SAFETY BENEFITS FROM CANDU REACTOR REPLACEMENT A CASE STUDY

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Abstract

Both total core replacement and core retubing have been used in the CANDU® industry. For future plant refurbishments, based on experience both in new construction and in recent refurbishments, the concept of total core replacement has been revisited. This builds on practices for replacement of other large plant equipment like boilers. The Bruce CANDU reactors, with their local shield tanks built around the Calandria and containment closely located around that Calandria Shield Tank Assembly (CSTA), are believed to be good candidates for core replacement.

A structured process was used to design a replacement CSTA suitable for Bruce A use. The work started with a study of opportunities for safety enhancements in the core. This progressed into design studies and related design assist safety analysis on the reactor. A key element of the work involved consideration of how verified features from later CANDU designs, and from our new reactor design work, could be tailored to fit this replacement core. The replacement reactor core brings in structural improvements in both calandria and end shield, and safety improvements like the natural circulation enhancing moderator cooling layout and further optimized reactivity layouts to improve shutdown system performance.

Bruce Power are currently studying the business implications of this and retube techniques as part of preparation for future refurbishments. The work explained in this paper is in the context of the safety related changes and the work to choose and quantify them.

1. Introduction

Both total core replacement for smaller reactors and core retubing have been used in the CANDU industry. In 2008 Bruce Power (B.P.) commissioned a feasibility study to see if it was technically feasible to replace a Bruce A station core and whether there would be advantages in this alternate approach. The study utilized experience both in new construction and in recent refurbishments. It built on practices for replacement of other large plant equipment like boilers. The Bruce CANDU reactors, with their local shield tanks built around the Calandria and containment closely located around that Calandria Shield Tank Assembly (CSTA), are believed to be good candidates for core replacement. The paper focuses on the work carried out to demonstrate that quantifiable safety benefits could be obtained with a modified design and with no negative consequences.

2. Project development

An important element of the proposed refurbishment of the Bruce units is keeping them in compliance with modern standards, especially with regard to safety. This process is typically documented and formalized against the requirements laid out in CNSC Regulatory Document (RD) 360 [1]. RD360 is multi-facetted and involves carrying out an Integrated Safety Review with 14 elements. The safety review is then collated in two main documents. These are

- The Global Assessment Report (GAR) that seeks to integrate all of the 14 elements and ensure that any interactions between them are considered
- The Integrated Implementation Plan (IIP) that identifies the program for any mitigation or improvement action to be completed

Internal drafts of the Bruce 3&4 Global Assessment Report and Integrated Implementation plan, which assumed fuel channel replacement, were used to understand plant wide issues that would relate to the CSTA replacements study. Given this work was concerned with just the reactor core, a subset of those steps and total scope was then used for the study, rather than an update of the complete plant unit document. It was recognized that replacement of the reactor core allows potential implementation of further safety improvements which would not be feasible to perform using a conventional fuel channel detube/retube approach, so the project added items to the proposed improvements contained in the current draft documents.

This paper highlights the process followed to show that the requirement toward continuing Safety improvements had been addressed if a new CSTA was to be constructed. It primarily focuses on the route towards acceptable design modifications followed, showing the amount of Analysis work that was required to confirm positive results and to ensure that there were no negative impacts from the proposed changes.

3. Process developed

A Bruce Power engineering department procedure was prepared that defined the steps that would be followed to produce an approved design basis for a modified CSTA.
Figure 1 shows the process Flow Chart for that procedure.

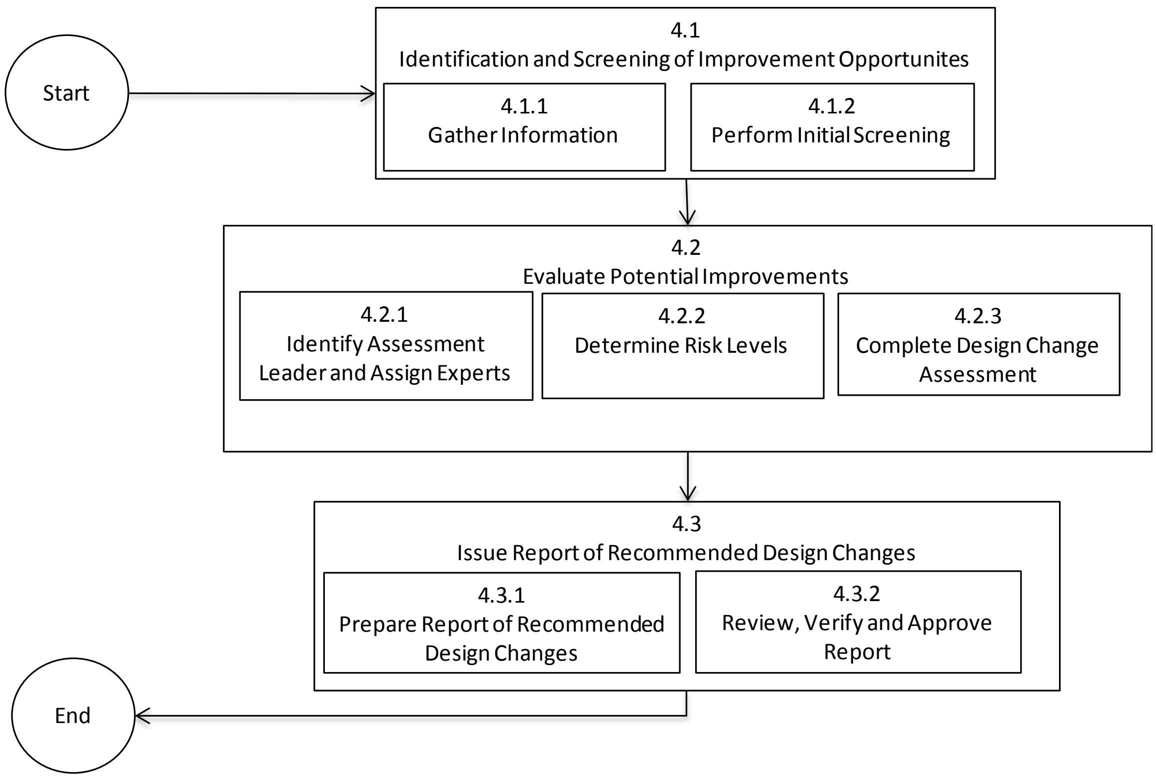

Figure 1 Process flow chart

The Key principles that must be met through performance of the process were:

- The plant, with the final list of improvements, must meet the relevant regulations and the current site licence,
- defence in depth is maintained and,
- safety margins are maintained or improved.

As per the process flow chart the assessment started with a safety related screening of opportunities for improvement. Identification of improvement opportunities was done by a rigorous evaluation of the elements which drive the safety performance as analyzed in the safety report. These were centered on areas where a change in core design could improve performance, and also by comparison with the safety performance of other CANDU reactors. The work considered an evaluation of the Bruce A accident scenarios with lowest safety margins as part of a feature-by-feature comparison of the Bruce A reactor core safety features with later CANDU reactors. A diverse panel of experts was used to look both at possible benefit of each potential improvement for Bruce A core, as well as the risk and cost. The expert panel contained a cross section of expertise with many tens of years of experience from Bruce Power, AECL and their contractors.

Concurrent with the safety review, OPEX was reviewed and operational and design improvements were considered. Operations data and design data from newer CANDU reactors was reviewed to assemble a list of potential hardware changes beyond those driven directly by Safety impact. From the OPEX, operations and design information and the review of safety performance, the information was fed into an activity to produce requirements for and then to design the replacement reactor. The most significant areas for improvement identified were SDS1 depth and moderator cooling

4. Safety analysis plan

Having collated a list of items with perceived safety benefits, and started reactor design work, a Safety Analysis Plan was produced by a group consisting of representatives from Bruce Power Nuclear Safety and Support (NSAS) and BP projects together with AECL® and Candesco.
The safety analysis plan mapped out the safety support requirements into 4 phases. The first series of calculations, design assistance in nature, were aimed at supporting the conceptual and detail design work to allow sizing and layout decisions to be made in the reactor core. The second series of calculations were more of a verification nature covering postulated accident cases and comparing the design assist results with the existing Bruce A core to document the improvements. The principle objective of these phases was to enable NSAS to support the proposition that the changes in the CSTA would not have any negative impact on the Safety Case and thus remove a major risk item affecting a decision to proceed with the replacement approach. Phases 3 and 4 would be completed if a decision was made to go ahead with a replacement approach, and would cover production of a revised Safety Report and any proposed performance changes such as increases in power or changes in life expectation.

5. Modelling safety performance

Modelling the performance of the coupled interaction of the various reactor and process systems around it is vital to both design and safety analysis work. A variety of codes are used, primarily those agreed as Industry Standard Toolset (IST) codes. [2] These verified codes make use of a mixture of theory, reactor data, and data from specialized facilities like the RD14 test loop at AECL Whiteshell and the ZED2 physics facility at AECL Chalk River Laboratories to model the performance of key systems and components in CANDU reactors. These include codes like:

- DRAGON IST which does three dimensional neutron transport calculations.
- WIMS IST A general purpose reactor physics program for core physics calculations. It contains verified lattice models of the reactor core.

- Reactor Fuelling Simulation Program (RFSP) which uses model inputs from WIMS or Dragon and is Capable of generating nominal power distributions and simulating reactor operations, including refuelling and burnup steps.

- Canadian Algorithm for Thermalhydraulic Network Analysis (CATHENA) Capable of analyzing two-phase flow and heat transfer in piping networks. This was used for moderator and end shield system performance analysis

- TUF a two phase flow thermohydraulics code with integrated neutron transport which is hard coded to reflect the configuration for a given reactor and concentrates on the heat transport system response.

- MODTURC, a three dimensional computation fluid mechanics code based on porous media approximations written for moderator thermohydraulics calculations

These codes are used individually and in concert using verified models and input data sets from Bruce Power with some new models and modified data sets created by AECL.

6. Safety calculations

In the safety support calculations the codes modelled the physics responses, the thermohydraulic responses, and coupled responses since the two responses directly affect each other. AECL and NSAS specialists cooperated to pick the most suitable toolsets to both allow comparison to current Bruce safety analysis but also provide the extra data that comes from some of the available design focused codes. For the new reactor core existing fuel channel modeling and input data sets were obtained from verified Bruce Power TUF models. Using WIMS some simple core optimization calculations were done before using DRAGON to come up with incremental changes to the modified core from the existing verified Bruce Power Models. This then allowed the reference Bruce Power RFSP model to be updated and run with TUF to get the updated response of the new core. Fuelling runs were performed for start and 8250 EFPD to get the bounding conditions for break analysis.

For the moderator performance work a new model was produced in MODTURC which was compared to the CANDU 9 quarter scale test facility measurements [3] and had sensitivity numbers run on variations of inlet and outlet nozzle positions. This model used as input the physical geometry modeled in CAD and RFSP runs to get both the heat lost from the fuel channels and also the nuclear heating into the moderator fluid from the reaction in the core. Simple network calculations and CATHENA were used to model the overall system flows around the moderator circuit and match the new calandria layout and piping to the existing system equipment to rematch the system flows to those currently existing. A CATHENA model of the end shield was also produced both to allow with a simple network calculation the system flows to be matched up as well as to be used for larger deterministic reactor level calculations of response of the new design.

Having built up these core models to allow steady state numbers to be worked out for key values like shutdown reactivity margin, fuel bundle and channel powers, and key temperatures like moderator steady state, they were then used for analyzing postulated accident transients. Bounding flux tilt and creep values were chosen appropriately for individual cases. This included an In core Loss Of Coolant Accident (LOCA) run where a break of a new uncrept pressure tube and subsequent calandria tube break was run through a coupled TUF and RFSP to look at the system response as the event progressed. This was compared to the existing unit performance to show both the significantly improved shutdown margin as the control system reacted to the event but also to compare responses as the event progressed. A large out of core accident was also simulated using conditions from later in life to see the progression of that accident and make sure the dynamic response of shutdown system 1 had not significantly changed. The new deck layout had moved the rods outward in the core giving more of them slightly further to travel for the important gate 1 way point. For this case the increased worth of the new shut off rod layout did not directly impact the event but simply gave a larger margin at the end. The impact of the small overall gate 1 change was also quantified and shown to be acceptable.

The analysis results were compared to the analysis of record for the current Bruce A to investigate the effect of the design changes. Overall it was shown the changes do give significant safety improvements on a number of postulated events with no negative effect on the plant safety as a result. As the design work is moved forward further detailed safety support calculations are planned on accident transients to support the formal stress analysis for final registration and build towards an updated safety report.

7. Some safety outcomes

7.1 SDS1 depth

The evaluation identified that SDS1 depth was limiting for some accident scenarios and so should be considered for improvement. This led into a conceptual evaluation of the shutoff rods and their layout to see if improvements could be achieved. In that evaluation initial simplified physics runs showed that, with several new layouts, a significant increase in worth could be gained via a movement of the absorber rod positions in the core, made possible by the removal of the booster assemblies

The more detailed safety support calculations included a comparison of many shutoff and control absorber positions. With these calculations the reactivity device layout was optimized to keep the zone control positions and the same detector signals to the operators already seen for units 1 & 2 while improving the step back symmetry and increasing shut off rod depth.

Figure 2 shows a comparison of the current and proposed configuration of SORs and MCAs. The top shows the proposed modified design while the bottom half shows the current Bruce A design.

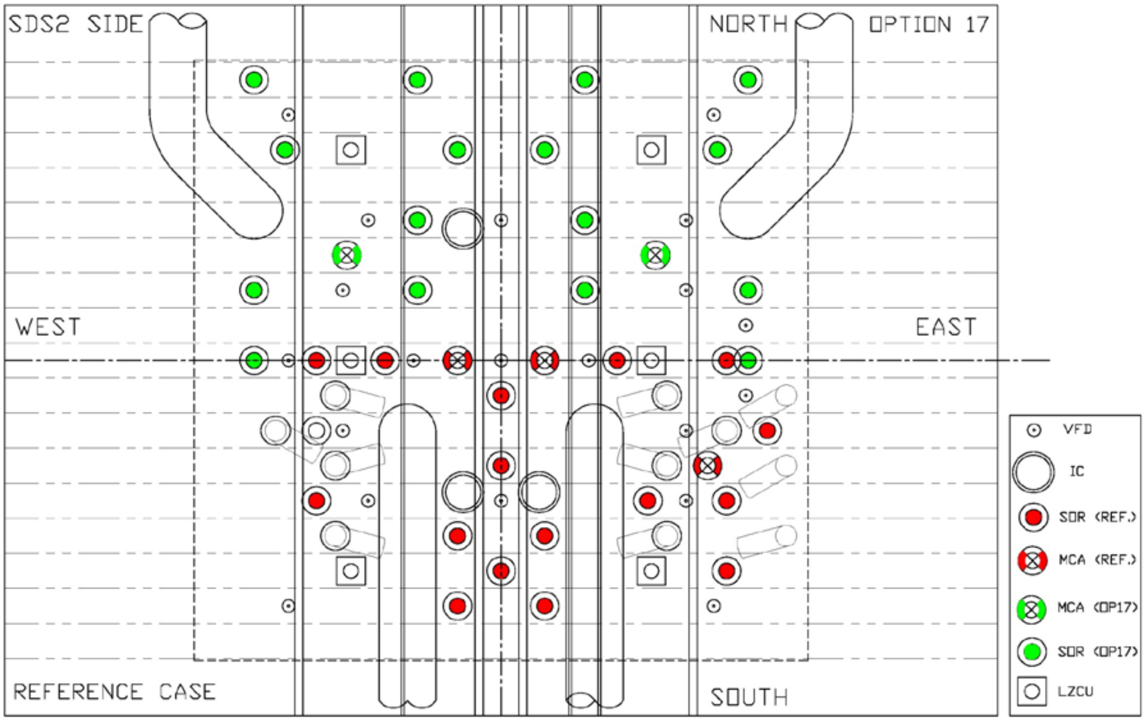

Figure 2 Comparison of reactivity deck equipment locations

7.2 Moderator cooling

Moderator cooling was originally designed for Bruce A to cool booster fuel first, and then circulate through the calandria vessel second. The booster fuel itself had been removed from use in Bruce A cores several decades ago but the flows remained as per the original design. Removal of the assemblies allowed the moderator cooling design to be redesigned using the latest verified concepts. This brought in a design that reinforced the normal nuclear heating currents to move coolant through the core thus bringing in a more efficient cooling, which in turn greatly reduced the temperature range through the core, and with that made the performance in postulated accidents more predictable and controllable. In past work [4][5] the original moderator temperatures were modeled based on temperature probes inserted into the Bruce 3 core and then used to verify a software model to fully analyze that original momentum based cooling scheme. The new buoyancy based scheme was proved [6] through a combination of a quarter scale model and associated with that a verified computer model. Figure 3 shows the predicted temperature distribution in the replacement calandria vessel as a result of the move to buoyancy based cooling.

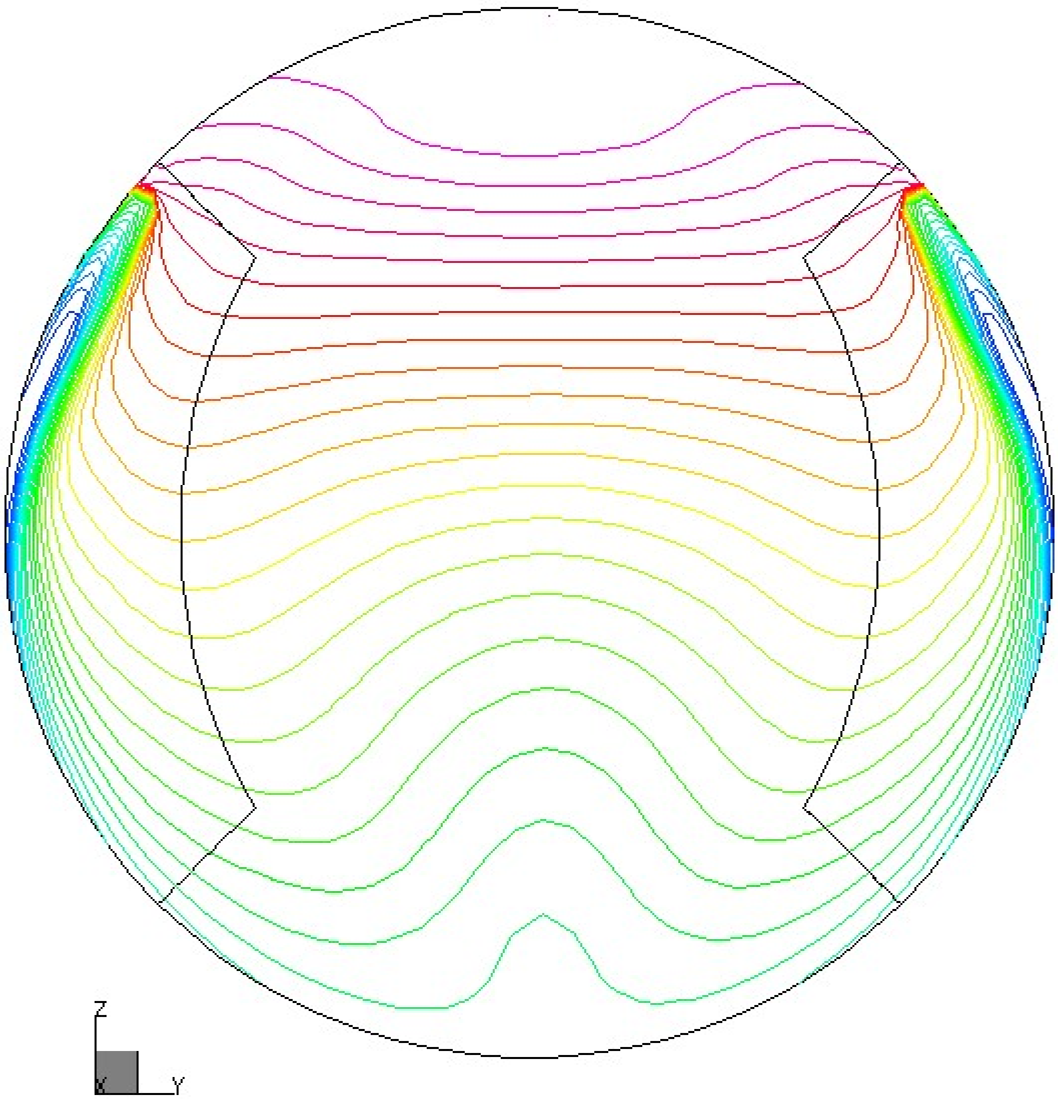

Figure 3 Improved moderator temperature distribution

The response of the moderator cooling circuit was modeled showing both several degrees of extra subcooling margin and with the smaller range of temperature in the moderator more even reactivity response and the ability related to this to obtain small improvements in reactivity coefficients that affect postulated accident progression.

8. Design

In addition to these changes the review process identified others for more detailed investigation, minor LISS nozzle clearances, and improved guide tube venting to mitigate hydrogen. The list of safety related changes was combined for this new CSTA with a larger list of many non safety related OPEX driven improvements to the CSTA. A further item from that first safety list on shield tank venting was adopted after detail design work. The additional items were primarily concerned with detail design improvements like changing the weld details on the nozzles on the calandria shell to improve alignment of thimbles containing reactivity devices, and a move to a simpler end shield structure with two sheets and a simpler cooling layout. A number of these detail design changes bring in operational, and to a lesser degree safety benefits. For example, improved access for operation and maintenance to the safety related items on the reactivity mechanism deck, and with the revised end shield structure a potential for thermosyphoning should coolant flow be interrupted. The safety assessment and rational behind the selections were recorded in a document which was reviewed with CNSC in 2010 gaining general acceptance of the work

Examples of some of the other changes are Fuel Channel design end fitting bearings, Fuel Channel restraints, LISS nozzle profile, RCU locator details, and Moderator level measurement.

Following documentation of the requirements the more detailed design work included both structural analysis and a more in depth safety support calculations. Using structural analysis the updated detail design of the calandria, end shield and shield tank was updated to be fully compliant with modern codes and standards. This provided the basis for a classification and preliminary registration submission for this replacement CSTA. A number of challenges were encountered during this work as the original construction and registration of the current vessels was completed prior to the introduction of the "modern approach to coding. The first of two main issues was associated with a currently designed moderator system being Class 1 equivalent and the new code not allowing bellows in a Class 1 circuit. Bellows are obviously required in the SOR guide tubes to allow for thermal movements. The second issue was a proposal to treat the replacement vessel as a component in its own right rather than its being the collection points of a number of discrete systems. Resolution of these issues remain on-going at the time of writing this paper.

Given the assembly of the CSTA was modified to suit the new vertical installation a revised access path into the shield tank during installation was needed. This was fashioned to be capped off after commissioning using a rupture disc to prevent pressurization of the shield tank and preserve its integrity in severe accidents.

The design and its supporting analysis was recorded in a set of arrangement drawings, and related design and design support documents. These included the specification and analysis required for provisional registration, documents to allow a detailed pricing of the replacement CSTA to be performed, and a classification document with related pressure boundary drawing for regulator discussion on how the replacement CSTA would be constructed to current codes and incorporated into the existing process systems.

9. Summary

A structured process was used to design a replacement CSTA suitable for Bruce A use [7]. The work started with a study of opportunities for safety enhancements in the core. This progressed into design studies and related design assist safety analysis on the reactor. A key element of the work involved consideration of how verified features from later CANDU reactor designs, and from new reactor design work, could be tailored to fit this replacement core. The replacement reactor core brings in structural improvements in both calandria and end shield, and safety improvements like the natural circulation enhancing moderator cooling layout and further optimized reactivity layouts to improve shutdown system performance. Bruce Power is currently studying the business implications of this replacement approach against retube techniques as part of preparation for future refurbishments. The work explained in this paper is in the context of the safety related changes and the work to choose and quantify them.

The paper demonstrates the level of detail considered to ensure that risks around any decision to proceed with a CSTA replacement program would be minimized as they relate to reactor Safety issues.

10. Acknowledgements

This work was carried out in AECL and Bruce Power. The Authors would like thank their colleagues who demonstrated exceptional cooperation between NSAS and AECL specialties, and performed the detailed analysis discussed here. A large team of several tens of staff was involved in production of the many documents and drawings and the verification reviews performed as part of it.

11. References

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