Evaluation of MONJU Core Damage Risk due to Control Rod Function Failure*

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Abstract

MONJU is a sodium-cooled, loop-type prototype fast breeder reactor with three primary cooling loops that can supply 280 MW of electricity. The limiting conditions of operation defined in the safety regulations for MONJU given the allowed outage time were evaluated by a probabilistic safety assessment technique in our previous study. If a function failure is found in a control rod, certain measures are required by the safety regulations. In this case, if it is confirmed within 24 h that no other control rods are stuck, reactor operation is allowed to continue. To assess the validity of the 24 h allowable time in view of core damage risk, it is necessary to analyze the conditions to be changed when a stuck rod is discovered. Furthermore, to develop a method for evaluating the probability of a control rod insertion failure, it is necessary to re-estimate the frequency of core damage under control rod insertion failure conditions. This paper describes a method for this re-estimation. The probability of an insertion failure of one control rod has been calculated by applying Jeffrey's noninformative prior distribution by considering insertion times based on the results of a mock-up test. The necessary rod insertion numbers for the main reactor shutdown and backup shutdown systems were considered. The results showed that a completion time of 24 h gives a safety margin comparable to that of the Incremental Conditioned Core Damage Probability, that is, an acceptable risk threshold represented by the U.S. Nuclear Regulatory Commission RG 1.177. Thus, the timeframe defined in the present safety regulations was concluded to be appropriate.

Key words: FBR, MONJU, PSA, AOT, LCO, Common Cause Failure, Control Rod Stuck

1. Introduction

If a piece of important equipment fails, we need to know whether the countermeasures taken after the discovery of failure and the time necessary to complete them are appropriate or not to control damage risk. In general, in the safety guidelines based on technical specifications, equipment related to safety functions (i.e., reactor shutdown, cooling, and containment) is defined as part of the safety system along with the countermeasures to mitigate function failure [1]. The countermeasures and the time to complete them are also established in the safety guidelines on the basis of an engineering approach.

So far, according to several methodological guidelines [2-5], the allowed outage time (AOT) for these safety systems, which contribute to the prevention or mitigation of core

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damage frequency (CDF), can be evaluated through the results of a level 1 probabilistic safety assessment (PSA). The limiting conditions of operation (LCO) in the safety guidelines specific to each nuclear power station define these countermeasures and AOTs. The influence of the failures of these systems is estimated by calculating Δ CDF in which the failure probability value for an intended system is 1.

Sotsu and Kurisaka estimated the contribution of AOT to the CDF for the Japanese prototype fast breeder reactor MONJU [6] [7]. MONJU is a sodium-cooled, loop-type prototype fast breeder reactor with three primary cooling loops that can supply 280 MW of electricity. The conclusion of their study was that the present AOTs established in the MONJU safety guidelines provide a sufficient safety margin. This assessment assumed that the target events had already occurred. The evaluation method assigned event probabilities of 1 to any subject included in the fault tree. The failure of a control rod at the insertion event was estimated with generic data acquired from control rod operating experience [8]. In this instance, the failure probability, which considers a common cause failure, should be analyzed further by changing the evaluation conditions.

The MONJU safety guidelines define the measures required at the time of discovery of one stuck control rod, which is a safety function failure. The guidelines allow the reactor to continue operation if surveillance testing within 24 h confirms the integrity of the other control rods. If the surveillance testing finds that another control rod is stuck, the guidelines demand manual reactor trip operation.

In the present study, CDF was applied as an index to evaluate the applicability of the 24 h completion time for surveillance testing. First, in the process to estimate the probability of a control rod insertion failure, the analysis conditions were changed by the fact that one stuck rod is discovered. It is necessary to calculate the conditioned core damage frequency for the control rod insertion failure probability that was recalculated appropriately for the modified plant status. We reviewed recent research for a reference pertaining to assumptions in the calculation process for control rod insertion failure probability and for re-evaluation under such a condition. There are some studies about control rod insertion failure probability that considered its multiplicity.

According to the International Atomic Energy Agency (IAEA) database for research reactors [9], the failure rate for a control rod was estimated as $3 \times 10^{-6}/\text{d}$ from one mechanical failure experienced in 15,940 demands from several research reactors. On the basis of the IAEA database for research reactors mentioned above, for the Greek research reactor Demokritos, Aneziris et al. estimated the failure rate of one rod as $2.28 \times 10^{-5}/\text{h}$ and that of all five rods as $2.28 \times 10^{-6}/\text{h}$ [10]. Bassi et al. [11] reported level 1 PSA for a gas-cooled fast reactor. In the study, the failure rate of a control rod and electromagnet was estimated as $1.27 \times 10^{-3}/\text{d}$ on the basis of three common databases: EIREDA, a compilation of feedback from French pressurized water reactors; T-BOOK 6, related to Scandinavian light water reactors, and the EG&G-gas, related to gas-cooled reactors. In addition, common cause failure parameters were also estimated from operational feedback from French and German light water reactors. On the basis of the IAEA database for research reactors, Faghihi et al. [12] evaluated the frequency of a stuck rod as $6.44 \times 10^{-5}/\text{h}$ to quantify the initiating event frequency [9]. Eide and Calley [13] evaluated the probability of failure upon insertion or detachment of the control and safety rods/diverse safety rods and drive assembly as $3 \times 10^{-5}/\text{d}$ on the basis of the Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR). Mladin analyzed the control rod failure rate by comparing the operational experience of the Romanian TRIGA steady state 14 MW reactor [14] with the IAEA reliability database for research reactors. From the results of Mladin's study, the control rod failure rate of the Romanian TRIGA reactor was estimated as about $1 \times 10^{-4}/\text{h}$. In particular, the control rod failure probability was estimated from actual failure experiences and it ranged from $1 \times 10^{-6}/\text{d}$ to $1 \times 10^{-3}/\text{d}$. However, no studies were found in

which the analysis conditions had to be changed and the result of re-evaluation involved control rod function failure experiences. On the other hand, there are methods to evaluate whether the completion time is appropriate. Incremental conditioned core damage probability (ICCDP) can be applied on the basis of conditional core damage frequency by considering the control rod insertion failure probability and a completion time of 24 h. The obtained value of ICCDP can be compared with the risk threshold in NRC RG 1.177 [3].

This paper describes a method for estimating conditional core damage probability after the observation of a stuck control rod. Moreover, the evaluation results of applying this method to MONJU are presented.

Abbreviations

AOT	: Allowed Outage Time
ATWS	: Anticipated Transient Without Scram
BSS	: Backup Shutdown System
CCRs	: Coarse Control Rods
CDF	: Core Damage Frequency
FCRs	: Fine Control Rods
ICCDP	: Incremental Conditioned Core Damage Probability
LCO	: Limiting Conditions of Operation
MSS	: Main reactor Shutdown System
PSA	: Probabilistic Safety Assessment

2. Evaluation Method

According to the PSA analytical models of MONJU, the control rod insertion failure probability value has been calculated as $1 \times 10^{-8}/d$ as described in section 2.1. Figure 1 illustrates the anticipated transient without scram (ATWS) event tree and the branch of control rod insertion. This failure probability value, which means the probability of failing to insert the necessary number of control rods, has been applied to the heading after reactor trip breaker opening or successful accident management for reactor trip.

Fig. 1 ATWS event tree for automatic reactor trip

This event sequence includes control rod insertion failure reaching core damage directly; that is, the heading of control rod insertion applies only to the ATWS event trees. In this study, variations of core damage risk are evaluated from the result of the CDF recalculation by changing this value of control rod insertion heading. The fundamental knowledge necessary to calculate this failure probability is described below.

2.1 Calculation of Control Rod Insertion Failure Probability

In the level 1 PSA of MONJU, the failure probability per insertion of one control rod is calculated on the basis of the insertion performance number, which is evaluated as follows.

As shown in Table 1, the insertion performance number includes the number of successful insertion results attained during the mock-up test of the control rod drive system. Based on the number of successful insertions, the average failure probability of insertion was calculated by applying Bayesian estimation along with Jeffrey's noninformative prior distribution.

This formula calculates the average value described below.

$$u = \frac{x + 0.5}{n + 1} \quad \dots \dots \dots (1)$$

where u is the average probability of one rod insertion failure, x is the number of failures, and n is the number of trials.

Table 1 Insertion failure probability of one rod

Data Source	Number of Failures:x	Number of Trials:n	Upper Limit	Average	Lower Limit	Variance
mock-up test Phase 1	0	2900	6.6E-04	1.7E-04	6.8E-07	5.9E-08
mock-up test Phase 2	0	6130	3.1E-04	8.2E-05	3.2E-07	1.3E-08
mock-up test for Joyo	0	1542	1.2E-03	3.2E-04	1.3E-06	2.1E-07
Total	0	10572	1.8E-04	4.7E-05	1.9E-07	4.5E-09

The necessary number of control rod insertions required to shutdown the reactor has been evaluated as eight of ten rods in the main reactor shutdown system (MSS) and as three of six rods in the backup shutdown system (BSS). These premises are determined from the result of evaluating the reactivity worth of various control rod insertion patterns targeted in the equilibrium core.

The probability of insertion failure considering a common cause failure was evaluated by the multiple Greek letter (MGL) method [15]. The simultaneous insertion failure rate of any number of control rods was obtained by Eqs. (2) and (3). MGL parameters have been determined from the α factor in NUREG/CR-5500 Vol. 2 [8], which uses the operating experience of Westinghouse-designed plants as a data source.

Operation testing of the control rods was assumed to not shift the timing of the test; that is, nonstaggered testing conditions were applied. The α factor was converted to the MGL parameter by using Eqs. (4) and (5).

The calculated MGL parameters are shown in Table 2. The header row of Table 2 indicates the total number of components and the first column represents the Greek letter corresponding to the number of components that fail simultaneously.

$$Q_k = \frac{1}{C_{k-1}^{m-1}} \times \left(\prod_{i=1}^k \rho_i \right) \times (1 - \rho_{k+1}) \times Q_t \quad \dots \dots \dots (2)$$

$$\rho_1 = 1, \rho_2 = \beta, \rho_3 = \gamma, \rho_4 = \delta, \dots, \rho_{m+1} = 0 \quad \dots \dots \dots (3)$$

$$\frac{k}{\alpha_t} \times \alpha_k = \left(\prod_{i=1}^k \rho_i \right) \times (1 - \rho_{k+1}) \quad \dots \dots \dots (4)$$

$$\rho_1 = 1, \quad \rho_2 = \beta = 1 - \frac{1}{\alpha_t} \times \alpha_1 = \frac{\sum_{i=2}^m i \times \alpha_i}{\sum_{i=1}^m i \times \alpha_i},$$

$$\dots, \quad \rho_k = \frac{\sum_{i=k}^m i \times \alpha_i}{\sum_{i=k-1}^m i \times \alpha_i} \quad \dots \dots \dots (5)$$

where

- Q_t = total component failure probability,
 Q_k = simultaneous component failure probability of k out of m system,
 α = α factor,
 ρ = MGL parameter.

Table 2 Converted MGL parameters

	2	3	4	5	6	7	8	9	10
$1-\beta$	8.93E-01	8.25E-01	7.21E-01	6.51E-01	6.11E-01	5.85E-01	5.67E-01	5.52E-01	5.40E-01
β	1.07E-01	1.75E-01	2.79E-01	3.49E-01	3.89E-01	4.15E-01	4.33E-01	4.48E-01	4.60E-01
γ		2.12E-01	3.54E-01	5.76E-01	6.98E-01	7.73E-01	8.24E-01	8.52E-01	8.72E-01
δ			2.75E-01	3.36E-01	5.52E-01	6.71E-01	7.45E-01	8.10E-01	8.52E-01
ε				3.11E-01	3.34E-01	5.57E-01	6.64E-01	7.29E-01	7.87E-01
μ					3.38E-01	3.19E-01	5.50E-01	6.43E-01	7.11E-01

It is also necessary to consider the correlation of uncertainty with each data point. Therefore, according to the following equation, the method of moments was used for the uncertainty propagation calculation by assuming a lognormal distribution.

The average lognormal distribution is

$$E(X) = e^{\left(\mu + \frac{\sigma^2}{2}\right)} \quad \dots \dots \dots (6)$$

where

$$\mu = \ln(X_{50}) \quad \dots \dots \dots (7)$$

$$\sigma = \frac{\ln \sqrt{X_{95}/X_{05}}}{1.645} \quad \dots \dots \dots (8)$$

- X_{50} = 50% probability value,
 X_{05} = 5% probability value,
 X_{95} = 95% probability value.

In this study, the lognormal distribution of 5, 50, and 95% values, obtained as the product of two data points, is the average value calculated after propagating uncertainty in relation to the above formula. These 5, 50, and 95% values are determined on the basis of an error factor, which is estimated by an engineering approach. The relationships of these values are described below.

$$X_{50} = \exp(\mu) = \sqrt{X_{05} \times X_{95}} \quad \dots \dots \dots (9)$$

$$EF = \sqrt{X_{95}/X_{05}} = \exp(1.645\sigma) \quad \dots \dots \dots (10)$$

From the above calculation results, the average value of 1×10^{-8} (/d) has been applied as the probability of one rod insertion failure.

$$F = \sum_{k=1}^m {}_{m-1}C_{k-1} \times Q_k (12)$$

where

P = event probability under the heading of control rod insertion,

F = fraction of common cause failure,

$P_M =$ insertion failure probability of one rod in the MSS,

$F_M =$ fraction of common cause failure in the MSS,

$P_B =$ insertion failure probability of one rod in the BSS,

$F_B =$ fraction of common cause failure in the BSS.

The correlation of uncertainty was considered with F_M and F_B when one stuck rod was discovered. Thus, the calculated failure probabilities of control rod insertion are shown in Table 4.

Table 4 Event probability under the heading of control rod insertion

LCO and Required Measure	Insertion failure probability (/d)
Ref.Case	1.0E-08
1 rod stuck discovered in MSS	5.0E-05
Reconfirmation of no other rods stuck	1.1E-07
2 rods stuck discovered in MSS	4.4E-04

According to NRC RG 1.177 [3], incremental conditional core damage probability (ICCDP) is defined as in Eqs. (13) and (14). Completion time can be considered as AOT in Eq. (13) because the core damage risk remains higher during this time. In addition, the risk threshold was determined as 5×10^{-7} ; therefore, the completion time equivalent to the threshold can be calculated.

$$\triangle CDF = R(p_i = 1) - R_0 \quad (13)$$

where $R(p_i = 1)$ is the conditional CDF and R_θ is the baseline CDF. Then,

$$ICCDP = \Delta CDF \times AOT / 8760 \quad \dots \dots \dots (14)$$

3. Analysis Result and Discussion

The result of the calculated CDF fraction $R(p_i = 1)/R_0$ is shown in Table 5. CDF varied depending on the conditions stated in the safety guidelines. The CDF when one rod was stuck increased by about 290 times. The completion time equivalent of less than 5×10^{-7} of ICCDP, which is the risk acceptance criteria indicated by NRC RG 1.177 [3], was estimated within 76 h by using the above CDF. If the measures stated in the current safety guidelines are not completed within 24 h, then the reactor must be shutdown, ordinarily within 6 h. The duration of operation required in this case is 30 h, which is sufficiently short compared to the 76 h. Therefore, the core damage risk to this completion time was evaluated as being below the acceptance criteria.

Table 5 CDF fraction

LCO and Required Measure	CDF fraction	Completion time to ICCDP<5E-7(h)
Ref.Case	1.0E+00	-
1 rod stuck discovered in MSS	2.9E+02	7.6E+01
Reconfirmation of no other rods stuck	1.3E+00	6.3E+04
2 rods stuck discovered in MSS	3.0E+03	7.2E+00

As shown in Figure 2, the solid line shows the variation in the case that no other stuck control rod was confirmed in the MSS within 24 h after discovering one stuck rod and reactor operation was continued.

Fig. 2 CDF fraction on LCO

As shown in Table 5, CDF increased by about 1.3 times when it was confirmed that no other control rods were stuck. In this case, the completion time equivalent of less than 5×10^{-7} of ICCDP was estimated as more than 60,000 h. Therefore, measures for reactor operation continuance were reasonable in terms of the risk acceptance criteria.

Moreover, when two stuck rods were discovered, the safety guidelines provide the measures to open the reactor trip breaker as soon as possible. The CDF in this case is to be evaluated in the event tree shown in Figure 3. The conditioned core damage probability per reactor trip breaker opening is equal to the control rod insertion failure probability, which has a value of about 4.4×10^{-4} from Table 4. This is because CDF was increased owing to the open state of the reactor trip breaker, which increases the common cause failure probability of control rod insertion. Hence, the ordinary plant shutdown procedures, which are the measures in this case, can be safer than a manual reactor trip. The conditional core damage probability in the case of an ordinary plant shutdown can be evaluated by the event tree shown in Figure 4. The accident sequence is a loss of decay heat removal function after reactor shutdown in that all heat cooling systems are available. This conditional core damage probability for each shutdown was estimated as approximately 0.01 times of normal operation.

Fig. 3 ATWS event tree for reactor trip due to other causes

Figure 2 shows the variation in the conditional core damage probability based on the above discussion. The dotted line indicates the discovery of two stuck rods followed by ordinary plant shutdown. In this case, the CDF due to reactor trip is about 1000 times larger than the ICCDP when ordinary plant shutdown is performed within 6 h. Therefore, the core damage risk can be decreased markedly by changing the measures from manual trip to ordinary plant shutdown operations.

Fig. 4 Decay heat removal event tree

Next, operational continuity of the reactor is considered. MONJU contains two kinds of control rods: automatic fine control rods (FCRs) and manual coarse control rods (CCRs). Therefore, it is necessary to consider both types of rods. FCRs will be withdrawn to compensate for reactor power due to burnup as needed. On the other hand, CCRs will be withdrawn by an amount equivalent to power compensation by FCRs week by week. Furthermore, control rod deviation will be limited to 20 mm or less by the safety interlock circuit. Thereafter, power compensation will be restricted within a week at least when an FCR becomes stuck. On the other hand, it will be difficult to maintain the rated power after two weeks in the case where a CCR is stuck.

Thus, the decision as to whether reactor operation must be continued should be made by considering the prospects of repairing a stuck rod within the allowed operation time.

4. Conclusion

The plant status variations that affect the LCO for control rod function failure by sticking were analyzed, and the core damage frequency was re-evaluated on the basis of the variations. The evaluation result obtained by using ICCDP indicates the adequacy of the required measure and its completion time. In addition, the influence of the current measures

on the conditional core damage probability was assessed, and some prospects were clarified for further consideration.

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