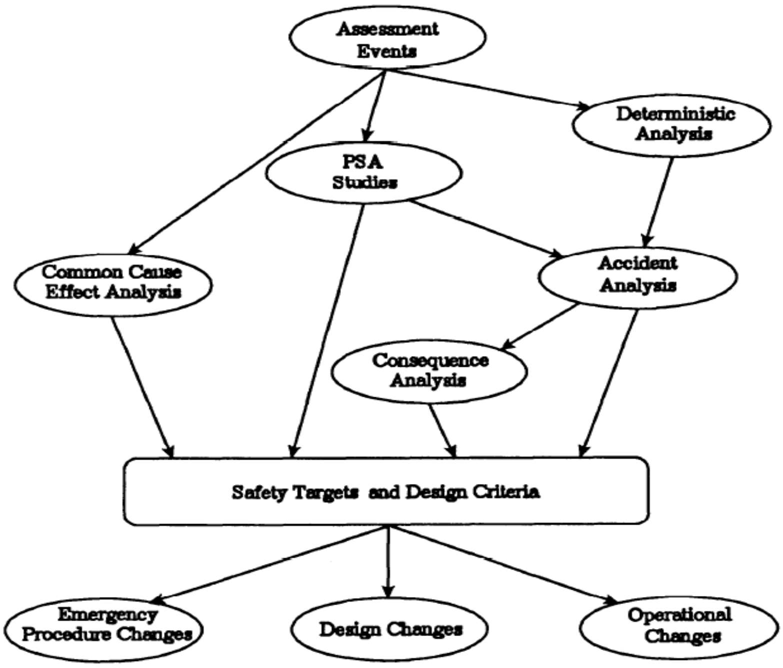

Fig 5. Safety Assessment Process

The areas for improvement identified by the safety assessment were grouped as follows:
1) Loss of reactivity control;
2) Loss of primary coolant;
3) Loss of primary coolant flow;
4) Loss of secondary coolant;
5) Common mode failures due to internal/external events (e.g. seismic, flood);
6) Confinement of radionuclides following fuel damage; and
7) Remote monitoring of the core.

2.3 Upgrade Selection

The completion of the safety assessment provided the NRU Upgrades Project with a number of initiating events where procedural changes (operational or emergency response) were not sufficient to meet the new safety goals established in Section 2.1. For these events the only means to meet the safety goal(s) was through physical changes to the plant. As with all engineering changes, the project had to consider a number of factors, the most significant of these being:

- Modification must result in a measurable net positive gain in safety;
- Any changes must be implemented in a reasonable time frame;
- NRU must remain operational with no extended outages;
- Modifications will not require extensive operator retraining; and
- The cost of modification must not be excessive.

The list above is typical of most engineering changes with the exception of the 3rd bullet 'NRU must remain operational with no extended outages'. This requirement was imposed due to importance of NRU as the major supplier of medical isotopes at the time[2]. The final result of the selection process was a list of design changes that satisfied all these requirements as shown in Figure 6.

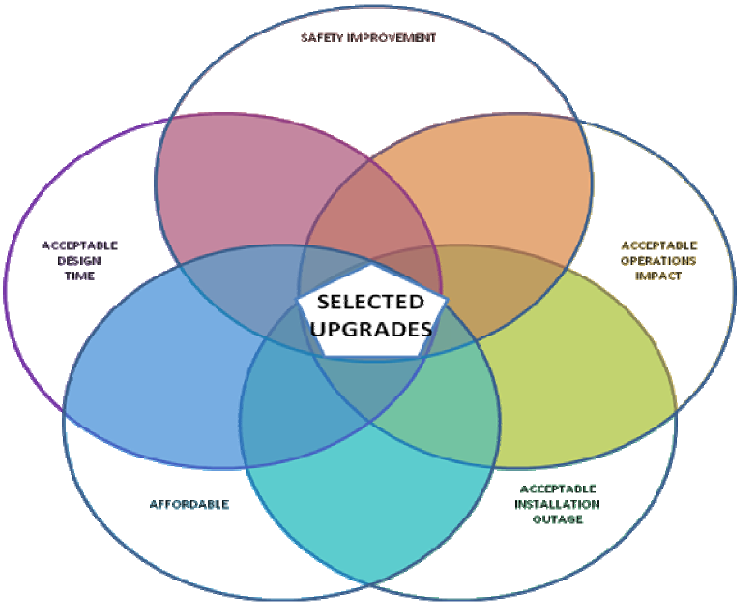

Fig 6. Upgrade Selection Considerations

Based on the considerations summarized in Figure 6, seven system upgrades were designed and implemented, these systems referred to as the NRU Upgrades are listed below:

1) Second Trip System (STS)
2) New Emergency Core Cooling System (NECC)
3) Emergency Power Supply (EPS)
4) Qualified Emergency Water System (QEWS)
5) Main Pump Flood Protection (MPFP)
6) Liquid Confinement Vented Confinement (LCVC)
7) Qualified Emergency Response Centre (QUERC)

2 In the 1990's NRU accounted for 90% of the world's supply of Mo-99.

2.4 Evaluation of the NRU Upgrades

The NRU Upgrades enhanced the safety of NRU by addressing the areas of improvement determined during the safety assessment as described in Section 2.2. These enhancements are summarized in Table 1.

Safety Upgrade	Enhancement
STS	Provides additional protection against loss of reactivity control that is hazards qualified[3].
NECC	Provides additional protection against loss of primary coolant accidents that is hazards qualified.
EPS	Provides additional protection against loss of external power by providing a power source that is hazards qualified. Provides additional protection against loss of primary coolant flow (typially associated with loss of power) Provides a power source for those NRU Upgrades that are hazards qualified.
QEWS	Provides additional protection against loss of secondary coolant accidents that is hazards qualified.
MPFP	Provides protection against internal flooding as a result of a common mode failure of the process water supply that is hazards qualified.
LCVC	Provides enhanced capability to confine radionuclides in the event of fuel damage.
QUERC	Provides a hazards qualified means to shut down and monitor the core should the control room be uninhabitable.

Tab 1. Summary of Enhancements Provided by the NRU Upgrades

As was stated in the comprehensive review of NRU in the 1990s, the NRU reactor was safe to operate with areas for improvement. The successful implementation of the NRU Upgrades addressed the areas offering the greatest improvement given the project constraints. The impact of the NRU Upgrades is shown in Figure 7, which demonstrates that the NRU Upgrades provides increased event coverage and also provides increased defense-in-depth by working with the existing NRU systems.

[3] Hazards qualified refers to systems that have been engineered to withstand process-type failures, earthquakes, internal fires, flooding, and tornados.

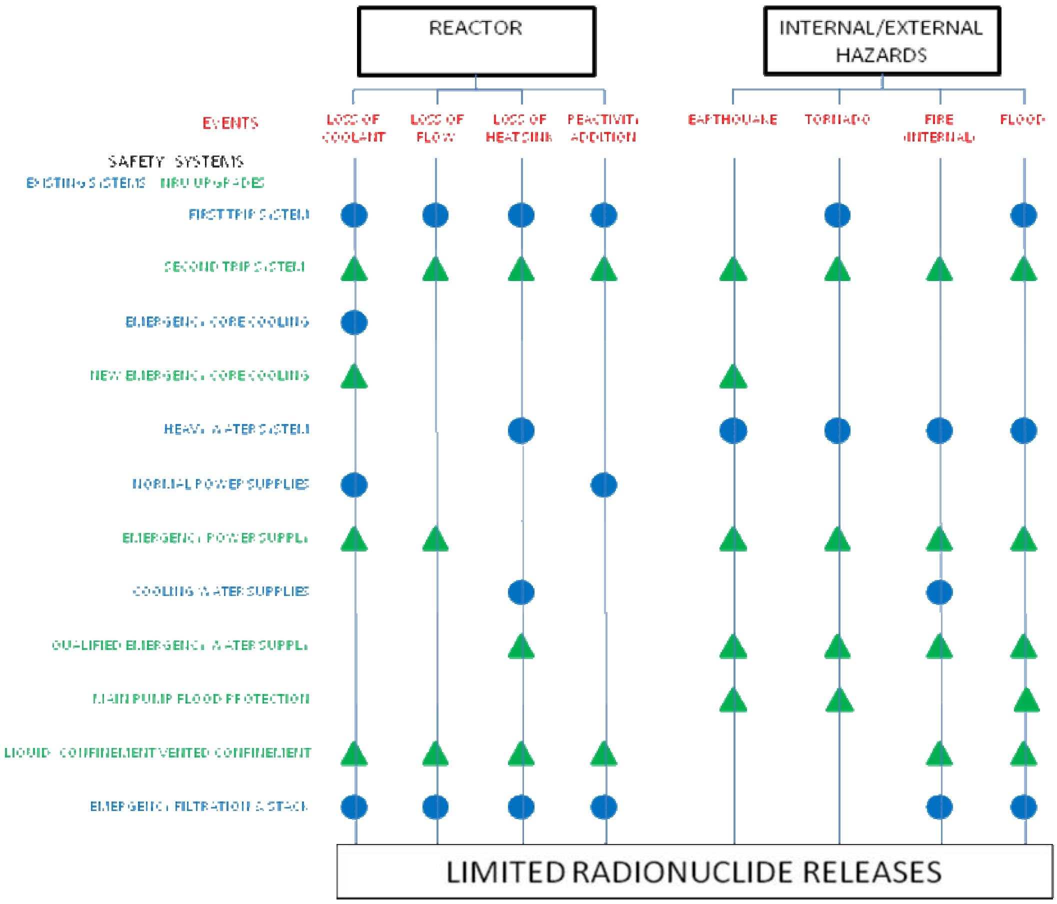

Fig 7. Event Coverage by the Existing NRU Safety Systems and the NRU Upgrades

The implementation of the NRU Upgrades enabled AECL to be confident that it could safely operate NRU beyond the year 2000. The combination of the existing NRU safety systems with the NRU Upgrades strengthened AECL's ability to demonstrate the ability to meet the four basic safety requirements for a reactor:

- It shall be possible to shut down the reactor and maintain it in that state indefinitely;
- The capability of removing decay heat from the fuel during this shut down period shall be maintained;
- The confinement structure shall continue to be capable of limiting radioactivity release; and
- Continuous monitoring of reactor safety functions shall remain available.

The NRU Upgrades Project demonstrates that improvements can be implemented on an operating reactor to improve its overall safety by addressing areas that were not part of the original design requirements. Furthermore, the NRU Upgrades Project demonstrates that physical changes in the field are often required to improve safety.

3 The Integrated Safety Review Project & Integrated Implementation Plan

The NRU Upgrade Projects had enabled AECL to demonstrate that NRU was still safe to operate resulting in the Canadian Nuclear Safety Commission (CNSC) granting Licenses to continue to operate NRU in 2003 and 2006. In 2008, AECL agreed with the CNSC to perform an ISR of NRU in support of continued operation up to 2016 [4]. The ISR follows the principles contained in CNSC Regulatory Document, RD-360, *Life Extension of Nuclear*

Power Plants [5], and the International Atomic Energy Agency (IAEA) Safety Guide NS-G-2.10, *Periodic Safety Review of Nuclear Power Plants* [6], taking into consideration that NRU is a research reactor and not a nuclear power plant.

The objective of the ISR was to determine the following:
a) The extent to which the plant conforms to modern standards and practices.
b) The extent to which the licensing basis will remain valid over the proposed extended operating life.
c) The adequacy and effectiveness of the arrangements that are in place to maintain plant safety for long-term operation.
d) The recommended improvements to be implemented to resolve safety issues that have been identified.

This objective was achieved by reviewing the Safety Factors described in [5], [6] and shown in Table 2.

Subject Area	Safety Factors	
Plant	1	Plant Design
	2	Actual Condition of Structures, Systems and Components
	3	Equipment Qualification
	4	Ageing
Safety Analysis	5	Deterministic Safety Analysis
	6	Probabilistic Safety Analysis
	7	Hazard Analysis
Performance and Feedback of Experience	8	Safety Performance
	9	Use of experience from other plants and research findings
Management	10	Organization and Administration
	11	Procedures
	12	Human Factors
	13	Emergency Planning
	14	Quality Management
Environment	15	Impact of nuclear and hazardous substances
Safeguards and Security	16	Safeguards
	17	Security

Tab 2. Safety Factors for NRU ISR

For each Safety Factor the following tasks were performed:
- Confirmation that the NRU Reactor meets the current licensing basis, and that there are existing programs to ensure that it will continue to meet the licensing basis until 2021.
- Identification of gaps between the current state of the NRU Reactor and the requirements or guidance provided in modern codes and standards.

The gaps identified in the Safety Factor reports were then combined with other known gaps that AECL had already committed to addressing resulting in a Global Assessment Report (GAR). These gaps were then assessed to determine which gaps required closure to facilitate safe operation of NRU up to, and beyond, 2021. In reviewing the gaps the ISR project was able to identify five groups into which all gaps could be consolidated. These five groups are referred to as the Global Issue Groups (GIGs) and are presented in Table 3. Within each GIG, gaps that were deemed important were identified as high priority, which resulted in a requirement for mandatory completion. Once identified, all gaps were prioritized and captured in the IIP. The IIP provides a closure schedule for all gaps; high priority gaps being addressed in the first two years, lower priority being addressed in years three to five, and long term improvements scheduled for beyond five years.

No.	GIG Title	Summary of Significant Improvements Identified in the GAR
1	Current Plant Condition and Plant Life Management	Physical improvements to plant equipment on a risk-informed priority basis to address aging effects; implement integrated plant life management processes to ensure systems are effectively monitored and maintained to mitigate aging-related degradation.
2	Managed System and Organization Effectiveness	Ensure adequate numbers of trained, qualified staff to manage, operate, maintain and support NRU; improve the management system and managed processes to ensure committed work is completed efficiently and effectively; continue with initiatives to improve safety culture.
3	Safe Operating Envelope and Safety Analysis	Implement managed processes for safety case development and interfaces with other processes; update the safe operating envelope to reflect the current safety case; address inconsistencies among safety case documents; update the NRU safety case; Severe Accident Management (SAM) program.
4	Training and Nuclear Programs	Implement improvements to radiation protection and worker safety processes, including physical improvements to reduce worker dose exposures; implement improvements to Systematic Approach to Training (SAT) based training; make incremental improvements to environmental protection; emergency preparedness.
5	Engineering and Design Changes Related to Modern Standards	Address Design Basis Recovery issues, address man-machine interface issues related to equipment tagging and labeling; address Criticality Safety Program issues; implement equipment qualification program.

Tab 3. Summary of Global Issue Groups

The ISR process adopted by AECL has been presented and is summarized in Figure 8.

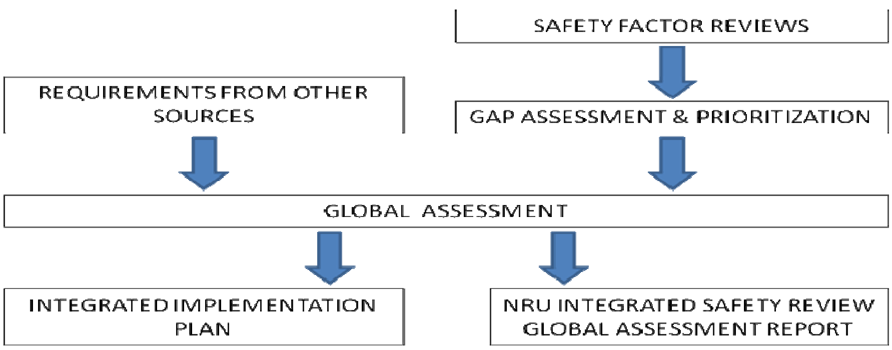

Fig 8. Summary of the ISR Process

3.1 Current Plant Condition and Plant Life Management

As an example of ISR /IIP process, this paper will now focus on GIG 1 Current Plant Condition and Plant Life Management. As shown in Figure 8, in addition to using the Safety Factor reviews, gaps were also identified from other sources. For GIG 1, these other sources were the Plant Life Management (PLiM) recommendations contained within Condition Assessments (CA) and Life Assessments (LA)[4], Phenomena Investigation Ranking Table (PIRT) Recommendations, and items captured on the NRU Risk List[5]. The gaps identified in GIG 1 fall into 2 areas:

1) GIG 1 Hardware – Physical Improvements to Plant
2) GIG 1 Programs – Equipment Reliability

The GIG 1 Hardware is focused on improvements to the current plant with GIG 1 Program focused on implementation of an integrated plant life management program to ensure physical improvements can be sustained. This paper will provide an overview of GIG 1 Hardware, providing an example of gap closure through the replacement of aged equipment.

3.2 GIG 1 Hardware

GIG 1 Hardware takes the gaps identified in the GAR and determines how the gap can be closed. The process is presented in Figure 9.

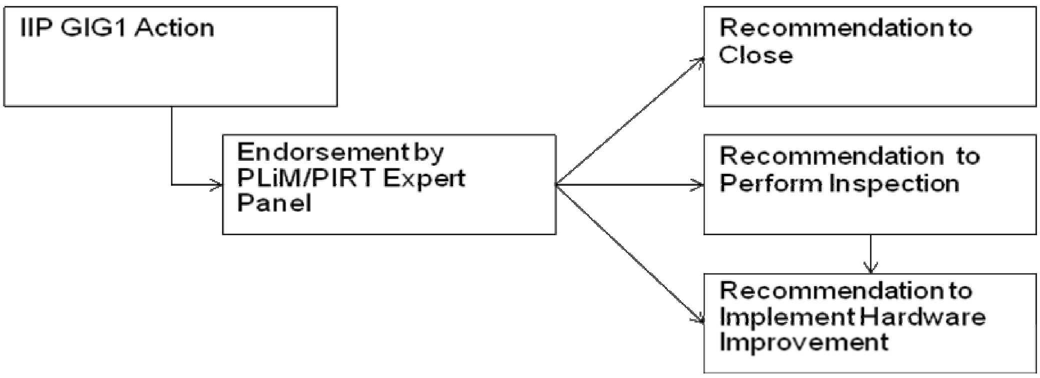

Fig 9. GIG 1 Hardware Process

The GIG 1 Hardware is focused around the need to address ageing components identified in the IIP. All IIP GIG 1 Hardware actions are reviewed by the PLiM / PIRT Expert Panel which consists of management representatives from Technical Support, Design Engineering, Safety and Licensing, and Senior Reactor Management, and is chaired by the Director of the IIP Hardware Project. The panel recommends that the action can be closed as presented, recommend that further inspections be conducted to determine the current condition of the equipment, or recommend implementing hardware improvements. The implementation of hardware improvements is not limited to replacing equipment, it may also include obtaining critical spares and/or modifications to maintenance strategies. Any recommendations that will enable the physical plant to be maintained in the future are also communicated to the GIG 1 Programs – Equipment Reliability Project..

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4 Condition Assessments (CA) are for those components that are considered replaceable, Life Assessments (LA) are for those components that are considered non replaceable and whose failure would result in the failure to continue operating.

5 The NRU Risk List was a list of known issues with specific equipment.

3.3 Case Study

The case study will focus on one of the GIG 1 hardware gaps identified during the ISR project. The case study provides details of the system assessed, the factors driving determining the priority and the opportunities that were realised by the project in addressing the gap.

3.3.1 NRU Electrical Power System

The NRU reactor electric power system integrates off-site power supplies into a multi-source power system. The off-site supplies originate from the grid, whereas the backup supplies of the multi-source system are provided within NRU. A major safety upgrade, consisting of a hazards-qualified Emergency Power Supply (EPS), was added to the NRU system (see section 2.3). A brief description of the NRU electric power system and how it responds to a loss of external power is described in this section.

The NRU electrical system uses the following nomenclature:-

- **Class I**: Non-interruptible direct current (DC) supplies with directly-connected batteries for essential auxiliaries, instrumentation, protection and control equipment.
- **Class II**: Non-interruptible alternating current (AC) supplies for essential auxiliaries, instrumentation, protection, and control equipment.
- **Class III**: Alternating current supplies to essential auxiliaries whose interruption may be tolerated for intervals of a few minutes. These essential auxiliaries are those necessary to maintain the reactor in a safe shutdown state.
- **Class IV**: Normal AC supplies to auxiliaries and equipment, for which interruptions of indefinite duration may be tolerated without compromising the ability to place and maintain the reactor in a safe shutdown state.
- **Emergency Power Supply (EPS)**: Hazards Qualified Class I, II, and III power.

When the external power supply to NRU is lost, NRU trips and DC power is provided to two pumps from a Class I power battery bank that is supported through the Class I rectifiers when the Class III reactor diesel generators are in operation. This ensures that forced cooling is always available to the fuel rods in the core. In addition to supplying supply power to the two pumps, the Class I power also supplies power to the Class II system through Motor-Generators sets. As described previously, Class II power is then used to power essential auxiliaries, instrumentation, protection, and control equipment.

3.3.2 Assessment of NRU Electrical Power System

The electrical power system was assessed as part of Safety Factor 2 of the ISR. The findings from this review were then combined with information contained in other documents such as Condition Assessments, PIRT Assessments and the NRU Issues list, and assessed as part of the Global Assessment. Following the completion of Global Assessment, the existing NRU Class I rectifiers were identified as a high priority activity and added to the GIG 1 Hardware project. The high priority ranking was based on a number of factors:

- The important role the rectifiers play in responding to a loss of external power[6]
- The existing rectifiers years (though still functioning without major incidents) had exceeded the manufacturers recommended service life of 15 years
- Advances in rectifier design has resulted in improved performance and reliability
- New rectifiers can eliminate the need for operator intervention, a recognized process weakness with the existing Class I rectifiers.

[6] NRU typically experiences 8 loss of external power events a year

The PLiM/PiRT Expert Panel endorsed the recommendation that the rectifiers be replaced with modern equivalents and that the replacement take place without significantly impacting reactor operations. (As previously mentioned, due to NRU's role as a supplier of medical isotopes, the implementation of any changes must not require extensive outages). Given the high priority ranking; replacement of the Class I rectifiers was a year one IIP activity.

3.3.3 Implementing Recommendations

Following the endorsement of the PLiM/PiRT Expert panel to replace the existing Class I rectifiers a design team was formed. The design team took the requirements for the existing Class I rectifiers and revised them to take advantage of some of the new features modern rectifiers are able to supply; two of these new requirements were:

1) No operator required to transfer loads from batteries to Class I rectifiers
2) No operator intervention required to prevent diesel overload

The new Class I rectifiers are equipped with Adaptive Current Limitation (ACL) feature which allows the rectifiers to remain connected to the Class III bus, since ACL ensures that the diesel cannot be overloaded. In addition, the new rectifiers ramp up to full load automatically over a preset period of time, without any operator intervention. Both of these features eliminate the need for operator action, thus reducing the potential opportunities for human error. In addition, the new rectifiers are able to compensate for temperature when charging the Class I batteries. Optimal charging of the Class I batteries further improves the reliability of the Class I system.

The need to not negatively impact NRU operations is being accomplished by taking advantage of the redundancy present in the NRU power system. The Class I system has two Class I rectifiers, either of which can carry 100% of the load enabling the rectifiers to be replaced one at a time. Figure 10 shows the existing and new Class I rectifiers within NRU.

EXISTING NEW

Fig 10. NRU Class I Rectifiers

3.4 Impact of the ISR on NRU

The GIG 1 Hardware case study that has been presented demonstrates how AECL is using the IIP resulting from the ISR process to identify, prioritise and make physical changes to the plant to address ageing issues. The case study has shown that making physical changes to the plant also provides an opportunity to increase safety by identifying how modern day equipment can be used to address operational constraints resulting from old technology. As with the NRU Upgrades Project, AECL has again demonstrated that physical improvements can be implemented without adversely impacting reactor operations.

This case study focused on the hardware improvements, but it must be recognized that GIG 1 Programmatic elements provide an equally important role, as they provide the means to maintain the gains realised by GIG 1 hardware.

4 Summary

This paper has provided an overview of the approaches taken by AECL to ensure that the NRU Reactor remains safe and reliable to operate now and for the near future (2021 and beyond).

The paper has demonstrated that in order to safely extend reactor life it is important to address both design issues (NRU Upgrades Project) and ageing management issues (ISR Project /IIP).

The ability of NRU to support both nuclear and medical industries has been the driving force behind NRU operation and why AECL is committed to safely extend the operating life of the reactor.

5 References

[1] Atomic Energy Control Board, *Requirements for the Safety Analysis for CANDU Nuclear Power Plants*, Consultative Document C-6, 1980.

[2] International Atomic Energy Agency, *Basic Safety Principles for Nuclear Power Plants*, *a report by the International Nuclear Safety Advisory Group*, No. 75-INSAG-3, 1988.

[3] Walker J R, et al, *Upgrading the NRU Research Reactor*, Nuclear Engineering International, 1993.

[4] Canadian Nuclear Safety Commission, *Protocol for National Research Universal Licensing Activities*, July 15, 2008, CCM-2008-000208, E-Doc#3259232.

[5] *Regulatory Document Life Extension of Nuclear Power Plants*, RD-360, February 2008.

[6] IAEA Safety Guide, *Periodic Review of Nuclear Power Plants*, NS-G-2.10, 2003.