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Measurement and analysis of feedback reactivity in the Monju restart core

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The purpose of this study is to develop a feedback reactivity measurement technique in the Japanese prototype fast breeder reactor Monju and to validate calculation methodology to forecast the nuclear feedback phenomena. A feedback reactivity measurement technique has been developed based on a reactivity model featuring components that depend on the reactivity coefficients, denoted as reactor power ($K_R$) and reactor vessel inlet temperature ($K_{in}$). This technique can precisely measure the two reactivity components simultaneously and was applied to the feedback reactivity experiment conducted in the Monju system start-up test in May 2010. A thorough evaluation considering all possible biases and uncertainties demonstrated that the reactivity coefficients can be evaluated with a measurement uncertainty smaller than 3%. The evaluated reactivity coefficients were simulated considering the temperature distribution in the core. The calculated and measured values of $K_R$ agreed within 1%, and the value of $K_{in}$ was consistent with that reported in a previous isothermal temperature coefficient experiment. The measured and calculated fuel subassembly outlet temperatures also agreed well within 0.2°C, which supports the validity of the temperature calculation.

Keywords: fast breeder reactor; Monju; reactor kinetics; reactor physics; Doppler effect

1. Introduction

The Japanese prototype fast breeder reactor, Monju, resumed the SST in May 2010 after an interruption of 14 years. The major reactor physics parameters, such as criticality [1], control rod worth [2], isothermal temperature coefficient, [3] and feedback reactivity, were measured. This paper describes the evaluation of the feedback reactivity experiment in which the self-stability characteristics of the Monju core were confirmed [4].

The feedback reactivity imposed by a power increase is a key parameter for the safe operation of nuclear reactors. The design prediction accuracy of the parameter can be validated by the total reactivity feedback effect measured during reactor operation. The feedback effect, however, consists of various components, such as the Doppler and core expansion effects; thus, component-wise measurements are required for detailed validation. In conventional measurements, the power coefficient, which is the change in reactivity mainly due to the Doppler effect, is evaluated based on the total feedback effect by subtracting the isothermal reactivity effect using the isothermal temperature coefficient measured at a zero-power condition [5]. In this procedure, the temperature dependence of the isothermal temperature coefficient is neglected or corrected via calculations. Another attempt to measure the component-wise feedback reactivity has been reported, which focused on the three components: reactor power, average coolant temperature, and core support plate temperature [6]. The results are physically reasonable; however, the measurement uncertainty exceeded 10%.

In the present study, a new technique is proposed to measure the component-wise feedback reactivity more efficiently and precisely. This method can evaluate the power and isothermal temperature coefficients simultaneously with reasonably small uncertainty. The corresponding calculation values are also evaluated by simulating the temperature distribution in the core under the experimental conditions of power and R/V inlet temperature, and the accuracy of the calculation is discussed.

2. The feedback reactivity experiment

2.1. Monju core specifications

The basic configuration and major specifications of the Monju core are given in Figure 1 and Table 1,
respectively. Monju is a prototype fast breeder reactor with a reactor power of 280 MWe (714 MWt) cooled by sodium. The Monju core consists of two fuel regions, inner and outer cores, surrounded by the blanket fuel and shielding. The core is fueled by plutonium and uranium mixed oxide, and the blanket is fueled by depleted uranium oxide.

The control rods in the Monju core consist of three types: CCR, FCR, and BCR. The number of control rods installed in the core is 10 for CCRs, 3 for FCRs, and 6 for BCRs. CCRs and FCRs are used for normal operation, and BCRs are used for emergency shutdown. The CCR loaded at the core center (denoted as CCR1 in Figure 1) was used in the present feedback reactivity experiment.

The neutron detectors are located outside the R/V. They are classified into the following three systems. The source range monitor is used at the start-up of the reactor, and the power range monitor is used for monitoring during power operation. The WRM signals, which are monitored every 1 s, are used in the feedback reactivity experiment.

### 2.2. Experiment

The feedback reactivity experiment evaluated in the present study was conducted during the SST in June 2010. The principal purpose is to demonstrate the self-stability feature of the core after a reactivity insertion at a zero-power level. Figure 2 shows the plant conditions during the experiment. Both the PHTS and the secondary system have three loops each. The reactor was operated at around 200°C with a primary coolant flow-rate set at 49% of the rated value. The air coolers in the ACS, which was the main heat sink during the experiment, were activated without active air ventilation. The secondary loop was operated with a coolant flow-rate set at 7% of the rated value. The R/V inlet temperature was measured at the inlet piping, outside the R/V, for each loop.

The experiment was conducted in the following procedure. First, the initial conditions of the WRM count rate (associated with the reactor power) and R/V inlet temperature were confirmed at a critical state with CCR1 withdrawn by about 400 mm and the other control rods fully withdrawn. A positive reactivity was then inserted by withdrawal of CCR1. The WRM count rate and R/V inlet temperature were monitored until the parameters became stable. The procedures were repeated for three cases with different reactivity insertions of $+2 \varepsilon$, $+4 \varepsilon$, and $+6 \varepsilon$. The time histories of the WRM count rate and R/V inlet temperature are shown in Figures 3 and 4, respectively.

After the reactivity insertion, the core temperature increases following the power elevation, which induces a negative reactivity feedback by the Doppler and core expansion effects. Thereafter, the core reactivity reduces
3. Evaluation of feedback reactivity

3.1. Two-component model of feedback reactivity

The reactivity change can be expressed by a combination of two effects: the Doppler and core expansion effects. The core expansion effect in the axial direction appears in all the materials composing the core, whereas that in the radial direction is mainly determined by the core support plate expansion.

These effects were formulated as follows:

$$\rho_{\text{feedback}} = DP + k_D^F \Delta T_F + k_S^E \Delta T_S + k_{Na}^E \Delta T_{Na} + k_C^E \Delta T_{in}.$$  \hspace{1cm} (1)

$$DP = K_F^D \ln \left(1 + \frac{\Delta T_F}{T_0 + 273.15}\right) + K_S^E \ln \left(1 + \frac{\Delta T_S}{T_0 + 273.15}\right) + K_{Na}^E \ln \left(1 + \frac{\Delta T_{Na}}{T_0 + 273.15}\right).$$  \hspace{1cm} (2)

The Doppler coefficient \((\Delta k/k)\degree C\) of each material (fuel, structure, and sodium), expressed by \(k^D = K^D/(T + 273.15)\), is assumed to be constant in the small temperature range of 200–210°C observed in the experiment. Therefore, Equation (2) can be expressed using...
the temperature changes as

\[ DP = k_D^p \Delta T_F + k_D^S \Delta T_S + k_D^{Na} \Delta T_{Na}. \]  

(3)

The thermal conductivity of materials can be also assumed to be constant in this small temperature range. Therefore, the temperature changes, \( \Delta T_F, \Delta T_S, \) and \( \Delta T_{Na} \), are proportional to \( \Delta P_{reactor} \) as

\[ \Delta T_F = \Delta T_{in} + a \Delta P_{reactor}, \]

(4)

\[ \Delta T_S = \Delta T_{in} + b \Delta P_{reactor}, \]

(5)

\[ \Delta T_{Na} = \Delta T_{in} + c \Delta P_{reactor}, \]

(6)

where \( a, b, \) and \( c \) are constants. \( \Delta P_{reactor} \) is converted from \( \Delta n_{WRM} \) using a conversion factor \( H_F \) as

\[ \Delta P_{reactor} = H_F \Delta n_{WRM}. \]  

(7)

The factor \( H_F \) will be evaluated in Section 3.2.3. Finally, the following equation is obtained:

\[ \rho_{feedback} = K_R \Delta P_{reactor} + K_{in} \Delta T_{in}. \]  

(8)

The parameters in Equation (8), \( K_R \) and \( K_{in} \), are the target values in the present study. \( K_R \) is the reactivity coefficient of the reactor power, and \( K_{in} \) is the coefficient of the R/V inlet temperature, which is equivalent to the isothermal temperature coefficient. The variables \( \Delta n_{WRM} \) and \( \Delta T_{in} \) are observables in the experiment.

At the maximum and minimum power points (hereafter referred to as peak power points) shown in Figure 3, the absolute value of the total feedback reactivity \( \rho_{feedback} \) is nearly equivalent to the positive reactivity inserted by the control rod operation. The inserted reactivity is determined by the asymptotic period method based on \( \Delta n_{WRM} \) at around the beginning of the experiment.

There are six sets of \( \Delta P_{reactor} \) and \( \Delta T_{in} \) corresponding to the peak power points in the three experimental cases with different reactivity insertions. As such, the feedback reactivity coefficients \( K_R \) and \( K_{in} \) can be determined by applying multiple regression analysis [7] to Equation (8).

Noted that \( \Delta T_{in} \) is measured at a point outside the R/V inlet, and a certain time delay should be considered for sodium transport from the inlet pipe to the core support plate. This time delay is estimated as 68 s based on the volume of the lower plenum (158 m³) and sodium flow rate (2091 kg/s).

### 3.2. Evaluation of observables

#### 3.2.1. Evaluation of inserted reactivity

The reactivity inserted by the control rod was evaluated using the asymptotic period method. The period was determined through fitting to WRM count rate change from 4000 to 8000 cps, and reactivity was evaluated with kinetic parameters shown in Table 2. The results are summarized in Table 3. Three corrections are considered in the evaluation, each of which is described below in detail. The latter two corrections correspond to slight reactivity changes with respect to time owing to the long duration of the experiment (several hours).

<table>
<thead>
<tr>
<th>Reactivity ((\times 10^{-5} \Delta k/k))</th>
</tr>
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</table>
| \(+2 \xi\)      | \(+4 \xi\)      | \(+6 \xi\)
| Original value | 7.48 | 7.48 | 13.80 | 13.80 | 19.68 | 19.68 |
| Feedback effect | 0.09 | 0.09 | 0.10 | 0.10 | 0.11 | 0.11 |
| Pu-241 decay | −0.02 | −0.06 | −0.01 | −0.05 | −0.01 | −0.05 |
| Rod-position shift | Shaft contraction | 0.06 | 0.15 | 0.08 | 0.27 | 0.07 | 0.27 |
| Structure expansion | −0.07 | 0.02 | −0.14 | 0.02 | −0.20 | 0.00 |
| Corrected value | 7.53 | 7.67 | 13.83 | 14.13 | 19.64 | 20.01 |

*Expansion of R/V wall and control rod shaft considering time delay.*
3.2.1. Feedback effect in the asymptotic period method. The reactivity was evaluated by the asymptotic period method using low count rate data, above which an obvious feedback effect appears, but is still slightly affected by a small feedback effect. This small reactivity feedback was evaluated by the product of \( K_R \), \( H_F \), and WRM count rate change. The value of \( K_R \) was taken from the calculation to be described in Section 4.1 and that of \( H_F \) is evaluated in Section 3.2.3. The WRM count rate change was considered from 3000 to 6000 cps, corresponding to the initial condition and to the middle of the WRM count range in the period method.

3.2.1.2. \(^{241}\)Pu decay. The reactivity change owing to \(^{241}\)Pu decay was calculated as \(-0.62 \times 10^{-3} \ (\Delta k/k)\text{day}^{-1} \) [3], and the elapsed time after the reactivity insertion.

3.2.1.3. Rod position shift. Before confirming the critical state at the beginning of each experimental case, the central control rod was moved from the fully inserted position to the critical position. Accordingly, the upper part of the control rod (drive shaft) was moved from under the sodium region (200°C) to the cover gas region (70°C). Thus, after a control rod withdrawal in the experiment, the length of the drive shaft contracted owing to the temperature change, slightly shifting the control rod in the upper direction. The resulting reactivity change was evaluated following a previously employed method [2]. The same effect on the other control rods was found to be negligible since the control rods were fully withdrawn.

In addition, the reactivity effect of relative position change of the control rod in the fuel region during a temperature change was corrected.

The control rod bottom position shifts to the lower direction by the expansion of the control rod drive shaft, and so does the fuel region bottom by the expansion of the R/V [8,9]. The temperature of the control rod drive shaft and R/V can be described by a first-order delay function as

\[
T_D(t) = T_D(t_{-1}) + \frac{dt}{\tau + dt}(T_B(t) - T_D(t_{-1})).
\] (9)

In order to consider the expansion, the flow condition in R/V was simplified in the temperature calculation. The R/V temperature was separately evaluated in two regions: the middle plenum and the upper part of R/V shown in Figure 5. The R/V inlet temperature was set to the base sodium temperature \( T_B \) for the middle plenum region, whereas the R/V outlet temperature was set for the upper part of R/V and the control rod shaft. The time constants \( \tau \) were calculated to be 47 s, 164 s, and 1084 s for the control rod shaft, the upper part of R/V, and the middle plenum, respectively, based on the heat capacity of the structure material, thermal conductivity, sodium flow rate, etc. (see Appendix).

3.2.2. Evaluation of the WRM count rate

The WRM count rate data used in the experiment was obtained from the main process computer in Monju, where the signal processing mode is automatically switched from the pulse mode to the Campbell mode at around 150,000 cps. The processing system is normally calibrated before the reactor power operation, but the feedback reactivity experiment was conducted before the calibration. Thus, the observed WRM count rate data was not continuous before and after the mode switching and required correction.

In the present study, the WRM count rate data obtained in the Campbell mode was corrected by referring to another source of WRM count rate data processed by a temporarily equipped system. The referred system processes the same WRM detector signal but with a different signal processing device. The system was used only in the pulse mode for a limited time span from reactivity insertion to the maximum power points in the experiment. The dead time correction was applied using a non-paralyzable dead time [10] of 0.4 \( \mu s \), which was estimated based on the width of the logic pulse of the processing device.

The correction was made in two steps. The first correction was made using a factor evaluated from the ratio of the WRM count rates before and after mode switching. Another correction factor, the ratio of the referred data to the signal corrected by the first step, was subsequently introduced as a function of the count rate. Figure 6 shows an example of the corrections in the + 6 \( \varepsilon \) case. The second correction from (b) to (c) in the figure was introduced in addition to the first correction from
Figure 6. WRM count rate correction for the +6 g case (for Ch 2).

(a) to (b) because the signal processing in the Campbell mode would cause a larger error.

3.2.3. Evaluation of the conversion factor from the WRM count rate to reactor power

The conversion factor from the WRM count rate to the reactor power, \( H_P \), was evaluated by analyzing the heat balance in the ACS, which was the dominant heat sink in the experiment. When the reactor power and sodium temperature are stable, the heat balance in the PHTS is expressed by

\[
P_{\text{plant}} + P_{\text{reactor}} = L_{\text{plant}} + L_{\text{ACS}}. \tag{10}
\]

The parameters \( P_{\text{plant}} \) and \( L_{\text{plant}} \) depend on the PHTS and are assumed to be constant during the experiments since the PHTS was steadily operated.

\( L_{\text{ACS}} \) can be evaluated by the enthalpy change \( \Delta H_{\text{ACS}} \) and sodium flow rate \( W_{\text{ACS}} \) of 195 kg/s in the ACS as

\[
L_{\text{ACS}} = \Delta H_{\text{ACS}} W_{\text{ACS}} \times 10^{-3}. \tag{11}
\]

\( \Delta H_{\text{ACS}} \) was determined by the sodium temperatures (°C) at the ACS inlet \( (T_1) \) and outlet \( (T_2) \) [11] as

\[
\Delta H_{\text{ACS}} = 1.437 (T_1 - T_2) - 2.903 \times 10^{-4} (T_1^3 - T_2^3) + 1.543 \times 10^{-7} (T_1^3 - T_2^3). \tag{12}
\]

The reactor power and sodium temperature were stable at the beginning and end of each experimental case. At the end of an experimental case, the parameters were judged stable when the WRM signal changed less than 1% during 1000 s. As such, a total of six sets of \( n_{\text{WRM}} \) \( (P_{\text{reactor}} = H_p n_{\text{WRM}}) \) and \( L_{\text{ACS}} \) values are available to evaluate the conversion factor. Figure 7 plots the relation between \( n_{\text{WRM}} \) and \( L_{\text{ACS}} \), and Table 4 shows related measured parameters in the ACS, where the temperatures are averages of the three loops, and the flow rates are sums of them. The error bar (approximately ±0.25 MW) in the figure reflects the uncertainties in the temperature and flow-rate measurement systems, which have little influence on \( H_P \) because of the systematic nature of the error. A good proportional relation is confirmed, and the conversion factor \( H_P \) was evaluated by fitting as 1.09 × 10^{-5} MW/cps.

Figure 7. Heat removal in the ACS and WRM count rate.

Table 4. Basic conditions of the ACS for evaluation of \( H_P \).

<table>
<thead>
<tr>
<th>Item</th>
<th>Initial state</th>
<th></th>
<th></th>
<th>Final state</th>
</tr>
</thead>
<tbody>
<tr>
<td>WRM count rate ((\times 10^4 \text{cps}))</td>
<td>+2 g</td>
<td>+4 g</td>
<td>+6 g</td>
<td>+2 g</td>
</tr>
<tr>
<td>Inlet temperature (°C)</td>
<td>198.5</td>
<td>198.2</td>
<td>198.5</td>
<td>201.3</td>
</tr>
<tr>
<td>Outlet temperature (°C)</td>
<td>197.5</td>
<td>197.3</td>
<td>197.7</td>
<td>197.7</td>
</tr>
<tr>
<td>Flow rate (kg/s)</td>
<td>195</td>
<td>195</td>
<td>195</td>
<td>195</td>
</tr>
<tr>
<td>Enthalpy change (kJ/kg)</td>
<td>1.30</td>
<td>1.17</td>
<td>1.03</td>
<td>4.86</td>
</tr>
<tr>
<td>Removed heat (MW)</td>
<td>0.254</td>
<td>0.228</td>
<td>0.200</td>
<td>0.948</td>
</tr>
</tbody>
</table>

\*a Before reactor operation.
\*b At the end of experiment.
\*c Average value for three loops.
\*d Total value for three loops.
3.2.4. Criticality judgment in the experiment

Precisely speaking, the reactivity is not zero at the peak power points shown in Figure 3 due to the delayed neutron contribution. The true critical points appear after the peak power points with a certain time delay. The time delay was evaluated as follows:

The reactivity $\rho(t)$ can be calculated by an inverse kinetics equation [12, 13] as

$$\rho(t) = \beta - \frac{\ell}{n_{WRM}(t)} \sum_{i=1}^{6} \lambda_i C_i(t) + \frac{\ell}{n_{WRM}(t)} \frac{dn_{WRM}(t)}{dt}.$$  

(13)

The term ($\ell/n_{WRM}$)($dn_{WRM}/dt$) is negligible compared with the other terms [12], and $C_i(t)$ can be calculated as follows [13]:

$$C_i(t + \Delta t) = \beta_i \frac{n_{WRM}}{\ell} \Delta t + C_i(t) - \lambda_i \dot{C}_i \Delta t.$$  

(14)

where $n_{WRM}$ and $\dot{C}_i$ are the mean values over a time increment $\Delta t(0.1 \text{ s})$. The value of $\dot{C}_i$ is calculated using the values of $C_i(t)$ and $C_i(t + \Delta t)$ in an iterative manner.

The reactivity can be determined by Equations (13) and (14) in accordance with reference [13], and the results were least-square fitted to a quadratic function around the criticality. The criticality point was determined from the fitted curve.

Figure 8 shows the WRM count rate and reactivity calculated by the inverse kinetics method at around the maximum power point in the $+6 \xi$ case. There is a time lag of about 39 s between the critical point and the maximum power point. This time delay was considered in determining $\Delta P_{\text{reactor}}$ and $\Delta T_{in}$.

The final values of the changes in reactor power ($\Delta P_{\text{reactor}}$) and R/V inlet temperature ($\Delta T_{in}$) are summarized in Table 5. The influence of the time delay appears in $\Delta T_{in}$, especially at the maximum power points, due to a relatively large variation in time, which has a significant impact on the evaluation of $K_R$, whereas the influence barely appears in $\Delta P_{\text{reactor}}$.

3.3. Uncertainty evaluation

The uncertainties in the feedback coefficients $K_R$ and $K_{in}$ were evaluated based on individual uncertainties of the temperature and WRM count rate. All
the uncertainties are classified into random and systematic components in the 1σ value (68% confidence level). The term “systematic” denotes here that the uncertainty sources are common to all peak power points. The uncertainty was evaluated on the following sources.

3.3.1. Kinetic parameters used in the period method

The reactivity evaluated by the period method depends on the kinetic parameters: the delayed neutron fraction and the decay constants of the delayed neutron precursors. The resulting uncertainty in the reactivity evaluated previously [2] was applied to the present study.

3.3.2. Fluctuation in the WRM count rate

The statistical fluctuation of the WRM count rate depends on the WRM count rate level, and was thus evaluated at each peak power point. The standard error $SE$ is expressed as

$$SE = \frac{SD}{\sqrt{n}}. \quad (15)$$

The uncertainty at the initial condition was evaluated using the WRM count rate data for 600 s ($n = 600$). At the peak power points, the uncertainty was evaluated using the data for 20 s ($n = 20$). The value SD was obtained in the least square fitting to a quadratic function for a gradual change of the WRM count rate. The total uncertainty of $n_{WRM}$ was obtained as the RSS value.

The fluctuation of the WRM count rate also affects the initially inserted reactivity evaluated by the period method. The resulting uncertainty was evaluated based on the fitting error.

3.3.3. Criticality judgment

The criticality point was judged by the output of the inverse kinetics method, as described in Section 3.2.4. The zero-reactivity point was determined from the fitted function at around the criticality point, which accompanies a certain range of reactivity fluctuation. The uncertainty of reactivity was evaluated based on the fitting error.

3.3.4. Dead-time correction

The WRM count rate was corrected using the temporary data acquisition system, as described in Section 3.2.2. Thus, the result accompanies systematic uncertainty associated with the dead-time correction. It was estimated as ±0.03 μs based on an observation of the logic pulse.

3.3.5. Fluctuation of core temperature

The standard error of the R/V inlet temperature was evaluated at the initial condition by using the stable data for 600 s ($n = 600$). The resulting uncertainty is around ±0.005°C.

3.3.6. Temperature measurement system

The R/V inlet coolant temperature was measured by a resistance thermometer and signal converters. The uncertainty in the thermometer is given in the device specifications as a function of temperature: ±0.5% × T + 0.3°C. The systematic nature of the uncertainty was assumed as previously noted [3].

3.3.7. Uncertainty of conversion factor $H_P$

The conversion factor $H_P$ was evaluated based on the heat removed by the ACS and WRM count rate, as described in Section 3.2.3.

The uncertainty in $H_P$ due to the dead time in $\Delta n_{WRM}$ was evaluated following the above evaluation given in (d) by which the uncertainties at the end of the experiment for the +2 ε, +4 ε, and +6 ε cases are ±135, ±337, and ±625 cps, respectively. These correspond to an uncertainty of 0.5% in $H_P$.

Details concerning the uncertainties in the heat removed by the ACS are listed in Table 6, in which the total value corresponds to the error bars given in Figure 7. The uncertainties in the temperatures at the inlet and outlet of the ACS are mainly determined by those of the signal transfer devices, which are ±1.2°C, yielding an uncertainty of about 180 kW in the heat removed by the ACS. The uncertainty in the sodium flow rate is determined mainly by that of the flow meter, which is ±1.9 kg/s, yielding an uncertainty of 1.0% under a constant flow-rate condition of 195 kg/s. The enthalpy correlation includes an uncertainty of 0.7% [11].

In addition, a fitting error of 1.3% was considered in the determination of $H_P$ to consider the possibility that the heat balance given in Equation (10) was not satisfied exactly at a measurement point. The total uncertainty of $H_P$ was evaluated as 1.9%.

The results of the uncertainty evaluation are summarized in Table 7. The component-wise uncertainties in $K_R$ and $K_{in}$ due to each uncertainty source were first evaluated. The total uncertainties in $K_R$ and $K_{in}$ were obtained by summing up the results for the peak power points while considering the nature of the uncertainty: simple addition for systematic uncertainties and RSS for random uncertainties.

The final measurement results of $K_R$ and $K_{in}$ were obtained as $-2.859 \pm 0.083 \times 10^{-5} (\Delta k/k)/MW$ and $-2.858 \pm 0.051 \times 10^{-5} (\Delta k/k)/°C$, respectively. The measurement uncertainties are smaller than 3%. The value of $K_{in}$ in this study is consistent with the result of...
Table 6. Uncertainties in the heat removed by the ACS.

<table>
<thead>
<tr>
<th>Uncertainty source</th>
<th>Initial state</th>
<th>Final state</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>+ 2 ȼ</td>
<td>+ 4 ȼ</td>
</tr>
<tr>
<td>Inlet temperature</td>
<td>±179.2</td>
<td>±179.1</td>
</tr>
<tr>
<td>Outlet temperature</td>
<td>±179.2</td>
<td>±179.1</td>
</tr>
<tr>
<td>Flow rate</td>
<td>±3.6</td>
<td>±3.3</td>
</tr>
<tr>
<td>Enthalpy correlation</td>
<td>±2.2</td>
<td>±1.9</td>
</tr>
<tr>
<td>Total</td>
<td>±253.4</td>
<td>±253.3</td>
</tr>
</tbody>
</table>

4. Calculation of feedback reactivity

4.1. Calculation of $K_R$ and $K_{in}$

The total feedback reactivity $\rho_{feedback}$ is calculated by the sum of the feedback reactivity for region $j$ as

$$\rho_{feedback} = \sum_{i,j} k'_{ij} \Delta T_j.$$  \hspace{1cm} (16)

The values of $k'_{ij}$ are assumed to be constant over the small temperature range observed in the experiment.

The feedback coefficients were evaluated according to the following three steps: (a) temperature coefficient calculation, (b) temperature calculation, and (c) evaluation of the feedback coefficients $K_R$ and $K_{in}$.

4.1.1. Temperature coefficient calculation

The temperature coefficients corresponding to the Doppler and core expansion effects were calculated for the fuel pellet, structure (stainless steel), and sodium. The calculation method was based on a deterministic calculation scheme depicted in Figure 9, which was basically the same as that employed in the reference [3]. The effective cross section was calculated from a 70-energy-group constant set based on JENDL-4.0 [14] with SLAROM-UF [15, 16]. The core calculation was performed using the three-dimensional Tri-Z model with DIF3D code [17]. Corrections were applied to the calculation results to minimize errors associated with the diffusion approximation, finite spatial mesh size, and finite energy groups. The geometrical model shown in Figure 1 was applied considering the experimental temperature condition of 200°C. The core fuel region in the calculation model was divided into 48 regions, the adequacy of which is discussed in Section 4.2.

Table 7. Uncertainties in measurements.

<table>
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<td></td>
<td></td>
<td>+ 2 ȼ</td>
<td>+ 4 ȼ</td>
<td>+ 6 ȼ</td>
<td>+ 2 ȼ</td>
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<td>+ 6 ȼ</td>
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<td>+ 4 ȼ</td>
<td>+ 6 ȼ</td>
<td>+ 2 ȼ</td>
</tr>
<tr>
<td>Reactivity</td>
<td>Kinetic parameters</td>
<td>S</td>
<td>±1.6</td>
<td>±1.6</td>
<td>±1.6</td>
<td>±1.6</td>
<td>±1.6</td>
<td>±1.6</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>WRM fluctuation</td>
<td>R</td>
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<td>±0.2</td>
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</table>

The core expansion coefficients were calculated from the density coefficients and geometrical expansion coefficients. The Doppler and density coefficients were calculated by the first-order perturbation theory with forward and adjoint flux at the initial critical condition. The thermal expansion property was based on the Monju design calculation [18, 19].

4.1.2. Temperature calculation

The temperature calculation was performed for the same regions defined in the temperature coefficient calculation. The temperature transient around the measurement points is so sluggish that a steady state can be assumed in the temperature calculation. The temperature distribution in each subassembly was calculated by the subchannel code COBRA4-MJ [20,21]. The temperature of sodium was evaluated by the subchannel calculation. The temperature inside the fuel pellet was calculated by a one-dimensional radial heat conduction model using appropriate thermal property data. The gap conductance was evaluated to be 0.176 W/(cm²·C) based on the Ross and Stoute model [22, 23], considering the gas thermal conductivity and gap distance between fuel and cladding. The fuel thermal conductivity was calculated based on the correlation model for mixed oxide fuel proposed by Morimoto [24], reflecting the actual ²⁴¹Am content as shown in Table 1. The flow-rate condition in the Monju core, in which the flow is divided into 11 orificing zones, was considered in the calculation.

The temperature was calculated starting from the bottom of the fuel (or the R/V inlet temperature). The calculated temperature distribution was averaged by volume to obtain the temperature for each material in the 48 regions. Figure 10 shows the temperature distributions at around the core center in the axial direction for the peak power points in the +6 ε insertion case. The temperature change from the initial value of about 200°C is mainly caused by fuel heating in the maximum power point case, whereas it is mainly caused by the R/V inlet temperature change in the minimum power point case. Figure 11 shows the temperature distribution in the radial direction at the core mid-plane for the +6 ε maximum power point. The temperature variation in each region is considerably smaller than that of the axial direction.

4.1.3. Evaluation of feedback coefficients

In the third step, the coefficients $K_R$ and $K_n$ were calculated based on the derived temperature coefficients and temperature variation. The effective temperature due to the crystalline binding effect [25] was additionally considered for the Doppler effect in the fuel
Expansion Fuel

−10

feedback reactivity model given in Equation (8), which

supports the validity of the simple representation of the

imental case were found to be nearly equivalent. This

activity component by the corresponding power change

−

Doppler Fuel


Table 8. Reactivity coefficients at 200°C condition.

Table 8. Reactivity coefficients at 200°C condition.

Table 9. Component-wise feedback reactivity by calculation.

Table 9. Component-wise feedback reactivity by calculation.

4.2. Influence of region division in the
temperature calculation

It is important to divide the temperature calculation region sufficiently finely to consider the actual temperature distribution in the core [26]. Table 9 shows the breakdown of the feedback reactivity for the reactivity components and materials. The Doppler and core expansion effects in the fuel occupy over 80% contribution to the total reactivity effect at the maximum power point, which implies the importance of the precise treatment of the temperature distribution in the fuel. In particular, the axial distribution for the maximum power point is the most important because a strong position dependence is observed in the temperature (Figure 10) and in the reactivity coefficient (Figure 12).

The dependence of the parameter $K_R$ on the division patterns was surveyed, as summarized in Table 10. The core fuel region was divided into as many as 12 regions for the axial direction and up to 2 regions for the radial direction. The parameter $K_n$ was determined by the R/V inlet temperature and therefore not influenced by the division patterns.
A larger effect was observed in the axial division, and the calculation result of $K_R$ almost saturates at 12 divisions. In contrast, the effect of the radial division is small, and the condition of two divisions appears to be sufficient. The difference in the effect can be explained by the temperature distributions shown in Figures 10 and 11.

4.3. Calculation uncertainty

The calculation uncertainties of $K_R$ and $K_{in}$ were evaluated for major sources of uncertainty. The parameter $K_{in}$ does not depend on the temperature calculation; thus, the uncertainty of the isothermal coefficient evaluated in a previous study [3] was applied, which was given to be 3.3% ($1\sigma$).

The parameter $K_R$ depends on the temperature calculation; thus, the uncertainty in the temperature calculation was additionally considered.

The uncertainty in the temperature coefficient was evaluated in the same manner as that of $K_{in}$ using the JENDL-4.0 nuclear data covariance [14], which was evaluated to be 3.8%. The uncertainty in the temperature calculation mainly derives from uncertainties in thermal property data for the fuel. For the maximum power point of the $+6\epsilon$ insertion case, temperature increases in the fuel and gap contribute to the total maximum fuel temperature rise from the R/V inlet by 21% and 63%, respectively. The thermal conductivity has an uncertainty of 3.0% based on the measurement values under 1000 K [27], resulting in an uncertainty of 0.6% for the fuel temperature change based on the contribution. The gap conductance has a larger uncertainty of 15.9% obtained from the measurement data [23], resulting in an uncertainty of 10.0% for the fuel temperature change based on the contribution. The contributions are nearly equivalent among the peak power points, and the above uncertainties are applied to all the cases.

A summary of the uncertainty evaluation is given in Table 11. The total uncertainty of $K_R$ was evaluated to be 10.7% ($1\sigma$).

The uncertainty in the total feedback reactivity is also shown in the table, which was evaluated considering the respective contributions from the $K_R$ and $K_{in}$ terms. The uncertainties of the measured parameters $\Delta T_{reactor}$ and $\Delta T_{in}$ are included in the uncertainty since the parameters are used only in the calculation.

5. Comparison of measurement and calculation results

5.1. Feedback reactivity coefficients

The C/E values of the feedback reactivity coefficients are 0.990 for $K_R$ and 1.075 for $K_{in}$ based on the results shown at the end of Sections 3 and 4.1. The overestimation in the calculated $K_{in}$ is consistent with that reported in the isothermal temperature coefficient evaluation [3], where the presence of a bias was suspected in the measured value. The underestimation in the calculated $K_R$ is comparable with the measurement uncertainty of 2.9% and considerably smaller than the calculation uncertainty of 10.7%. This implies that the calculation accuracy of $K_R$ is reasonable, and the calculation uncertainty can be smaller.
Table 11. Uncertainties in the feedback reactivity calculation.

<table>
<thead>
<tr>
<th>Item</th>
<th>Relative uncertainty ((%); 1(\sigma))</th>
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<tr>
<td></td>
<td>+ 2 (\epsilon)</td>
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<tr>
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<tr>
<td>(\rho_{feedback})</td>
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<tr>
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<tr>
<td>(\Delta T_{in})</td>
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<td>Gap conductance</td>
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<td>SUM</td>
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</table>

\(^aK_f:\) Thermal conductivity of mixed oxide fuel.
\(^bT/C:\) Temperature coefficient.

It should be noted that the measured values of \(K_R\) and \(K_{in}\) are independent of each other because the temperature change owing to a reactor power increase is clearly separated from the R/V inlet temperature change, as defined in Equations (4)–(6). In contrast, the calculated values of \(K_R\) and \(K_{in}\) are correlated through common components such as the Doppler reactivity. In this aspect, the overestimation in \(K_{in}\) would be attributed to the unidentified bias in the measured value, as discussed in reference [3], or to the calculation of the core support plate expansion coefficient, which is unique to \(K_{in}\) and has the largest contribution as shown in Table 9.

5.2. Fuel subassembly outlet temperature

The S/A outlet temperature was also measured in the experiment. The data are not essential to the feedback reactivity evaluation but can be used to validate the temperature calculation.

The S/A outlet temperature data at the representative eight positions, shown in Figure 13, were used in the present study. The fuel S/A outlet temperature was measured at 45 cm above the top of the S/A.

The S/A outlet temperature has a certain time delay relative to the reactor power because of the response of the thermocouple and sodium mixing. Figure 14 shows the time history of the S/A outlet temperature and the WRM signal at the end of the experiment for the + 6 \(\epsilon\) case. The S/A outlet temperature begins decreasing 13 s after the WRM count rate decreases owing to control rod insertion. This time delay was considered in the R/V inlet temperature setting of the temperature calculation.

Figure 15 compares the measured and calculated results for the S/A outlet temperature at the maximum power point in the + 6 \(\epsilon\) case. The uncertainty is evaluated as \(\pm 0.3\,^\circ\text{C} (1\sigma)\) according to the specifications of the signal converters (\(\pm 0.2\,^\circ\text{C}\)) and thermocouples (\(\pm 0.2\,^\circ\text{C}\)). Both results are in good agreement within 0.2\(^\circ\text{C}\) at all positions, which validates the coolant temperature calculation model employed in the present study. The slight overestimation can be attributed to the

![Figure 13. S/A outlet temperature measurement points.](image)

![Figure 14. Time delay of the S/A outlet temperature at the end of the experiment (+ 6 \(\epsilon\)).](image)
5.3. Total Feedback Reactivity

Figure 16 compares the measured and calculated values of the total feedback reactivity and shows good agreement. When closely examined, the values for the minimum power points exhibit a larger discrepancy of up to 7.5% relative to that of 3.1% for the maximum power points. The errors in $K_R$ and $K_{in}$ are propagated to the total feedback reactivity by their contributions shown in Figure 17. The larger errors in the total feedback reactivity for the minimum power points can be attributed to the larger contribution of $K_{in}$.

6. Conclusion

The two inherent safety parameters, reactor power coefficient $K_R$ and reactor vessel inlet temperature coefficient $K_{in}$, were evaluated in the SST using the Monju restart core with 1.5% Am bearing fuels. A new feedback reactivity measurement technique featuring the coefficients $K_R$ and $K_{in}$ has been developed to evaluate the coefficients simultaneously with reasonably small uncertainty. The evaluation yields the coefficients as $K_R = -2.859 \pm 0.083 \times 10^{-5}$ $(\Delta k/k)/$MW and $K_{in} = -2.858 \pm 0.051 \times 10^{-5}$ $(\Delta k/k)/^\circ C$.

The result of the present analyses unveiled the adequacy of the calculation methodology for the two feedback parameters. The C/E values for the two parameters $K_R$ and $K_{in}$ were determined to be 0.990 and 1.075, respectively. The parameter $K_R$ agreed with the calculated value within the range of uncertainty, which suggests that the temperature and temperature coefficient in the core fuel are treated correctly in the calculation. The larger discrepancy in $K_{in}$ is consistent with that reported in a previous evaluation of the isothermal temperature coefficient. A comparison was also made for the fuel subassembly outlet temperature, and good agreement was confirmed, which supports the validity of the coolant temperature calculation. Therefore, it seems natural to conclude that the calculation methodology of $K_R$ and $K_{in}$ was validated by the measurement of the prototype reactor.

A further study of the measurements and analyses regarding feedback reactivity of the Monju core should be conducted in order to examine the safety characteristics of sodium-cooled fast reactors. The present evaluation was conducted at a zero-power level with a sodium temperature of 200 $^\circ$C, although the developed technique would be applicable to a slightly higher power and sodium temperature. Such a measurement is planned for the coming SST, which would enhance the reliability of the method, and consequently reduce the calculation uncertainties, especially in the temperature calculation.
### Nomenclature

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<thead>
<tr>
<th>Symbol</th>
<th>Description</th>
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<tr>
<td>$C_i$</td>
<td>Concentration of the $i$th delayed neutron group precursor</td>
</tr>
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<td>$D_P$</td>
<td>Doppler reactivity ($\Delta k/k$)</td>
</tr>
<tr>
<td>$H_{cp}$</td>
<td>Conversion factor (MW/cps)</td>
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<tr>
<td>$\Delta H_{TCS}$</td>
<td>Enthalpy change in ACS (kJ/kg)</td>
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<td>$k_f^{}$</td>
<td>Expansion coefficient ($\Delta k/k/\degree C$)</td>
</tr>
<tr>
<td>$K_D$</td>
<td>Doppler constant ($\Delta k/k$)</td>
</tr>
<tr>
<td>$k_f^{}$</td>
<td>Doppler coefficient (($\Delta k/k)/\degree C$)</td>
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<tr>
<td>$k_f^j$</td>
<td>Temperature coefficient of type $i$ and region $j$ (($\Delta k/k)/\degree C$)</td>
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<tr>
<td>$K_{in}$</td>
<td>Reactivity coefficient of R/V inlet temperature ($\Delta k/k$/\degree C)</td>
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<td>$\ell$</td>
<td>Prompt neutron lifetime (s)</td>
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<tr>
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<td>Heat removed by the ACS (MW)</td>
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<td>Base sodium temperature ($\degree C$)</td>
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<tr>
<td>$T_D$</td>
<td>R/V or control rod shaft temperature ($\degree C$)</td>
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<tr>
<td>$\Delta T$</td>
<td>Temperature change ($\degree C$)</td>
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<tr>
<td>$\Delta T_{in}$</td>
<td>R/V inlet temperature change ($\degree C$)</td>
</tr>
<tr>
<td>$t_i$</td>
<td>Time at $i$th data point (s)</td>
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<td>$dt$</td>
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<td>$W_{ACS}$</td>
<td>Sodium flow rate in ACS (kg/s)</td>
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### Greek letters

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<td>$\beta$</td>
<td>Effective delayed neutron fraction</td>
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<tr>
<td>$\beta_i^{}$</td>
<td>Delayed neutron fraction of the $i$th group</td>
</tr>
<tr>
<td>$\lambda_i^{}$</td>
<td>Decay constant of the $i$th delayed neutron group precursor (1/s)</td>
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<tr>
<td>$\rho_{feedback}$</td>
<td>Total feedback reactivity ($\Delta k/k$)</td>
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<td>$\tau$</td>
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### Subscripts

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<tr>
<td>$F$</td>
<td>Fuel</td>
</tr>
<tr>
<td>$Na$</td>
<td>Sodium</td>
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<tr>
<td>$S$</td>
<td>Structure</td>
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### Acronyms

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<th>Description</th>
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<tr>
<td>ACS</td>
<td>Auxiliary cooling system</td>
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<tr>
<td>BCR</td>
<td>Backup control rod</td>
</tr>
<tr>
<td>CCR</td>
<td>Coarse control rod</td>
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<tr>
<td>FCR</td>
<td>Fine control rod</td>
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<tr>
<td>PHTS</td>
<td>Primary heat transport system</td>
</tr>
<tr>
<td>RSS</td>
<td>Root-sum-square</td>
</tr>
<tr>
<td>R/V</td>
<td>Reactor vessel</td>
</tr>
<tr>
<td>S/A</td>
<td>Fuel subassembly</td>
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<td>System start-up test</td>
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<td>WRM</td>
<td>Wide range monitor</td>
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### Acknowledgements

We would like to thank the entire staff in the Monju fuel fabrication sections. We wish to thank Mr. H. Nishi for fruitful discussions concerning the evaluations. We are also indebted to Mr. T. Miyagawa, Mr. T. Suzuki, and Mr. Y. Okawachi for advice regarding the problems encountered during the study and experiment.

### References

The time constant $\tau$ describing the first-order delay response of the R/V temperature can be expressed by

$$\tau = \frac{C_p \cdot \rho \cdot V}{S \cdot h},$$  \hspace{1cm} (A1)

where $C_p$ is the specific heat at constant pressure of structure, $\rho$ is the density of structure, $V$ is the volume of structure, $S$ is the surface area, $h$ is the heat transfer coefficient of surface, $kNa$ is the thermal conductivity of sodium, $D$ is equivalent hydraulic diameter, and $Nu$ is the Nusselt number.

The time constant of the control rod drive shaft can be evaluated in the same manner, replacing the heat transfer coefficient by

$$h = \frac{1}{h_{CRS}} + \left(\frac{\phi_1}{k_{str}}\right) \ln \left(\frac{\phi_1}{\phi_e}\right),$$  \hspace{1cm} (A3)

where $h_{CRS}$ is the heat transfer coefficient of the control rod drive shaft, $\phi_1$ is the radius of the control rod drive shaft, $k_{str}$ is the thermal conductivity of the structure, and $\phi_e$ is the effective radius of the control rod drive shaft corresponding to the average temperature.

The time constant $\tau$ and parameters used in the evaluation are shown in Table A1. The temperature was calculated with these time constants. Figure A1 shows the time history of temperature change for each position. Response in the temperature change in R/V upper part indicates large delay because the flow into the middle plenum was lower than 1% of total sodium flow in R/V. The temperature change for each case is shown in Table A2.
<table>
<thead>
<tr>
<th>Item</th>
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<th>Control rod drive shaft</th>
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<th>R/V wall (middle plenum)</th>
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<td>$D$</td>
<td>m</td>
<td>$9.95 \times 10^{-2}$</td>
<td>$4.60 \times 10^{-1}$</td>
<td>3.06</td>
</tr>
<tr>
<td>$h$</td>
<td>W/m$^2$C</td>
<td>$1.04 \times 10^3$</td>
<td>$1.28 \times 10^3$</td>
<td>$1.94 \times 10^2$</td>
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<tr>
<td>$C_p$</td>
<td>J/kg°C</td>
<td></td>
<td>$5.27 \times 10^2$</td>
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<tr>
<td>$\rho$</td>
<td>kg/m$^3$</td>
<td></td>
<td>$7.91 \times 10^3$</td>
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<tr>
<td>$k_{Na}$</td>
<td>W/m°C</td>
<td></td>
<td>81.8$^a$</td>
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</tr>
<tr>
<td>$\tau$</td>
<td>Sec</td>
<td>47</td>
<td>164</td>
<td>1084</td>
</tr>
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*Common values for all positions.*

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<tbody>
<tr>
<td>Control rod drive shaft</td>
<td>+1.4</td>
<td>+2.5</td>
<td>+2.2</td>
<td>+4.5</td>
<td>+2.6</td>
<td>+6.2</td>
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<tr>
<td>R/V wall in upper part</td>
<td>+1.3</td>
<td>+2.5</td>
<td>+1.9</td>
<td>+4.6</td>
<td>+2.1</td>
<td>+6.2</td>
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<tr>
<td>R/V wall in middle plenum</td>
<td>+0.1</td>
<td>+1.7</td>
<td>+0.1</td>
<td>+2.9</td>
<td>0.0</td>
<td>+3.7</td>
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</tbody>
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