# FISSION GAS RELEASE MODEL FOR PHWR FUEL OPERATING AT SUPERCRITICAL WATER REACTOR TEMPERATURES

**R.I. Lounsbury[1] and M. Tayal[2]**
[1] Suretech Development Limited, Deep River, Ontario, Canada
[2] Nueva Science Incorporated, Oakville, Ontario, Canada
lounsbur@sdl.on.ca, tayal2000@gmail.com

## 1. Introduction

Large nuclear electricity generating reactors are one of the few current concentrated sources of electrical power that are not greenhouse gas emitting. While large-scale hydraulic power also provides a concentrated electrical energy source, most suitable hydraulic locations have already been developed. The remaining available hydraulic energy sources are often very remote from the location of concentrated electrical demand necessitating extensive electrical transmission infrastructure. In comparison, large nuclear generating stations can be constructed wherever there is an available cooling water source, which is often close to most cities and industrial centres.

Large nuclear generating stations would appear to be an ideal option for meeting the increasing electricity demand from transportation and industrial electrification and artificial intelligence facilities. One challenge is the high, and at times uncertain, capital cost of building large nuclear generating stations, and the long time required from initial project commitment to first electricity generation. The high capital cost can be partially mitigated by building in smaller units using emerging small modular reactor (SMR) technology. While the initial capital cost of SMRs may be lower it is uncertain whether the unit cost of electricity will be competitive with large nuclear power plants. The long project execution time for large nuclear generating stations may only be partially addressed by SMRs as evidenced by the long project time for the first SMR plants.

One option to address the high capital cost and long project times of large nuclear new build projects is to consider power upgrades to existing large nuclear power plants. Power upgrades, where technically feasible, offer the potential for lower capital cost than a new large reactor and shorter project execution times. PHWR generating stations have the potential for significant power upgrading by refitting a higher temperature and pressure primary heat transport system (PHTS) in conjunction with new fuel channels and fuelling machines. The physical work required for such a refit is very similar to the work undertaken during PHWR refurbishments, of which several have been conducted on a now predictable cost and schedule.

The advantages to obtaining increased nuclear generating capacity through refitting existing PHWR plants are:

- The construction time is much shorter than 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

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. The greater power output combined with similar operating costs provides a significant return on investment potential over the life of the refitted plant.

Operating and shutdown PHWR plants are present in Canada, India, South Korea, Romania, China and Argentina. These provide a substantial fleet of reactors to which the refit technology might eventually be deployed. It also could form a basis for a future new build large reactor product that operates at higher temperature and pressure than current PHWRs.

## 2. Refit reactor fuel operating environment, margins and modelling

The PHWR refit has been conceived as a modular replacement of the reactor fuel channels, fuelling machine and PHTS. This restricts changes to other interfacing systems thereby limiting the scope of work during a refit and simplifying new system integration. The refit systems and changes are described in a past conference paper [1].

While the changes to the overall plant are limited by the modular concept, the changes to the fuel bundle and its operating environment are significant. The most significant changes are the higher fuel temperature and the use of single crystal sapphire fuel cladding and fuel bundle constructions in lieu of Zircalloy-4 used for current PHWR fuel. The fuel pellet is similar to existing PWHR pellets except for the fuel pellet diameter, the use of slight enrichment for some pellets and slight changes to the dish and chamfer. The pellet diameters are close to those of CANFLEX fuel previously tested in the Point Lepreau nuclear generating station and NRU research reactor. Slightly enriched fuel has been used extensively for experiments in the NRX and NRU research reactors to achieve power reactor power densities when the fuel is operated in the lower flux research reactors. Experiments have been conducted in NRU on fuel with various dish and chamfer / taper geometries to study the effects of variations [2].

The current version of the refit core uses a combination of natural and slightly enriched uranium fuel pellets which differs from current PHWRs that use natural uranium only. It is anticipated that in future it may be possible to have a natural uranium only refit reactor core by optimizing some fuel channel components to reduce parasitic neutron absorption. The fuel channel optimization will benefit from operating experience, and therefore the initial refit deployment is planned to use a mixture of natural and slight enrichment to provide confidence in achieving adequate reactivity margins in combination with economical fuel utilization.

The effect of the sapphire fuel cladding and higher temperature and pressure coolant conditions on fuel performance and failure modes has bee studied [3]. Sixteen different damage mechanisms were identified and assessed. In this paper, we have focused primarily on internal gas over pressure leading to pressure overstress of the fuel cladding/endcap junction.

The fuel internal pressure is a result of several competing and time dependent processes. These include surface heat flux, radial power distribution, gap conductance, fuel thermal conductivity, cladding temperature, burn-up and time dependent fission gas production and relocation and fuel cladding irradiation growth. These are competing processes, some with positive, and others with negative feedback. It is therefore not straightforward to predict the limiting conditions that will give the lowest fuel internal pressure margins.

The determination of the fuel elements with limiting pressure stress margins requires a survey of a relatively large number of elements in the reactor core operating at different combinations of conditions. This is accomplished using a fuel and fuel channel heat transfer model that models the fuel elements in pairs of interconnected fuel channels along with the hydraulic behaviour of end-fittings and interconnector pipes. The model has ten axial increments to capture local effects such as end-flux peaking which influence PCI. This model needs to be run for a significant number of channel pairs to be confident of capturing the fuel elements with limiting fuel pressure stress margins.

Computational demands are made manageable by having relatively simple radial fuel models that rely on thermal deposition and conductivity adjustment factors to account for variation in radial fuel power deposition with burn-up [4]. This modelling approach precludes the use of traditional PHWR fission gas release models [5] that require radial compartments to model fission gas production, agglomeration, transport and release.

Once potentially limiting fuel elements are identified by analysis using the fuel channel pair model described above, a more detailed fuel model, with 100 radial increments and 100 axial nodes, will be used.

## 3. Simple fission gas release model

Various fuel fission gas release models have been developed [6,7] for different fuel and reactor types. Most of these models focus on mimicking or simulating known physical processes from fissioning of the heavy metal isotope through to release of volatile gas product from the fuel matrix. While the same processes may exist for different fuels and reactors they do not necessarily operate at the same rate or by the same mechanisms. A model that provides good predictions for one reactor and fuel type may not provide good predictions for another reactor even if the fuel material is similar.

The conceptual design of the refit reactor core requires a model that is mathematically simple to implement such that it can be applied across large parts of the core. It also needs to provide confidently conservative results to ensure that non-viable design options are not indicated by the analysis. The approach taken was to have a mathematical model that modelled known PHWR fuel fission gas release characteristics without explicitly modelling all of the underlying processes.

The starting point of the model was the premise that the fission gas potentially available for release may be related to the fission gas fuel swelling. This premise is based on the fuel swelling deformation requiring the elemental fission product elements to diffuse and collect as a gas in bubbles prior to creating pressure and volume to expand the fuel pellet. A simple empirical model for the fission gas swelling is available for LWR fuel for use in the NRC FRAPCON code [8].

$$S_g = 8.8 \times 10^{-56} (2800 - T)^{11.73} e^{-[0.0162 (2800 - T)]} e^{[-8 \times 10^{-27} B]} B_s$$

Where:

$S_g$ - fractional volume change due to gas fission products (m3 volume change/m3 fuel)
T - temperature (K)
B - total burnup of fuel (fissions/m3)
$B_s$ - burnup during a time step (fissions/m3)

The model for fuel pellet radial strain converts burn-up from fission / m3 to MWh/kgU at a ratio of 1 fission / $m^3$ = 4.11 x $10^{-24}$ MWh/kgU and assumes the strain to be 1 / 3 of the volumetric swelling.

Pellet radial strain due to fission gas:

$$\varepsilon_{rgfp,x} = 0.333 \times 8.8 \times 10^{-56} (2800 - (T_{fu,avg,x} + 273))^{11.73} e^{(-0.0162(2800 - Tfu,avg,x + 273))} e^{-8 \times 10E\text{-}27 Bx / 4.11 \times 10E\text{-}24)} (B_x - B_{x\text{-}10}) / 4.11 \times 10^{-24}$$

The fission gas release during a jump or drop in power is calculated by applying coefficients related to fuel pellet temperature change and remaining fission gas in the pellet:

Power jump:

$$\Delta FGR = \underbrace{((\varepsilon_{r,mp,gfp} + \varepsilon_{r,ep,gfp}) /2)}_{\text{Term A}} \underbrace{(0.6}_{\text{Term B}} \underbrace{(1 + (T_{fu,in,mp,a} - T_{fu,in,mp,b}) / 50)}_{\text{Term B'}} \underbrace{((667.7\ B) - FGR_{last}))}_{\text{Term C}}$$

Power drop or shutdown and return to same power:

$$\Delta FGR = \underbrace{((\varepsilon_{r,mp,gfp} + \varepsilon_{r,ep,gfp}) /2)}_{\text{Term A}} \underbrace{(0.2}_{\text{Term B}} \underbrace{((667.7\ B) - FGR_{last}))}_{\text{Term C}}$$

Where

$\varepsilon_{r,mp,gfp}$ and $\varepsilon_{r,ep,gfp}$ - gaseous fission product swelling strain for pellet mid plane and end plane
$T_{fu,in,mp,a}$ and $T_{fu,in,mp,b}$ – fuel centre temperatures after and before power jump
B – burn-up just before power change
$FGR_{last}$ – the released fission gas inventory prior to the power jump or drop

The term A in the equations is the average of fuel midplane and end plane fission gas swelling. This term is intended to represent the gas release driving force from pressurized gas cavities within the fuel pellet. Term B is a magnitude coefficient that is intended to represent magnitude of release resulting from internal stress changes in the fuel pellet caused by small power increases or power decreases. Term B' is an increasing temperature multiplier that reflects that larger power/temperature increases induce greater pellet internal stress changes giving rise to larger fission gas releases. The coefficients for terms B and B' were calibrated based on past PHWR fission product release experience. Direct proportionality was assumed for these terms given the absence of detailed power jump magnitude information needed to determine another relationship. Term C represents the remaining fission gas inside the fuel pellet. At the time this fission gas production coefficient was determined there was no fission product source term analysis for the refit fuel and fission gas production rates from existing PHWR fuel were used instead.

All parameters in the model were selected with the goal of obtaining conservative predictions for the variety of conditions experienced by the refit reactor fuel.

## 4. Prediction results for anticipated refit reactor operating conditions

The predictions in this section are for the most recent evolution of the refit reactor core design. Over the past 15 years different potential core configurations have been analyzed resulting in convergence on an equilibrium core that is expected to have margins that allow a 100% or more increase in electrical output from an existing PHWR reactor.

The core design aspects that are expected to have the greatest influence on fission gas release are: channel coupling, two bundle shift, 2320 MW power to coolant and target core average exit burn-up of 250 MWh/kgU to 280 MWh/kgU.

Channel coupling involves cooling channel pairs in series with the coolant first passing through a 'forward' channel and then though a 'return' channel. This arrangement is done primarily for thermalhydraulic reasons. The forward fuel channels are located in the radial centre of the core and generally have higher channel powers whereas as the return channels are on the core radial periphery. The return channels generally have higher coolant temperatures and generally higher maximum fuel temperatures than the forward channels.

Fueling is limited to a maximum of two bundle shift to limit the potential for unacceptable PCI. This is necessary due to the use of a ceramic cladding material that does not creep or yield in the operating temperature regime. The fuel pellets require that both fuel densification and cladding swelling progresses before the pellets are exposed to maximum power. It is possible that four- or eight- bundle shift can be used for low power channels however this has not been fully explored due the higher power channels being critical the establish reactor power and accident margins.

The 2320 MW power to coolant provides the heat and thermodynamic efficiency needed for 100% or more increase in power output. The burn-up target ensures both reasonable fuel utilization and helps skew power towards the channel inlet to counter the opposite effect from deceasing coolant density along the channel length.

Table 1: Predictions for channel pair with average maximum power return channel fuel bundle (results for elements with highest internal pressure – just after fuel bundle shift)

| | Forward Channel K07 – Bundle 6 outer ring element | Return Channel C07 – Bundle 6 outer ring element |
|---|---|---|
| Average Coolant Temperature (°C) | 367 | 469 |
| Average cladding outer surface temperature (°C) | 504 | 686 |
| Average Fuel outer surface temperature (°C) | 641 | 779 |
| Average fuel centre temperature (°C) | 1,985 | 2,108 |
| Peak (5 cm average) fuel centre temperature (°C) | 2,065 | 2,187 |
| Internal Gas Pressure (MPa) | 15.6 | 21.9 |
| STP fission gas volume (mL) | 0.9 | 1.9 |
| Percent fission gas release (%) | 1.7 | 3.5 |

The outer ring of elements in the forward channel fuel bundle in position 6 have the highest internal pressure elements in the fuel channel. These elements had a linear fuel rating at loading of 27.1 kW/m followed by power ramps of 16.9 kW/m and 11.6kW/m. This element had a burn-up at the final power ramp of 96.0 MWh/kgU and a linear power of 43 kW/m. The fission gas at reactor exit was 2.2 mL corresponding to a 1.5 % release at an exit burn-up of 264 MWh/kgU.

The outer ring of elements in the return channel fuel bundle in position 6 have the highest internal pressure elements in the fuel channel. These elements had a linear fuel rating at loading of 18.8 kW/m followed by power ramps of 17.1 kW/m and 12.7 kW/m. This element had a burn-up at the final power ramp of 94 MWh/kgU and a linear power of 39.3 kW/m. The fission gas at reactor exit was 6.3 mL corresponding to a 4.1 % release at an exit burn-up of 270 MWh/kgU.

The outer ring of elements in the forward channel fuel bundle in position 6 have the highest internal pressure elements in the fuel channel. These elements had a linear fuel rating at loading of 21.7kW/m followed by power ramps of 16.4 kW/m and 15.8 kW/m. This element had a burn-up at the final power ramp of 114 MWh/kgU. The fission gas at reactor exit was 9.3 mL corresponding to a 5.0 % release at an exit burn-up of 327 MWh/kgU.

Table 2: Predictions for channel pair with high maximum power return channel fuel bundle (results for elements with highest internal pressure – just after fuel bundle shift)

| | Forward Channel P18 – Bundle 6 outer ring element | Return Channel P19 – Bundle 7 outer ring element |
|---|---|---|
| Average Coolant Temperature (°C) | 353 | 444 |
| Average cladding outer surface temperature (°C) | 485 | 621 |
| Average Fuel outer surface temperature (°C) | 623 | 721 |
| Average fuel centre temperature (°C) | 1,955 | 1,936 |
| Peak (5 cm average) fuel centre temperature (°C) | 2,029 | 1,980 |
| Internal Gas Pressure (MPa) | 14.6 | 18.4 |
| STP fission gas volume (mL) | 0.8 | 3.1 |
| Percent fission gas release (%) | 1.3 | 8.2 |

The outer ring of elements in the return channel fuel bundle in position 7 have the highest internal pressure elements in the fuel channel. These elements had a linear fuel rating at loading of 7.0 kW/m followed by power ramps of 21.2 kW/m, 18.2 kW/m and 5.3 kW/m. This element had a burn-up at the final power ramp of 162 MWh/kgU and a linear power of 37.2 kW/m. The fission gas at reactor exit was 5.8 mL corresponding to a 16.8 % release at an exit burn-up of 340 MWh/kgU.

The maximum predicted fission gas release is above the range of that for current PHWRs which have a maximum normal operation release of 8% [9]. This is expected given the higher fuel operating temperature and fission gas release processes that increase with temperature such as diffusion and grain growth.

The internal gas pressures are much higher than for current PHWRs. The higher internal pressure is compensated during operation by the higher coolant system pressure – 24 MPa – 25 MPa. Current predictions show the element internal gas pressure remaining below the coolant system pressure for all normal operating conditions.

The fuel channel with the higher fission gas release is the average power return channel C07 that has a channel power of 5.52 MW, which is lower than the powers of the other higher power channels, P19 at 5.61 MW, P18 at 5.57 MW and K07 at 6.5 MW. This may appear counterintuitive that a lower power channel would give rise to higher fission gas release. One of the contributors to this effect is coolant temperature which is higher for the return channels and lower for some higher power channels which are overcooled. The finding that a relatively low power channel gives rise to the more limiting fuel element internal pressure conditions demonstrates the need for a modelling tool that can survey a broad range of elements in the reactor core for fuel performance analysis.

## 5. Prediction results for instrumented experiment

Experiments have been performed with instrumentation to measure the real time internal pressure increase in test fuel elements. These experiments are relatively rare compared to end-of-life fission gas release data but they provide greater insight into the dynamics of fission gas release. Since the simplified fission gas release model focusses on modelling the 'burst gas release' phenomena observed for PHWR fuel, an instrumented experiment is considered to be the most appropriate type of real data to use for assessing the model developed for the refit.

Fortunately, some instrumented fission gas experiments do exist. The ones used in this paper were performed by Atomic Energy of Canada Limited (AECL) in 1969 and 1970 and reported in 1974 [10]. The report describes two experiments, designated ARY and ARZ, that both involve the operation of a single experimental element at constant power followed by a power increase and then shutdowns and start-ups and minor power adjustments. The irradiation power history and internal pressure changes of both elements during and following the power jump is shown in Figure 1.

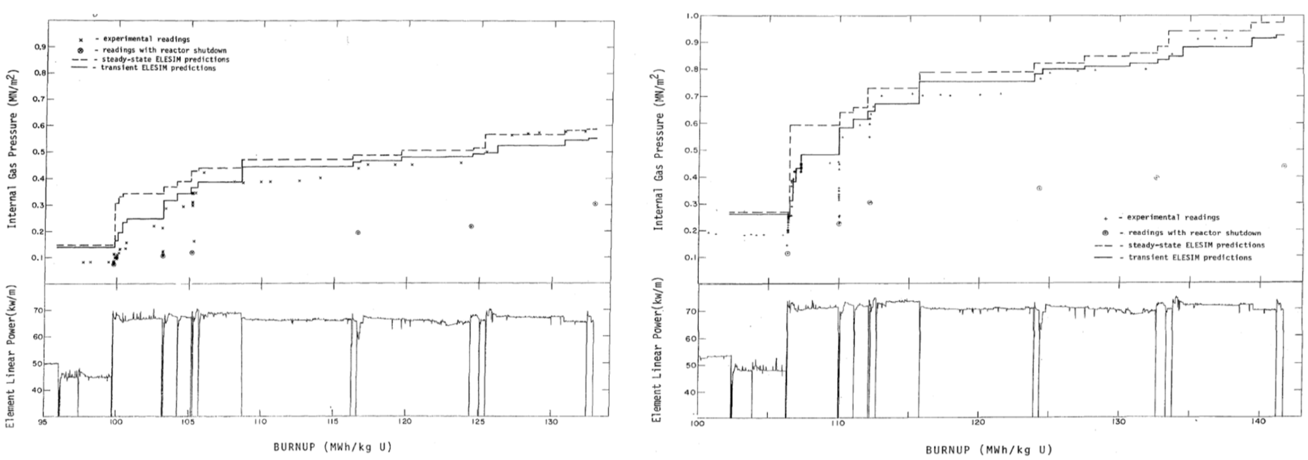


Figure 1 Partial power and internal pressure history of elements ARY and ARZ

It should be noted that the solid and dashed lines on the pressure plot in the figures are predictions of an AECL computer code, ELISIM. The information of interest for this paper is the 'x's and encircled 'x's that are the experiment pressure measurements. The values of the steps of near constant values 'x's, as read by the authors are provided in the Table 3.

These experiments were performed in the National Research Experimental, NRX, reactor located at Chalk River Laboratories. The experimental assemblies were first installed in B-14 and then D-18 cooling channels of the NRX reactor.

The physical dimensions of the fuel element and components, the fuel and stainless-steel sheath material, the fill gas and the in-reactor cooling conditions were modelled in a modified version of the refit reactor fuel and fuel channel cooling model. The modifications to the model were kept to the minimum needed to model these experiments. The principal changes were to fuel element geometry and cooling, substituting stainless steel sheath properties for those of the single crystal sapphire refit fuel cladding and different fill gas composition.

The initial predictions were performed for the zero burn-up initial power rise. These predictions were performed to assess the thermal modelling of the fuel and its cooling independent of the presence of fission gas. The refit reactor model predictions were performed using various fill gas assumptions when it became apparent that modelling using the intended argon fill gas was not giving representative results. The results are in Table 3.

Table 3: Comparison of model predictions to experiment measurements for different element fill gas assumptions

| | ARY experiment predictions | | | | ARZ experiment predictions | | | |
|---|---|---|---|---|---|---|---|---|
| | Meas.[1] values | Helium Fill Gas | Argon Fill Gas | Real Fill Gas | Meas.[1] values | Helium Fill Gas | Argon Fill Gas | Real Fill Gas |
| Linear Power (kW/m) | 46 | 46 | 46 | 46 | 49 | 49 | 49 | 49 |
| Coolant Temperature (°C) [1,2] | 17 | 18 | 18 | 18 | 17 | 18 | 18 | 18 |
| Sheath Outer Temperature (°C)[2] | | 56 | 56 | 56 | | 58 | 58 | 58 |
| Fuel Outer Temperature (°C)[2] | | 124 | 300 | 246 | | 122 | 248 | 204 |
| Element Internal Pressure (MPa)[1,2] | ~0.09 | 0.101 | 0.111 | 0.108 | ~0.18 | 0.119 | 0.201 | 0.177 |

[1] - Value derived from measurement during experiment
[2] - Value predicted by calculation of computer model

The element linear power was determined from continuous reading flux meters and normalized based on the integrated burn-up as determined from chemical burn-up analysis [10]. The coolant temperature is an estimate of the temperature over the duration of the irradiation. The actual coolant temperature varies seasonally with river water temperature from 3 °C to 20 °C. The sheath and fuel temperatures are predicted values from the refit reactor model. The element internal pressure is a value obtained by the authors from visually estimating the average of the numerous soak period pressure measurements shown in Figure 3 of the experiment report. The averaging of the soak period measured pressures was done due to the substantial scatter in measured pressures during the first 100 days following the start of irradiation.

While helium is the standard gas fill used for refit reactor calculations it is not the fill gas used in the experiments. The experiment fill gas is argon. When argon is used as the fill gas in the refit model, the predicted fuel outer surface temperature is 300 °C. The contact conductance model is slightly different from that in the heat transfer guide to account for the refit having a much harder fuel cladding material than zircalloy-4 used in current PHWRs. The experiment report contains a post irradiation gas composition analysis that indicates that other non-fission gases were present in the fuel element. When these gases are explicitly included in the refit reactor model the fuel outer surface temperature lowers to 200 °C to 250 °C.

The actual sheath outer surface temperature would be expected to be higher than that calculated from gap and contact conductance models since these models are based on intimate contact between the stainless-steel sheath and the fuel pellets. In practice the contact between the stainless steel and fuel pellet surface will be discontinuous since there will be bridging gaps that form near surface discontinuities. The surface discontinuities come from small variations in fuel pellet diameter and from fuel pellet 'hourglass ridging' cause by internal fission heating of the fuel. For zircalloy-4 fuel sheathing in a PHWR reactor, the sheathing will collapse onto the fuel due to thermal and irradiation creep resulting in intimate contact between the fuel sheath and pellet surface over the length of the element. The experiment sheath is not expected to behave this way. It is stainless steel and it will be operating at lower temperature than in a power reactor. Under these conditions significant thermal and irradiation creep are not expected. Elastic deformation of the stainless-steel sheath would be expected to leave sheath to pellet gaps near discontinuities that will reduce gap conductance and increase the temperature rise between the sheath and fuel pellet.

The fuel centre temperature is predicted using the Fink-Lucuta model [11] for irradiated uranium dioxide fuel. Irradiation effects generally reduce thermal conductivity and this model predicts lower fuel thermal conductivity than the non-irradiated fuel models over the temperature ranges of interest.

The refit model predicted element internal pressures are higher than the ARY experiment measured pressure for all cases, although the real gas fill results are closer to the measured value. The refit model predicted element internal pressures are higher than the ARY experiment measured pressure for the argon filled cases and lower for the remaining cases. The refit reactor model prediction real gas fill predicted pressure is closest to the measured value. The reasons for the difference could be errors in the component dimensions used to calculate free volume and errors in estimating the temperature of the fill gas at various locations.

The cause of the errors was investigated by running the refit reactor model at zero power and comparing the predicted element internal pressure with that measured for shutdown conditions. Shutdown pressure measurements in the report are only available just before the power jump at 99.5 MWh/kgU for the ARY experiment and 106.5 MWh/kgU for ARZ experiment. At these burn-ups there may have been some deformation of the fuel cladding and fuel elements affecting the free volume in the fuel element. The internal pressure had increased very little over this irradiation period suggesting little fission gas release. The refit reactor model calculated internal pressure for the ARZ experiment was 0.078 MPa compared to a measured value of ~0.08 MPa. The refit reactor model calculated internal pressure for the ARZ experiment was 0.126 MPa compared to a measured value of ~0.13 MPa. The close agreement of these values suggests that the operating pressure differences are primarily due to thermal rather than free volume assumptions.

The next phase of comparison is to compare the predictions for the various burn-up steps in the experiment. The ‘soak’ period before the temperature jump is not modelled explicitly. This period has numerous power maneuvers while maintaining a generally declining power profile. The measured pressure is relatively constant over the soak period indicating little or no fission gas release. This contrasts with the power jump and post power jump period that shows pressure changes and apparent fission gas release for most power maneuvers. It can be inferred that activation of fission gas release requires a power rise event at burn-up of sufficient magnitude to disrupt the fuel pellet structure creating paths for fission gas release. The current version of the refit reactor model does not simulate this activation effect and models releases for all power maneuvers. While this does not follow the observed experience of this experiment, it is conservative since released fission gases increase internal pressure and are more insulating than the helium fill gas causing fuel temperature increases.

The comparisons of the measured values and refit reactor model predictions are in Figure 2. The refit reactor model predicted pressure increase for both experiments is much higher for the power jump than the measured pressure increase. This reflects an over conservatism for the refit reactor model prediction of fission gas release for such power rises. The predicted fission gas STP volume at the end of irradiation for the ARY experiment is 26.5 mL compared to the measured value of 7.195. The predicted fission gas STP volume at the end of irradiation for the ARY experiment is 93.425 mL compared to the measured value of 11.195 mL. The predicted fission gas release is a factor of 3 and 8 larger than the measured values for the respective experiments.

This large increase was not expected given the initial calibration of the refit reactor model fission gas release coefficients. This led to exploration of relations other than direct proportionality for terms A, and B' as discussed in the section below.

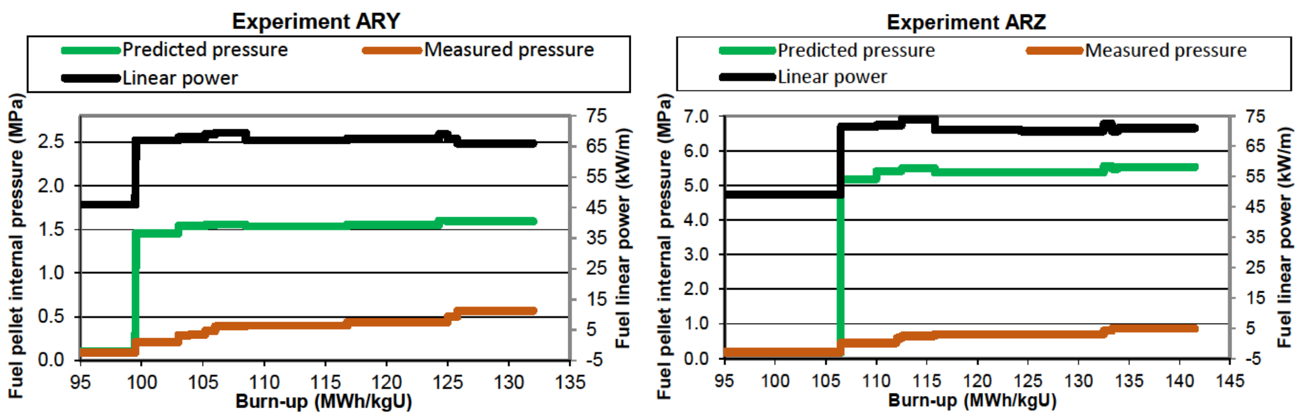


Figure 2 Predicted and measured element internal pressures and element power steps

## 6. Potential modifications to model for less conservative results

The previous model was modified by extracting the burn-up step term from the fuel pellet fission gas swelling Term A and applying an exponent and by applying an exponent to Term B'. The Term B coefficients were modified to give a good match for the low power increases and power decreases. Lastly, a term D was added reflecting 'delayed' release on a following power change of 1/ 4 the value from the previous power change. It was found that the following relation provided reasonably good agreement while still giving conservative predictions:

Power jump:

$\Delta FGR = ((\varepsilon_{r,mp,gfp,fx} + \varepsilon_{r,ep,gfp,fx})\ /2)\ (\Delta B^{0.3333}\ /\ 5)\ (40.6\ (1 + (T_{fu,in,mp,a} - T_{fu,in,mp,b})\ /\ 25)^{0.25}$

Term A Term B Term B'

$((1289.26\ m_{fu}) - FGR_{last})) + 1\ /\ 4\ FGR_{last}$

Term C Term D

Power drop or shutdown and return to same power:

$\Delta FGR = ((\varepsilon_{r,mp,gfp,fx} + \varepsilon_{r,ep,gfp,fx})\ /2)\ (\Delta B^{0.3333}\ /\ 5)\ (20\ \ ((1289.26\ m_{fu}) - FGR_{last})) + 1\ /\ 4\ FGR_{last}$

Term A Term B Term C Term D

Where

$m_u$ – Mass of uranium in the fuel

$\varepsilon_{r,mp,gfp,fx}$ and $\varepsilon_{r,ep,gfp,fx}$ – fission gas swelling strain calculated with the $B_s$ term fixed at 8 MWh/kgU

The results of using this model to simulate the ARY and ARZ experiments are provided in Figure 3. The model predicted fission gas release for ARY is 8.479 mL compared to the measured value of 7.195 mL. The model predicted fission gas release for ARZ is 17.016 mL compared to the measured value of 11.195 mL. The predicted fission gas releases predicted by the modified model are a factor of 1.2 and 1.5 greater than the measured values.

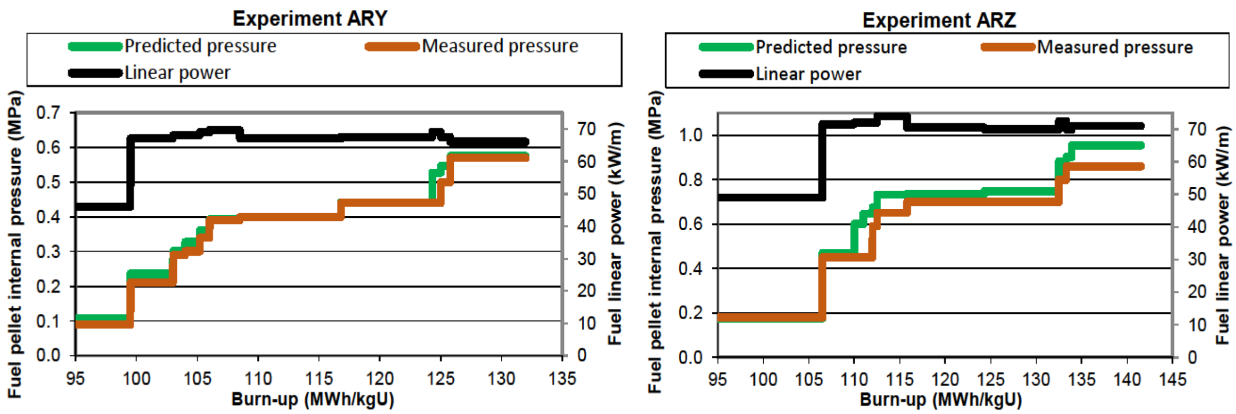


Figure 3 Modified model predicted and measured element internal pressures and element power steps

## 7. Discussion

The refit reactor model predicts much higher fission gas release than indicated in the experiment. The modified model provides closer but still conservative predictions for the two experiments.

The conditions for the experiment differ somewhat from the conditions of the refit reactor. The average post power jump fuel temperatures are 1360 °C (ARY) and 1505 °C (ARZ). These compared to refit reactor forward and return high bundle power outer ring elements average fuel temperatures of 1320 °C and 1450 °C. The fuel pellet radial temperature gradient, which is indicative of stress, is 107.8 °C/mm and 122.7 °C/mm for the experiments, and varies between 105 °C/mm and 125.7 °C/mm for medium to high power refit fuel elements. These ranges almost overlap however the burn-up and power steps differ most substantially. The experiment burn-up steps are an initial large step of about 100 MWh/kgU and a series of smaller steps 0.5 MWh/kgU to 9 MWh/kgU whereas refit high power steps are about 30 MWh/kgU to 40 MWh/kgU. The experiment power jumps are about 21 kW/m or in a much lower range of less than 4.4 kW/m, whereas the refit power increases in the high-power region are in the range of 5 kW/m to 25 kW/m. The refit burn-up and power steps fall into an intermediate range not covered by the experiment.

It is uncertain whether the modified model would provide conservative fission gas release predictions for the range of conditions experienced by the refit reactor. Two experiments having rather similar power and burn-up profiles are not sufficient to calibrate the model and validation will be required. To address this, we plan further comparisons of the model with additional experiments.

## 8. Conclusions

The refit reactor fission gas release model shows the limiting fuel element internal pressures are dependent on more than just burn-up and element linear power. An analysis of a variety of fuel channels will be required to identify the fuel with the highest element pressures. When the refit reactor model is compared against two instrumented experiments it gives a very conservative prediction of fission gas release. A modified model that gives a less conservative prediction of the experiments is proposed. It will require further calibration and validation to determine that it is conservative over the range of refit reactor conditions.

## 9. Acknowledgements

We wish to acknowledge support from AECL, and Noel Harrison in particular, for locating relevant fuel experiments, providing assistance in interpreting experimental results and review of this paper. Lastly, we would like to acknowledge the value of instrumented fuel experiments conducted in AECL reactors that provide uniquely valuable information on fuel behaviour.

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