Evaluation of MONJU Core Damage Risk with Change of AOT Using Probabilistic Method*

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Abstract

MONJU is a sodium-cooled, loop-type prototype fast breeder reactor with three primary cooling loops which can supply 280 MW of electricity. Limiting conditions of operation (LCO) defined in the safety regulations in MONJU given the allowed outage time (AOT) are evaluated using a PSA technique. The result indicates the possibility of limit extension and some prospects that we should examine.

Key words: FBR, MONJU, PSA, AOT, LCO

1. Introduction

The LCO are defined in the safety regulations, and mandatory measures and completion times there for when these LCO are not satisfied are determined by previously set LWR guidelines⁽¹⁾. The technical basis for the completion times is provided, upon which engineering judgment was applied considering the operation management and maintenance policy of MONJU.

There are two operating statuses of the system for which LCO is provided. One is the emergency systems being on stand-by, and the other is operating state of the systems used regularly. A failure of these usual systems due to the same cause worsens the performance both of power production operation and of the emergency systems, which reduces the potential effectiveness of the safety function during an accident.

Disruption of the operation status or the stand-by status due to function failure of a single train or component among these multiplexed systems cannot disturb power operation directly. However, there is potential reduction in the effectiveness of the above safety functions, and this reduction can be limited by appropriately providing the AOT while the power operation is continued.

In the level 1 probabilistic safety assessment (PSA) for MONJU, the systems which have important functions to prevent core damage are modeled. Several systems for safety limit regulation make significant contributions to prevention of core damage also. Therefore, the conditioned core damage probability assuming a multi-function failure due to failure of one train or component of these systems is evaluated. The effectiveness of measures for preventing such events is appraisable quantitatively by this method. In addition, the safety during an extended outage time can be evaluated as to whether the risk increment is at an allowable level, in the same way.

Evaluation according to this principle has already been applied in actual operation and nuclear regulation in the United States. This paper shows that a similar technique is

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applicable to the Japanese prototype fast breeder reactor MONJU.

2. Outline of Existing Evaluation

Looking at evaluation guidelines up to now, the general rule indicator Regulatory Guide 1.174⁽²⁾ for changes in the authorized conditions of every individual plant and R.G. 1.177⁽³⁾ which modifies the technical specifications were established by USNRC, NEI06-09⁽⁴⁾, a risk informed technical specification guideline, was issued by NEI, and NUMARC93-01⁽⁵⁾ was developed by private industry.

R.G. 1.174 is a comprehensive guideline that established a general standard for changes in authorized conditions using the PSA results for individual plants. The decision-making process for changes in the authorized conditions, evaluation approach, quality assurance and the documentation are shown in this guideline.

In this evaluation approach, an allowable risk standard is provided. This standard is decided based on core damage frequency (CDF) and large early release frequency (LERF) baseline data and Δ CDF and Δ LERF, which are the extra risk caused by the change in authorized conditions.

In R.G.1.177 is another standard for temporary risk change, in which incremental conditional core damage probability (ICCDP) and incremental conditional large early release probability (ICLERP) are used to formulate another allowable risk standard when the AOT is changed in response to a permanent change which limits risk changes to temporary changes during repair or maintenance.

The value of ICCDP was obtained from changes in CDF and the periods during which these new CDF values continued. ICCDP signifies the yearly average of the excess risk that is increased during the outage periods after one-cause multiple failures.

There are a number of nuclear power plants with systems for which application for AOT extension has been made after considering risk information, e.g. Diesel Generators (D/G) in the Columbia, Diablo Canyon, and Comanche Peak nuclear power plants, the low pressure ECCS in the Browns Ferry nuclear power plant, the auxiliary feed water system in the South Texas Project nuclear power plant, and the basic service water system pump in the Byron and Braidwood nuclear power plants.

At the JAEA Oarai Research and Development Center which has been chiefly engaged in the PSA of MONJU, a study has been performed to develop a method for setting the allowed outage time in a fast reactor using risk information⁽⁶⁾. In this study, it is suggested that the evaluation model consider the case where the outage ends when repair of the failure is completed. This conditioned core damage probability (CDP) is assumed to be a function with AOT as a variable.

When this method is applied, it is necessary to use PSA to adequately calculate the conditional core damage probability in each one-cause multiple failure of the system causing shut down of the reactor, and the mean time to repair the component and thus restore the system as well.

Core damage frequency is used as an index of the risk in this report. It is evaluated based on the calculated Δ CDF upon one-cause multiple failure of the system. Repair completion time is limited to that in the safety regulations as analysis condition, and the estimation results of the two parameters mentioned above, namely ICCDP and CDP, are used. The calculation and evaluation method is as follows.

3. Estimation Method

3.1 Estimation of Delta CDF

The calculation formula of Δ CDF assuming a single train outage is as follows.

$$\Delta CDF = R(p_i = 1) - R_0 \quad \bullet \dots\dots\dots(1)$$

Where,

$R(p_i=1)$: conditional CDF

R_0 : baseline CDF

In R.G. 1.174, ΔCDF exceeding $1E-5$ is not allowed. In the safety regulations in MONJU, LCO is defined not for each component but rather for each train of the systems, so that the p_i of Eq. (1), which is the basic event considered here, is an intermediate event of the system fault tree.

Required times for detection of a single train failure of the system and for completion of operations that prevent reactor trip by the failure are defined because the systems which have LCO in the safety regulations interact. A single failure is an event in which operational failure of a system or circuit is detected and that system or circuit is bypassed, allowing inspection or repair. There is no limitation regarding instrument faults in which the instrument operates when it should not, e.g. malfunction with false signal generated. A component with this kind of fault still achieves its function. There are two kinds of measures demanded when there is deviation from the LCO. One is the restoration of the system undergoing one-cause multiple failure and the other is to improve the reliability of the system overall. These measures involve both the systems affected in the one-cause multiple failure and the other intact systems. The ΔCDF was by the method mentioned above, considering the demanded measures in the safety regulations.

3.2 Estimation of ICCDP

In the preceding example, ICCDP values were used in an evaluation of the possibility of an AOT extension. ICCDP with the extended AOT is calculated and compared with a numerical standard i.e. risk threshold. However, this standard is not used to determine whether the current AOT is appropriate or not.

According to the contents of the AOT extension application by the South Texas Project⁽⁷⁾ as an operative example, the evaluation was performed at the time the risk threshold $5E-7$ is reached, or after 30 days which is the backstop AOT, whichever is shorter.

$$ICCDP = \Delta CDF \times AOT_b / 8760 \quad \bullet \dots\dots\dots(2)$$

Where,

AOT_b : backstop AOT

Therefore, the evaluation of ICCDP for each LCO of the safety regulations, comparing these with the index value of $5E-7$, is performed in this report. The calculation method is expressed by Eq. (2)

3.3 Estimation of CDP

The formula for calculating CDP is as follows.

$$CDP = R(p_i = 1) \times \tau_R \times \{1 - p(\tau)\} + U \times p(\tau) \quad \bullet \dots\dots\dots(3)$$

$$p(\tau) = e^{(-\tau / \tau_R)} \quad \bullet \dots\dots\dots(4)$$

Where,

τ : AOT

τ_R : mean time for repair of component

U : conditional core damage probability for each shut down of the reactor

When CDP is calculated based on Eq. (3), repair time τ_R and also component failure rate are parameters which are peculiar to the component, and it is necessary to estimate τ_R appropriately for each event evaluated. The mean time to repair each component of the auxiliary cooling system (ACS) is shown in Table 1.

Table 1 Mean times for repair of each component

Components	MTTR (h)	Data Source
Mortor driven pump (sodium system excluded)	40	WASH-1400
Refrigerating machine		
Blower		
Valves (sodium system excluded)		
D/G	25	Operation data of Joyo EBR-II FFTF
Instrumentation/Electric device	20	
	6	
Damper/Vane	10	

The first term of Eq. (3) expresses conditional core damage probability during power operation after a single outage, taking restoration operations into consideration. The second term is conditional core damage probability for each shut down which is limited to a single outage without restoration. The required measure, manual shut down operation after power drop, is considered here. It is necessary to estimate the conditional core damage probability during a decay heat removal event sequence after reactor shut down. The event tree shown in Fig. 1 for decay heat removal where all cooling loops are operative can be used to calculate this conditional core damage probability.

Fig.1 Event tree for decay heat removal

4. Result and Discussion

4.1 Δ CDF Estimation Result

In R.G. 1.174, Region II is defined as the calculated increase in CDF per reactor year (/RY), and is in the range of $10E-6$ to $10E-5$ /RY. Applications will be considered only if it can be reasonably shown that the total CDF is less than $10E-4$ /RY. From the estimated Δ CDF and the base line CDF of about $1E-7$ /RY which considers only internal events, there are 4 items which may be placed in Region II. All of other items are placed in Region III.

Region III was defined as where the calculated increase in CDF is very small, set here to be less than $10^{-6}/(\text{RY})$. A change will be considered regardless of whether there is a calculation of the total CDF. As shown in Table 2 and Fig. 2, the result of four items in Region II and the top three items of Region III are as follows.

i) “Voltage low” signal of emergency high voltage bus line for D/G start up instrumentation: Region II

A single outage of this system blocks the total black out signal. The total black out signal is generated by the “voltage low” signal of all emergency bus lines. Therefore, the natural circulation cooling function of more than 1 loop of ACS activated by this total black out signal necessarily fails, and this function failure makes the ΔCDF about $3\text{E-}6/(\text{RY})$.

ii) ACS start up signal for D/G start up instrumentation: Region III

The single outage condition involving sodium leakage in other two loops causing this signal necessarily causes the failure of the forced circulation function in more than 1 loop. In addition, this condition causes one of accident management (AM) functions for Loss of Reactor liquid Level (LORL) to certainly fail as well. This AM operation needs the control function of all 3 loops of the ACS to prevent thermal shrinkage of sodium after 1 loop leaks in this LORL event, and these 2 failures make the ΔCDF about $5\text{E-}7/(\text{RY})$.

iii) Decay heat removal system and system for removal of other residual heat: Region II

A single outage of this system increases the probability of Protected Loss of Heat Sink (PLOHS). Namely, the probability of failure of the forced circulation function and the natural circulation function in more than 1 loop is increased by this outage. The ΔCDF becomes about $8\text{E-}6/(\text{RY})$.

iv) Emergency high voltage bus line: Region II

A single outage of this system increases the probability of the LORL and PLOHS event sequences. This is added to the probability of failure particularly of the decay heat removal by the Maintenance Cooling System (MCS) and the support function of the electromagnetic pump of primary sodium overflow system. This condition makes the ΔCDF about $3\text{E-}6/(\text{RY})$.

v) Emergency low voltage bus line: Region II

A single outage of this system affects the plant as well as the high voltage bus line mentioned above. However, the resulting increase in the failure probability of other related systems is not greater than that of the high voltage bus line, because it is downstream of the other high voltage circuits in MONJU. This condition makes the ΔCDF about $3\text{E-}6/(\text{RY})$.

vi) Reactor auxiliary component cooling water system: Region III

A single outage of this system mainly increases the probability of PLOHS. The reasons are that such outage disturbs the MCS function for decay heat removal and the electromagnetic pump of primary sodium overflow system which is related to LORL. This condition makes the ΔCDF about $6\text{E-}7/(\text{RY})$.

vii) Reactor auxiliary component cooling sea water system: Region III

A single outage of this system affects the plant as well as the reactor auxiliary component cooling water system mentioned above. Moreover, the failure increases the probability of loss of electric power because this sea water system is a heat sink of the heating, ventilation, and air-conditioning (HVAC) systems of the electric equipment room. This condition makes the ΔCDF about $7\text{E-}7/(\text{RY})$.

Table 2 Estimation results of conditional core damage risk

LCO (completion time defined)		AOT (h)	Backstop AOT(h)	Δ CDF (1/Ry)	ICCDP	U-RtR	CDP
RPS actuation instrumentation		28~75	747	Negligibly small			
ESF actuation signal instrumentation		42~84	747				
D/G start up instrumentation	D/G start up logic circuit	42	756	1.3E-08	4.7E-09	3.2E-09	1.5E-10
	Voltage low signal of emergency high voltage bus line	6	756	3.3E-06	3.6E-07	2.3E-09	3.3E-09
	ACS start up signal	42	756	5.3E-07	9.0E-08	3.3E-08	5.3E-10
Decay heat removal system and other residual heat removal system		8	756	8.2E-06	8.9E-07	-1.1E-09	9.1E-09
Off-site electric power		276	756	6.0E-08	1.1E-08	4.2E-09	1.8E-10
Emergency alternating current source		8	756	1.2E-08	4.6E-09	2.9E-09	2.4E-09
Emergency direct current source		276	756	2.4E-10	3.4E-09	3.2E-09	1.4E-10
Emergency bus line	high voltage	44	756	3.0E-06	4.8E-07	1.5E-07	2.4E-08
	low voltage	44	756	2.8E-06	4.6E-07	1.5E-07	2.4E-08
Reactor auxiliary component cooling water system		276	756	6.0E-07	1.1E-07	4.2E-08	3.7E-09
Reactor auxiliary component cooling sea water system		276	756	7.1E-07	1.2E-07	4.2E-08	4.2E-09

Fig.2 Δ CDF in various leading cases

Looking at these results, all of the calculated increases in CDF are less than 10^{-5} /RY. Thus, any of these single outages are considered to satisfy the relevant rules of risk-informed regulation with respect to CDF.

The estimated Δ CDF has the following effects on the Reactor Protection System (RPS) actuation instrumentation and Engineering Safeguard Features (ESF) actuation signal instrumentation: the increase in CDF by 1channel failure and the reduction in CDF by the required measure, i.e. turning the channel on to keep generating a signal when the 1 channel failure has occurred, are negligibly small. The reason is that the contribution of the reactor trip signal failure is not greater than the common cause failure of reactor trip breakers, and the influence of a 1channel failure is not a dominant factor in reactor trip failure probability.

Additionally, the Δ CDF was estimated considering the measures required for the other two intact trains of the decay heat removal system as shown in Fig. 3. According to the safety regulations, the above increase in CDF of $8E^{-6}$ (/RY) within 8 hours is allowed. The

required measure for the other two intact trains is to change the surveillance interval from one month to two days. The estimated Δ CDF where this measure is carried out is reduced. The regulations allow this plant status for 10 days, and if the restoration of failed train is not completed in that time, the regulations require that 1 intact train be kept in operation. This measure reduces the CDF further and this status is allowed for 30 days by the regulations. If the restoration is not finished within these 30 days, the reactor should be shut down. The conditional core damage probability at that point is the U which is used in Eq. (3).

Fig.3 CDF of decay heat removal system under various conditions

4.2 ICCDP Estimation Result

The ICCDP is proportional to the AOT if the Δ CDF is fixed. According to the precedent example, ICCDP can be used to as an indication of the risk variation parameter used for AOT extension application. The maximum extended backstop AOT is 30days, and the values of ICCDP are calculated using this AOT in this report. The conditional core damage probability U must be considered if there is shut down with restoration uncompleted, and is added to ICCDP. The evaluation was done by comparing the ICCDP with the risk threshold value of $5E-7$. The results show that there is one item which crossed the threshold. Below are this one item and five items which are near-threshold.

- i) “Voltage low” signal of emergency high voltage bus line for D/G start up instrumentation
In case of a single outage of this system, the allowed restoration completion time in the regulations is 6 hours. ICCDP of the backstop AOT in this case is estimated at about $4E-7$. The risk threshold is estimated to be crossed after about 44 days.
- ii) Decay heat removal system and system for removal of other residual heat
In case of a single outage of this system, the completion time established in the regulations is 8 hours. ICCDP of the backstop AOT in this case is estimated to be about $9E-7$. The risk threshold value is estimated to be crossed after about 18 days. These estimations do not consider the required measures to be performed on the other intact train. The evaluation results considering these measures are as follows
- iii) Emergency high voltage bus line
In case of a single outage of this system, the allowed restoration completion time

established in the regulations is 44 hours. ICCDP of the backstop AOT in this case is estimated to be less than $5E-7$. The risk threshold value is estimated to be crossed after about 49 days.

iv) Emergency low voltage bus line

In case of a single outage of this system, the allowed restoration completion time established in the regulations is 44 hours. ICCDP of backstop AOT in this case is estimated at less than $5E-7$. The risk threshold value is estimated to be crossed after about 53 days.

v) Reactor auxiliary component cooling water system

In case of a single outage of this system, the allowed restoration completion time established in the regulations is 10 days. ICCDP of the backstop AOT in this case is estimated at about $1E-7$. The risk threshold value is estimated to be crossed after about 243 days.

vi) Reactor auxiliary component cooling sea water system

In case of a single outage of this system, the allowed restoration completion time established in the regulations is 10 days. ICCDP of the backstop AOT in this case is estimated at about $1E-7$. The risk threshold value is estimated to be crossed after about 207 days.

In case of a single outage of the decay heat removal system and the system for removal of other residual heat, the measure for reliability improvement of the two intact trains is required to be completed within 8 hours. This measure reduced CDF as above mentioned. The various ICCDP values for the different conditions should be evaluated.

Fig.4 ICCDP of decay heat removal system under various conditions

As shown in Fig. 4, ICCDP which crosses the risk threshold in just one case. In this graph, ICCDP1 is for a single outage, ICCDP2 considers the surveillance interval improvement performed in this case, and ICCDP3 considers improvement where a single train is activated. Among these results, only ICCDP1 crossed the risk threshold, and then only in the range of the backstop AOT; the time at that point is 18 days as mentioned in ii)

above.

Consequently, there are no items which should not be considered when making any extension of the AOT in the safety regulations. Furthermore, ICCDP estimation results indicate that the present AOT has a safety margin.

4.3 CDP Estimation Result

As shown in Table 2, the maximum CDP based on the AOT which is established in the safety regulations is $2E-8$. This CDP is conditional core damage probability per failure of the system. As expressed in Eq. (3), the calculation formula of this CDP, the condition for this formula to be monotonically increasing is that the derivative is positive. The first derivative function of Eq. (3) is as follows.

$$\frac{d(CDP)}{d\tau} = -\frac{\{U - R(p_i = 1)\tau_R\}P(\tau)}{\tau_R} \dots\dots\dots(5)$$

Therefore, the condition for the CDP to be monotonically increasing is $U - R(p_i = 1)\tau_R < 0$. CDP estimation reveals that the case in the safety regulations which satisfies this condition is a single outage of the decay heat removal system and the system for removal of other residual heat. The CDP of all of the other cases monotonically decrease. Thus, the other items have core damage risk which is small enough to allow extension of the AOT.

$R(p_i = 1)\tau_R$ is the asymptotic value of CDP with increase of AOT. There are two cases where there is greater CDP than this asymptotic value, i.e. emergency high/low voltage bus line. The AOT of these items is 44 hours but it remains possible that the CDP will be reduced by the extension of this time and the possible restoration of these systems.

On the other hand, the CDP of a single outage of the decay heat removal system is calculated without considering the reliability improvement of the intact train. The CDP is a monotonically increasing function; however, the value of CDP is almost equal to the asymptotic value in this condition. Therefore, the CDP value of becomes saturated previously before reaching the present AOT.

Fig.5 Relation between CDP and AOT after single outage of decay heat removal system

Figure 5 shows the relation between CDP and AOT after a single outage of the decay heat removal system. The mean time to repair, τ_R , with the CDP described above was 10 hours, the same as the mean repair time of the damper or vane of the air cooler. Sensitivity

of this parameter to CDP was estimated with the τR value set at 6 hours for instrumentation/electric devices and at 40 hours for the blower. According to Eq. (5), a small τR can reverse the magnitude relation of U and $R(pi=1)$ τR . As the result of this estimation, the case in which τR is 6 hours turns the CDP into a monotonically decreasing function. The case of τR being 40 hours had larger CDP than the case of 10 hours. These results suggest how the CDP can be suppressed to make it possible to shut down the reactor immediately when the repair needs a couple of days.

5. Conclusion

The systems which influenced CDF in the facilities where AOT is established in the safety regulations were chosen, and ΔCDF was calculated to estimate amount of the influence. The possible time of the examination was estimated based on risk information.

Based on the estimation results above, ICCDP used for examination of possible AOT extension was estimated, and it was judged that it is possible to extend all present AOT.

Evaluation was performed based on the conditional core damage probability per failure, and some prospects for improvements that we should examine in the future were revealed.

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