Probability of adventitious fuel pin failures in fast breeder reactors and event tree analysis on damage propagation up to severe accident in Monju

Yoshitaka Fukano[a*], Kenichi Naruto[b], Kenichi Kurisaka[a], and Masahiro Nishimura[a]
[a]Japan Atomic Energy Agency, Tsuruga, Japan
[b]NESI Inc., O-arai, Japan

Abstract: Experimental studies, deterministic safety analyses and probabilistic risk assessments (PRAs) on local fault (LF) propagation in sodium cooled fast reactors (SFRs) have been performed in many countries because LFs have been historically considered as one of the possible causes of severe accidents. Adventitious fuel pin failures were considered to be the most dominant initiators of LFs in these PRAs because of high frequency of occurrence during reactor operation and possibility of subsequent pin-to-pin failure propagation. Therefore event tree analysis (ETA) on fuel element failure propagation initiated from adventitious fuel pin failure (FEFPA) in Japanese prototype fast breeder reactor Monju was performed in this study based on state-of-the-art knowledge on experimental and analytical studies on FEFPA and reflecting latest operation procedure at emergency in Monju. Probability of adventitious fuel pin failures in SFRs which is the initiating event of this ETA was also updated in this study. Probability of FEFPA to the peripheral sub-assemblies was quantified to be 1.7×10^{-12} in Monju based on this ETA. It was clarified that FEFPA in Monju was negligible and could be included in core damage fraction of the anticipated transient without scram and protected loss of heat sink in the viewpoint of both probability and consequence.

Keywords: PRA, ETA, SFR, LF

1. INTRODUCTION

Local fault (LF) accidents have been considered as one of the possible causes of core-disruptive accidents or severe accidents in sodium cooled fast reactors (SFRs) for a long time. The fuel element failure propagation (FEFP) was considered to be of greater importance in safety evaluation because fuel elements are generally densely arranged in the subassemblies (SAs) of SFRs and power densities in this reactor type are higher compared with those in light water reactors (LWRs) as shown in Table 1. Therefore probabilistic risk assessments (PRAs) [1-3], deterministic safety analyses and experimental studies on LF accident have been performed in many countries historically.

Table 1 Comparison of power densities among FBR, PWR and BWR

	FBR	PWR	BWR
Power density (kW/l)	350~1000	~100	~50

Table 2 Frequency of initiating events of LFs for CDFR

Initiating event	Frequency (/ry)
Adventitious fuel pin failure	35
Inlet blockage	10^{-3}
Outlet blockage	10^{-3}
Wrapper split	2×10^{-2}
Overrated sub-assembly loaded	3×10^{-2}
Partially blocked sub-assembly loaded	2×10^{-3}
Oil in sub-assembly	10^{-1}
Non-oil debris in sub-assembly	10^{-1}

Table 2 shows frequency of initiating events of LFs for British commercial demonstration fast reactor (CDFR) [1]. Among the different initiators of LFs, adventitious fuel pin failure was most dominant one because of high frequency of occurrence during reactor operation and possibility of pin to pin failure propagation. Therefore event tree analysis (ETA) of FEFP from adventitious fuel pin failure (FEFPA) is necessary in SFRs. ETA of FEFPA in Japanese prototype fast breeder reactor Monju (Monju) was performed in this study based on latest knowledge of experimental

----

* Contact author: fukano.yoshitaka@jaea.go.jp

and analytical studies on FEFPA and reflecting latest operation procedure at emergency in Monju. Probability of adventitious fuel pin failures in SFRs was also updated based on the state-of-the-art review of open papers concerning fuel pin failure experiences in SFRs, because probabilities of fuel pin failures used in existing PRA [1-3] were based on experiences up to 1985.

2. Updated frequency of initiating event

In order to quantify the frequency of adventitious fuel pin failure, fuel pin failure experiences in SFRs were widely investigated based on open papers [4-13].

Table 3 shows the number of failed fuel pins and related data in SFRs based on this investigation. It should be noted that fuel pin failure experiences in the SFRs of which nominal full power were less than 100 MWth were excluded from this table because these experiences are used in the ETA for Monju of which nominal full power is 714 MWth.

Table 3 The number of failed fuel pins and related data in SFRs

(n) Reactor	JOYO (1) Mk-II	JOYO (2) Mk-III	(3) Phenix	(4) Super Phenix	(5) PFR	(6) FFTF
(A) The number of irradiated (driver) fuel pins (-)	43434	16510	166521	98644	98000	47500
(B) The number of failed (driver) fuel pins (-)	0	0	29	0	22	1
(C) Mean residence time (years)	0.7	1.1	1.9	1.8	1.0	1.8
(D) Equivalent full power years (years)	5.0	1.9	10.1	0.9	4.1	6.2
(E) Total number of fuel pins in equilibrium core (-)	8509	10795	22351	98644	23400	15841
(F) Average achieved burnup (GWd/t)	42	68.5	100	60	150	70
(G) Nominal full power (MWth)	100	140	563	2990	650	400

The largest frequency of fuel pin failure in Monju was decided to use conservatively in the ETA because frequency of fuel pin failure could be obtained by the following several methods.

(1) Method 1: The frequency of fuel pin failure in this method (P_1) was calculated by following equation and A, B and C in **Table 3**. The arithmetic average of failed fuel pin and mean residence time were used in this method. It should be noted that the average weighted by irradiated fuel pins was used for mean residence time.

$$P_1 = \frac{\sum B_n}{\sum A_n} \div \frac{\sum A_n C_n}{\sum A_n} \tag{1}$$

The frequency of fuel pin failure in Monju of this method (P_{M1}) can be calculated by the following equation;

$$P_{M1} = P_1 \times N_{SA} \times N_{Pin} \tag{2}$$

N_{SA} : Total number of SAs in the core

N_{Pin} : Total number of fuel pins in one SA

(2) Method 2: The frequency of fuel pin failure in this method (P_2) was calculated by following equation and B, D and E in **Table 3**. The frequency of fuel pin failure for each reactor and the average of irradiated fuel pins weighted by equivalent full power years (EFPYs) were used in this method.

$$P_2 = \frac{\sum(\frac{B_n}{D_n E_n} \times D_n E_n)}{\sum D_n E_n} \quad (3)$$

The frequency of fuel pin failure in Monju of this method (P_{M2}) can be calculated by the following equation assuming that mean load factor of Monju is 71 %.

$$P_{M2} = P_2 \times N_{SA} \times N_{Pin} \times 0.71 \quad (4)$$

(3) Method 3: The frequency of fuel pin failure per burnup (P_3) was calculated by following equation and B, D, E and F in **Table 3**. In addition to method 2, average achieved burnups of irradiated fuel pins were used in this case.

$$P_3 = \frac{\sum(\frac{B_n}{D_n E_n} \times D_n E_n)}{\sum D_n E_n F_n} \quad (5)$$

The frequency of fuel pin failure in Monju of this method (P_{M3}) can be calculated by following equation assuming that the average achieved burnup of Monju is 80 GWd/t.

$$P_{M3} = P_3 \times N_{SA} \times N_{Pin} \times 0.71 \times 80 \quad (6)$$

(4) Method 4: Frequency of fuel pin failure per reactor power (P_4) was calculated by following equation and B, D, E and G in **Table 3**. In addition to method 2, nominal full power of each reactor was used in this case.

$$P_4 = \frac{\sum(\frac{B_n}{D_n E_n} \times D_n E_n)}{\sum D_n E_n G_n} \quad (7)$$

The frequency of fuel pin failure in Monju of this method (P_{M4}) can be calculated by following equation assuming that the nominal full power of Monju is 714 MWth.

$$P_{M4} = P_4 \times N_{SA} \times N_{Pin} \times 0.71 \times 714 \quad (8)$$

Table 4 shows calculated frequency of fuel pin failure for each method based on the above-mentioned equations. The frequency of fuel pin failure in Monju in method 1 was decided to be conservatively used in the following ETA because the frequency was the largest in all cases.

Table 4 Frequency of fuel pin failure by each method

Method	Method 1	Method 2	Method 3	Method 4
Frequency of fuel pin failure	7.2×10^{-5} [/y/pin]	9.1×10^{-5} [/EFPY/pin]	9.9×10^{-7} [/EFPY/pin/(GWd/t)]	1.0×10^{-7} [/EFPY/pin/MWth]
Frequency of fuel pin failure in Monju (/ry)	2.4	2.2	1.9	1.8

3. Event tree analysis for failure propagation from fuel pin failure

The damage propagation from fuel pin failure up to whole core damage and automatic reactor trip by delayed neutron detectors (DNDs) were considered to be main events in the existing PRA [1-3]. Not only automatic rector trip by DNDs but also various reactor shutdown means by several kinds of detectors such as precipitators or NaI detectors in cover gas (CG) method were equipped in Monju as shown in **Figure 1**.

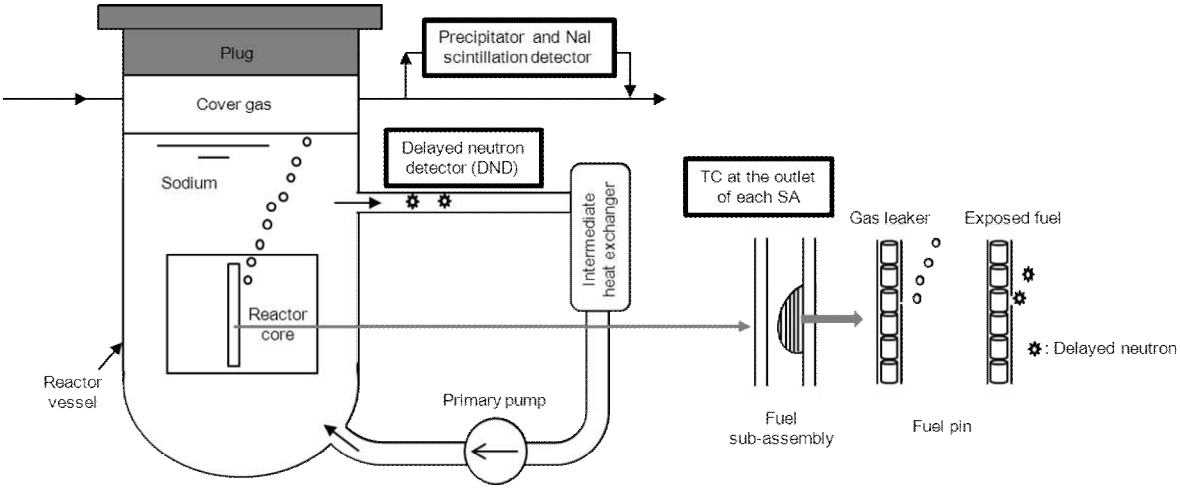


Figure 1 Schematic drawing of failed fuel detectors in Monju

The main flow after fuel pin failure in the operation procedure, alert and reactor trip level of detectors of fuel pin failure are shown in **Figure 2** and **Table 4** respectively.

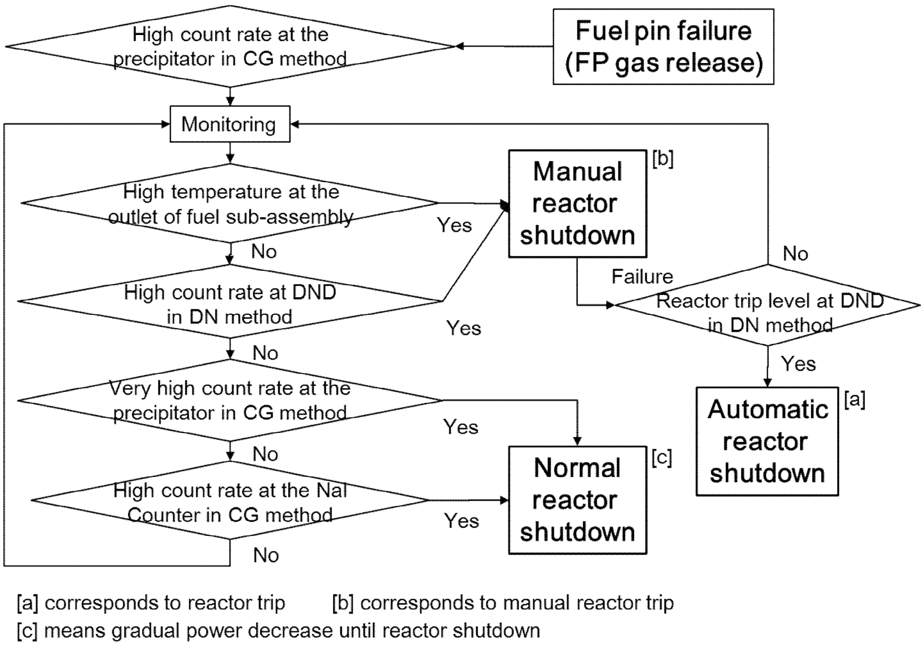


Figure 2 Main operation procedure after fuel pin failure in Monju

Table 4 Alert and reactor trip level of detectors for fuel pin failure in Monju

Name of alert or reactor trip	Alert or reactor trip threshold
Alert of high count rate at the precipitators in CG method	FP gas release (0.01% of one fuel pin)
Alert of very high count rate at the precipitators in CG method	Fuel pin failure (0.01% of total fuel pins)
Alert of high count rate at the NaI counter in CG method	Fuel pin failure (0.02% of total fuel pins)
Alert of high count rate at DNDs in DN method	Breached cladding area (over 200mm^2)
Alert from TC at the outlet of sub-assembly	More than 66% of flow blockage within one sub-assembly
Reactor trip at DNDs in DN method	Breached cladding area (over 5,000mm^2)

The ET reflecting the operation procedure after the fuel pin failure and the latest knowledge on experiments and analyses were presented in **Figure 3**. The main characteristics of this ET compared with existing PRA are:

(i) Various measures for failed fuel pin detection and reactor shutdown based on the operation procedure after the fuel pin failure in Monju;

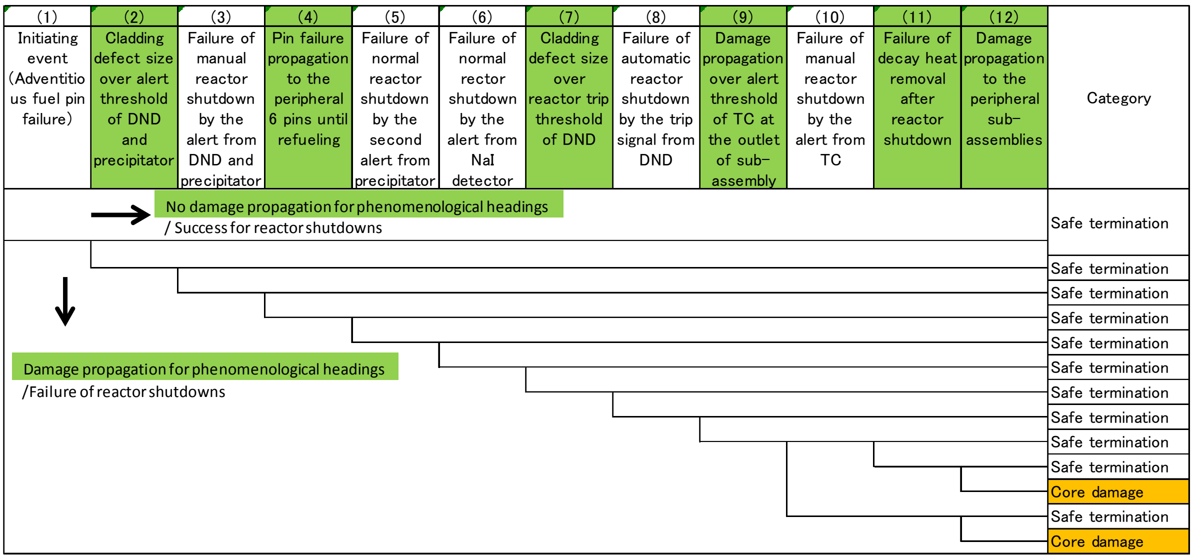
(ii) More detailed development of the ET headings based on the state-of-the-art knowledge on experiments and analyses;

(iii) The possibility that fuel pin failure does not expand to detectable scale even at the end of cycle;

(iv) Removal of damaged SA by refuelling after reactor shutdown owing to detection of fuel pin failure;

(v) The possibility that decay heat is not removed in terms of coolable geometry even after reactor shutdown.

Figure 3 Main ET for FEFPA



Although the quantification of branch probabilities for phenomenological headings was determined through the engineering judgment based on the knowledge on experiments and analyses, it was standardized using **Table 5** in order to keep consistency in this ETA.

Table 5 Branch probability ranks

Probabilistic rank	Qualitative representation	Representative value	Range of application
1	Indeterminate	0.5	0.7～0.3
2	Unlikely	0.2	0.3～0.1
	Likely	0.8	0.7～0.9
3	Highly Unlikely	0.05	0.1～0.01
	Highly Likely	0.95	0.9～0.99
4	Extremely Unlikely	<0.01	<0.01
	Extremely Likely	>0.99	>0.99
5	Impossible	ε	ε
	Certain	1-ε	1-ε

Each branch probability is described below;

3.1. Cladding defect size over alert threshold or reactor trip threshold in DN method [headings (2) and (7)]

After the fuel pin failure and subsequent FP gas release detection at the precipitators in CG method, manual reactor shutdown will be initiated if cladding defect size exceeds 200 mm^2 which is the alert threshold in DN method. Furthermore automatic reactor shutdown will be initiated if cladding defect size exceeds 5,000 mm^2 which is the reactor trip level in DN method.
Figure 4 shows defect sizes of failed fuel pins in the experiments on run beyond cladding breach (RBCB) [14]. There existed no data which shows cladding defect size over 200 mm^2 even after 200 days RBCB in Mol-7B experiment [15]. Table 6 shows burn-ups of driver fuel pins at the time of the adventitious fuel pin failure [6, 12]. The mean burn-up of the driver fuel pins was approximately 7.2 at.% from this table and was higher compared with that of Mol-7B experiment which was approximately 6.5 at.%.

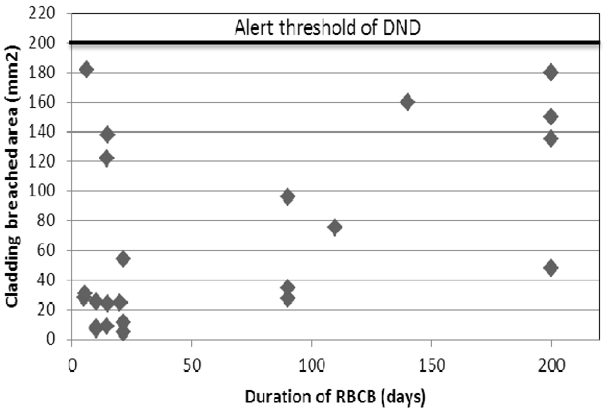


Figure 4 Cladding breached areas in RBCB experiments

Table 6 Fuel burn-ups at driver fuel pin failures

Reactor	No.	Cladding material	Fuel burn-up at fuel pin failure (at.%)
Phenix	1	316CW	8.9
	2	316CW	9.1
	3	15/15Ti	0.6
	4	15/15Ti	5.7
	5	15/15Ti	5.3
	6	15/15Ti	9
PFR	1	316CW	1.2
	2	316CW	10.7
	3	316CW	5.6
	4	316CW	7.6
	5	316CW	10.7
	6	316CW	10.4
	7	316CW	3.14
	8	316CW	9.54
FFTF	1	316SS	10
Average			7.17

Conservatively assuming that the burn-up at the time of the adventitious fuel pin failure in Monju is 6.5 at.% which is same as that in Mol-7B experiment, cladding defect size never exceeds 200 mm^2 at the maximum burn-up of approximately 8 at.% in Monju because cladding defect size was less than 200 mm^2 even at the maximum burn-up of 13 at.% in Mol-7B experiment. Furthermore increase rate of cladding breached area is approximately 10 mm^2/day at most and it will decrease to zero as time goes on [16].Consequently cladding defect size never exceeds 5000 mm^2 until refuelling even assuming initial cladding defect size of 200 mm^2. Therefore the probabilities of cladding defect size over alert threshold and reactor trip threshold by DN method were judged to be highly unlikely (0.05).

3.2. Fuel pin failure propagation to the peripheral 6 pins until refuelling [heading (4)]

Normal reactor shutdown will be initiated after the detection of 0.01 % of total driver fuel pin failures at the precipitators in CG method and after the detection of 0.02 % of total driver fuel pin failures at the NaI detectors in CG method. More than 3 or 6 fuel pin failures are necessary for normal reactor shutdown by precipitators or NaI detectors respectively because the number of driver fuel pin in the core are 33,462 pins in Monju.
Analyses for following two possible causes of FEFPA were performed in this study:
(1) Thermal transient due to FP gas release from adjacent fuel pin;
(2) Flow reduction due to flow blockage.

3.2.1. Thermal transient due to FP gas release from adjacent fuel pin

Figure 5 shows cladding temperature history analysed by FALL code [17-19] of which main analytical conditions are shown in Table 7. Although cladding temperature increased up to approximately 750 degree C during FP gas release in the case that angles with gas blanketing are 360 degree assuming multiple fuel pin failures, duration of FP gas release was estimated at most 104s even under the condition that the reactor operation is kept without removing the initial defect pin from the core. Necessary duration for cladding creep failure of the fuel pin at the end of cycle is approximately 130 hours at 750 degree C from Figure 6 which was used in the licensing document of Monju [17]. Therefore cladding failure due to FP gas release from adjacent fuel pins was judged to be impossible (ε) in Monju.

Table 7 Main analytical conditions for FALL code analysis

Axial position of gas blanketing	Top of fissile column (TFC) / Peak power node (PPN)
Power density of fuel (W/cm^3)	938 at TFC / 1720 at PPN
Angles for gas blanketing (degree)	180 / 360
Released gas temperature (degree C)	660 at TFC / 535 at PPN
Coolant inlet temperature (degree C)	655 at TFC / 525 at PPN
Coolant outlet temperature (degree C)	660 at TFC / 535 at PPN
Number of axial cell	1
Number of radial cell	15
Fuel pellet	10
Fuel-Cladding gap	1
Cladding	3
Coolant	1
Numbers of azimuthal cell	
Angles with gas blanketing	3 for 180degree / 1 for 360degree
Angles without gas blanketing	3 for 180degree / 0 for 360degree

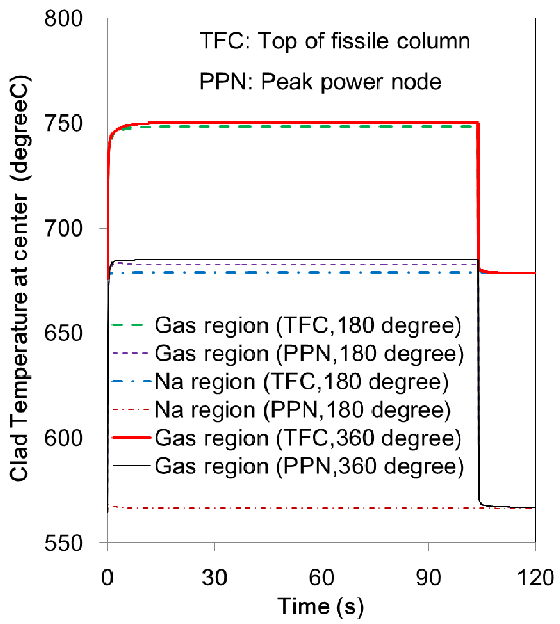


Figure 5 Cladding temperature history

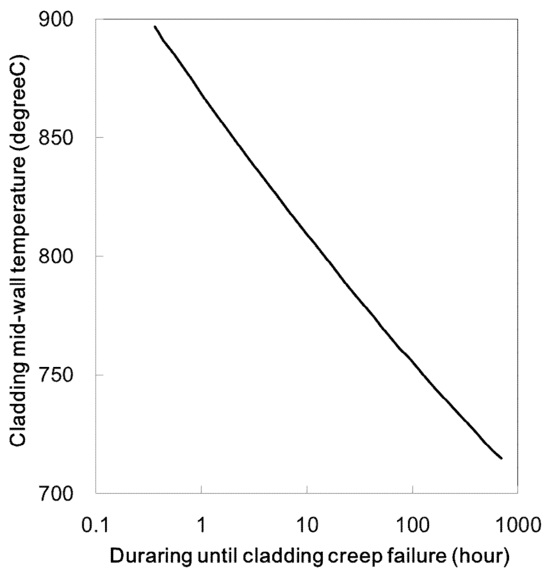


Figure 6 Duration until cladding creep failure

3.2.2. Flow reduction due to flow blockage

Total flow blockage was approximately 38% of free sodium channel at the end of the RBCB in the Mol-7B experiment [15] in the higher power and higher coolant temperature conditions compared with Monju. Seventeen fuel pins of total 18 irradiated fuel pins were failed in this experiment. Therefore flow blockage by each fuel pin was approximately 2.2 % of total flow area. It should be noted that flow blockage rate by each fuel pin in Monju is much smaller because there is 169 fuel pins in one SA. Figure 7 shows cladding and coolant temperatures in case of flow blockage analysed by SEETHE code [17, 18] of which main analytical conditions are shown in Table 8. Up to approximately 30 % of flow blockage, coolant temperatures and cladding temperatures were below boiling point and 830 degree C respectively which were the safety criteria for fuel pin failure in the safety assessment for Monju [17]. Therefore fuel pin failure propagation was judged conservatively to be highly unlikely (0.05) at the 2.2 % blockage of total flow area.

Table 8 Main analytical conditions for SEETHE code analysis

Radial position of blockage	Center
Axial position of blockage	Axial center of the core
Blockage rate (%)	9.5 / 14.3 / 19.0 / 25.4 / 31.7 / 39.7 / 47.6 / 57.1 / 66.6 / 77.7
Number of radial mesh	14
Number of axial mesh	50
Axial height for calculation (m)	1.0
Nominal power condition	
Power of the sub-assembly (kW/m)	39.4
Coolant inlet temperature (degree C)	487
Coolant inlet velocity (m/s)	549
Decay heat power condition	
Power of the sub-assembly (kW/m)	28.0
Coolant inlet temperature (degree C)	468
Coolant inlet velocity (m/s)	0.429

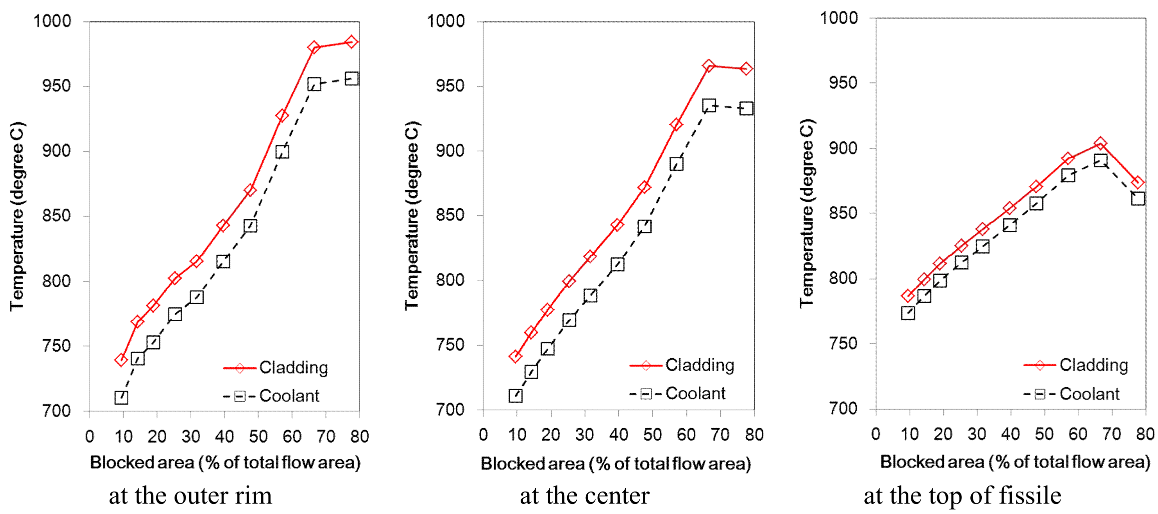


Figure 7 Cladding and coolant temperature in the downstream of blockage at nominal power

Therefore fuel pin failure propagation to the peripheral 6 pins until refuelling was conservatively judged to be highly unlikely (0.05).

3.3. Damage propagation over alert threshold by TC at the outlet of SA [heading (9)]

Manual reactor shutdown will be initiated if coolant flow blockage exceeds 66% of total flow area in one SA which is the alert threshold of TC at the outlet of SA. Although there is no RBCB experiment with cladding defect size over 5,000 mm^2, probability of damage propagation from cladding defect of which size was over 5,000 mm^2 until flow blockage rate of 67% was quantified based on the analyses and related experimental data.

As described in Sec. 3.2.1, damage propagation is impossible (ε) based on the analysis on FP gas release even assuming multiple pin failures.

The remaining possibility of damage propagation is only in the case of molten fuel ejection into the coolant channel due to flow blockage induced by fuel sodium reaction product. In terms of possibility of molten fuel ejection, there was neither fuel melting nor molten fuel ejection into the coolant channel in the existing RBCB experiments. In addition, Figure 8 shows radial temperature distribution within the fuel calculated conservatively by FALL code in case of 67% of coolant flow blockage Monju. Areal melt fraction was approximately 5% which was far below the molten fuel ejection threshold of at least 20 % [19]. This result shows that there is no molten fuel ejection before heading (9). On one hand, it should be noted that damage propagation will be highly unlikely even in case of small amount of molten fuel ejection [20]. Therefore damage propagation over alert threshold by TC at the outlet of SA was judged to be conservatively highly unlikely (0.05).

3.4. Failure of decay heat removal after reactor scram [heading (11)]

Coolant temperature profile after the reactor scram in case of 67% of coolant flow blockage was calculated by SEETHE code as shown in Figure 9. Maximum coolant temperature was 680 degree C and far below the boiling point of coolant. Fuel and cladding temperature were also calculated by FALL code as shown in Figure 8. Fuel and cladding temperature was far below the fuel melting point and 830 degree C respectively which are the conservative criteria for fuel pin failure. Therefore failure of decay heat removal after reactor scram is judged to be unlikely to occur (0.2).

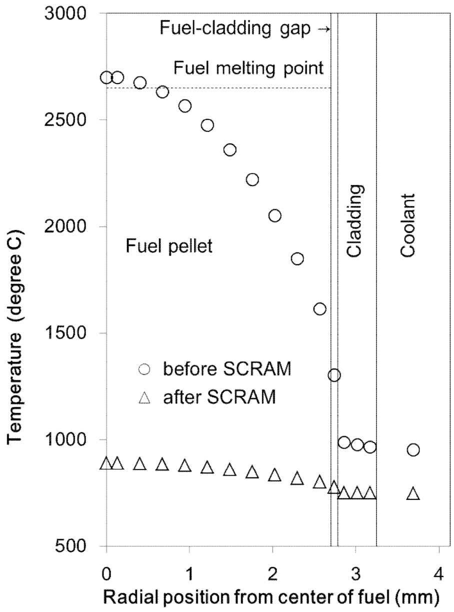


Figure 8 Fuel and cladding temperature profile calculated by FALL code

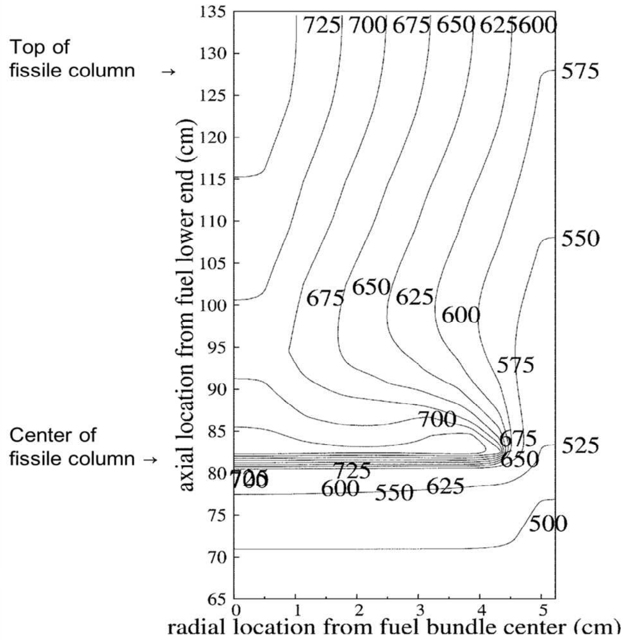


Figure 9 Coolant temperature profile calculated by SEETHE code

3.5. Damage propagation to the peripheral SAs [heading (12)]

Although damage propagation is unlikely in case of small amount of fuel melting [18], damage propagation is conservatively judged to be likely (0.8) if large amount of fuel melting due to more than 66% of flow blockage. It should be noted that damage propagation might be terminated in the inter SA gap or control rod guiding tube (CRGT) before propagation to the peripheral six SAs.

3.6. Fault tree analysis for other headings [heading (3), (5), (6), (8), and (10)]

There is functional dependency among headings (3), (5), (6), (8), and (10) in Figure 3 because those headings share some support and common systems. It is necessary to consider this functional dependency in the event tree quantification. So, those shared systems were identified as shown in the upper part of Table 9. Since failure of these shared systems causes dependently functional failure in some of those headings, combination of failures of the support and common systems was also developed in support event tree. Then, the functional dependency was considered by combining the main and support event trees with the event tree linking (ETL) method by using RISKMAN® code. In order to obtain the branch probability in those event trees, the fault tree analysis (FTA) was performed in addition to the consideration in the section 3.1 through 3.5. This FTA includes the quantification of human error probability (HEP) on the basis of allowable time estimation for operators. HEP during the allowable time is estimated based on time reliability curves from technique for human error rate prediction (THERP) [21]. Main results in this FTA as shown in Table 9 were applied to calculate the accident sequence probability with the ETL method.

Table 9 Results of FTA

<table>
<tr><th colspan="5">Headings</th><th>Probability</th></tr>
<tr><td rowspan="10">Support and common systems for frontline systems</td><td>I</td><td colspan="3">Unavailability of support systems (power supply) for manual reactor shutdown</td><td>2.30E-06</td></tr>
<tr><td>II</td><td colspan="3">Unavailability of common system for manual reactor shutdown by the alert from precipitators or NaI detectors (Ar gas sampling line)</td><td>7.60E-04</td></tr>
<tr><td>III</td><td colspan="3">Unavailability of common component for precipitators (3 signal line and calibration error)</td><td>4.01E-04</td></tr>
<tr><td>IV</td><td colspan="3">Unavailability of common component for precipitators (2 signal line)</td><td>1.69E-08</td></tr>
<tr><td>V</td><td colspan="3">Common cause failure and calibration error in DN method for manual and automatic shutdown</td><td>4.01E-04</td></tr>
<tr><td>VI</td><td colspan="3">Unavailability of common system in DN method for manual and automatic shutdown (detectors, control circuits, and power supply for A loop)</td><td>2.71E-06</td></tr>
<tr><td>VII</td><td colspan="3">Unavailability of support system in DN method for manual and automatic shutdown (power supply for B loop)</td><td>1.95E-06</td></tr>
<tr><td>VIII</td><td colspan="3">Unavailability of support system in DN method for manual and automatic shutdown (power supply for C loop)</td><td>1.95E-06</td></tr>
<tr><td>IX</td><td colspan="3">Failures of components necessary only for manual shutdown</td><td>2.91E-06</td></tr>
<tr><td>X</td><td colspan="3">Failures of components for reactor shutdown (reactor trip breakers and control rods)</td><td>6.41E-08</td></tr>
<tr><td rowspan="9">Main event tree headings (Frontline systems)</td><td rowspan="2">(3)</td><td rowspan="2">Failure of manual reactor shutdown by the alert from DNDs and precipitators</td><td>(3)-2</td><td>Failures of DN detectors</td><td>1 out of 2:7.60E-07
2 out of 2:5.77E-13</td></tr>
<tr><td>(3)-3</td><td>Cognitive and decision error against alert</td><td>4.26E-04</td></tr>
<tr><td>(5)</td><td>Failure of normal reactor shutdown by the second alert from precipitators</td><td>(5)-2</td><td>Cognitive and decision error against alert</td><td>4.26E-04</td></tr>
<tr><td rowspan="2">(6)</td><td rowspan="2">Failure of normal rector shutdown by the alert from NaI detectors</td><td>(6)-1</td><td>Failures of NaI detectors</td><td>1.16E-06</td></tr>
<tr><td>(6)-2</td><td>Cognitive and decision error against alert</td><td>8.27E-04</td></tr>
<tr><td>(8)</td><td>Failure of automatic reactor shutdown by the trip signal from DNDs</td><td>(8)-1</td><td>Failures of DN detectors</td><td>1 out of 2:7.60E-07
2 out of 2:5.77E-13</td></tr>
<tr><td rowspan="2">(10)</td><td rowspan="2">Failure of manual reactor shutdown by the alert from TC</td><td>(10)-1</td><td>Failures of components related to alert signal lines</td><td>9.94E-04</td></tr>
<tr><td>(10)-2</td><td>Cognitive and decision error against alert</td><td>4.26E-04</td></tr>
</table>

4. RESULTS AND DISCUSSIONS

Table 10 shows the results of ETA for FEFPA compared with those in existing PRA. Probability of damage propagation to the peripheral SAs was estimated to be 1.7×10^{-12} based on the ETA in this study. It should be noted that probability of whole core damage is extremely small because there are following other detection and reactor shutdown systems after damage propagation to the peripheral SAs:

-Manual reactor shutdown owing to the alert from primary argon monitor;

-Automatic reactor shutdown owing to high neutron flux level.

The probability of adventitious fuel pin failure in this study is much smaller than that in existing PRA in the following reasons:

-The probability of fuel pin failure used in this study was derived from pin failure experiences after 1985 in addition to those before 1985 which were used in existing PRA.

-Fuel pin failure experiences in small size reactors of which nominal full power were less than 100 MWth are excluded in this study for Monju of which nominal full power were 280 MWth;

The probability of damage propagation to the peripheral SAs was also much smaller than that in existing PRA because of the following reasons:

-The probability of adventitious fuel pin failure in this study was small.
-Various detection and reactor shutdown systems were taken into account in the ETA of this study;
-Probability of damage propagation within the SA was reduced reflecting latest experimental and analytical knowledge.

Table 10 Result of ETA for FEFPA in Monju compared with those in existing PRA

	Vaughan (CDFR)	Schleisiek (SNR 300)	JNES (Monju)	This study (Monju)
Probability of adventitious fuel pin failure [pin/ry]	35	No estimation (1)	4.4	2.4
Bulk boiling in one SA [ry]	-	6.5×10^{-6}	-	-
Fuel melting in a substantial part of the incident SA [ry]	1.9×10^{-6}	-	-	-
Damage propagation to the peripheral SA [ry]	-	1.0×10^{-7}	-	1.7×10^{-12}
Untripped States (Limited damage) [ry]	2.0×10^{-7}	-	-	-
Damage propagation more than 37 SA [ry]	-	-	5.6×10^{-8}	-
Whole core accident [ry]	1.9×10^{-9}	-	-	-

Table 11 shows the core damage fraction (CDF) of the anticipated transient without scram (ATWS) and protected loss of heat sink (PLOHS) in Monju [22].

Table 11 CDF from FEFPA compared with those from ATWS and PLOHS

	Without scram	With scram (failure of decay heat removal)
CDF from FEFPA (/ry)	$\sim9.6\times10^{-13}$	$\sim7.3\times10^{-13}$
CDF from ATWS and PLOHS (/ry)	$\sim3\times10^{-8}$ in ATWS	$\sim5\times10^{-8}$ in PLOHS

The probability of damage propagation to the peripheral SA without or with scram is smaller than CDFs of ATWS and PLOHS. Furthermore the consequence of whole core accident from adventitious fuel pin failure without or with scram is not greater than that of ATWS or PLOHS because almost all the SA will damaged at ATWS or PLOHS. Therefore FEFPA can be included in CDF of ATWS or PLOHS in the viewpoint of both probability and consequence.
It should be noted that the CDF of FEFPA in the future FBR can become larger than that in Monju because higher fuel burnups may induce more fuel swelling and denser fuel pin arrangement may induce fuel melting.

5. CONCLUSIONS

ETA from adventitious fuel pin failure in Monju was performed in this study based on latest knowledge on experimental and analytical studies for failure propagations and reflecting latest operation manual at emergency in Monju. Probability of damage propagation to the peripheral SAs was quantified to be 1.7×10^{-12} in Monju based on the ETA. Therefore probability of whole core damage is much smaller than this value because there are other detection and reactor shutdown systems after damage propagation to the peripheral SAs. It was clarified in this study that damage propagation from adventitious fuel pin failure in Monju can be included in CDF of ATWS or PLOHS in the viewpoint of both probability and consequence.

Acknowledgements
The authors are grateful to M. Sotsu, Y. Morohashi and S. Suzuki of JAEA for their useful suggestions and comments for FTA. The authors would like to express their gratitude to N. Yoshioka of NESI Inc. who assisted the calculation using FALL and SEETHE codes.

References

[1] K. Schleisiek, “Risk Oriented Analysis of Subassembly Accidents”, Proceedings of International Topical Meeting on Fast Reactor Safety, 1985 May 12-16, Guernsey (UK), pp. 141-149.

[2] G. J. Vaughan, “Event Tree Analysis of the Sub-assembly Accident”, Proceedings of International Topical Meeting on Fast Reactor Safety, 1985 May 12-16, Guernsey (UK), pp. 457-463.

[3] Japan Nuclear Energy Safety Organization, “Study on analytical method for local subassembly fault in the fast reactor”, Japan: Japan Nuclear Energy Safety Organization; JNES/SAE05-108; 2005 [in Japanese].

[4] S. Suzuki, “Summary of the JOYO Mk-III upgrade,” Japan: Japan Atomic Energy Agency; JNC TN9200 2003-003; 2003 [in Japanese].

[5] Japan Atomic Energy Commission, “Atomic energy white paper”, Japan: Japan Atomic Energy Commission; 2012 [in Japanese]

[6] H. Plitz, G. C. Crittenden, A. Languille, “Experience with failed LMR oxide fuel element performance in European fast reactors”, J.Nucl.Mat. 1993;204,238-243.

[7] SFEN, “Power Plant Safety and Fuel performance - The PHENIX Reactor: Assessment of 35 Years’ Operation-“, France: SFEN; 2009

[8] IAEA, “Status of liquid metal cooled fast reactor technology”, International Atomic Energy Agency; IAEA-TECDOC-1083; 1999

[9] IAEA, “Liquid Metal Cooled Reactors: Experience in Design and Operation”, International Atomic Energy Agency; IAEA-TECDOC-1569; 2007

[10] IAEA, “Fast reactor database 2006 update”, International Atomic Energy Agency; IAEA-TECDOC-1531; 2006

[11] Office for Nuclear Regulation, “MOX FUEL MANUFACTURE AT SELLAFIELD”, UK; Office for Nuclear Regulation; 2000 (http://www.hse.gov.uk/nuclear/mox/mox2.htm)

[12] R. B. Baker, F. E. Bard, R. D. Leggett, A. L. Pitner, “Status of fuel, blanket, and absorber testing in the fast flux test facility”, J.Nucl.Mat., 1993;204,109-118.

[13] Y. Maeda, Y. Kashimura, T. Suzuki, K. Isozaki, H. Hoshiba, R. Kitamura, T. Nakano, M. Takamatsu, T. Sekine, “Periodic Safety Review of the Experimental Fast Reactor JOYO - Review of the Activity for Safety –“, Japan: Japan Atomic Energy Agency; JNC TN9440 2005-001; 2005 [in Japanese].

[14] S. Miyakawa, “Mechanism of fuel release to coolant from breached oxide fuel in liquid metal fast reactor”, Transactions of the Atomic Energy Society of Japan, 1994;36,879-888.

[15] P. Weimar, W. Ernst, “Mol-7B - an 18-pin bundle operating 200 days beyond breach”, Nuclear Technology. 1982 April; 57:81-89.

[16] K. Haga, “Status of research on local fault and prospect to future work,” Japan: Japan Atomic Energy Agency; JNC TN2410 87-002; 1987 [in Japanese].

[17] Japan Atomic Energy Agency. “The licensing document for the construction permit of the prototype FBR Monju”, Japan: Japan Atomic Energy Agency; 1980, revised 2006 [in Japanese].

[18] R. Nakai, M. Itoh, K. Terata, Y. Kani, K. Maeda, H. Endo, S. Kondo, K. Aizawa, Y. Ohmori, “Computer codes for safety analysis of LMFBR”, Japan: Japan Atomic Energy Agency; PNC TN241 81-28; 1981 [in Japanese].

[19] Y. Fukano, J. Charpenel, “The adventitious-pin-failure study under a slow power ramp”, Proc. 12th Int. Conf. on Nuclear Engineering (ICONE12); 2004 April 25–29; Arlington (VA). [CD-ROM]

[20] Y. Fukano, “Comprehensive and consistent interpretation of local fault experiments and application to hypothetical local over-power accident in Monju”, J.Nucl.Sci.Technol., 2013;50,950-965.

[21] A. D. Swain, H. E. Guttman, “Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications”, USA: Nuclear Regulatory Commission NUREG/CR -1278; 1983.

[22] Tsuruga Head Office, Japan Atomic Energy Agency, “Summary of AM report in Monju”, 2008 (Available at http://www.jaea.go.jp/04/turuga/jturuga/press/2008/03/p080317.pdf) [in Japanese]