DEVELOPMENT, QUALIFICATION TESTING AND COMPUTER CODE REQUIREMENTS FOR A REFITTABLE SUPERCRITICAL WATER PRIMARY HEAT TRANSPORT SYSTEM

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Abstract

The power output of current Pressurized Heavy Water Reactors (PHWRs) can be doubled by refitting them with a higher temperature and pressure primary heat transport system [1]. The refit involves a modular replacement of the primary heat transport system and fuel channels that is similar to work undertaken during current refurbishments. The refit is enabled by new fuel channel materials and a new channel closure and fueling machine that require development testing. This paper examines the testing, computer code and standard development needed as input to design the first refit. The types of test facilities and computer code development needed are discussed along with possible sources of supply, while maintaining a focus on capabilities currently available in Canadian industry. The paper provides a roadmap to finalize technology, analytical methods and standards for first refit implementation.

1. Introduction

The demand for electricity is increasing as a result of increased electrification of transportation and heating. There is also a potential further increased demand for electricity to produce electrolytic hydrogen for heavy transportation, steel making and other process industries. This will require substantial new generating sources. The options for new non-carbon dioxide generation are limited, particularly in northern latitude locations that are sub-optimal for solar and where there may be no steady wind sources. Where there are existing PHWRs, a refit to increase power output is an ideal candidate for new generation as described below.

The power output of existing PHWR / Candu type plants can be doubled through a modular replacement of the primary heat transport system (PHTS) [1,2], with a higher temperature (max 625 °C) and pressure (Max. 25 MPa) PHTS, referred to hereafter as a PHWR refit. There are several advantages to obtaining increased nuclear generating capacity through refitting existing PHWR / Candu plants:

- The construction time is much shorter that of large plant new builds (3 versus 10 years)
- No new site is required, thereby enhancing land use and simplifying approvals
- Project management relies on methods proven during refurbishments
- No new operating organization is required

There are other options for new nuclear generation such as small modular reactors (SMRs) or updated versions of the existing large nuclear power plants. These can have drawbacks not encountered by the PHWR refit. SMRs have relatively low power output requiring many units for regions of concentrated electricity demand. While construction may be faster than large nuclear power plants, the economics of building and maintaining a large number of plants is uncertain. The PHWR refit has only a slightly increased number of components compared to existing PHWRs. With double the power output, per MWh operating and maintenance costs will improve over existing PHWRs. Updated versions of large power reactors (e.g. EPR, AP-1000) require long new build construction times and have encountered significant cost and schedule overruns on their first plants. The PHWR refit can be constructed in much shorter times while benefitting from project management experience developed during PHWR refurbishment.

While a PHWR refit has considerable advantages, it requires investment to bring the technology to an engineering-ready status. The investment is required for the development of component manufacturing, development and qualification testing, new analytical tools and standards specific to the new technology. The remainder of this paper will describe the specifics of what is required and the potential for Canadian industry participating in this development work.

2. Organization of Development

Development of the refit involves a combination of activities that can be performed at the same time (in parallel), that can be performed one after each other (in series) and some which are inherently iterative. An efficient development process needs to consider how to organize activities so that iteration is minimised, and continuing investment oversight can be easily exercised.

The development process can be viewed as progressive investment to achieve usable product and reduce the risk that technologically difficult obstacles remain with the project. Ideally, the beginning of development should focus on the lowest cost activities that provide the greatest reduction in technology risk (i.e. the risk that difficult and costly-to-solve problems will be encountered).

The planned duration of the development project is also important. A shorter planned development necessitates that more development be performed in parallel. This can increase iteration and development project costs.

The descriptions in this paper envisage an order to development activities that should reduce iterations and costs. This order is illustrative only. There are other ways to organize development depending on investment approaches and factors such as the availability of facilities and development partners.

3. Manufacturing Process Development

Manufacturing process development needs to be done early in the development program to make the as-manufactured items available for other types of development testing. There are three reactor in-core components that use materials that are new to in-core applications and require new or altered manufacturing processes. These are the fuel cladding, the pressure tube and the pressure tube insulator. The first two of these components are expected to require substantial manufacturing process development. Other components that will require manufacturing process development are the fuel pellets, pressure tube to end fitting transition piece, the end fitting inboard hub and, potentially, the outlet feeder tubes. The first three manufacturing development activities are best conducted up-front and concurrently to condense the development schedule.

The fuel cladding and fuel bundle structural material is single crystal sapphire. This material was selected for its unique combination of properties: relatively high strength, high elastic constant, very high oxidation corrosion resistance, largely known reactor irradiation properties, relatively high thermal conductivity, good thermal and irradiation expansion compatibility with uranium dioxide fuel, relatively low thermal neutron absorption and high thermal radiance transparency [2].

Other materials that have been examined include stainless steels, nickel alloys and silicon carbide. While the choice of a metal would seem natural given the successful use of metals in coal fired supercritical water plants, nuclear reactors face an even more challenging corrosion environment due to in-core radiolytic dissociation of supercritical water [3]. The use of sapphire, an oxide ceramic, provides the potential for better corrosion resistance [4] than currently studied metals. This significantly adds to our confidence in the fuel cladding's corrosion resistance prior to building a prototype plant. Additionally, none of the alternate materials provide sapphire's thermal radiation transparency that is desired for the PHWR refit safe dry channel decay heat transfer [1,5].

While single crystal sapphire properties are well suited to a supercritical water reactor application, single crystal sapphire cannot be easily manufactured to tight dimensional tolerances. Since it is a ceramic single crystal, it must be grown using crystal growth methods. Tubular shapes are grown using the edge defined film fed growth (EFG) method. This produces a tube with facetted surfaces around the circumference. Following EFG growth, a part can be ground to achieve a smooth circular surface, but this is very expensive since sapphire is a very hard material.

High temperature fine forming will be used to produce high tolerance cladding tubes at production scale. This requires refinement of the EFG process to produce cladding tube blanks and development of the high temperature fine forming process. A laser eutectic joining process will need to be developed to join end cap to the cladding tubes, to join fuel elements to the end plates, and to join bundle appendages to the bundles. These processes could be developed in Canada but, at least the first two processes, would be more efficiently developed in cooperation with the current EFG sapphire manufactures. The two large scale EFG sapphire manufacturers are located in the United States and in Japan. Laser joining of sapphire has been developed in Germany for solder type joints.

The pressure tube material is silicon carbide fibre reinforced silicon carbide composite. This material was selected due to its combination of high directional strength, low neutron absorption, largely known reactor irradiation properties, and reasonably good thermal conductivity. These properties make it well suited for a PHWR pressure tube material. While silicon carbide composites have been manufactured into fuel cladding for experimental purposes [6], there is limited experience with large scale reactor components. A zirconium alloy, Excel, has also been examined for pressure tubes. This material is the planned pressure tube material for the Canada Gen IV supercritical water reactor (SCWR) [7]. Excel was not selected for the PHWR refit due to its higher neutron absorption and the potentially higher directional strength of silicon carbide composites.

The refit pressure tube will need to withstand 25 MPa internal pressure, resulting in high hoop stress. The proposed fibre lay-up maximises hoop tensile strength through an alternating off-set filament lay-up. The pressure tube manufacturing process will need to be trialled and iterated with various tests of pressure tube properties. While pressure tube testing is expected to be performed at existing Canadian facilities, the manufacturing trials are best conducted at existing silicon carbide composite manufacturing facilities located primarily in the United States and Japan.

The pressure tube insulator material is fused silica. This material was chosen for its low thermal conductivity, its high thermal (infrared) radiation transmissivity, and its potential for high irradiation dimensional stability. Yttria stabilized zirconia was also considered. This is an insulator considered for the Canada Gen IV SCWR [7]. Yttria stabilized zirconia has successful irradiation experience as an insulator in the blowdown test facility experiments. Yttria stabilized zirconia was not chosen for the PHWR refit because it does not have the thermal radiation transparency of fused silica that is desired for safe dry channel decay heat transfer [1,5].

Manufacturing development is required to produce the required insulator dimensions, density and develop a coating process. The manufacturing process development will need to be iterated with pressurization and irradiation testing to achieve the desired dimensional stability, coating reliability and optimize irradiated thermal radiation transparency. The manufacturing process development is best done at existing manufacturers. These are located in Germany and the United States. Pressurization and irradiation testing can be performed by entities in Canada, with accelerated fast neutron irradiation done in the United States, if required.

In addition to the three previous in reactor components, a more limited manufacturing development will be required for the fuel pellets, the pressure tube to end fitting transition piece, end fittings and outlet feeders.

The refit reactor fuel pellets are slightly enriched uranium dioxide pellets. The microstructure and density are expected to be similar to existing PHWR fuel pellets with minor geometrical changes. The most significant difference is likely to be the tighter manufacturing tolerances required, particularly diametral tolerance. While tight tolerance fuel has been manufactured for experiments it has not been used for production PHWR fuel. The manufacturing development consists of economical production scale manufacturing of fuel pellets to tighter tolerances and greater quality control testing. An increase in automation is expected to be part of this development. Fuel manufacturing development is best conducted by existing Canadian fuel manufacturing facilities supported by research laboratories.

The pressure tube to end fitting transition piece is made from Excel zirconium alloy, an existing alloy with good strength and irradiation properties. The development work is required to form a mechanical joint with the pressure tube hub, the end shield bore and the inboard hub of the end fitting. The development is required because the tube thickness is significantly greater than the current pressure tubes that are rolled into the end fitting hub. The preferred joining method is rolled joining or a variation thereof. Possible approaches are successive rolling, potentially with induction annealing between rolls. Alternative methods are hydrostatic expansion and explosive expansion. The development work is best performed by a Canadian nuclear manufacturer.

The end fitting inboard hub needs to be higher strength than the current end fitting to accommodate in-situ rolling of the Excel transition piece into the hub. This higher strength could be achieved with a different stainless-steel alloy and heat treatment than the current end fitting. The material selection also needs to consider the pressure boundary, sealing and fuelling machine grappling functions of the end fitting. The manufacturing development of the end fitting inboard hub is best performed by existing Canadian manufacturers and laboratories familiar with end fittings.

The outlet feeders are exposed to irradiated high pressure and high temperature supercritical heavy water fluid at high flow velocities. It may be possible to use existing alloys intended for ultra-supercritical water coal plants for the outlet feeders. The maximum refit PHWR feeder temperatures are less than those encountered in ultra supercritical water plants, however the effects of water irradiation may render these materials unsuitable due to high corrosion rates. If high corrosion rates are encountered during testing, then either new materials or internal coatings will need to be considered for the feeders.

4. Materials Property Testing

Most materials planned for the refit have well established material properties. The exceptions are the materials subject to manufacturing process development where the as-manufactured properties may vary from the standard property data available for 'off-the-shelf' forms of the materials. Also, material properties of the heavy water coolant will need to be supplemented at much higher pressure and temperature than in existing PHWRs.

The three materials subject to extensive manufacturing process development (i.e fuel cladding, pressure tube and insulator materials) will require a determination of the full range of design relevant properties for both the irradiated and non-irradiated condition.

There is one material property that is absent for off-the-shelf sapphire: neutron-irradiation-induced creep. There are studies of sapphire growth and deformation under neutron irradiation but these studies do not involve applied stress. There is also considerable data on thermal creep of sapphire under load but these studies do not involve neutron irradiation. It is a priority to test as-manufactured sapphire for neutron irradiation creep. The cladding is almost always under compressive load during neutron irradiation in the reactor. Any neutron irradiation creep that occurs is likely to counteract the substantial neutron irradiation growth. Thermal creep of sapphire in the principal stress axes is only present at temperatures well above the operating temperatures in the reactor.

The testing of non-irradiated properties of the as-manufactured reactor core materials may be performed at the manufacturers' facilities or at laboratories in Canada, whichever is more convenient. The irradiation of components is discussed in the following section and is best performed at facilities in Canada or the United States depending on the desired fast neutron fluence. Post irradiation testing is best performed at Canadian laboratories.

The fuel pellets are expected to have the same material properties as the existing PHWR fuel pellets with the possible exception of the fuel emissivity. Since a tighter tolerance is needed, the fuel pellet grinding may result in a different surface finish affecting emissivity. Fuel pellet material property measurements are best made at a Canadian fuel manufacturer of nuclear laboratory.

The transition piece ductility and hardness will be affected by the mechanical joining process. These properties are best measured by the manufacturer developing the mechanical joining process.

The end fitting inboard hub hardness and yield strength will be affected by heat treatment and these properties will need to be measured. Additional properties may also need to be measured if the selected alloy does not have existing AMSE section III uses. The end fitting properties measurements are best performed by the manufacturer or a Canadian materials laboratory.

Heavy water properties were measured by Atomic Energy of Canada Limited (AECL) for heavy water conditions in existing PHWRs. The supercritical refit heavy water coolant operates at higher temperature than the properties originally measured by AECL. While conceptual design calculations have been performed using extrapolated properties, accurate engineering analyses will require heavy water property measurements at higher temperatures than those currently available. These measurements could be performed at a Canadian materials laboratory or at a Canadian university with appropriate equipment.

5. Component Development and Testing

The supercritical refit requires a number of new components that are not present in existing PHWRs. These components need to be developed to establish a viable detailed design and reliable operation under the supercritical refit operating conditions. The components that require the greatest development work are expected to be the in-core components that will require irradiation testing to prove out in-reactor performance. Since irradiation testing will require a significant amount of time and experimental resources, it is a priority to commence testing of these components early in the development program. The description of component testing is generally ordered with the high priority component development activities described first.

The fuel element development testing will need to be preceded by manufacturing process development for the fuel cladding tubes, the fuel pellets and the fuel cladding tube to endcap joining process. Once the manufacturing is complete, the fuel elements will need to undergo mechanical and thermal testing. The mechanical testing will determine dimensional uniformity and properties such as compressive buckling resistance, impact resistance, and potentially tensile strength at elevated temperatures. The thermal testing will test properties such as thermal irradiance at high temperatures, onset of thermal creep and thermal creep failure criteria.

Irradiation testing of fuel elements and other reactor components will require experimental facilities. There are currently no experimental facilities in Canada for fuel irradiation testing and there are very few facilities available internationally. Under these circumstances, it is proposed that fuel elements could be irradiation tested in an insert in the calandria of an operating Canadian reactor. While this may be perceived to be a significant intrusion to electricity production in the reactor, it is not that different from isotope production routinely performed in Candu / PHWR reactors. The insert would remain in the reactor for a significant period of time, similar to existing isotope production targets, and the insert would be fully sealed to prevent the release of any materials to the moderator in the unlikely event of a failure.

The fuel element irradiation insert is limited in terms of the operating regimes that can be tested. The fast neutron flux to pellet power ratio is lower than what it would be for a fuel bundle irradiation. This means that the irradiation growth of the fuel cladding will be slower than it is in an actual fuel bundle. Notwithstanding these limitations, it is expected that the fuel irradiation insert should permit the testing of a wide range of fuel power, temperature and burn-ups. Following testing, the fuel will need to be examined at a post irradiation examination facility. The fuel element pre-irradiation testing, fuel element irradiation and fuel element post irradiation examination are best performed at Canadian power reactor facilities and nuclear laboratories.

The pressure tube needs to be tested for mechanical and corrosion properties and irradiation endurance. The mechanical tests will require a substantial test facility. Fortunately, test facilities for qualifying pressure tubes already exist in Canada. These will, however, need to be upgraded to operate at supercritical water pressures. Tests will need to be performed for burst strength, axial buckling load and adjacent pressure tube interaction. The latter test should be performed with heated supercritical water and blowdown characteristics similar to those predicted for the refit end fitting and PHTS. The corrosion tests would examine any changes to the pressure tube surface in the presence of high-pressure supercritical water. These would not include the effect of irradiation on both the pressure tube and water. It may be possible to test irradiation corrosion behaviour using a carrier bundle in a power reactor or, alternately at a research reactor.

Irradiation mechanical testing of a full-scale pressure tube cannot be performed except through in-situ irradiation in an operating power reactor. It will, however, be necessary to establish the suitability of the pressure tube before its first use in a power reactor, particularly the ASME requirements for pressure boundaries [8]. This is expected to require irradiation testing of pressure tube material samples under various loading and environmental conditions. Non-irradiated pressure tube testing can be performed in existing Canadian pressure tube test facilities. Irradiated testing of specimens will need facilities that can achieve a high fast flux fluence. No such irradiation facilities exist in Canada and the irradiations will likely need to be performed in the United States. Post irradiation examination can be performed in nuclear laboratories in Canada.

The insulator will need to be tested for mechanical properties, thermal properties and corrosion properties as well as irradiation endurance. The mechanical properties include tensile and compressive strength, dimensional stability during pressurization, wear resistance and impact resistance. The thermal properties include through diameter thermal conductivity, reflectivity, transparency and emissivity for both the dry and wetted surface conditions and thermal expansion coefficient. The corrosion properties include material loss on coated and uncoated surfaces when exposed to pressurized heavy water, and flow related material losses to the coated inner surface at various flow velocities and temperatures.

The irradiation endurance testing of a full-scale insulator tube cannot be performed except through in-situ irradiation in an operating power reactor. It will be necessary to test smaller scale specimens. Long-term endurance testing will require a high fast neutron flux test facility, likely in the United States. This testing would confirm dimensional stability under neutron irradiation and allow the post irradiation testing of mechanical, thermal and optical properties. It would also be desirable to test corrosion properties in flowing irradiated heavy water. This might be performed using a carrier bundle in an operating power reactor. In the carrier bundle concept, the central element of a power reactor fuel bundle would be replaced with a sample holder with corrosion specimen samples and the bundle would undergo a full irradiation cycle in the power reactor. The sample holder would then be de-mounted in the fuel bay.

Non-irradiated insulator testing can be performed in existing Canadian laboratories. Fast flux irradiation of specimens would be performed in a United States facility with post irradiation examination in a nuclear laboratory in Canada. Corrosion carrier irradiation could be performed in a Canadian power reactor with post irradiation examination in a nuclear laboratory in Canada.

The shield plug provides a hydraulic function by limiting coolant flow during a blowdown caused by a loss-of-coolant-accident (LOCA) event. These hydraulic features have flow resistances that cannot be readily calculated, particularly for two-phase flow that may exist during blowdown. These tests will require a supercritical water supply similar to that needed for pressure tube burst interaction testing. It is expected that the most economic approach is to use the same test facilities. The shield plug hydraulic testing can be performed at Canadian nuclear test facilities.

The channel closure needs to be tested for mechanical functionality, leak tightness and reliability. This test requires a supercritical water source. The supercritical water testing is best done at the same Canadian facility used for insulator and pressure tube testing.

The out of reactor components need to be tested for various characteristics listed in Table 1. Most of this testing can be performed in Canada.

TABLE 1: Out-of-reactor component development testing

Component	Characteristics to be tested
Fuelling Machine	Mechanical movements, grappling, sealing, thermal behaviour, corrosion
Primary Coolant Pump	Heavy water hydraulic performance, sealing, pressure regulation
Primary Side Turbine	Bearings, rotor access, sealing, thermal hydraulic testing/simulation
Steam Generator	Condensing heavy water heat transfer, boiling vibration, dry out
Superheater desuperheater	Pressure loss, heat transfer, thermal hydraulic testing/simulation
Control System	Control stability, plant system interactions
Shut down System Instruments	Instrument response, environmental reliability, trip input scanning

6. Integration Testing

Integration testing tests the interaction between various components beyond what is possible to determine from testing components in isolation. It is possible to perform this type of testing during commissioning but it can lead to significant start-up delays for a first-of-a-kind plant, as phenomena interactions are discovered and adjustments need to be made to a plant. Another approach is to use advanced simulation software to uncover interactions between systems that may not otherwise be apparent. Integration testing and systems software simulation provide a means to reduce the risk of serious delays during plant start-up.

A potential for component interactions exists for the in-core components and connected assemblies. Advanced simulation software such as computational fluid dynamics (CFD) may be used to uncover interactions, however testing can provide an important additional degree of confidence. It is possible to test these items short of building the full refit by installing two test channels in an existing operating power reactor and connecting them to a loop frame totally separate from the existing reactor PHTS. While this intrudes with the operation of the power reactor to produce electricity, there are examples of successful multi-mode operation of nuclear reactors. Existing Ontario power reactors have been producing cobalt-60 in parallel with electricity production for many years. The experimental 37M bundle was tested in an Ontario power reactor. The NRU research reactor produced the majority of the world's supply of molybdenum-99 while conducting and extensive research program including severe fuel damage tests.

The most opportune time to install two test channels in a power reactor would be during one of the programmed reactor refurbishments. This would require qualified supercritical water pressure tubes at the time of test channel installation. Installation of the loop in the reactor can be simplified using modularized skids to limit the in-situ pipe work needed during loop installation.

Integration testing can test the fuel under the full range of operating conditions with the correct combination of burn-up, temperature, linear power and fast flux. Hydraulic and thermal hydraulic stability of the fuel bundle can be confirmed under flow conditions including the channel entrance and exit effects. Fuelling operations can be tested under on-power conditions.

The full-scale pressure tube and insulator can be tested under the combination of coolant flow and pressure, coolant irradiation, fast flux and temperature gradients present during operation. The thermalhydraulic characteristics of the channels can be fully characterized for inputs to PHTS loop simulations. The test channels and testing would be implemented in a Canadian reactor. The post irradiation examination would most conveniently be conducted at a nuclear laboratory in Canada.

The PHTS loop is too large to be integration tested apart from installation in a prototype refit. The loop does however have the potential to experience component interaction phenomena such as the pressure waves from the pump vanes interacting with fuel bundles that was encountered during Darlington nuclear generating station commissioning. The most practical way to explore these types of interactions is through advanced simulation software. Simulation software exists for nuclear and non-nuclear plant cooling and steam system design. It is best to use a variety of software to have the greatest chance of uncovering effects that might be otherwise difficult to predict.

7. Analytical Computer Code Development

In addition to testing, there is a need to have analytical tools to support the initial prototype refit design. Current PHWR's have a full complement of analytical computer programs to support design and safety analysis. This industry standard toolset exists to augment predictions available from general purpose engineering software such as for stress and thermal analysis. The refit PHTS will require new or augmented computer codes to predict phenomena that is unique to supercritical coolant operation. The principal codes are described in this section.

The thermahydraulic computer codes used for current PHWRs may lack adequate heavy water properties at the supercritical water temperatures of the refit. It will be necessary to incorporate these properties or resulting correlations into existing thermal hydraulic codes. New thermal hydraulic codes may be needed for predicting the coolant blowdown behaviour and temperature details close to fuel bundle end plates and appendages. One possibility, is to consider use of computational fluid dynamics (CFD) predictions for supercritical water regimes where the current codes have limitations.

Current PHWR's use various codes for fuel behaviour prediction. These codes are based on zircaloy-4 fuel sheathing and are not capable of predictions for the single crystal sapphire clad fuel. These fuel codes would either need to be modified or new codes developed from scratch for the refit fuel.

The conceptual design of the refit has used a fuel and fuel channel heat transfer model that incorporates supercritical heavy water properties extrapolated to the refit operating temperature. The single crystal sapphire clad fuel is modelled in simplified form using simple empirical relations. While this model is suitable for approximate calculations, it is too inaccurate to be used for defining final design dimensions.

Given the significant differences between the refit and current PHWR fuel, it may be appropriate to develop a refit fuel code from scratch, potentially adopting some of the concepts from the existing simplified model. Supercritical water does not have a critical heat flux (CHF) or dry-out phenomenon. The safety of the fuel is determined by other parameters such as excessive internal pressure or excessive interface pressure from fuel pellet to clad interaction. Some of the key aspects that will need to be modelled can be gleaned from a past study on the use of single crystal sapphire cladding [2].

The refit fuel is designed for efficient high temperature radiative heat transfer to the pressure tube. Thermal radiation is emitted from the fuel and will be transmitted, absorbed and reflected through the semi-transparent sapphire cladding. A similar set of phenomena are present for thermal radiation incident on the semi-transparent insulator surface. No current computer code is capable of modelling these multiple modes of heat transfer in parallel. Work is currently underway on developing such a model focusing on the refit element, bundle and fuel channel geometry.

The reactor physics core behaviour can be modelled using existing nuclear industry computer codes, both deterministic and Monte Carlo based. However; the modelling approaches may be different than for current PHWRs. One difference between the PHWR refit and existing PHWRs is the increased neutron absorption by fuel channel components, particularly the fuel cladding and insulator. Other differences are the large variations in coolant density and temperature along the fuel channels. The reduction in density of supercritical water coolant, compared to existing PHWR coolant conditions, increases reactivity. This is exploited in the PHWR refit design [1] by having a two-zone core with two types of channels Forward and Return. The Return channels being hotter and with lower coolant density on the core radial periphery and Forward channels in the central zone to flatten the radial power profile and improve operating and safety margins.

The deployment of the supercritical water reactor technology as a refit constrains the reactor physics design to existing reactor parameters such as lattice pitch and calandria diameter. For simplicity, it is desired to keep the reactor regulating and monitoring systems analogous to the existing PHWR reactor to the extent possible. Ideally, the same or slightly modified flux measurement, reactivity control and shutdown systems can be used.

The current PHWR refit reactor physics analysis focuses on using WIMS-AECL 3.1 with ENDF/B-VII.1 nuclear data library for initial reactivity and fuel burn-up calculations. MCNP 6.2 with ENDF/B-VII.1 and ENDF/B-VIII.0 is used for code-to-code validation with WIMS. Also, the MCNP half- and full-core models are used for radial and axial power and heat distributions and reactivity confirmation. In the refit, it is important to understand localized heat generation which influences several fuel parameters that are important to operating and safety margins. A more detailed half-bundle MCNP model is developed for detailed examination of bundle junction end flux peaking, fuel pellet radial power distributions and fuel pellet fission product and fuel material compositions.

The RFSP code has not been used yet, in part due to uncertain validation for supercritical water conditions. While MCNP has not been validated for these conditions, it is a code with a long pedigree for analysing many diverse reactor physics problems. MCNP has been evaluated against RFSP for reactivity and power profiles for CANDU type analyses [8]. The ZED-2 research reactor at Chalk River Laboratories is a valuable asset that can be used to validate the reactor physics models using the new materials and coolant conditions present in the supercritical refit.

The physics methodology is currently in the process of evaluation and finding the most efficient approaches for optimizing the refit core.

8. Standards Development

Standards are a key component on the engineering design process. Current standards have been developed around existing applications, including current PHWRs. The refit has differences from existing PHWRs that result in situations not addressed by the standards or requirements that may be inappropriate. The standards that may require modified or new requirements to accommodate refit technology are primarily American Society of Mechanical Engineers (ASME) pressure boundary codes and Canadian Standards Association (CSA) Nuclear series standards. The non-nuclear sections of the ASME pressure boundary code already address supercritical water power generating applications (e.g. for existing North American coal fired supercritical generating stations [10]).

The pressure tube is made from silicon carbide fibre composite. The ASME requirements for this material are currently being developed [8] for high temperature reactor applications. The refit pressure tube application differs somewhat from these applications in terms of the use of an internal insulator, temperature and stress. Notwithstanding these differences, the current ASME code development work is expected to be applicable to the pressure tube with the exception of the internal insulator.

The CSA nuclear standards will apply to a PHWR refit in a similar way to current PHWRs. A few of the standards may require some augmented requirements that are specific to a PHWR refit. The standards that are likely to need new or modified requirements are identified in Table 2.

TABLE 2: CSA Nuclear Standards that may Require Changes to Address a PHTS Refit

CSA Series	Standard Title	Potential Changes for Refit Technology
N285	General requirements for pressure-retaining systems and components in CANDU nuclear power plants/Material Standards for reactor components for CANDU nuclear power plants	1) New material requirements for pressure tube; 2) potential use of consequence of failure analysis for classification; 3) consideration of end fitting / shield plug as a restriction; 4) inspection requirements for pressure tube
N286	Management system requirements for nuclear facilities	None anticipated
N287	Standards for nuclear power plants concrete containment and safety-related structures	None anticipated
N288	Environmental management of nuclear facilities	None anticipated
N290	Standards for reactor control, safety systems and programs	1) Use of distributed control; 2) scanning of single channel trips; 3) unavailability of single channel trip

9. Summary

The testing, code development and standards development activities for the refit of a supercritical water PHTS into an existing PHWR have been identified. The completion of these activities should permit the design of the first refit to proceed using conventional nuclear engineering methods. While the activities identified in the preceding sections involve a substantial scope of work, many of the capabilities required already exist in Canadian industry and institutions. A list of these activities and their potential sourcing is provided in Table 3.

TABLE 3: Potential Locations for Performing Reactor Refit Development Activities

Development Activity	Work Location		Development Activity	Work Location	
	Canada %	Other		Canada %	Other
Fuel Cladding Manufacturing	50%	USA, Japan	Fuelling Machine Testing	100%	
Pressure Tube Manufacturing	50%	USA, Japan	Primary Coolant Pump Testing	-	USA
Insulator Manufacturing	50%	Germany	Turbine Simulation and Testing	-	USA, Europe, South Korea
Fuel Manufacturing	100%		Steam Generator Testing	100%	
Transition Piece Joining	100%		Superheater-desuperheater	-	USA
End fitting Manufacturing	100%		Control Testing	100%	
In-core Materials Properties	70%	USA	Shutdown Systems Testing	100%	
Ex-core Materials Properties	100%		Fuel Channel Integration	100%	
Heavy Water Properties	100%		PHTS simulation	100%	
Fuel Element Testing	100%		Analytical Code Development	100%	
Pressure Tube Testing	70%	USA	ASME Code Development	-	USA
Insulator Testing	70%	USA	CSA Standards	100%	
Shield Plug Testing	100%		Reactor physics analysis and testing	100%	
Channel Closure Testing	100%		Thermal hydraulics analysis and testing	100%	

The duration required for the development program depends on the level of investment and will be affected by uncertainties arising from any unexpected test and analysis results. It is expected that development could be completed within seven years provided that development is not constrained by financial, facility or personnel resources. The critical path is the design construction and operation of key experimental facilities and the resulting test programs. The estimated durations are based on past experience with test facilities and programs such as to blowdown test facility research program at Chalk River Laboratories.

Other important activities can be conducted in parallel with the development activities described herein. The design work and cost estimate for an initial prototype refit can be conducted in parallel with the development work such that all information is available for a final investment decision at the end of the development program. For a situation where existing units are in lay-up, such as current Pickering A units 2 &3, all existing components that require removal for the refit can be removed, the facility decontaminated and a complete engineering baseline (measurement of existing reactor component dimensions and condition) can be established for the refit.

A preliminary estimate of the development cost of PHWR refit has been prepared. It is based on execution costs of individual activities, project management and allowance for uncertainties. Some of the development costs can potentially be shared with other entities -- such as manufacturers that can benefit from future sales. The estimate did not take credit for such sharing – largely because its extent is currently unknown. The total rough estimated cost of PHWR refit development is of the order of one billion Canadian dollars.

While the above cost may seem very large, the payback and return on investment is very substantial if the refit technology is deployed to multiple units such as those at the Pickering A four-unit station.

The development program described herein is based on the use of enriched uranium fuel which is a deviation from existing operating PHWRs. The enriched fuel departs from one of substantial benefits of existing PHWRs: the use natural uranium fuel that is relatively low cost and easily sourced. The initial use of enriched uranium fuel is a strategy intended to reduce development risk and complexity by using the existing Canflex fuel geometry and avoiding the need for a highly structured insulator that may require substantial testing to achieve the required robustness.

Concepts are under development for a natural uranium version of the supercritical refit. As an indication of the extent of reduction in neutron absorption required, preliminary results indicate that a 0.78% enrichment could achieve the same fuel exit burn-up for Pickering A as achieved during current Pickering A operations. i.e the amount of reduction in neutron absorption required is relatively modest (to achieve a reduction from 0.78% enrichment to 0.71% enrichment) and should be achievable with more advanced design concepts.

10. Conclusions

A 100% increase in power output from an existing PHWR can be achieved through a development program to create technology for a modular refit of the PHTS. The development program is substantial and would require significant investment; however, many of the capabilities for development already exist in Canada. For an entity with a substantial number of PHWRs, such as Ontario, the development investment would be readily paid off by increased electricity generation, which could amount to 14 000 MW, if all 20 units were eventually refitted.

The current status of the Ontario PHWR program is that refurbishments have already been completed or committed for 12 of the 20 units, with a detailed refurbishment study underway for the 4 Pickering B units. In this context, a super critical refit would be most relevant for the four Pickering A units that are prototype units for which a business case for conventional refurbishment may be more difficult than the other Ontario units. A supercritical refit would also be relevant for a second round of Ontario refurbishments for further life extension and power increase. The business case for such second round of refurbishment would be determined by the comparative merits of other options for increasing Ontario electric power generating capacity. Given the doubling of power output with only a marginal increase in operating cost, the business case for a supercritical refit is expected to be attractive in comparison to other electric generation alternatives.

The supercritical refit concept can also be deployed as a new build. In this scenario, it would be integrated with the most current products, such as the AtkinsRéalis Candu MONARK.

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