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DETAILED BR2 STEADY-STATE AND DECAY POWER DISTRIBUTIONS DURING 1963 A/400/1 FLOW TEST

B. Dionne, C. P. Tzanos, and J. Matos
Research and Test Reactor Conversion Department,
Nuclear Engineering Division, Argonne National Laboratory,
9700 S. Cass Avenue, IL 60439 - USA

ABSTRACT

Comparisons between measured and predicted cladding temperatures for a 1963 flow test (test A/400/1) at BR2 were performed to establish the credibility of the RELAP model for the flow test and extend the validation database of RELAP. Studies showed that more detailed steady-state and decay power distributions for the various core regions are needed to accurately predict the cladding temperatures during the transient. To provide this essential input to RELAP, a sophisticated methodology was developed to predict the neutron and gamma power distributions at steady-state, and the alpha & beta as well as gamma decay power distributions during the transient. Detailed MCNP5 and ORGIEN-2 models were developed for BR2 "configuration 4" to: i) extract reactor-specific cross sections from MCNP5 for predicting the Li-6 concentration in the Be matrix using BERYL, as well as xenon concentrations and burnup using ORIGIEN-2, ii) predict power decay curves and decay gamma spectrum at the time of the test using ORIGEN2, iii) calculate neutron and photon power distributions at steady-state, iv) calculate decay photon power distributions. This information was then used to obtain the required detailed power distributions during the transient. Good agreement between the measured and calculated peak cladding temperatures is obtained.

1. Introduction

To support the safety analyses required to convert the BR2 research reactor from highly-enriched uranium (HEU) to low-enriched uranium (LEU) fuel, RELAP [1] simulations of a number of loss of flow tests performed at BR2 has been undertaken. This work is focused on test A/400/1, performed in 1963, which is characterized by a steady-state peak heat flux of 400 W/cm^2, total loss of flow without loss of system pressure, and opening of a by-pass flow valve a few seconds after the initiation of the transient.

During test A/400/1, the cladding temperature peaks immediately after the pump is shut off, then it comes down significantly as the reactor power drops, and a second peak is reached when the flow in the fuel elements is almost reversed. Initial RELAP simulations, based on the conservative assumption that all decay heat is deposited in the fuel, predicted cladding temperatures at the second peak that were much higher than the measured temperatures. Sensitivity analyses with RELAP indicated that this overprediction was sensitive to the decay heat split between the fuel and the other reactor regions. Therefore, additional efforts were made to obtain a better estimate of the decay power distributions in the BR2 core during test A/400/1.

To obtain an improved decay power distribution for the instrumented fuel element (F-346) and the other regions of the core (average channel, plugged channels, bypass channel, etc.), a detailed MCNP5 [2] model of the 1963 core configuration was developed. This model is first used to obtain the steady-state power distribution prior to reactor shutdown. Subsequently an ORIGEN-2 [3] model was developed to predict fuel burnup, decay heat curves, and decay photon spectra throughout the transient. The predicted photon source was then used to calculate the redistribution of the decay power after reactor shutdown.

This paper is organized as follows. Section 2 presents a short description of the BR2 reactor as configured in 1963. Section 3 presents the methodologies used to obtain two sets of power distributions labeled in this work as “conservative” and “best estimate.” Section 4 presents the computational results (power distributions, energy spectra, etc.) from the “best estimate” methodology and the impact of this methodology on predicted peak cladding temperatures. Finally, conclusions are presented in Section 5.

2. BR2 Reactor Configuration during Test A/400/1

2.1 Overall description

BR2 is a thermal water-cooled reactor moderated by water and beryllium. The core is located inside an aluminum pressure vessel. The beryllium consists of a matrix of hexagonal prisms with a central bore forming a channel. Each channel can contain a fuel element, a control/regulating rod, an experimental device, or a beryllium plug. Figure 1 shows a schematic of the reactor[1].

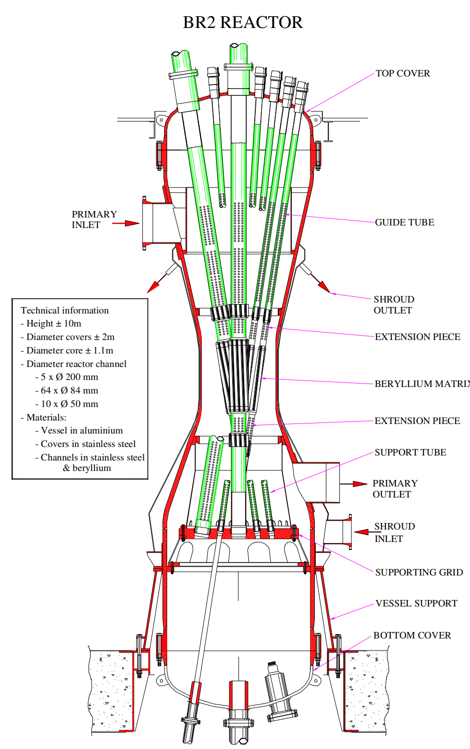

Figure 1 BR2 reactor schematic

Each fuel element is composed of 6 concentric fuel plates divided by aluminum stiffeners into three sectors. The Sylcor VIa elements used in 1963 differ from the current fuel elements mainly by the composition of the fuel meat. However, the exact 1963 fuel composition is not well defined in the literature [4, 5, 6, 7]. Incomplete or inconsistent information is provided in these references for the following

[1] This schematic of the reactor is a courtesy of BR2

parameters: i) weight percent of U in the fuel meat, ii) porosity, iii) exact volume fraction of each component of UAl_x, and iv) exact enrichment.

Therefore, it was necessary to make certain assumptions. Table 1 gives the assumptions made for the 1963 alloy fuel composition. Using these parameters, the calculated ^{235}U plate loading is within 0.4% of the reported [4] nominal loading for the Sylcor VIa fuel.

Table 1 Best estimates of 1963 BR2 Sylcor VIa HEU fuel composition

Assumption
UAl_x alloy (assumed all UAl_4)
90% enriched
U-234 content ~1%, and U-236 content ~0.3%
24 w/o of uranium in alloy
1% porosity (typical for alloy)
244g of U-235 per element

The flexibility of the BR2 core design allows for a variety of core loadings. At the beginning of the 1963 flow test series, the core was loaded with 14 fresh fuel elements (4 were instrumented), 7 control rods and 2 regulating rods in a configuration labeled "configuration 4". This configuration is shown in Figure 2.

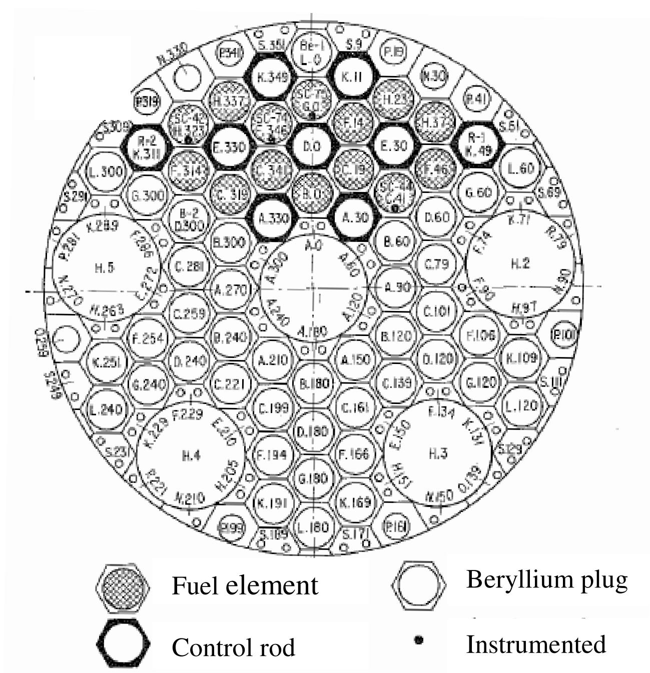

Figure 2 BR2 core "configuration 4"

2.2 Conditions of the Test A/400/1

The goal of test A/400/1 was to study the thermal performance of the core after a total loss of flow without loss of system pressure, and opening of a by-pass flow valve a few seconds after the initiation of the transient. This test was performed after about 6.5 days of reactor operation, with the reactor operating at steady state at 24 MWth (peak heat flux of 400 W/cm^2) just before the initiation of the transient. The detailed power history prior to the test, i.e., from the beginning of the test series (22-09-63, 6h30) to the time of test A/400/1 (28-09-63, 22h57) [8] initiation, is illustrated in Figure 3.

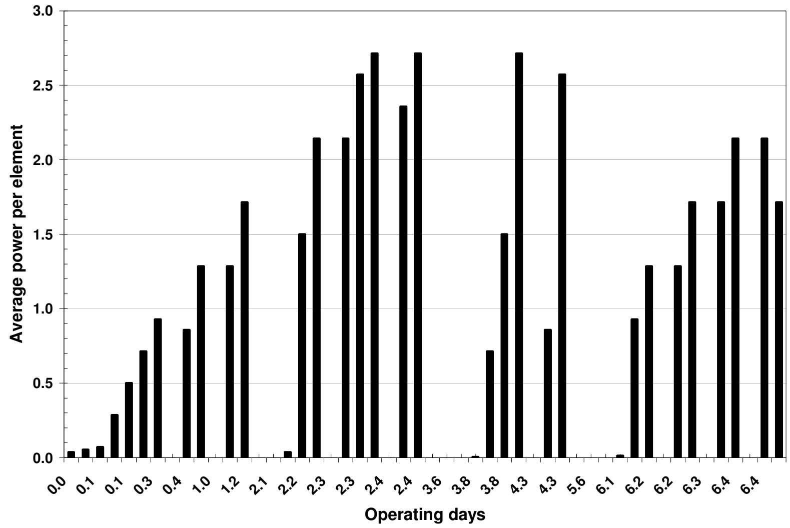

Figure 3 Detailed reactor power history up to test A/400/1

Note that just prior to the test A/400/1, the core was critical with the control rods withdrawn to 551mm [9] and the regulating rods to 250mm [10]. As shown in Figure 2, four fuel elements were instrumented with 4 thermocouples embedded in the cladding of one sector at different heights. In this work, the label "instrumented" fuel element will refer to element F-346 (see Figure 2).

3. Description of the Methodologies

3.1 "Conservative" methodology

This methodology is labeled as "conservative" since the main assumption that all the decay power is deposited in the fuel is, as will be shown later, quite conservative. The following list presents the main assumptions used in this approach:

1. The instrumented channel steady-state power is calibrated to match the measured channel power, and the remaining power is assumed to be deposited in the other fuel channels.
2. A single axial power distribution was estimated[2] from various power and neutron flux measurements.
3. No power redistribution is assumed after reactor shutdown.
4. The decay power for the fuel regions was calculated using the ANSI/ANS 5.1-1979 [11] decay curve.
5. In RELAP, the instrumented fuel assembly was represented by an average fuel plate.

[2] No measurement of the axial profile was made for the instrumented channel (channel F-346) for the test A/400/1 conditions. Axial power profile was therefore generated to match the steady-state cladding temperature profile measured in channel F-346.

3.2 "Best estimate" methodology

To obtain a better estimate of the steady-state and decay power distributions, it was necessary to develop a methodology which would improve on each of the assumptions of the "conservative" approach. The following list presents the main features of this "best estimate" methodology:

1. The codes MCNP, ORIGEN-2 and BERYL [12] were used to obtain a detailed representation of the BR2 core at the time the test A/400/1 was performed.
2. Steady-state power distributions were calculated for each region of the core using the models developed in step 1.
3. The models and results from steps 1 and 2 were used to calculate the distribution of decay heat generated in each region by the absorption of gamma rays.
4. Then, results from steps 1 to 3 were used to generate decay power distributions.
5. Power distributions from steps 2 and 4 were used in a refined RELAP model representing explicitly fuel plates in the sector of the instrumented element where the thermocouples were located.

These features of the methodology are described in more detail in the following sections.

3.2.1 MCNP, ORIGEN-2 and BERYL fresh core models

A model of BR2 "configuration 4" was first developed for a fresh core, i.e., with control rods withdrawn to 450mm [13], no burnup, xenon-free, and a clean matrix. This model was used to obtain reactor-specific cross sections and reaction rates for the ORIGEN-2 and BERYL codes. The detailed power history shown in Figure 3, and these two codes were used to generate information about fuel burnup, xenon-135 concentrations, and beryllium matrix poisoning (He-3, Li-6, and H-3), as well as curves for total decay heat and heat generated from gamma rays, and the gamma ray energy spectrum. The current fresh core model has a bias to criticality of about 0.7% $\Delta k/k$ with the control rods located at 450mm. Note that reference [13], which provides typical control rod positions for fresh and depleted cores, does not unambiguously define the state of the core for the reported control rod positions.

Considering the small fuel burnup predicted at the initiation of test A/400/1 (0.4 atom % of ^{235}U), a simplified element average burnup was used because: i) a small burnup will not significantly alter the steady-state power sharing between fuel elements, ii) a small burnup will not significantly affect the steady-state axial power distribution, iii) for the short duration (60 seconds) of the transient of interest, the decay power distribution is mainly a function of the steady-state power distribution (short-lived fission products) and not of the burnup distribution. This approach should be re-examined in cases were any of the three previous arguments would not hold.

To evaluate the impact of control rod position on element-averaged one-group cross sections and reaction rates, a sensitivity study was performed. This study showed that a variation in the critical rod height of 15mm (i.e., about 0.75% $\Delta k/k$ from critical) had a negligible impact (<1%) on cross sections and reaction rates. Therefore, predicting an exact critical rod height for this fresh, xenon-free model is not essential. Further study of this effect might be needed if a detailed axial burnup profile is needed.

In this work, different concentrations of He-3, Li-6 and H-3 (based on the matrix power history up to the time of test A/400/1 initiation [14]) were applied to each fuel element and control rod beryllium hexagon. For the other beryllium structures modeled in RELAP, region-averaged concentrations based on the breakdown described in Figure 4 were used. The control rod followers and beryllium "sleeve" of the regulating rod were treated independently. Because the concentration of poison in the beryllium matrix was relatively low, its impact on the spectrum, and consequently on cross sections and reaction rates, was not studied. This aspect would need to be further studied for a beryllium matrix with a longer lifetime.

3.2.2 MCNP model of the core at the time of test A/400/1

Fuel burnup, xenon-135 and beryllium poisoning information were used to update the fresh core model and obtain a model representative of the BR2 core at the time of test A/400/1. The control and regulated rod positions were adjusted to match the conditions described in Section 2.2. Using this updated MCNP model, a coupled neutron-photon transport calculation was performed and the steady-state power distribution was tallied for each region modeled in RELAP (bypass channels, plugged channels, instrumented channel and average channel), as shown in Figure 4.

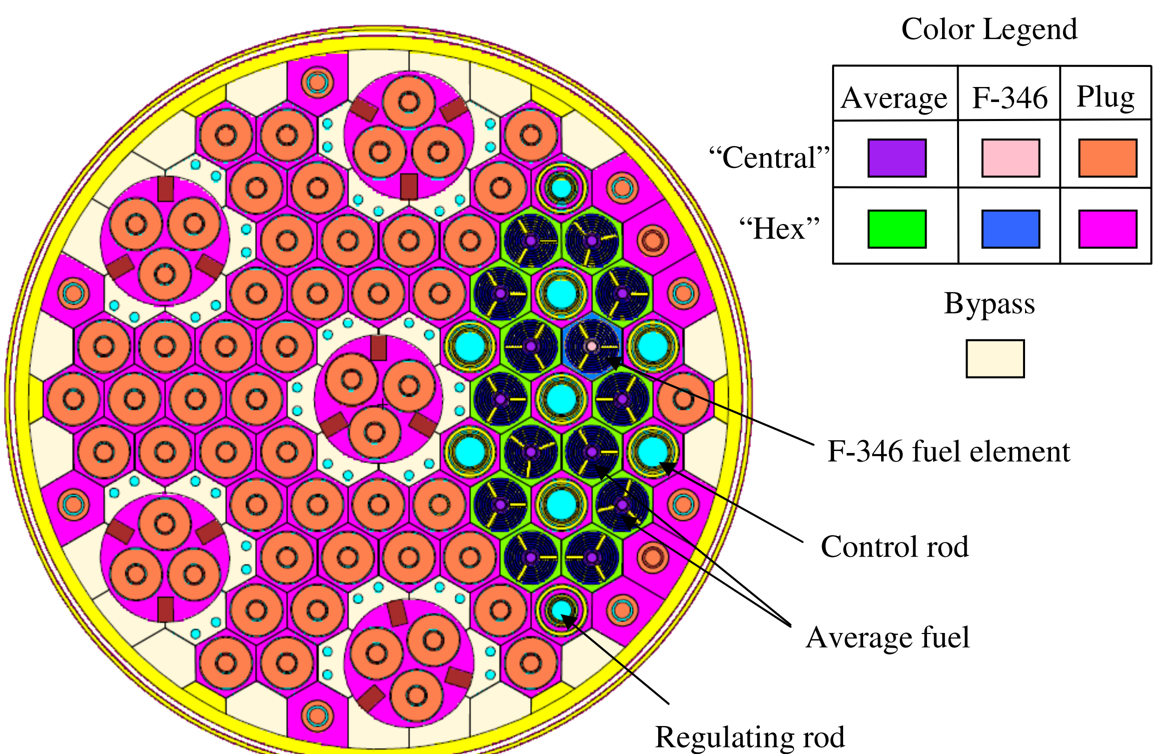

Figure 4 Regions modeled in RELAP

To capture the power peaking effect due to the orientation of the fuel element with respect to the core center, the power was explicitly tallied for the sector containing the thermocouples. As previously mentioned, the RELAP model was refined to explicitly represent the fuel plates in this sector.

3.2.3 Gamma decay power distribution

The gamma decay power distribution was estimated from a photon transport calculation which used a source defined by the fission products and actinides gamma energy spectrum from ORIGEN-2, and the normalized steady-state power as a spatial distribution. This approach assumes that for the low burnup and short transient of interest, most gammas will be produced from short-lived fission products whose distribution depends on the instantaneous power profile just prior to reactor shutdown. This assumption should be re-examined in other cases. The control rod positions were not readjusted to reflect the shutdown conditions, because it is expected that this effect has a negligible impact on the gamma decay power distribution, especially since the activation of the control rods was neglected.

3.2.4 Generation of the decay power distribution

To use the information obtained from the first three steps (see beginning of Section 3.2) to generate the decay power distribution as a function of time after reactor shutdown, the following assumptions were made: i) the photon spectrum remains constant, ii) the decay power associated with sources other than gammas is all deposited in the fuel and distributed as the neutron power, iii) the decay heat due to photon absorption is distributed as defined in step 3 (see beginning of Section 3.2). This approach assumes that for the length of the transient of interest (~60 seconds) the alpha and beta contributions to the decay power come mainly from the short-lived fission products whose distribution depends more on the instantaneous power than on the burnup distribution. Equations 1 and 2 gives the relationships used to generate the decay power distribution for the fuel and non-fueled regions.

For each non-fueled region *i*, the normalized time-dependent power at each axial node *j* was calculated from:

$$f_{i,j} = f_{\gamma\, deposited\, in\, region\, i,j} \cdot f_{photon\, decay\, power}(t) \cdot P_{core\, decay\, power}(t) \big/ P_{steady-state\, of\ region\ i} \qquad (1)$$

For each fueled region *i*, the normalized time-dependent power at each axial node *j* was calculated from:

$$f_{i,j} = \left[f_{\gamma\, deposited\, in\, region\ i,j} \cdot f_{photon\, decay\, power}(t) + \left(1 - f_{photon\, decay\, power}(t)\right) \cdot f_{\alpha\beta\, deposited\, in\, region\ i,j} \right] \cdot \frac{P_{core\, decay\, power}(t)}{P_{steady-state\, of\ region\ i}} \qquad (2)$$

where $f_{\gamma\, deposited\, in\, region\, i,j}$ is the time-averaged normalized gamma power in node *j* of region *i*, $f_{photon\, decay\, power}(t)$ is the time-dependant fraction of gamma decay power, and $f_{\alpha\beta\, deposited\, in\, region\ i,j}$ is the normalized steady-state power distribution.

3.2.5 Neglected contributions to the decay heat

Photonuclear data for the (γ, n) reactions in beryllium are not provided with the standard release of MCNP5 (version 1.51) used in this work. Therefore, the photo-neutron contribution to the decay power could not be calculated using this methodology. This contribution should be relatively small since only slightly above 4% of the decay photons are above the 1.666 MeV threshold for the (γ,n) reaction in beryllium. This assertion is confirmed indirectly by a previous study [15] that calculated, using the delayed groups for neutrons and photon-neutrons, the combined decay heat contribution from those two processes to be less then 0.15% of total power after 20 seconds.

In this analysis, all the energy released from the decay (gamma and/or beta) of isotopes created through neutron capture was not taken into account. It is difficult to evaluate this contribution, because it depends strongly on the level of impurities in the various core materials. It is expected that for a short transient and a relatively new core, as the case is for test A/400/1 (about 9 months after startup), this contribution should be small.

4. Computational Results

4.1 ORIGEN-2 results

For the duration of the transient of interest (60 seconds after shutdown), ORIGEN-2 predicts that the photon spectrum will remain relatively unchanged (<2%) for the energy groups contributing more than 0.1% to the total decay gamma source. Only photons in the two fastest energy groups change significantly over that period. The small contribution of those two energy groups is obvious in Figure 5, which shows the time-average decay gamma spectrum predicted by ORIGEN-2.

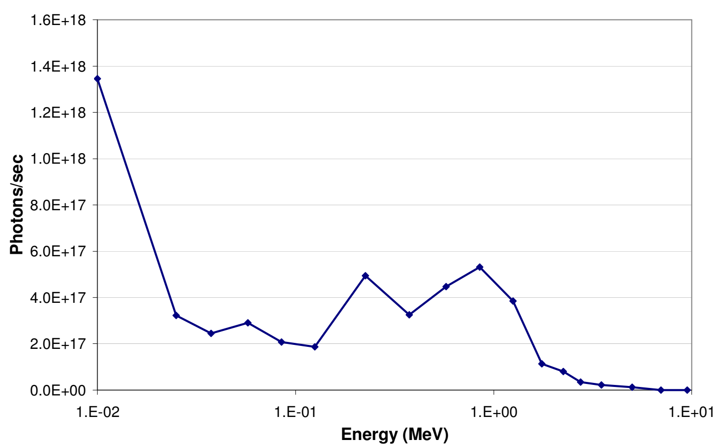

Figure 5 Average decay gamma spectrum predicted by ORIGEN-2

Since the spectrum remains relatively unchanged, the time-average gamma energy spectrum was used to calculate a single time-averaged gamma decay power distribution for each region for the whole transient.

The total and gamma decay power curves were also obtained from ORIGEN-2. Figure 6 shows the fraction of decay power attributed to gamma decay as a function of time, as well as a comparison between power decay curves from the ANSI/ANS 5.1 - 2005 standard [16] and ORIGEN-2.

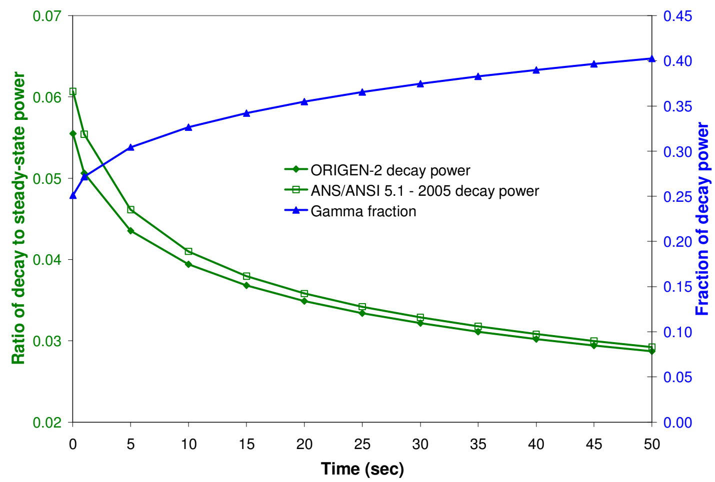

Figure 6 ORIGEN-2 relative decay power curve and gamma fraction over 60 seconds

Figure 6 shows that the decay power predicted by ORIGEN-2 (with the reactor-specific cross sections) and that predicted by the ANSI/ANS standard are in close agreement. This is especially true at the time the cladding temperature peaks i.e., ~40 seconds after reactor scram.

4.2 MCNP5 results

To calculate the power distributions, an MCNP model for test A/400/1 was developed. This updated MCNP model uses the ORIGEN-2 fuel compositions, the BERYL beryllium poison concentrations, the control rods withdrawn to 551mm and the regulating rods to 250mm. The bias to critical of this model remains similar to the fresh core model, i.e., about 0.7% Δk/k.

4.2.1 Steady-state power distributions

Table 2 gives the total power in each of the regions modeled in RELAP.

Table 2 MCNP5 steady-state power in regions of interest

Region	Total power (MWth)	Fraction
Beryllium hex of instrumented channel	0.019	0.0008
Beryllium hex of average channel	0.190	0.0079
Beryllium "hex" of plugged channel	0.409	0.0171
Beryllium bypass channel	0.075	0.0031
Fuel average element	21.05	0.8793
Fuel instrumented element	1.910	0.0798
Central plug of instrumented channel	0.004	0.0002
Central plug of average channel	0.032	0.0013
"Central plug" of plugged channel	0.310	0.0104

From Table 2, it can be observed that about 4% of the steady-state power is deposited in regions other than the fuel. The calculated F-346 channel power (1.91 MWth) agrees quite well with the measured channel power of 1.906 MWth [5].

Figure 7 shows the normalized axial power distribution in fuel elements calculated with the "best estimate" methodology and the "conservative" methodology.

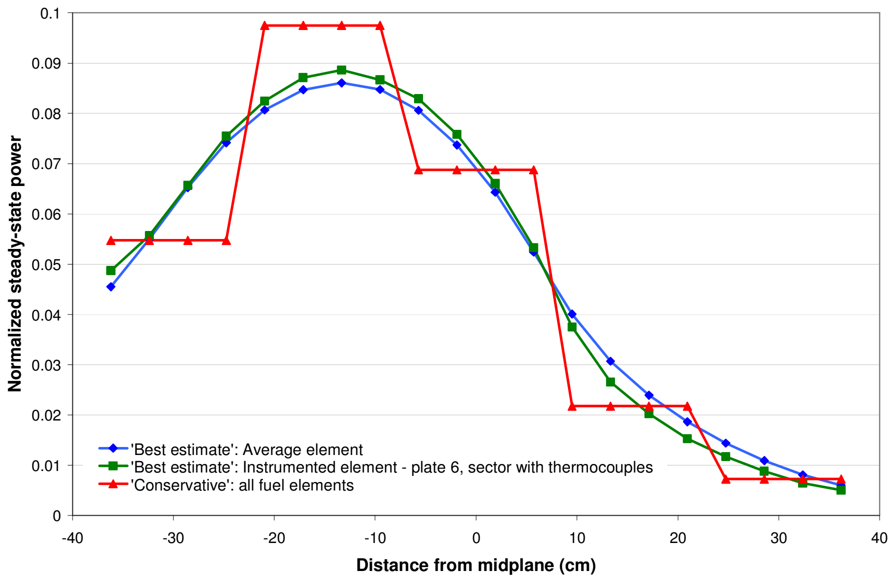

Figure 7 Comparison of "best estimate" and "conservative" normalized steady-state power distributions in fuel elements

Figure 8 shows the normalized axial power distributions from the "best estimate" methodology for the non-fueled regions illustrated in Figure 4.

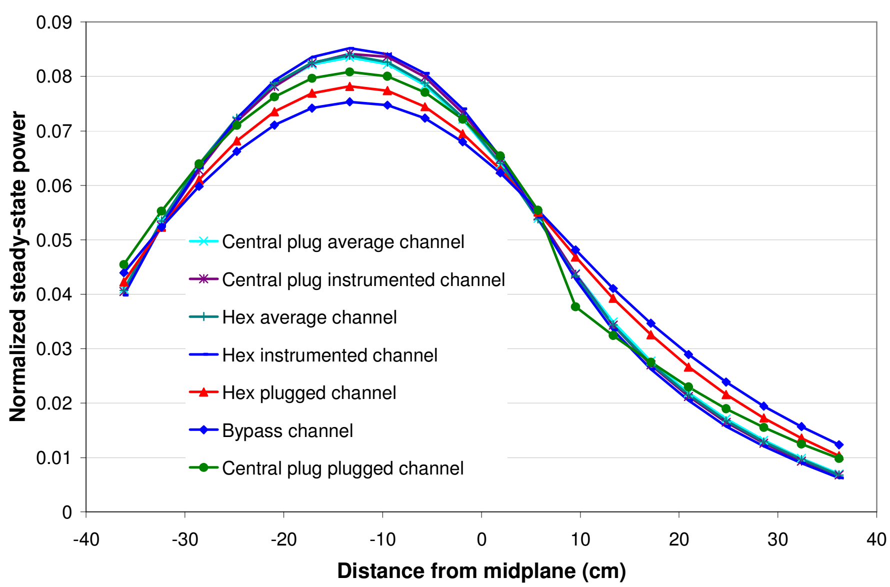

Figure 8 MCNP5 steady-state power axial profiles in no-fueled regions

4.2.2 Gamma and total decay power distributions

Using the photon source defined from the information shown in Figures 5 and 7, a photon transport calculation was performed to calculate the gamma decay power distribution. Table 3 gives the fraction of gamma decay power deposited in each RELAP region (see Figure 4).

Table 3 MCNP5 gamma decay power for the various regions of interest

Region	Fraction
Beryllium hex of instrumented channel	0.014
Beryllium hex of average channel	0.146
Beryllium "hex" of plugged channel	0.286
Beryllium bypass channel	0.047
Fuel average element	0.282
Fuel instrumented element	0.018
Central plug of instrumented channel	0.005
Central plug of average channel	0.032
"Central plug" of plugged channel	0.170

It can be seen from Table 3 that only about 30% of the gamma decay power remains in the fuel. Considering the gamma fraction of the total decay power, the “best estimate” methodology predicts that only about 70% of the total decay power will remain in the fuel near flow inversion (~40 seconds from reactor scram) as opposed to about 96% at steady-state (see Table 2). As shown in Section 4.2.4, taking credit of this effect results in a lower peak cladding temperature.

Since a significant fraction of the decay power comes from gammas, in addition to the redistribution among the various regions of the core, the fuel axial power distribution could also be affected. The axial decay power distribution for the instrumented fuel element sector is shown in Figure 9. This figure also shows the ratio of the normalized decay power distribution, calculated using Equation 2, to the normalized steady-state power distribution.

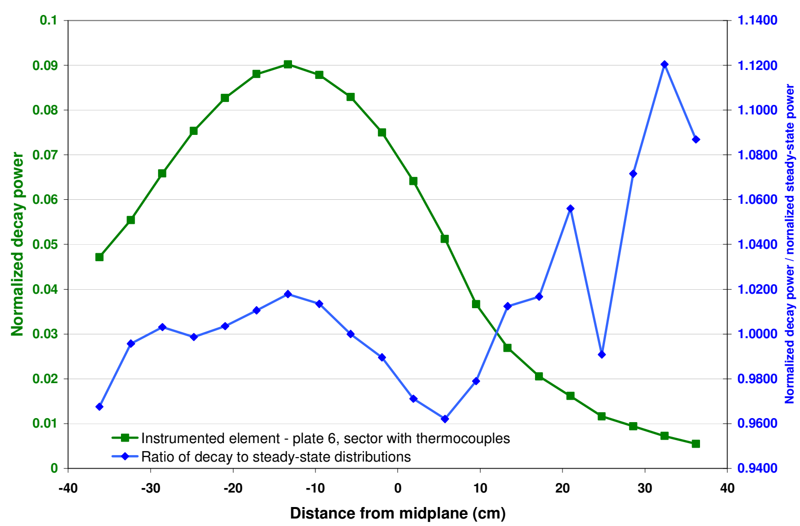

Figure 9 MCNP5 gamma decay power axial profiles in all non-fueled regions

A small axial shift toward the top of the fuel element is predicted by the "best estimate" methodology. However, this axial shift in power has a small impact (a few tenths of a degree C) on the predicted peak cladding temperature, compared to that predicted by a simulation using the normalized steady-state power distributions instead.

4.2.4 RELAP peak cladding temperatures

Using the steady-state and decay power distributions from both methodologies, the cladding temperature was calculated by RELAP at various heights in the instrumented fuel element (F-346). Figure 10 shows measured and predicted peak cladding temperatures (300mm below the mid-plane) during the test A/400/1.

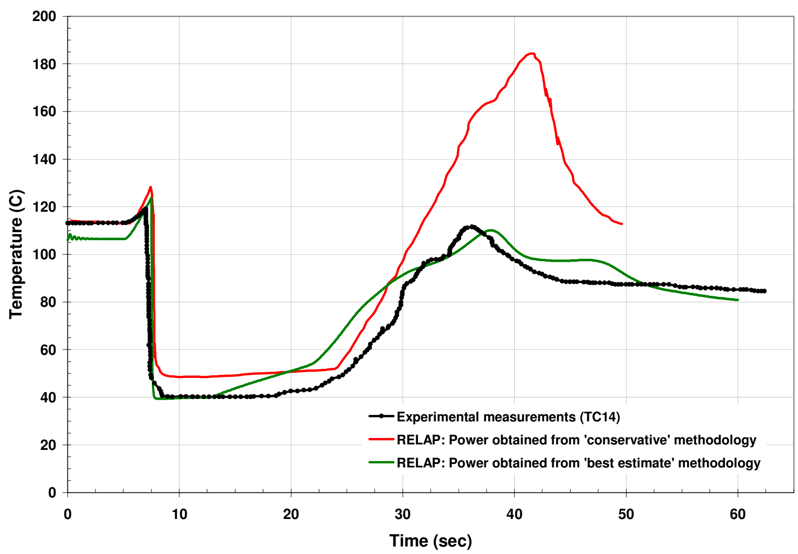

Figure 10 Experimental and predicted peak clad temperature in fuel element F-346

With the "best estimate" methodology the RELAP predictions are in a significantly better agreement with measurements.

5. Conclusions

To support the BR2 HEU to LEU conversion safety analyses, a RELAP model has been developed. To establish the credibility of this model and extend the validation database of RELAP, simulations of the 1963 flow test A/400/1 have been preformed. To improve the agreement between code predictions and measurements it was necessary to improve the prediction of the decay power distribution after reactor shutdown. Even though a more "conservative" approach can be deemed acceptable from a safety perspective, it can provide predictions that differ significantly from experimental measurements.

A "best estimate" methodology was developed to predict the neutron and gamma power distributions at steady-state, as well as the alpha/beta and gamma decay power distributions during the transient. Detailed MCNP5 and ORGIEN-2 models were developed for BR2 "configuration 4" to: i) extract reactor-specific cross sections from MCNP5 for predicting the Li-6 concentration in the Be matrix using BERYL, as well as xenon concentrations and burnup using ORIGIEN-2, ii) predict power decay curves and decay gamma spectrum at the time of the test using ORIGEN2, iii) calculate neutron and photon power distributions at steady-state, iv) calculate decay photon power distributions.

The power distributions calculated using the "best estimate" methodology significantly improve the agreement between peak cladding temperatures predicted by RELAP and experimental measurements.

6. Acknowledgements

Acknowledgement must be given to contributions of S. Kalcheva and S. Heusdains who provided the initial MCNP and RELAP input files, respectively, used as a basis to develop the models used in this work.

7. References

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