ON HOW A BEST ESTIMATE PLUS UNCERTAINTY ANALYSIS CAN IMPROVE THE DESIGN OF A RESEARCH REACTOR

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ABSTRACT

Thermal-hydraulic system codes have been used to evaluate the behaviour of research reactors in the steady state and during transients. The existence of uncertainties has forced the analyst to use a conservative approach to absorb deviations and guarantee that design criteria are satisfied. The best estimate method (BEPU) is an alternative providing a more realistic simulation of the case being studied. The best estimate calculation is complemented by an uncertainty analysis to obtain upper and lower bounds for relevant figures of merit. RELAP5 is used to model the behaviour of OPAL research reactor. Both, the conservative and BEPU approaches are considered.

The study is divided into sections, including: a) validation of the model, the determination of set point values for reactor trips and the evaluation of an initiating event considered for reactor design. The BEPU methodology results in a more efficient reactor design with improvements in its operation.

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1. Introduction

Thermal-hydraulic system codes have been specifically developed to evaluate transients in Nuclear Power Plants (NPP) and they have been successfully used in the design of Nuclear Research Reactors (RR). Most best-estimate thermal-hydraulic system codes, such as RELAP5 solve the mass, energy and momentum equations for the liquid and the vapour phases and they make use of validated empirical correlations to close the problem. Sources of uncertainties, including geometrical deviations, the lack of knowledge in the physical phenomena involved, the existence of undefined or uncertain boundary conditions, the deviations in the empirical correlations and the extensive use of approximate mathematical methods, have forced the designer to use conservative values, boundary conditions and hypotheses to absorb all deviations and guarantee that all design criteria are satisfied.

The best estimate modelling is an alternative approach providing a more realistic simulation of the case being studied, with a precision commensurate with the current knowledge of the phenomena involved. The best estimate calculation is complemented by an uncertainty analysis to obtain upper and lower bounds for the relevant figures of merit. The Best Estimate Plus Uncertainty (BEPU) methodology has been implemented to assess transients in NPP but its application has not been extended to the analysis of RR yet. In the design of NPP, the BEPU approach has proved useful as it has resulted in more economic designs and improved reactor performance.

The present study uses the thermal-hydraulic system code RELAP5 to model the behaviour of OPAL reactor in the steady state and during transients. Both, the conservative and the BEPU approaches are considered, thus resulting in two alternative ways to assess reactor behaviour. For the BEPU calculation approach, the “input error propagation” technique has been adopted. Uncertainties in operating parameters have been settled based on engineering judgment, measurements and experience. The study is divided into different sections, including a) a validation of both calculation approaches by comparing calculated values of figures of merit, such as coolant flows with experimental values, b) a comparison between the set point values of parameters used for reactor trips as calculated by the two methodologies and, c) the evaluation of a loss of flow...
accident (LOFA) to determine the size of the inertia flywheel of the primary cooling system pumps.

The analysis performed shows that the BEPU calculation approach may result in a more efficient reactor design with improvements in its operation. In particular, set points can be defined in such a way that the spurious triggering of a safety system due to a noisy signal is reduced. It also allows an optimization of the moment of inertia of the cooling pumps. Finally, the importance of validating the BEPU calculation approach also becomes relevant as it has been used for safety analysis and in the licensing of NPP and it could be extended to RR.

1. **Description of the cooling circuit**

The reactor considered for the analysis is the 20 MW OPAL reactor. During normal operation, the heat generated in the core is removed by demineralized light water flowing in an upward direction in a forced circulation cooling regime. The coolant is provided by a set of pumps which are part of the Primary Cooling System (PCS). Heat exchangers remove the heat to the Secondary Cooling System (SCS) and a decay tank has been added to allow activated nitrogen to decay.

During normal operation at low powers (≈ 400 kW) or following a pump stop, the power in the core is removed in the natural circulation cooling regime. Two set of flap valves have been placed in each inlet pipe for such purpose. The flap valves open when the pressure difference between the inlet pipes and the reactor pool is reduced, allowing the water in the pool to flow upwards through the core. The pumps in the PCS contain an inertia flywheel which provides the flow required to guarantee a smooth transition from the forced to the natural circulation cooling regime.

2. **Reactor design based on thermal-hydraulic design criteria**

The reactor design must guarantee that the fundamental safety function of heat removal is satisfied, so to preserve the integrity of the fuel. Thermal-hydraulic design criteria based on limiting physical phenomena have been established. These design criteria are considered for the reactor design and also to establish some of the set points values leading to the triggering of trips resulting in reactor shutdown.

The limiting phenomena considered in reactor design are Departure From Nucleate Boiling (DNB) and Flow Redistribution (RD). While different in nature, both of them result in vapour blanketing and a degradation of heat transfer which may lead to fuel damage. At low flows, these phenomena are known as Burn-Out (BO) and Boiling Power (BP) respectively.

Thermal-hydraulic design criteria have been established considering a margin to these limiting physical phenomena. These margins are summarized in Tab 1 and they are conservatively evaluated in a “hot channel”. The hot channel is the hottest channel in the core and it acts as an envelope to all the cooling channels. A peaking factor (PF) accounting for the non-homogeneous power distribution among the reactor core is considered to calculate the power in this hot channel.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Normal Operation</th>
<th>Design criteria</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Forced circulation</td>
<td>Natural circulation</td>
</tr>
<tr>
<td>DNBR=q^<em>_{DNB}/q^</em>_{max}</td>
<td>≥ 2.0</td>
<td>N/A</td>
</tr>
<tr>
<td>RDR=P_{RD}/P_{max}</td>
<td>≥ 2.0</td>
<td>N/A</td>
</tr>
<tr>
<td>BOR=q^<em>_{BO}/q^</em>_{max}</td>
<td>N/A</td>
<td>2.0</td>
</tr>
<tr>
<td>BPR=BP/P_{max}</td>
<td>N/A</td>
<td>2.0</td>
</tr>
</tbody>
</table>

**Tab 1: Thermal-hydraulic design criteria**

In Tab 1:

q^*_{DNB}: Heat flux leading to DNB
**$P_{RD}$**: Power leading to RD

**$q''_{BO}$**: Heat flux leading to BO

**BP**: Boiling Power

**$q''_{max}$**: Maximum heat flux, calculated as $PF* q''_{ave}$, being $q''_{ave}$ the average heat flux which is the ratio between the thermal power in the core and the total heat transfer area

The $q''_{DNB}$ is calculated by the Mirshak correlation (1). The $P_{RD}$ is calculated by the Whittle and Forgan correlation with the French formulation recommended by Fabrega (2). For low flows (coolant velocities < 1.3 m/s) the $q''_{BO}$ is calculated by using the Fabrega correlation (3) while the BP is calculated as the power required to achieve the saturation temperature.

### 3. Thermal-hydraulic design

The thermal-hydraulic design of a nuclear reactor relies on calculation codes. These codes make use of empirical correlations which introduce uncertainties to the calculations. Deviations from nominal operating parameters and fabrication tolerances are also a source of uncertainty. Therefore, both, the selection of an adequate calculation code and the treatment of uncertainties in the thermal-hydraulic analysis are two factors to be considered.

A calculation code which results adequate to perform a thermal-hydraulic calculation is that which can efficiently predict the behaviour of the reactor. A validation procedure is usually required and it implies a comparison between the measured and calculated value of a thermal-hydraulic relevant figure of merit.

In reference to the uncertainty treatment, two different approaches have been used: the conservative and the best estimate approach.

In the **conservative approach**, uncertainty factors, boundary conditions and modelling hypotheses are chosen in such a way that a pessimistic model of the reactor behaviour is obtained. The values to be chosen in the conservative approach depend on the operating condition of the reactor or the transient under analysis and also on the figure of merit chosen to evaluate the reactor behaviour. For the thermal-hydraulic design, these figures of merit are the design criteria in Tab 1. The conservative approach aims at absorbing all the deviations and uncertainties to guarantee that the design criteria are satisfied.

The **best estimate approach** considers boundary conditions, parameters and hypotheses leading to a realistic prediction of the reactor behaviour. This approach requires an uncertainty analysis to obtain upper and lower bounds for the thermal-hydraulic figure of merit under evaluation, so to account for the uncertainties in the model. This calculation methodology is known as Best Estimate Plus Uncertainty (BEPU) and the results are given in terms of probabilities and confidence levels.

The thermal-hydraulic analysis can be used in reactor design for different purposes. The present study is divided into three parts:

- A validation of the calculation tool and methodology
- A study of the reactor behaviour in the steady state to determine the set point values of the parameters used to trigger the actuation of the First Shutdown System (FSS)
- The determination of the moment of inertia of the pumps in the PCS based on the reactor behaviour to a Loss of Flow Accident (LOFA)

All the items mentioned above are analysed considering both, the conservative and the BEPU methodologies and results are compared.

### 4. Description of the study

The present section describes the calculation tool, the model and the methodologies used to perform the study.

#### 1.1 Calculation tool

The calculation tool used is the thermal-hydraulic system code RELAP5 v. 3.4, with uncertainty package. RELAP5 solves the mass, energy and momentum equations for the
liquid and the vapour phases and it makes use of empirical correlations to solve the problem. The cooling system is modelled by a series of volumes and junctions connecting them. The heat transfer to and from the cooling fluid is modelled by components known as heat structures. Specific components usually found in plants, such as pumps, are also included. The mass and energy equations are solved in the centre of the volumes while the momentum equation is solved in the junctions.

The uncertainty package is an additional module which has been added to perform uncertainty analysis based on a “base case” (best estimate) in which no uncertainties are considered. This additional module allows the user to specify the uncertainty distribution for a specific parameter, including both, input parameters and parameters calculated by the source code such as heat transfer coefficients or water properties. Details of the methodology used to perform the uncertainty analysis are given in this section.

1.2 Calculation model

The nodalization used to model the reactor and the PCS is schematically illustrated in Fig 1. It includes the components inside the reactor pool: core, chimney and inlet pipes and the main components of the cooling circuit such as pumps, decay tank and heat exchangers. The reactor core is modelled by two pipes which represent the hot (HFA) and an average (AFA) channel. Heat structures have been attached to these two pipes to model the heat transfer from the fuel to coolant. Flap valves have also been included on the inlet pipes, so to model the natural circulation cooling regime. The SCS is considered as a boundary condition.

Fig 1. Nodalization of PCS
1.3 Calculation methodologies

The conservative and the BEPU calculation approaches are used to perform the study and the conclusions will be based on the comparison of the results obtained by both methodologies.

1.1. Conservative calculation approach
A single calculation run is performed and the input parameters, boundary conditions and hypotheses are chosen in such a way that the reactor behaviour, as calculated by the model, is a pessimistic one in terms of a pre-established thermal-hydraulic figure of merit.

1.2. BEPU calculation approach
An uncertainty distribution and the parameters describing such distribution are defined for a group of “relevant” parameters. The calculation consists of:
- A “best estimate” or a “base case” calculation in which uncertainties are not considered in the modelling of the reactor.
- An uncertainty analysis based on the input error propagation technique. In this case, weighting factors are randomly generated for each parameter, considering the uncertainty distribution defined by the user. These weighting factors are applied to the nominal value of the variable, thus modifying the input value. The new values are randomly combined generating a number of inputs or cases to be run.

The result of the BEPU approach is a “best estimate” prediction for a thermal hydraulic relevant figure of merit, with the upper and lower bounds achieved for such figure of merit. Results are given in terms of probabilities and confidence levels, which are a function of the number of calculations performed and which can be estimated by using Wilk’s formula (4, 5). The uncertainty module included in version 3.4 of RELAP 5 makes use of such formula to generate the number of cases to be run according to the desired probability and confidence level specified by the user. It also allows the user to specify the number of runs to be performed regardless of Wilk’s formula.

1.4 Analysis

As mentioned before, the study is divided into different parts. Each of them is described in the present section:

1.4.1 Validation of the model and calculation methodologies
Measurements for a Loss of Normal Power Supply (LNPS) Test at 20 MW were performed during the commissioning of OPAL reactor. The experimental values of core flow are used to validate the calculation tool, model and methodologies. This is done by comparing the measured values with the values of core flow as calculated by RELAP5 during the steady state normal operation and during the LNPS event for the conservative and the BEPU calculation approaches.

The event under analysis consists of a pump stop, actuation of the FSS and interruption of the SCS at t=0, when the loss of power takes place. Flap valves open when the pressure difference between the inlet pipes and the reactor pool falls below a given value, thus changing from the forced to the natural circulation cooling regime. The FSS is modelled by means of a table describing the negative reactivity inserted as a function of time.

1.4.2 Determination of set points
The values of set points for some of the parameters triggering the FSS are determined by modelling the steady state with the conservative and the BEPU approach. The set point values for reactor power, core flow and inlet temperature are evaluated in this study, as the FSS can be triggered either by high reactor power, by low core flow or by high inlet temperature.
For the conservative evaluation, the steady state of the reactor is modelled considering conservative values of input parameters and boundary conditions. A series of calculations are performed varying the value of the set point parameter accordingly (i.e., increasing reactor power, decreasing coolant flow, increasing inlet temperature) until the thermal-hydraulic design criteria fails to be accomplished. The procedure is similar for the BEPU evaluation. The steady state value of the set point parameter in the best estimate case is varied until the upper or lower bound of the uncertainty analysis fails to comply with the design criteria.

1.4.3 Determination of the moment of inertia of the pumps

A LOFA is the event studied to determine the moment of inertia of the pumps in the PCS. The pumps are stopped at t=0, together with the pumps in the SCS. The coolant in the PCS falls according to the coast-down curve of the pump, which depends on its moment of inertia. The FSS is triggered either by a low flow or by a high inlet temperature trip. The forced circulation cooling regime is therefore maintained until the flap valves open, when the pressure difference between the inlet pipe and the reactor pool falls below a given value. The moment of inertia of the pumps is specified in such a way that, at the time at which the flap valves open and the natural circulation cooling regime establishes, the decay power is low enough so as to satisfy the thermal-hydraulic design limits. The LOFA is modelled by using both, the conservative and the BEPU approaches.

1.4.4 Input data

The calculation model at the best estimate steady state is adjusted to a thermal power of 18.8 MW, a core flow of 1900 m$^3$/h, and an inlet temperature of 37 °C. Tab 3 summarizes the input parameters on which an uncertainty distribution has been considered, the type of distribution and the parameters characterizing it.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Uncertainty distribution</th>
<th>Characterization of distribution</th>
</tr>
</thead>
<tbody>
<tr>
<td>Heat transfer coefficient for modes 2 and 3 (*)</td>
<td>Uniform</td>
<td>Min / max value: 0.8 / 1.2</td>
</tr>
<tr>
<td>Reactor power</td>
<td>Normal</td>
<td>Mean: 1; Deviation: 0.025</td>
</tr>
<tr>
<td>Reactivity insertion</td>
<td>Normal</td>
<td>Mean: 1; Deviation: 0.010</td>
</tr>
<tr>
<td>Rated flow of pumps</td>
<td>Normal</td>
<td>Mean: 1; Deviation: 0.025</td>
</tr>
<tr>
<td>Friction coefficients (***)</td>
<td>Uniform</td>
<td>Min / max value: 0.3 / 1.7</td>
</tr>
<tr>
<td>Pump torque and coefficients</td>
<td>Normal</td>
<td>Mean: 1; Deviation: 0.05</td>
</tr>
<tr>
<td>Moment of inertia of pumps</td>
<td>Normal</td>
<td>Mean: 1; Deviation: 0.1</td>
</tr>
<tr>
<td>Inlet temperature to the SCS</td>
<td>Uniform</td>
<td>Min / max value: 0.9968 / 1.003 (***).</td>
</tr>
</tbody>
</table>

(*) According to RELAP5 heat transfer map  
(**) Loss coefficients on the junctions in the core  
(***) Resulting in a +/- 1°C deviation

Tab 3: Parameters with uncertainty distribution and its characterization

For most of the cases, the uncertainty distribution has been determined and characterized based on engineering judgement and experience. For the case of reactivity insertion, the errors committed in neutronic calculations are considered. The uncertainty distribution for the heat transfer coefficient is based on the comparison between published experimental values with the values as calculated by the correlations used in RELAP5.

For the steady state, a total number of 59 runs are performed in the uncertainty analysis. This means that, for a particular thermal-hydraulic figure of merit, there is a 95% probability that its value falls within the calculated upper and lower bounds, and this is guaranteed with a 95% confidence level, according to Wilk’s. For the transients, i.e., the LNPS and the LOFA the number of runs performed is increased to 90, thus increasing the confidence level to
99%. For the conservative evaluation, the maximum and minimum values corresponding to each parameter are combined to obtain the most pessimistic behaviour of the reactor in terms of a particular thermal-hydraulic figure of merit obtained as calculation output.

1.4.5 Evaluation of results

For the validation process, the thermal-hydraulic figure of merit considered is the core flow. For the conservative approach, the input parameters are combined in such a way that the core flow falls below the experimental values, resulting in a reduction of the time taken for the flap valves to open. For the BEPU approach, it is expected that the experimental values fall within the upper and lower bounds. The margins to the limiting thermal-hydraulic design criterion, RDR for the steady-state normal operation and BPR for the transient, are also evaluated for completeness.

For the analysis of set points in the steady-state, the thermal-hydraulic figure of merit considered is the RDR, being RD the limiting phenomenon. In case of the LOFA, the thermal-hydraulic figure of merit considered is the BPR, being the BP the physical phenomenon conditioning the result.

5. Results

1.5 Validation of the model and calculation methodologies

Fig 2 shows the calculated and measured values of the core flow at steady-state normal operation and during the LNPS.

![Figure 2: Calculated and measured core flow during a LNPS](image)

For the steady-state normal operation, the experimental values of core flow falls within the upper and lower bound of the BEPU analysis. For the conservative case, the calculated core flow falls below the experimental value. In reference to the RDR, the minimum value of the lower band in the BEPU analysis is equal to 2.8. This value falls to 2.5 for the conservative case. Consequently, all thermal-hydraulic design criteria are satisfied at steady-state normal operation, regardless the calculation approach.

In reference to the LNPS, flap valves open at 114 seconds according to measurements. For the best estimate prediction, this value is equal to 110 seconds and the measured values of
core flow fall within the uncertainty band, thus validating the calculation model and methodology. In reference to the BPR, the minimum value of the lower bound exceeds the design limit (1.53). The time taken for the flap valves to open according to the conservative prediction is equal to 74 seconds. The core flow as calculated by this approach falls below the experimental values. In reference to the minimum BPR, this is equal to 1.35, meaning that the thermal-hydraulic design limit is not accomplished.

1.6 Determination of set points at steady state

Tab 4 summarizes the results obtained for the set point values as determined by both, the conservative and the BEPU prediction. The design values, which have been established with a different methodology and calculation tool, have been included for completeness.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Set point value</th>
<th>Design value</th>
<th>Conservative</th>
<th>BEPU</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal power</td>
<td>115% of nominal value</td>
<td>120% of nominal value</td>
<td>130% of nominal value</td>
<td></td>
</tr>
<tr>
<td>Core flow</td>
<td>90% of nominal value</td>
<td>80% of nominal value</td>
<td>75% of nominal value</td>
<td></td>
</tr>
<tr>
<td>Inlet temperature (°C)</td>
<td>Nominal value + 9</td>
<td>Nominal value + 16</td>
<td>Nominal value + 18</td>
<td></td>
</tr>
</tbody>
</table>

Tab 4: Set point values as calculated by the different methodologies

1.7 Determination of the moment of inertia of the pumps

Fig 3, Fig 4 and Fig 5 show the evolution in the BPR throughout the transient under analysis for a moment of inertia of pumps equal to 70 kg.m², 60 kg.m² and 50 kg.m² respectively.

![Graph showing BPR evolution](image)

Fig 3: Evolution for the BPR during a LOFA for a moment of inertia of pumps of 70 kg.m²
As observed in results, a single minimum value for the BPR is obtained for the conservative and for the best estimate cases. Based on this observation, an increase in the moment of inertia of pumps results in an increase in the BPR. However, this conclusion changes when the minimum value of the lower bound is considered. As observed, there is a period of time in which the minimum value of BPR remains almost constant and an increase in the moment of inertia of pumps does not necessarily result in an increase in the minimum value of the BPR. Results are summarized in Tab 5.
6. Conclusion

The conservative and BEPU calculation approaches have been used to evaluate the design of OPAL reactor. The evaluation is made in terms of the most limiting thermal-hydraulic design criterion. The model and calculation methodologies were validated by comparing calculated with experimental values. While the BEPU methodology efficiently predicts the behaviour of the reactor at steady state and during the LNPS, the conservative estimation is too pessimistic for the evaluation of the transient. Both methodologies were used to determine the set point values of parameters triggering the FSS. The BEPU approach results in less conservative values, thus reducing the possibility of spurious actuation of the FSS. In reference to the analysis of a LOFA used to determine the moment of inertia of the pumps in the PCS, the conservative calculation approach would result in designs far exceeding the 70 kg.m², making it not viable. The results obtained from the BEPU calculation approach shows that the moment of inertia perhaps could be reduced from 70 kg.m² to 60 kg.m², although more calculations are required. It can also be seen that such moment of inertia may improve reactor performance, as the lower bound in this case exceeds the design limit while the lower bound for the moment of inertia of 70 kg.m² falls below such value, this behaviour could be, probably, the result of uncertainties combination.

7. References