Uncertainty Evaluation of Anticipated Transient without Scram Plant Response in the Monju Reactor Considering Reactivity Coefficients within the Design Range

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Abstract: This paper describes the methods and results of an uncertainty evaluation of a significant plant response analysis of reactor trip failure events, specifically anticipated transients without scram in the Japanese prototype fast breeder reactor Monju. Unprotected loss of heat sink (ULOHS) has a relatively large contribution to the core damage frequency due to reactor trip failure. The uncertainty in the allowable time to core damage in this event has so far been estimated by considering the range of reactivity coefficients. There are some cases where it is considered that core damage will be avoided. Specifically, if the primary heat transport system (PHTS) pump inlet sodium temperature stays below 650 °C for 1 h, the avoidance of core damage due to a ULOHS event is assumed. This is the temperature at which the probability of cavitation in the static pressure bearing begins to increase. In this study, a success scenario was investigated in two aspects: identification of influential input parameters and estimation of the probability of success. In the parameter identification, input parameters that satisfy the pump inlet temperature being below 650 °C are clarified by treating the reactivity coefficients and reactor kinetics parameters as variables that can be taken to be within the design range. In the probability estimation, the results are fitted to a lognormal distribution function, from which the output variable was found to fall between 640 and 679 °C with a probability of 90%, the probability of the temperature being 650 °C or lower was 0.23, and the average and mode value was 659 °C.

Key words: FBR, Monju, ATWS uncertainty.

1. Introduction

Monju is a sodium-cooled, loop-type prototype fast breeder reactor that uses mixed oxide (MOX) fuel with an electrical power of 280 MW, designed and built by the Japan Atomic Energy Agency (JAEA). Monju achieved first criticality in 1994 but has so far only reached 40% of its rated operational power. In 2018, an application to change the license from power generation to decommissioning was accepted by the nuclear regulation authority. Currently, the spent fuel is being withdrawn from the reactor core. Although operation of Monju has been abandoned, the knowledge and data obtained during the construction and operation should be retained for use in the future design of fast reactors. They can be used to enhance safety analysis methods for fast reactors.

This paper describes the methods and results of an uncertainty evaluation of a significant plant response analysis of a reactor trip failure event, i.e., an anticipated transient without scram (ATWS) event in Monju. Among ATWS events, an unprotected loss of heat sink (ULOHS) will have a relatively large contribution to the core damage frequency. This kind of event has significant allowable time until core damage is caused because the core coolant flow can be maintained [1]. Fig. 1 shows a diagram of the cooling system, and Fig. 2 shows a diagram of the...
reactor vessel. In-vessel retention of the molten corium after core damage may not always be possible because of the high temperature of the coolant that results from the loss of the heat sink. Therefore, the uncertainty in the allowable time to core damage in this event has so far been estimated by considering the range of reactivity coefficients [1].

Possibility of the core damage in the ULOHS event is investigated in other fast reactors as follows.

Experimental Breeder Reactor II (EBR-II), which uses metal fuel, underwent a ULOHS event as a test to confirm the ability to avoid core damage. According to the results, which were reviewed by Sofu, T. [2], core and system damage were avoided because of the negative reactivity that was induced by the core support grid being spread radially with increasing core inlet temperature.

Phénix, which used MOX fuel, underwent a natural convection test with drying out of the steam generator before a reactor scram as an end-of-life test [3]. The core inlet temperature increased by about 40 °C after the loss of heat sink and before the scram, and the thermal power simultaneously decreased by about 60%. This significant decrease in thermal power was explained by a reactivity feedback effect resulting from the thermal expansion of the diagrid that bears the core assemblies.

Sofu, T., et al. [4] analyzed a transient in the ASTRID reactor, which was being developed as a sodium-cooled, MOX-fueled fast reactor. This transient was a ULOHS sequence with convergence of the inlet and outlet temperature of the core. According to this analysis, the fission power decreased as a result of negative radial core expansion reactivity feedback.
Yamano, H., et al. [5] developed a phenomenological event tree for ULOHS events in the Japan Sodium-cooled Fast Reactor, which was designed by JAEA. According to their study, ULOHS events can result in a wide variety of plant responses. Reactivity feedback behavior resulting from thermal expansion of the core support structure is also a necessary factor for explaining the phenomena that occur in this type of event. After a parametric analysis study, the authors summarized that the large uncertainty in the plant response phase could be clarified by examining the dynamic plant response data from existing fast reactor plants such as Jōyō and Monju.

All of the above studies show a preferable feature of the feedback behavior that can mitigate the progression of an event.

However, once the core damage occurs, in-vessel retention of the molten corium after core damage is presumed to be difficult to achieve because of the high temperature of the coolant due to loss of the heat sink. Core damage occurs as a result of the loss of flow in the primary heat transport system (PHTS) resulting from PHTS pump failure after a rise in the sodium temperature during a ULOHS event. Structural integrity analysis of the PHTS pump demonstrated that the pump will fail if the sodium temperature reaches 750 °C at the inlet to the pump.

Before reaching 750 °C, the probability of cavitation in the static pressure bearing begins to increase at 650 °C [1]. In other words, it is considered that core damage due to the ULOHS event would be avoided if the PHTS pump inlet sodium temperature stays below 650 °C for a sufficiently long time in terms of reactor operation, 1 h for example.

In this paper, conditions that satisfy the pump inlet temperature being below 650 °C are examined for the case of a ULOHS event lasting for 1 h. The reactivity coefficients and reactor kinetics parameters are treated as input variables that can be taken as being within the design range. Moreover, the distribution of the analysis result and the confidence interval are estimated.

2. Extraction of Important Factors by a Phenomena Identification and Ranking Table

First, the process of extracting input variables in the analysis of ULOHS events is described. As a result of previous research on uncertainty analysis, Kikuchi, N. and Mochizuki, H. [6] used sensitivity analysis results to narrow down input variables. From this, 43 input variables were extracted, and these were subjected to uncertainty propagation analysis. Currently, as a “best estimate plus uncertainty” method, a phenomena identification and ranking table (PIRT) method [7] used in the code scaling, applicability, and uncertainty method [8], is extensively used.

As an example of application to fast reactor design, in the analysis evaluations by Kang, S., et al. [9] and Kang, S. J., et al. [10], input variables were extracted by narrowing down physical phenomena using a PIRT. Regarding ULOHS events, fuel heat transfer, radial core expansion, fuel and cladding tube axial expansion, control rod drive shaft expansion, and Doppler reactivity are rated as highly important. In this study, we used a PIRT to evaluate important physical phenomena and establish their rankings in the events to be evaluated. Table 1 shows a PIRT listing of important physical phenomena in several events. Super-COPD, a plant dynamics analysis code that was developed for obtaining best estimates of the plant response of the Monju reactor, was used for this analysis.

In the ULOHS event, the reactivity feedback behavior and the heat transfer behavior of from the fuel pins to the coolant have been evaluated as important phenomena in the PIRT. However, the sensitivity of the heat transfer at a low flow rate has been evaluated to be insignificant [1], and thus the variability is considered only for the reactivity feedback behavior in this paper.
**Table 1  Phenomena Identification Ranking Table for each event.**

| Items                                      | UTOP | ULOF | ULOHS | LOHRS |
|--------------------------------------------|------|------|-------|-------|
| Decay heat behavior                        | L    | L    | L     | H     |
| Feedback reactivity behavior               | H    | H    | H     | L     |
| Heat transfer behavior of fuel assembly    | H    | H    | H     | L     |
| Natural convection flow behavior of RV     | L    | L    | L     | H     |
| Heat transfer behavior of IHX               | L    | L    | L     | H     |
| RV upper plenum mixing / thermal stratification behavior | L    | L    | L     | H     |
| Heat transport behavior of PHTS            | L    | L    | L     | H     |
| Flow coast down behavior of PHTS           | L    | H    | L     | H     |
| Heat transfer behavior of SG               | L    | L    | L     | L     |
| Heat transport behavior of SHTS            | L    | L    | L     | L     |
| Heat transport behavior of WS              | L    | L    | L     | L     |
| Heat transport behavior of ACS             | L    | L    | L     | H     |
| Plant control system behavior              | L    | L    | L     | L     |
| Reactor safety protection system behavior  | L    | L    | L     | H     |

SHTS: secondary heat transport system; SG: steam generator (SH and EV); WS: water and steam system; UTOP: unprotected transient overpower; ULOF: unprotected loss of flow; ULOHS: loss of heat removal system.

### 3. Range of Uncertain Variables

Next, it was assumed that two factors, the reactivity coefficients and the reactor kinetics parameters, which govern the reactivity feedback behavior, are input variables that can take the ranges shown in Appendix 8 of the installation permission application. Table 2 shows the variables and their ranges. The variables depend on fuel burnup and core configurations assumed in the Monju design.

Six reactivity coefficients were applied: Doppler coefficient, fuel temperature coefficient, cladding tube temperature coefficient, wrapper tube temperature coefficient, coolant temperature coefficient, and core support plate temperature coefficient. Two reactor kinetics parameters were also applied: prompt neutron lifetime and effective delayed neutron fraction.

The 22 cases created in this way were analyzed using the plant transient response analysis code, and the relationship between the PHTS pump inlet sodium temperature as an output variable and the reactivity coefficients and the reactor kinetics parameters as the input variables was examined. The analysis and examination were executed by linking the general-purpose multi-objective optimization design support program mode FRONTIER with the Super-COPD code that was developed for use in best estimates of the plant dynamics of the Monju reactor.

### 4. Results and Discussion

Fig. 3 shows the PHTS pump inlet sodium temperature behavior in all 22 cases. In the ULOHS event, the temperature increases with time and the time to the maximum temperature in all cases were 3,600 s. The maximum temperature reached within the analyzed time is discussed as the output variable. As shown in Figs. 4 and 5, the reactor power decreases and the ULOHS event is dominated by negative reactivity due to the expansion of the core support plate from the start of the event.

Table 3 and Fig. 6 show the relationship between each input variable and the output variable. In Fig. 6, the range of reactivity coefficients and the reactor kinetics parameters yielding low output variables is shown by a group of blue lines. Values are more negative for the fuel temperature coefficient and core support plate temperature coefficient, which increases the negative reactivity. On the other hand, values are
more positive (or less negative) for the Doppler coefficient, cladding tube temperature coefficient, wrapper tube temperature coefficient, and coolant temperature coefficient, which decrease the negative reactivity. As evaluated in the previous studies, the contribution of the core support plate temperature coefficient is dominant. Among the 22 cases, there are many cases where the output variable is below 660 °C, and for values above 660 °C, the above-mentioned trend in the characteristics of the input variables seems to be reversed. Only the cases where the output variable is kept below 650 °C are shown in Fig. 7. From this result, it is possible to specify the reactivity coefficients and the reactor kinetics parameter ranges in which the PHTS pump inlet sodium temperature is below 650 °C during a ULOHS event.

Next, the distribution of output variables will be described. Recently, a statistical estimation of uncertainty in analysis results computed considering a range of parameters was conducted. Horie, H., et al. [11] evaluated the confidence level of the analysis result by varying eight input parameters using a statistical approach. They estimated the coolant temperature in ATWS events in the 4S reactor, which is being designed as a small sodium-cooled fast reactor for decentralized power suppl. Vierow, K., et al. [12] estimated the uncertainty by plotting the cumulative distribution function based on analysis.

### Table 2 Input parameters.

| Reactivity Coefficients | Unit | Low burnup | High burnup | Minimum | Maximum |
|-------------------------|------|------------|-------------|---------|---------|
| Doppler                 | ×10⁻⁷/Tds/dT | BOIC: -5.50E+00 | -3.54E+00 | - 3.50E+00 | - 3.90E+00 | BOIC: -6.00E+00 | - 6.00E+00 | - 6.00E+00 | - 6.00E+00 | - 6.00E+00 |
|                         |      | EOIC: 4.00E+00 | 3.40E+00 | 4.00E+00 | 4.00E+00 | BOEC: -5.00E+00 | - 5.00E+00 | - 5.00E+00 | - 5.00E+00 | - 5.00E+00 |
| Fuel temp.              | ×10⁻⁷/kk'/℃ | BOIC: 7.00E+00 | 7.00E+00 | 3.00E+00 | 3.00E+00 | BOIC: 8.00E+00 | 8.00E+00 | 3.00E+00 | 3.00E+00 | 3.00E+00 |
| Cladding tube temp.     | ×10⁻⁷/kk'/℃ | BOIC: 3.00E+00 | 3.00E+00 | 2.30E+00 | 2.30E+00 | BOIC: 8.00E+00 | 8.00E+00 | 3.00E+00 | 3.00E+00 | 3.00E+00 |
| Wrapper tube temp.      | ×10⁻⁷/kk'/℃ | BOIC: 1.70E+01 | 1.70E+01 | 1.70E+01 | 1.70E+01 | BOIC: 2.20E+01 | 2.20E+01 | 1.70E+01 | 1.70E+01 | 1.70E+01 |
| Sodium temp.            | ×10⁻⁷/kk'/℃ | BOIC: 2.30E+01 | 2.30E+01 | 2.30E+01 | 2.30E+01 | BOIC: 8.20E+00 | 8.20E+00 | 3.00E+00 | 3.00E+00 | 3.00E+00 |
| Core support plate temp. | ×10⁻⁷/kk'/℃ | BOIC: -1.21E+01 | -1.21E+01 | -1.21E+01 | -1.21E+01 | BOIC: -1.12E+01 | -1.10E+01 | -1.10E+01 | -1.11E+01 | -1.09E+01 |
| Prompt neutron life time | μs   | BOIC: 3.80E-03 | 3.80E-03 | 4.17E-03 | 4.17E-03 | BOIC: 4.10E-03 | 4.10E-03 | 4.10E-03 | 4.10E-03 | 4.10E-03 |
| Effective delayed neutron fraction | | BOIC: 3.30E-03 | 3.30E-03 | 3.30E-03 | 3.30E-03 | BOIC: 3.30E-03 | 3.30E-03 | 3.30E-03 | 3.30E-03 | 3.30E-03 |
Fig. 3  PHTS pump inlet sodium temperature behavior.

Fig. 4  Reactor power behavior.
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Fig. 5 Reactivity effects of input variables.

Table 3 Input/output parameter (analysis result).

| Case | Doppler | Cladding tube temp. | Fuel temp. | Wrapper tube temp. | Sodium temp. | Core support plate temp. | Prompt neutron life time(s) | Effective delayed neutron fraction | Max PHTS inlet temp. | Max temperature achieved time(s) |
|------|---------|---------------------|------------|-------------------|--------------|-------------------------|-----------------------------|-------------------------------|---------------------|---------------------------------|
| 1    | -7.36E-03 | 3.90E-07           | -2.96E-06  | 2.70E-07          | 1.20E-07     | -1.08E-05               | 4.48E-07                   | 3.99E-03                    | 6.89E+02   | 3.60E+03                        |
| 2    | -6.50E-03 | 5.50E-07           | -2.96E-06  | 3.20E-07          | 8.80E-07     | -1.08E-05               | 4.35E-07                   | 3.86E-03                    | 6.76E+02   | 3.60E+03                        |
| 3    | -6.81E-03 | 3.00E-07           | -3.37E-06  | 2.30E-07          | -1.70E-07    | -1.19E-05               | 4.17E-07                   | 4.04E-03                    | 6.75E+02   | 3.60E+03                        |
| 4    | -6.07E-03 | 6.50E-07           | -3.01E-06  | 3.50E-07          | 1.34E-06     | -1.10E-05               | 4.32E-07                   | 3.79E-03                    | 6.70E+02   | 3.60E+03                        |
| 5    | -6.92E-03 | 7.00E-07           | -3.17E-06  | 3.90E-07          | 1.88E-06     | -1.12E-05               | 4.11E-07                   | 3.41E-03                    | 6.69E+02   | 3.60E+03                        |
| 6    | -6.06E-03 | 7.60E-07           | -3.24E-06  | 3.90E-07          | 1.87E-06     | -1.14E-05               | 4.13E-07                   | 3.41E-03                    | 6.68E+02   | 3.60E+03                        |
| 7    | -6.01E-03 | 4.70E-07           | -3.35E-06  | 2.90E-07          | 6.00E-07     | -1.19E-05               | 4.07E-07                   | 3.90E-03                    | 6.65E+02   | 3.60E+03                        |
| 8    | -5.62E-03 | 5.50E-07           | -3.37E-06  | 3.10E-07          | 9.80E-07     | -1.19E-05               | 4.01E-07                   | 3.83E-03                    | 6.60E+02   | 3.60E+03                        |
| 9    | -5.43E-03 | 8.80E-07           | -3.22E-06  | 4.20E-07          | 2.38E-06     | -1.14E-05               | 4.05E-07                   | 3.36E-03                    | 6.60E+02   | 3.60E+03                        |
| 10   | -5.41E-03 | 8.60E-07           | -3.08E-06  | 4.20E-07          | 2.29E-06     | -1.12E-05               | 4.11E-07                   | 3.45E-03                    | 6.60E+02   | 3.60E+03                        |
| 11   | -5.33E-03 | 7.90E-07           | -3.54E-06  | 3.60E-07          | 1.79E-06     | -1.22E-05               | 3.83E-07                   | 3.42E-03                    | 6.58E+02   | 3.60E+03                        |
| 12   | -5.50E-03 | 7.10E-07           | -3.60E-06  | 3.60E-07          | 1.73E-06     | -1.23E-05               | 3.63E-07                   | 3.41E-03                    | 6.57E+02   | 3.60E+03                        |
| 13   | -5.13E-03 | 9.30E-07           | -3.23E-06  | 4.40E-07          | 2.62E-06     | -1.14E-05               | 3.97E-07                   | 3.30E-03                    | 6.56E+02   | 3.60E+03                        |
| 14   | -5.12E-03 | 9.60E-07           | -3.25E-06  | 4.50E-07          | 2.74E-06     | -1.14E-05               | 4.48E-07                   | 3.36E-03                    | 6.55E+02   | 3.60E+03                        |
| 15   | -5.04E-03 | 9.40E-07           | -3.23E-06  | 4.40E-07          | 2.69E-06     | -1.14E-05               | 3.93E-07                   | 3.29E-03                    | 6.54E+02   | 3.60E+03                        |
| 16   | -4.92E-03 | 9.60E-07           | -3.24E-06  | 4.50E-07          | 2.82E-06     | -1.14E-05               | 3.90E-07                   | 3.25E-03                    | 6.52E+02   | 3.60E+03                        |
| 17   | -4.97E-03 | 7.70E-07           | -3.44E-06  | 3.80E-07          | 2.01E-06     | -1.21E-05               | 3.82E-07                   | 3.46E-03                    | 6.49E+02   | 3.60E+03                        |
| 18   | -4.96E-03 | 8.20E-07           | -3.60E-06  | 3.90E-07          | 2.23E-06     | -1.24E-05               | 3.78E-07                   | 3.36E-03                    | 6.49E+02   | 3.60E+03                        |
| 19   | -4.66E-03 | 9.00E-07           | -3.62E-06  | 4.20E-07          | 2.59E-06     | -1.24E-05               | 3.87E-07                   | 3.36E-03                    | 6.45E+02   | 3.60E+03                        |
| 20   | -4.68E-03 | 8.70E-07           | -3.59E-06  | 4.10E-07          | 2.49E-06     | -1.24E-05               | 3.69E-07                   | 3.30E-03                    | 6.45E+02   | 3.60E+03                        |
| 21   | -4.58E-03 | 8.90E-07           | -3.59E-06  | 4.20E-07          | 2.56E-06     | -1.24E-05               | 3.67E-07                   | 3.29E-03                    | 6.44E+02   | 3.60E+03                        |
| 22   | -4.46E-03 | 9.20E-07           | -3.61E-06  | 4.30E-07          | 2.70E-06     | -1.24E-05               | 3.63E-07                   | 3.24E-03                    | 6.42E+02   | 3.60E+03                        |
Fig. 6  Relationship between input variables and output variable.

Fig. 7  Input variables that satisfy output variable being below 650 °C.
results of the stored energy of the core material in a loss of forced circulation event in a pebble bed modular reactor. Watanabe, O., et al. [13] performed a statistical safety evaluation on the maximum fuel cladding temperature during a loss-of-off-site-power event in a sodium-cooled fast reactor design. Kang, D. G. [14] analyzed large break loss of coolant accidents in a typical three-loop nuclear power plant to evaluate the uncertainty in the peak of cladding temperature. The probability density and cumulative probability were calculated from the analysis of 124 cases and the margins to the acceptance criteria were evaluated.

In the present study, a statistical approach was similarly introduced to estimate the probability that the core damage would occur in the ULOHS event with input parameters that vary in the Monju design range.

First, the distribution function of the output variable was determined so that the Kolmogorov-Smirnov (KS) value becomes the highest. The KS value was calculated with a function installed in mode FRONTIER. Table 4 summarizes fitting results to various distribution functions.

Fig. 8 compares the empirical distribution function of the output variable and the log-normal distribution. The formula for the empirical distribution function of the output variable is as:

\[ F_n(x) = \frac{1}{n} \sum_{i=1}^{n} 1_{x_i < x} \]  

where \( F_n(x) \) is the empirical distribution function, \( x \) is the output variable, and \( 1_A \) is the indicator function, such that \( 1_A = \begin{cases} 1 & (A \text{ is true}) \\ 0 & (A \text{ is false}) \end{cases} \).

The formula for the cumulative distribution function of the log-normal distribution is as:

\[ F(x; \mu, \sigma) = \frac{1}{2} \text{erfc} \left( -\frac{\ln x - \mu}{\sigma \sqrt{2}} \right) \]  

where \( F(x; \mu, \sigma) \) is the cumulative distribution function, \( x \) is a continuous random variable, \( \mu \) is the mean of the variable’s natural logarithm, \( \sigma \) is the standard deviation of the variable’s natural logarithm, and \( \text{erfc} \) is a complementary error function, such that \( \text{erfc} = \frac{2}{\sqrt{\pi}} \int_{x}^{\infty} e^{-t^2} dt \).

From the results shown in Fig. 8, it can be seen that the distribution of the output variable and the log-normal distribution are in good agreement. Fig. 9 shows the cumulative distribution function and probability density function of the log-normal distribution. The formula for the probability density function of the log-normal distribution is as:

\[ f(x) = \frac{1}{\sqrt{2\pi\sigma x}} \exp \left( -\frac{(\ln x - \mu)^2}{2\sigma^2} \right) \]  

where \( f(x) \) is the probability density function, \( x \) is a continuous random variable, \( \mu \) is the mean of the variable’s natural logarithm, and \( \sigma \) is the standard deviation of the variable’s natural logarithm.

Table 5 shows the mean value of the output variable, the 5th and 95th percentiles, and the probability of the inlet temperature being 650 °C or lower. From these results, under the conditions of reactivity coefficients...
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Fig. 8 Comparison of log-normal and empirical distributions of PHTS pump inlet sodium temperature.

Fig. 9 Cumulative distribution function and probability density function of PHTS pump inlet sodium temperature.

Table 5 Statistical parameters estimated with lognormal distribution.

| PHTS Pump Inlet Sodium Temp. (°C) | Avarage | Mode | 5%ile | 95%ile | Probability of 650°C or less |
|-----------------------------------|--------|------|-------|--------|-----------------------------|
| Average                           | 659    | 659  | 640   | 679    | 0.23                        |

and reactor kinetics parameters within the design range, the output variable falls between 640 and 679 °C with a probability of 90%. The probability of the temperature being 650 °C or lower is 0.23, and the average and mode value is 659 °C.

The results indicate that the temperature being below 650 °C for 1 h cannot be satisfied with the input variables used in the Monju design. However, the variation of input variables employed in this paper is mainly due to fuel compositions. Therefore, by examining the influence of fuel compositions, the average value and the mode value may be able to satisfy 650 °C or lower.

Regarding evaluation methods, uncertainty in the input variables can be considered in a more sophisticated manner using the Monte Carlo method with an input variable table created using random numbers. In addition, the uncertainty in the reactivity coefficients has been scrutinized by reflecting measurement results [15]. Future studies will take this information into account, and the range of the input variables will be reevaluated.

5. Conclusion

The input conditions that satisfy the condition of the pump inlet temperature being 650 °C or lower have been clarified by treating the reactivity coefficients and reactor kinetics parameters as variables that can be taken to be within the design range. From these results, the output variable was found to fall between 640 and 679 °C with a probability of 90%, the probability of the temperature being 650 °C or lower was 0.23, and the average and mode value was 659 °C.

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