RESEARCH ACTIVITIES RELEVANT TO THE APPLICATION OF LEU AT THE HOR*

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Scope

Synoptic information is given concerning research activities at the Hoger Onderwijs Reactor (HOR), the results of which may be meaningful to the application of high-density, low enriched uranium fuel (LEU). The HOR, a 2 MW pool-type reactor, is one of the main research tools of the Interuniversity Reactor Institute (IRI), located at the campus of the Delft University of Technology, Netherlands. In the framework of a research program on the application of LEU at the HOR, the activities of both the Reactor Operations Group and the Reactor Physics Group of IRI are partially merged in a concerted effort to assess the consequences of core conversion. Part I - contributed by the Reactor Operations Group - deals with experimental results on fuel temperature measurements of a standard MTR-type fuel element, fitted with thermocouples in the fuel cladding. Part II - contributed by the Reactor Physics Group - gives an outline of programmed experimental research on the thermohydraulics of MTR-type fuel elements with an emphasis on detecting normal and off-normal cooling conditions.

Part I HOR Fuel Temperature Measurements

I.1 Introduction

In order to assess safety margins and to verify theoretical calculations, the Reactor Operations Group in cooperation with the Reactor Physics Group of the Interuniversity Reactor Institute at Delft, Netherlands, has performed fuel temperature measurements at the HOR for different experimental conditions of the reactor core. The HOR, a pool-type reactor, can be operated in two different modes, viz. with natural convection cooling up to a power level of 0.5 MW and with forced convection cooling up to a power level of 2 MW. Measurements have been performed for both modes of operation, but for the sake of brevity only results for the forced cooling mode are reported here. Further measurements are programmed for 1983, focussing on the influence of coolant flow rate

+ Supervising professor, coordinating Reactor Physics Group activities
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variations and verification with theoretical results of the COBRA-3C/RERTR code. Although the instrumented fuel element is of the HEU-type, the measurements are considered to be useful to assess the actual departing conditions for core conversion and to verify computer code calculations with experimental results.

I.2 Experimental Setup

For studying the thermohydraulic phenomena in the core of the HOR and for experimental verification supporting safety assessments, IRI has the disposal of a fuel element instrumented with thermocouples in the fuel cladding (Instrumented Fuel Assembly, IFA). The fuel element is of the standard MTR-type, as normally in use at the HOR. The fuel is of the aluminide type, highly enriched in U-235 (93%). There are 19 flat fuel plates per standard element with a nominal loading of 10 grams U-235 per fuel plate. The central plate and one of the outer plates of IFA are fitted with Chromel-Alumel thermocouples of dimensions 0.17x0.50 mm in the cladding. The central fuel plate has an additional feature: The coolant channel inlet and outlet temperatures are monitored by pairs of thermocouples - protruding 1.5 mm in the coolant gap -, located at the top and bottom of the fuel plate, just a few millimeters away from where the fuel region starts.

Fig. 1 gives the location of the thermocouples in IFA. The fuel meat has a nominal thickness of 0.50 mm and the cladding thickness is nominally 0.30 mm. However, for manufacturing purposes the two intrumented fuel plates of IFA have a cladding thickness of 0.38 mm. The coolant gap is nominally 3.10 mm with locally allowed minima of 2.85 mm, but the gaps adjacent to the instrumented fuel plates are nominally 3.02 mm with minima of 2.77 mm. IFA can be loaded into any grid position of the core, except of course at the control element positions.

The thermocouple signals are either processed by an automatic data logging system with cold junction compensation for measuring steady state temperatures and long term behavior or they are processed by special amplifiers for studying fluctuation phenomena by noise analysis. Correlation techniques are used to derive coolant channel flow velocities.

I.3 Experimental Results

Several measurement runs have been made under different experimental conditions. One series of measurements, reported here, has been performed with the Instrumented Fuel Assembly loaded into the central grid position D5 of core no 49-02. Fig. 2 gives the core configuration. The instrumented outer fuel plate was facing the fuel element at grid position D4. The measurements were performed under relatively clean core conditions at power levels of 0.5, 1, 1.5, 2 and 2.3 MW respectively. At a power level of 2 MW the shim/safety rods were about 61% withdrawn from the core. Besides, the influence of reducing the core coolant flow by deliberately leaving out one of the grid plate plugs has been investigated. These plugs are used to prevent a core flow short-circuit by grid positions not occupied with fuel elements. The coolant flow is forced downwards the core, passes a suction head under the grid plate and is circulated by the primary coolant pump through the heat exchanger to be returned to the pool by means of a diffuser. The nominal coolant flow rate was about 61 liter per second, whereas the effective core coolant flow rate is estimated to be lower by some 5%.

Figures 3 and 4 show the measured temperature profiles in vertical direc-
tion along the fuel plate under steady state conditions for different power levels for the central and outer fuel plate respectively. The pool water temperature was between 25 and 35 degrees centigrade during the measurements. Table I shows the maximum fuel cladding temperature measured in the central and outer fuel plate together with the results calculated with the COBRA-3C/ RERTR code.

The following presumptions have been made for the calculations:

- Heat transfer, bulk temperature - one phase: Dittus-Boelter
  two phase: Bergles & Rohsenow
- Void and bubble detachment - Bowring model
- Channel inlet temperature - 30 degrees centigrade
- Bulk velocity
  - central channel : 50 cm/s
  - inter-element channel : 40 cm/s

Table I

<table>
<thead>
<tr>
<th>Power (MW)</th>
<th>Inlet Temp. (°C)</th>
<th>Measured</th>
<th>Calculated</th>
<th>Measured</th>
<th>Calculated</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Central Plate</td>
<td></td>
<td>Outer Plate</td>
<td></td>
</tr>
<tr>
<td>0.5</td>
<td>25</td>
<td>41</td>
<td>44</td>
<td>48</td>
<td>48</td>
</tr>
<tr>
<td>1.0</td>
<td>27</td>
<td>54</td>
<td>57</td>
<td>67</td>
<td>64</td>
</tr>
<tr>
<td>1.5</td>
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<td>35</td>
<td>83</td>
<td>88</td>
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<td>92(97)</td>
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<td>110(113)</td>
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<tr>
<td>3.0</td>
<td>-</td>
<td>103(107)</td>
<td></td>
<td>120</td>
<td>120(120)</td>
</tr>
</tbody>
</table>

( ) - value for inlet temperature of 40 °C

a - subcooled boiling, no bubble detachment; FIR = 1.7 Levy
b - subcooled boiling, no bubble detachment; FIR = 1.5 η=37

At power levels on the order of 2 MW, the measured temperature signals at steady state conditions were slowly fluctuating in the seconds to minutes time domain with amplitudes of 2-3 degrees centigrade. In general, it took several minutes to reach steady state conditions after a change in power level.

1.4 Discussion of Results

From the measurement results the following can be concluded: Obviously, the highest fuel cladding temperature occurs at the outer fuel plate. Due to the wider inter fuel element coolant channel gap, the maximum thermal neutron flux density of the outer fuel plate is about 3 to 4% higher compared to the central fuel plate. The power density of the outer plate is accordingly higher. This partly explains the higher temperatures reached in the outer plate. However, most of the difference in maximum temperature between the central and the outer fuel plate must be attributed to the less favourable cooling condi-
tions in the inter fuel element coolant channel gap. The coolant bulk velocity is estimated to be lower by about 20% in the inter element channel; this is confirmed by cross correlation analysis of the fluctuating parts of thermocouple signals. Furthermore, there exists a rather large temperature gradient in horizontal direction across the outer fuel plate (thermocouples 13, 14 and 15). At a power level of 2 MW the temperature at the boundary of the fuel region is about 25 degrees lower compared to the maximum temperature at the centre of the outer fuel plate. The inter fuel element coolant channel flow distribution is supposed to be far less uniform than in the fuel element itself. The shape of the inter element channel is less well defined and horizontal cross flow and leakage from the core boundary over the total height of the core are complicating factors, so it is not at all easy to make a quantitative assessment of the inter element channel coolant flow velocity distribution.

The agreement between measurement and calculation is rather good, taking into account the uncertainties. For the central fuel plate the measured maximum fuel cladding temperature is systematically lower than the calculated one. Fig. 3 shows temperature profiles that are definitely more flat than one would expect from the power density profile. Thermocouple no 6 is suspected of not functioning well, giving biased readings (absolute readings being too low). In fact, temperature noise analysis of the thermocouple signals shows nearly complete coherence in the low frequency range for the signals of couples no 6 and no 7. The signal leads of couple no 6 are embedded in the fuel cladding running just above and very near to the sensing position of couple no 7. It is anticipated that the signal leads to couple no 6 are damaged just in the neighbourhood of couple no 7. Earlier measurements did not show this discrepancy.

The effect of leaving out one grid plate plug can be clearly seen from Fig. 3 and Fig. 4. The maximum fuel plate temperature at a power level of 2 MW increases by about 3 degrees centigrade, so the consequences are rather moderate.

The coolant channel inlet temperature for the central fuel plate was monitored by thermocouple no 2. In the course of the experiments, with the power raised in steps, the measured inlet temperature increased by about 10 degrees centigrade, as can be seen from Fig. 3. However, this variation of coolant channel inlet temperature does not influence the maximum cladding temperature to the same extent. Thermocouple 10 is suspected to give biased readings due to a location too near to the fuel region.

Future work will focus on the influence of changes in the steady state value of coolant channel flow velocity on cladding temperature by systematically investigating temperature profiles with increased and reduced primary pump flow rate in combination with further attempts to deduce flow velocities by cross correlation techniques. The measurement results will be compared with calculated results of the COBRA-3C/RERTR code. Also, temperature measurements will be performed for grid positions at the core boundary.

Part II A Research Program on Thermohydraulics of MTR-Type Fuel Elements and Detection of Cooling Anomalies

II.1 Introduction

In the Reactor Physics Group of the Interuniversity Reactor Institute, experiments have been started to investigate boiling phenomena in a MTR-type
fuel element. The measurements will be performed in two nearly identical experimental loops. One of these will be positioned in the periphery of the core of the HOR, the other is meant for ex-core operation. These loops will be installed in the first half of 1983. At present preliminary experiments are under way in a small out-of-core experimental loop.

II.2 Experiment Description

The main component in all the formentioned loops is a dummy fuel element with three electrically heated flat 'fuel' plates and four coolant channels with a gap width of 3 mm. The thermocoax heating wires in the plates are embedded in such a pattern, that the axial power density distribution is cosine shaped and thus resembles the situation in an actual reactor core. For both fabrication and physical reasons (heat distribution over the plate surface) the fuel plates have a thickness of 4 mm.

Each plate contains 5 thermocouples at different axial positions; coolant inlet and outlet temperatures are measured with two thermocouples each. In addition 3 measurement tubes are provided for insertion of neutron and gamma detectors (relevant only for in-core experiments) and an acoustical detector is positioned in the coolant loop for boiling detection. In the ex-core loop and the presently operated small-scale loop, the dummy element is provided with windows which enable the application of optical methods for studying boiling phenomena. Maximum heating power for the first series of experiments is 10 kW; this means that the power density is in the same range as the average power density in our 2 MW reactor. The maximum heat flux density amounts to 60 kW/m² which is a factor 2.5 below the hot spot value of the HOR. This implies that for boiling studies we are restricted to high inlet temperatures or low flow rates. The use of higher power densities in follow-up experiments depends on procurement of additional funds for this type of research.

II.3 Purpose and Program

The main purpose of the investigations with the out-of-core loop is the study of one-phase and two-phase flow in narrow heated channels. Studies will be made with both upward and downward coolant flow with varying inlet temperatures, flow rate and heating power. In the out-of-core loop average temperatures and temperature fluctuations will be measured and different flow regimes will be studied. Boiling will be detected by optical and acoustical means as well as with noise analysis on thermocouple signals and axial pressure differences in the channels. Flow and bubble velocities will be inferred from noise analysis on thermocouple signals and optical transmission signals. Emphasis will be on accurate detection of incipient boiling, bubble detachment and the development of flow instabilities, both in upward and downward flow. All results will be compared with COBRA-3C/RERTR calculations.

The in-core loop will be used to study the influence of different flow conditions on signals of neutron and gamma detectors at various distances from the dummy element. The main purpose here is to investigate the possibility of timely detection of boiling and flow instabilities with remote detectors. Both stationary and transient conditions will be studied. The philosophy is that ex-core measurements will be used for computer code verification and preparation and verification of incore experiments. With respect to measurement techniques the loops are partly complementary; optical measurements are not possible in the core, radiation detection is restricted to the core, temperature
and acoustical measurements will be performed in both loops.

II.4 Possible Program Extension

It is hoped that the outlined experiments will provide useful information for the analysis of thermohydraulics in MTR-type reactors under normal and abnormal (loss of flow, partial blockage) cooling conditions. Another subject of relevance to safety is LOCA-analysis. Three dimensional analysis of after heat dissipation in a completely dry reactor core is under way in the framework of LEU-studies*. A more complex situation is a partially submerged core, where free convection of air is blocked by the water in the lower part of the core. In this case the heat dissipation depends on radiation, conduction and the production and convective transport of water vapour in the narrow channels, which is a rather complicated physical situation that seems to be less amenable to numerical analysis. We therefore consider it mandatory to perform LOCA-measurements of electrically heated MTR-type fuel elements. However, due to limited manpower and funds we are presently not in a position to include LOCA-analysis in our experimental program.

* IAEA Guidebook, Draft no 2, March 1982: Research Reactor Core Conversion, Safety and Licensing Issues
THERMOCOUPLE LOCATIONS ON OUTER FUEL PLATE OF INSTRUMENTED HOR FUEL ELEMENT

<table>
<thead>
<tr>
<th>Location</th>
<th>Description</th>
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<tbody>
<tr>
<td>1-4</td>
<td>Fuel Region 600.5 MAX</td>
</tr>
<tr>
<td>5-11</td>
<td>583.5 MIN</td>
</tr>
<tr>
<td>12-17</td>
<td>12 mm within min. heat boundary</td>
</tr>
<tr>
<td>18-20</td>
<td>Coolant temperature extending 15 mm in</td>
</tr>
<tr>
<td>21-23</td>
<td>Coolant gap</td>
</tr>
<tr>
<td>24-26</td>
<td>12 mm within min. heat boundary</td>
</tr>
</tbody>
</table>

- **583.5 MIN in coolant gap**
- **12 mm within min. heat boundary**

**Fig. 1** HOR Instrumented Fuel Element

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THERMOCOUPLES

Chromel-Alumel 0.170, 0.500 mm

The instrumented fuel element has to be provided with pin sleeves on both sides.

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**Fig. 1** HOR Instrumented Fuel Element
FIG. 2 HOR CORE CONFIGURATION

DATE: 22-6-1982
CORE NO: 49-02
CORE LOADING: 3956 g 235U

CRITICAL ROD POSITION AT A POWER LEVEL OF 2 MW AND REACTIVITY WORTH OF THE SHIM/SAFETY RODS

<table>
<thead>
<tr>
<th>ROD</th>
<th>Rod position(%)</th>
<th>Reactivity(%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>2</td>
<td>61.28</td>
<td>2.8</td>
</tr>
<tr>
<td>3</td>
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<td>3.3</td>
</tr>
<tr>
<td>6</td>
<td>-</td>
<td>-</td>
</tr>
</tbody>
</table>

REFLECTOR TEMP.: 33 °C (2 MW)
EXCESS REACTIVITY: 4.4% (COLD, CLEAN CORE)

FISSION CHAMBER: FC: 245
SAFETY DETECTOR A: 74
SAFETY DETECTOR B: 101
SAFETY DETECTOR C: 41
SAFETY DETECTOR D: 6

GRID POSITION: D4
FUEL ELEMENT NO: D-26
BURNUP PERCENTAGE: 13
FIG. 3 CENTRAL FUEL PLATE TEMPERATURE PROFILES

GRID PLATE PLUG AT POS. D1 LEFT AWAY
FIQS 4 OUTER FUEL PLATE TEMPERATURE PROFILES

[Graph showing temperature profiles with vertical positions and power levels indicated]