Measurements and Calculations of the Fuel Temperature Coefficient of Reactivity for Hinkley Point 'B' AGR

- by -

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SUMMARY

Tests have been carried out on one of the Advanced Gas-cooled Reactors at Hinkley Point to determine the fuel temperature coefficient of reactivity, an important safety related parameter. Reactor neutron flux was measured during transients induced by movement of a bank of control rods from one steady position to another. An inverse kinetics analysis was applied to the recorded flux transient to determine the reactivity change as the fuel temperature changed, and the variation of mean fuel temperature was derived from the flux transient by a multiplane thermal hydraulics code representing an AGR fuel channel. The fuel temperature coefficient was then obtained from the slope of a plot of core reactivity against fuel temperature. The uncertainty to be applied to the derived temperature coefficient has been shown to be \( \sim \pm 10\% \) at the 1 \( \sigma \) level. The experimental technique has been found to be simple to apply on a commercial reactor and has given consistent results over a range of reactor operating conditions.

Calculations of fuel temperature coefficients of reactivity (based on the lattice code ARGOSY) have been carried out and reactor averaged values deduced for comparison with experiment. The calculated and measured coefficients agree to within one standard deviation over a range of core irradiations and power levels.
1. INTRODUCTION

The fuel temperature coefficient of reactivity ($a_u$) in an Advanced Gas-cooled Reactor (AGR) plays an important part in limiting the temperature transient caused by reactivity faults. The temperature coefficient is negative by virtue of the $^{238}$U Doppler effect but reduces in magnitude with the build-up of $^{239}$Pu during burn-up. Because of uncertainties in the predictions of these counteracting effects, particularly at high burn-ups, the resulting uncertainty on the total coefficient may be large. It is therefore desirable to check the calculated reactivity feedback against measurements on the type of plant in question. This is being done on one of the AGR's at Hinkley Point using a novel transient technique which separates out fast and slow acting feedback mechanisms. This paper presents a review of measurements made to date and the results of these measurements are compared with predictions.

2. REACTIVITY FEEDBACK IN AGR

In an AGR the only significant temperature feedback mechanisms occur in fuel and graphite. The fuel consists of 14 mm diameter UO$_2$ pins clad in stainless steel. Assemblies of 36 pins are supported by steel grids inside two concentric graphite sleeves. The sleeves act as a thermal and physical barrier between the hot upward flowing CO$_2$ coolant in the fuel channel and the downward 're-entrant' flow of CO$_2$ at lower temperature which cools the bulk graphite moderator.

At the start of life, Doppler broadening of the $^{238}$U resonances provides strong negative reactivity feedback from fuel temperature changes. However, as burn-up proceeds, the build-up of $^{239}$Pu in the fuel contributes an increasingly positive component to the fuel temperature feedback. This arises because neutron spectrum hardening, associated with the change of the temperature of the oxygen in the fuel,
causes an increase in fission rate in $^{239}\text{Pu}$ relative to absorptions in the lattice (owing to the non-$1/\nu$ thermal cross-section of $^{239}\text{Pu}$). The overall mean core fuel temperature coefficient of reactivity in AGR remains negative over reactor life because burnt-up fuel with high $^{239}\text{Pu}$ content is continually replaced by new fuel.

Reactivity feedback also arises from changes in moderator temperature. These effects are generally small in AGR because the re-entrant coolant keeps the bulk moderator cool and the thermal time constant of the graphite is long. There is a fast acting component which arises from temperature changes in the inner graphite sleeve which is in contact with the hot main coolant gas stream.

The presence of these moderator feedback mechanisms makes it difficult to measure the fuel temperature feedback coefficient by the usual steady state methods. Consequently a transient technique, which uses the thermal time constants of the various components to separate out these feedback effects, has been developed.

3. DESCRIPTION OF MEASUREMENT TECHNIQUE

The measurement technique used at Hinkley Point consists of applying a small positive reactivity ramp to the reactor by moving control rods, followed by a pause for the reactor to respond and then a negative reactivity ramp to restore the reactor to its initial configuration.

In the Hinkley Point AGR the control rods are grouped in four banks, each being symmetrically distributed over the core. Two of the banks consist of 'black' rods which are fully withdrawn during normal full power operation. The other two banks consist of a total of 37 'grey' regulating rods, each of which is normally under individual auto-control to maintain the local coolant gas outlet temperature at a preset level. Consequently the grey rods are usually distributed about their normal mean bank position of about 50% inserted. The two grey rod banks can be moved under manual control as one bank or two separate banks.
The test consists of raising one or both banks of grey rods rapidly by about 100 mm to add up to $\sim 50$ mN of reactivity$^a$. The rods are then held in the raised position for sufficient time for significant fuel temperature feedback to develop (typically 20 - 60 seconds) and then returned rapidly to their original position to ensure that the change in bulk moderator temperature is minimised and that the effect on the rest of the plant is small. The resulting neutron flux transient is recorded using the installed linear power instrumentation (an ionisation chamber on the boiler shield wall at mid-core height). The analysis of this recorded flux transient with a computer program APRECOT developed at Berkeley Nuclear Laboratories gives the fuel temperature coefficient of reactivity directly.

4. ANALYSIS TECHNIQUE

The method of analysis has been described in detail elsewhere (1). Briefly, the reactivity of the core is derived from the neutron flux using the well known inverse kinetics analysis and fuel temperature changes are derived from the recorded flux transient by solving a set of time dependent thermal hydraulics equations (extracted from the 1-D AGR fault study program KINAGRAX (2)). The important feature of the technique is that for the two periods when the control rods are stationary reactivity is governed solely by the feedback mechanisms of which the dominant one is fuel feedback. Thus a plot of reactivity against fuel temperature for the times when the control rods are stationary should give two straight lines whose slopes are the fuel temperature coefficient of reactivity. The small reactivity changes which occur during the transient due to changes in graphite temperatures are accounted for by applying corrections to the transient reactivity based on the calculated graphite temperature changes and on calculated temperature coefficients of reactivity. A least squares fitting routine in the APRECOT program
gives both the fuel temperature coefficient and a measure of the scatter on the experimental points.

5. TESTING OF ANALYSIS TECHNIQUE

The analysis route described above has the advantage of ease of use: it gives the fuel temperature coefficient directly from the measurement of neutron flux and requires no knowledge of the control rod reactivity insertion rate. There are however assumptions implicit in this route, of which the major one is that the neutron flux distribution does not change significantly during the transient. This assumption has been tested (1, 3) by simulating typical transients using the 1-D code KINAGRAK and the 3-D code SKARK (4). Transient fluxes calculated at different radial and axial oxidations were analysed and the values of $\alpha_u$ obtained were compared with those derived from the mean flux transient. These studies have shown that the uncertainty introduced by changes in flux shape is less than 2%.

Further testing of the route has taken the form of simulations of particular experiments using KINAGRAK and the 1-D thermal hydraulics code SKARC (5). In this work the transient gas temperatures in particular instrumented fuel channels were modelled and compared with temperatures recorded in those channels during the experiments. Agreement to better than 1°C in a transient of amplitude 20°C was seen throughout the transient. This increases confidence in the thermal hydraulics model used in the experimental analysis. KINAGRAK as a fault study method has been checked extensively in a code to code comparison (6) and by detailed analytical and numerical studies of modelling assumptions.

6. ESTIMATION OF ERRORS

Uncertainties fall into three categories:-

(a) experimental data errors; illustrated by the scatter of experimental points and the reproducibility of results.
(b) model data e.g. errors in heat transfer data, delayed neutron fraction.

(c) modelling errors, e.g. assumption of constant flux shape.

6.1 Experimental Errors

The errors quoted on measured temperature coefficients, as derived from the least squares fitting vary from ± 1% to ± 3% showing the high statistical accuracy of the measurements. Furthermore, the agreement between individual tests carried out on the same day is good, usually being within the quoted 1σ uncertainty.

6.2 Model Data

A study of the sensitivity of the analysis to the various items of data has been carried out (7). This study showed that the important sources of error were: initial power, coolant flow, flux shape, heat transfer coefficients and delayed neutron fraction. Realistic estimates of the uncertainties of these parameters together with the results of the sensitivity study were used to derive the uncertainty in temperature coefficient. In addition the uncertainty due to errors in the correction terms for graphite temperature feedback were derived. The study is summarised in Table I. The total uncertainty due to model data uncertainties was shown to be ~ 7%.

6.3 Modelling Errors

It is difficult to quantify in any rigorous way the uncertainties by errors in modelling. The assumption of constant flux shape has been tested and shown to introduce uncertainties of ~ 2%. The thermal hydraulics part of the analysis route has been extensively tested as part of a fault study route validation and has been shown to give gas temperatures which agree with measurements made during fuel temperature coefficient tests. It is therefore believed, with some confidence that no major errors of modelling exist.
6.4 Summary of Errors

Of the three identified sources of error, experimental and model data errors have been quantified at 3% and 7% respectively. The third source of error has been quantified in part at 2% but contains other components which are not quantifiable. It is considered that placing an overall uncertainty of ± 0.10 mN/°C (corresponding to ~ 10%) will cover any additional unquantified errors due to modelling.

7. RESULTS

Nine sets of tests have been carried out to date at different power levels and core burn-ups (7, 8, 9).

7.1 Low Power Tests

These tests are usually carried out during a reactor start-up. A convenient pause in the AGR start-up procedure occurs when the reactor heat is first fed to the boilers. The power level at this time is ~ 200 MW (th). The results of all the low power tests are given in Table II. It can be seen that, with the exception of the May 1977 tests, there is a high level of consistency within the sets of tests. Tests carried out at this power level can be affected by the presence of long term reactivity changes, e.g. due to xenon or unsteady initial plant conditions. The effect of such reactivity drifts is generally small (~ 0.02 mN/hour) and can lead to errors of a few percent in fuel temperature coefficient. However, the May 1977 tests took place during a decaying xenon transient, about 20 hours after a reactor shutdown from full power. This gave rise to reactivity drifts of ~ 2 mN/minute which is sufficient to account for the discrepancies seen in this set of tests. Subsequent analyses contained a correction for slow linear drifts in reactivity where necessary.
Figures (1) and (2) illustrate the outcome of one test (the first of the January 1977 set). Figure (1) shows the control rod perturbation applied and the resulting neutron flux transient. The graph shows both experimental points and the results of a KINAGRA simulation. Figure (2) shows the plot of core reactivity corrected for graphite temperature changes against mean fuel temperature. The two linear parts of the plot corresponding to the times at which control rods were stationary can be seen quite clearly.

7.2 High Power Tests

These tests are carried out at the normal operating power level ($\sim$ 1200 MW (th)) and so may be carried out at any time convenient to the reactor operators. These tests are not, in general, susceptible to long term transients since steady initial conditions can usually be achieved. The results of all the tests are given in Table III. It can be seen that, with the exception of the second test in September 1979, there is a high level of consistency within the sets. The discrepancy seen in that test was due to insufficient time being allowed after the first test for the reactor to settle back into a steady state. The results of the final May 1977 test are illustrated in Figure (3), the two linear parts of the plot being clearly visible once more.

8. Calculation of Temperature Coefficients of Reactivity

AGR temperature coefficients are usually calculated using the lattice code ARGOSY (10). ARGOSY derives data for up to three neutron energy groups: fast, resonance and thermal; the resonance group calculation, which is the most important consideration in computations of temperature coefficients, relies on a tabulation of resonance integral. The tabulation is set up using the NR approximation for neutron collisions in the moderator region but in the fuel region the $\lambda$-method (11) is used. Temperature coefficients calculated using ARGOSY have been
shown to be consistent with those derived from the more sophisticated transport code WIMS (12).

The ARGOSY code derives group constants as a function of fuel and moderator temperatures, fuel enrichment and burn-up. Polynomial fits to these group constants may be set up within ADOS (AGR Data Operating System) which may then be used to set up data for reactor codes. A data handling code AKTS derives nuclear data for the SKAR suite of programs from the polynomial fits using irradiations, fuel types (i.e. enrichments), temperature and flux shapes provided by Hinkley Point Station staff. A further code, ALPHA (13), has been developed which is based on AKTS; this derives maps of fuel and moderator temperature coefficients and then spatially condenses these maps using suitable weighting functions, to give fuel and graphite temperature coefficients averaged over planes of fuel elements and over the whole core.

9. AVERAGING OF TEMPERATURE COEFFICIENTS OF REACTIVITY

The weighting functions used in ALPHA have been derived from a consideration of the 3-D neutron diffusion equation. The derivation is described in detail elsewhere (13) and results in

\[
\frac{\alpha_u}{\Delta T_F} = \frac{1}{\int \psi k_o \Sigma_a \alpha_u \Delta T_F \phi \, dV} \int \psi k_o \Sigma_a \phi \, dV
\]

It can be seen that the reactivity change due to temperature changes is weighted with flux (\(\phi\)), multiplication coefficient (\(k_o\)), absorption cross-section (\(\Sigma_a\)) and a weighting factor (\(\psi\)). First order perturbation theory indicates that the weighting factor \(\psi\) should be the adjoint flux \(\psi^+\). In practical situations the fuel temperature change is not known, but, in situations where the coolant flow and flux shape are constant, \(\Delta T_F\) may be taken to be proportional to the difference between the local fuel temperature and the coolant inlet temperature.
The exact value of $\overline{a}_n$ depends upon the, entirely arbitrary, way in which $\overline{\Delta T}_f$ is defined. In this work $\overline{\Delta T}_f$ has been volume weighted radially but $\phi^+$ weighted axially in order to be consistent with the experimental analysis.

10. COMPARISON OF MEASUREMENTS AND THEORY

Measurements of the fuel temperature coefficient of reactivity have been made at Hinkley Point 'B' at regular intervals. The earlier tests were carried out at ~ 15% power but as confidence in the technique increased tests were carried out at normal operating conditions. The results of all the tests carried out and analysed to date are shown in Figure (4). Low power results are corrected to full power conditions using theoretical correction factors derived from ARGOSY. The figure also shows values of fuel temperature coefficient of reactivity calculated using ALPHA. It can be seen that the measured values confirm the predictions, within the estimated uncertainties, both on absolute level and on variation with power level and burn-up.

The fuel temperature coefficient of reactivity for Hinkley Point 'B' has also been measured using a control rod oscillation technique (16). The results obtained from these oscillation tests confirm those presented here.

11. CONCLUSIONS

A transient technique for measuring fuel temperature coefficient of reactivity has been developed. This technique uses the difference between thermal time constants of the various reactor components to discriminate between their feedback contributions. The 1σ level uncertainty to be applied to the derived coefficient has been shown to be $\sim \pm 10\%$. The method is simple to apply to an operating power reactor and has been shown to give consistent results. Theoretical predictions of the fuel temperature coefficient show good agreement with the measured values in absolute level and in variation with power level and burn-up.
ACKNOWLEDGEMENT

This paper is published by permission of the Central Electricity Generating Board.
REFERENCES


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5. Telford, A. R. R., 1980, CEGB Report RD/B/N4900, Analysis of Instrumented Stringer Data Recorded during $\alpha_u$ Tests at Hinkley Point 'B'.


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| Figure (4) | Comparison of Measurements with Theoretical Predictions. |</p>
<table>
<thead>
<tr>
<th>Parameter</th>
<th>$1\sigma$ uncertainty</th>
<th>Uncertainty in $\sigma_u$ (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Initial power</td>
<td>3%</td>
<td>4</td>
</tr>
<tr>
<td>Coolant flow</td>
<td>4%</td>
<td>2</td>
</tr>
<tr>
<td>Flux shape</td>
<td>-</td>
<td>3</td>
</tr>
<tr>
<td>Can to coolant</td>
<td>3%</td>
<td>1</td>
</tr>
<tr>
<td>Fuel to can</td>
<td>$\leq 3$</td>
<td>3</td>
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<tr>
<td>Heat Transfer Coeffs.</td>
<td>10%</td>
<td>2</td>
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<tr>
<td>Inter sleeve</td>
<td>$\leq 2$</td>
<td>1</td>
</tr>
<tr>
<td>Delayed Neutron fraction</td>
<td>4%</td>
<td>2.5</td>
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<tr>
<td>Graphite feedback coefficients:</td>
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<td></td>
</tr>
<tr>
<td>inner sleeve</td>
<td>0.1 mN/°C</td>
<td>1</td>
</tr>
<tr>
<td>bulk moderator</td>
<td>1.0 mN/°C</td>
<td>1</td>
</tr>
<tr>
<td>Total uncertainty (adding in quadrature)</td>
<td>7%</td>
<td></td>
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</tbody>
</table>
### TABLE II
Low Power Measurements of $\alpha_u$

<table>
<thead>
<tr>
<th>Date of Test</th>
<th>Source Reference</th>
<th>Core Burn-up (GWD)</th>
<th>$\alpha_u$ (mN/°C) $^b$</th>
</tr>
</thead>
<tbody>
<tr>
<td>March 1976</td>
<td>7</td>
<td>0</td>
<td>-1.61 - 1.64</td>
</tr>
<tr>
<td>January 1977</td>
<td>7</td>
<td>100</td>
<td>-1.27 - 1.28</td>
</tr>
<tr>
<td>&quot;</td>
<td>&quot;</td>
<td>&quot;</td>
<td>-1.19 - 1.25</td>
</tr>
<tr>
<td>&quot;</td>
<td>&quot;</td>
<td>&quot;</td>
<td>-1.20 - 1.20</td>
</tr>
<tr>
<td>&quot;</td>
<td>&quot;</td>
<td>&quot;</td>
<td>-1.17 - 1.15</td>
</tr>
<tr>
<td>May 1977</td>
<td>7</td>
<td>196</td>
<td>-1.06 - 1.64</td>
</tr>
<tr>
<td>&quot;</td>
<td>&quot;</td>
<td>&quot;</td>
<td>-1.07 - 1.32</td>
</tr>
<tr>
<td>&quot;</td>
<td>&quot;</td>
<td>&quot;</td>
<td>-1.11 - 1.27</td>
</tr>
<tr>
<td>August 1979</td>
<td>8</td>
<td>530</td>
<td>-1.04 - 0.96</td>
</tr>
<tr>
<td>&quot;</td>
<td>&quot;</td>
<td>&quot;</td>
<td>-1.02 - 0.99</td>
</tr>
<tr>
<td>&quot;</td>
<td>&quot;</td>
<td>&quot;</td>
<td>-1.02 - 0.97</td>
</tr>
<tr>
<td>&quot;</td>
<td>&quot;</td>
<td>&quot;</td>
<td>-1.02 - 0.98</td>
</tr>
<tr>
<td>July 1980</td>
<td>9</td>
<td>856</td>
<td>-1.01 - 1.02</td>
</tr>
<tr>
<td>&quot;</td>
<td>&quot;</td>
<td>&quot;</td>
<td>-1.03 - 0.98</td>
</tr>
</tbody>
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$^b$ The two values of $\alpha_u$ given are those derived from the two periods when control rods are stationary.
<table>
<thead>
<tr>
<th>Date of Test</th>
<th>Source Reference</th>
<th>Core Burn-up (GWD)</th>
<th>$\alpha_u$ (mW/°C)$^c$</th>
</tr>
</thead>
<tbody>
<tr>
<td>May 1977</td>
<td>7</td>
<td>184</td>
<td>-0.79 - 0.81</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>-0.79 - 0.81</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>-0.81 - 0.86</td>
</tr>
<tr>
<td>March 1978</td>
<td>7</td>
<td>280</td>
<td>-0.78 - 0.79</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>-0.79 - 0.81</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>-0.79 - 0.80</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>-0.77 - 0.82</td>
</tr>
<tr>
<td>September 1979</td>
<td>8</td>
<td>560</td>
<td>-0.70 - 0.71</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>-0.67 - 0.80</td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>-0.68 - 0.73</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>-0.71 - 0.71</td>
</tr>
<tr>
<td>July 1980</td>
<td>9</td>
<td>862</td>
<td>-0.67 - 0.66</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>-0.64 - 0.65</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>-0.64 - 0.66</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>-0.60 - 0.62</td>
</tr>
</tbody>
</table>

$^c$ The two values of $\alpha_u$ given are those derived from the two periods when control rods are stationary.
Footnote

The units of reactivity used throughout this paper are milliniles (mN). One millinile is equal to $10^{-5}$ in absolute neutron multiplication, 50 mN is equivalent to $\sim 0.1$ for the AGR core at the time of the tests described.
Measurements and Calculations of the Fuel Temperature Coefficient of Reactivity for Hinkley Point 'B' AGR

Dr. A. R. R. Telford

Figure (3)

- Kinagrax simulation.
- Linear power
- Control rod position

Reactor power (MW)

0.0 10.0 20.0 30.0 40.0 50.0 60.0 70.0 80.0 90.0 100.0 110.0 120.0 130.0

Time (secs)

0.0 12.0

10.0

8.0

6.0

4.0

2.0

0.0

-2.0

-4.0

Rod movement (cm)
Measurements and Calculations of the Fuel Temperature Coefficient of Reactivity for Hinkley Point 'B' AGR

Dr A.R.R. Telford

Figure (2)
Measurements and Calculations of the Fuel Temperature Coefficient of Reactivity for Hinkley Point B AGR.

Figure (3)

Reactivity, mN

- Rods stationary
- Rods moving in
- Rods moving out
- Mean fuel temperatures
Measurements and Calculations of the Fuel Temperature Coefficient of Reactivity for Hinkley Point 'B' AGR

Dr R. R. Telford

Figure 4

![Graph showing the fuel temperature coefficient of reactivity vs total core burn up (GWD). The graph includes points for high power and low power results, as well as theoretical predictions.]

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Theoretical predictions.

High power results

Low power results

+ Theoretical predictions.
High power results

Theoretical predictions.

Measurements and calculations of the fuel temperature coefficient of reactivity for Haldon Pool 18, AGR.