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Stochastic simulation of fission product activity in primary coolant due to fuel rod failures in typical PWRs under power transients

M. Javed Iqbal, Nasir M. Mirza *, Sikander M. Mirza

Department of Physics and Applied Mathematics, Pakistan Institute of Engineering and Applied Sciences (PIEAS), P.O. Nilore, Islamabad 45650, Pakistan

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Abstract

During normal operation of PWRs, routine fuel rods failures result in release of radioactive fission products (RFPs) in the primary coolant of PWRs. In this work, a stochastic model has been developed for simulation of failure time sequences and release rates for the estimation of fission product activity in primary coolant of a typical PWR under power perturbations. In the first part, a stochastic approach is developed, based on generation of fuel failure event sequences by sampling the time dependent intensity functions. Then a three-stage model based deterministic methodology of the FPCART code has been extended to include failure sequences and random release rates in a computer code FPCART-ST, which uses state-of-the-art LEOPARD and ODMUG codes as its subroutines. The value of the ^{131}I activity in primary coolant predicted by FPCART-ST code has been found in good agreement with the corresponding values measured at ANGRA-1 nuclear power plant. The predictions of FPCART-ST code with constant release option have also been found to have good agreement with corresponding experimental values for time dependent ^{135}I, ^{135}Xe and ^{89}Kr concentrations in primary coolant measured during EDITHMOX-1 experiments.

1. Introduction

The pressurized water reactors (PWRs), comprising more than two-thirds majority of operating power reactors worldwide, still maintain 10–100 times higher steam generator dose rates as compared with the competing gas- or sodium-cooled reactors. Consequently, the maintenance period gets prolonged, which not only reduces its effectiveness but also has economic repercussions estimated to be several million dollars per such reactor annually [1]. The corrosion products, coolant activation and fission product activity in the primary loops of PWRs have been identified as the dominant contributors toward the high dose rates in these systems [2–4]. The recent trend towards development of high burnup cores and extending the fuel cycle lengths of both existing and planned reactors has aggravated this problem further.

Several theoretical efforts have been made towards study of release modeling for the development of primary coolant radioactivity monitoring systems. Koo et al., have developed a model for the release of unstable fission products from defective fuel rods into the coolant of PWRs [5]. Similarly, Ivanov proposed a model for the release of fission gases out of porous fuel and has found close comparison between theoretical and experimental values of the characteristic time for flow changes [6]. Andrew et al., have used an artificial neural network based model for estimating fission product releases during severe accident conditions for both CANDU as well as LWR fuel [7]. Based on these models, computer programs were also developed for the estimation of both the number and the degree of failures in operating reactors.

Recently, the activity of ^{129}I and ^{137}Cs in the primary coolant and CVCS resin of a operating PWR have been estimated by Hwang [8] employing Lewis's model for ^{129}I in CANDU reactors [9], and gave a steady state analytic solution using a scaling factor approach. They considered the ratio of the release rate of ^{129}I and ^{137}Cs from the fuel

* Corresponding author. Tel.: +92 51 9290272/3/4; fax: +92 51 9223727.
E-mail address: nasirmm@yahoo.com (N.M. Mirza).

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matrix as a constant scaling factor of purification resin or CVCS resin. Several valuable experimental data have been reported for fuel failures at ANGRA-1, nuclear power plant [10]. This reactor had to be prematurely shut down in cycle 4 due to high ^{131}I activity in the primary coolant where, inspections revealed failures of one sixth of the total fuel assemblies. Similarly, Hartwell et al., have designed an on line, multi-spectrometer fission product monitoring system to support advanced gas reactor fuel testing and qualification in the advanced test reactor and presented their measured and modeled total activity results for the first TRISO-coated fuel particle failure during the NRA-1A experiment [11].

When fission occurs, part of the fission products released in the fuel diffuse to the fuel-clad gap through the complex network of tunnels in solid fuel matrix. These radioactive fission products remain trapped in the fuel-clad gap until the clad puncture occurs resulting in their leakage to the primary coolant. The removal of these isotopes from primary coolant can occur through multiple paths as shown in Fig. 1. In the out-of-core region, they are removed by ion-exchangers and CVC Systems (Fig. 1). Radioactive decay, leakages and filters are also other available paths for removing them from the primary circuit. Inside the core region, these radionuclides are removed by neutron activation and natural radioactive decay.

In our previous work [12], kinetic simulations of fission product activity in primary circuits of a typical PWR under power transients, were performed. A detailed two-stage model based methodology was developed and implemented in a computer coder FPCART. For normal constant power operation and averaged out release rate of fission fragments from fuel rod to coolant, results for over thirty-nine fission products show that the activity due to fission products in fuel region of PWRs is dominated by ^{134}I which is followed by ^{134}Te and ^{133}I. Although, the value of ^{134}I in the fuel region leads other isotopes, the most important of them all is ^{131}I, hence its activity values has been given due importance in these simulations. The value of the fission products activity in the fuel region predicted by FPCART code has been found to agree very well with the corresponding values found by using the ORIGEN-2.0 code [13]. The predictions of FPCART code have also been found in good agreement with the corresponding values found in ANS-18.1 Standard as well as with some available power-plant operation data with 2.4% deviation in the value of specific activity of the dominating fission product ^{134}I, and in case of ^{131}I it is 6.4% off as compared to standard and falls within operational data limits.

Owing to great number of fuel rods in the core of an operating PWR and large number of contributing parameters/causes of rod failure, many of them associated with

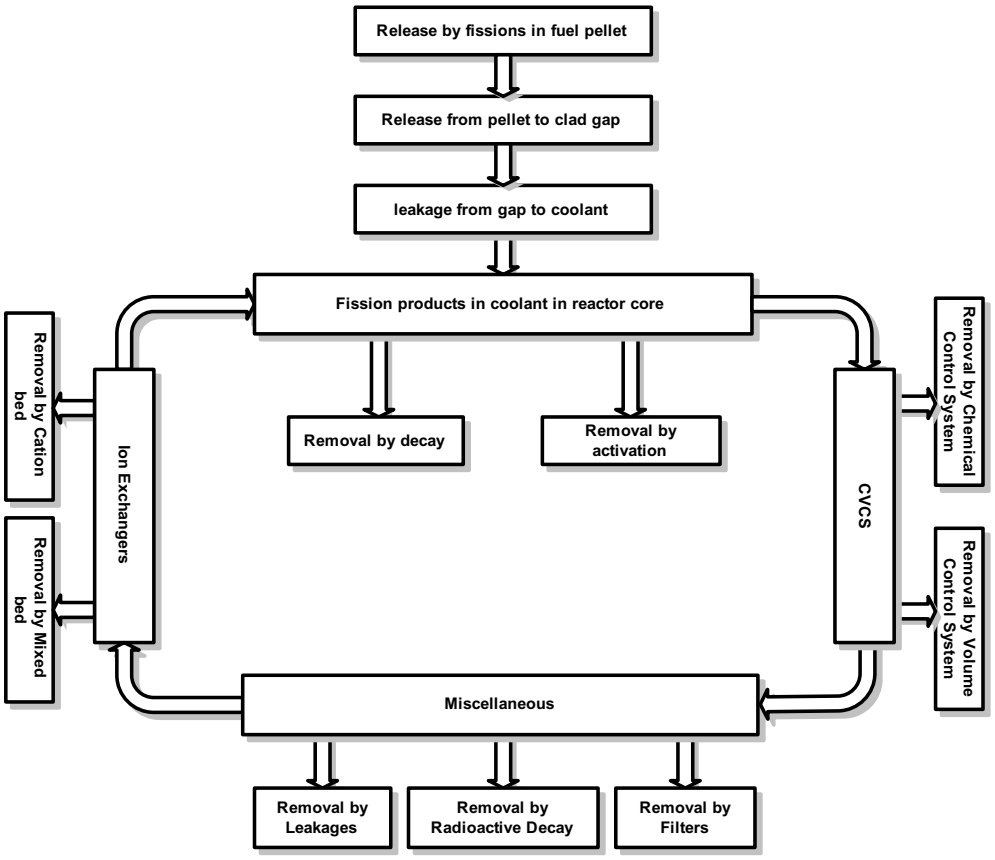

Fig. 1. Schematic diagram of production and loss cycle of fission product activity in primary coolant circuit of PWRs.

wide uncertainty and have truly random variability limits stochastic methods are well applicable. The fuel failure occurrence times are not predefined; this property separates such sequence of events from a time series events. Due to random occurrence of such events, stochastic simulation technique has been chosen to model fuel failure event sequences. In this work, a stochastic model has been developed which extends the computer program FPCART for the estimation of fission product inventory in primary coolant for realistic power perturbations. In the first part a stochastic approach is developed after generation of the fuel rod failure event sequences by sampling the time dependent intensity. In the second part, the FPCART deterministic method has been extended to incorporate the random fuel failures to calculate fission product activity in the fuel region and in the fuel-clad gap region. This method is based on Runge–Kutta–Fehlberg approach with adaptive step sizing to solve coupled ordinary differential equations. Then finally fission product activity in the primary coolant region is determined. The method also simulates the behavior of fission product activity under power perturbations in a typical PWR.

The details of this methodology along with its computer implementation are presented in the following sections. The mathematical details of the governing differential equations are given in Section 2. The numerical scheme used for computer simulations is described in Section 3 and finally the results of these simulations are presented.

2. Mathematical model

In this work, a typical 300 MWe PWR has been considered. The design specifications of such a system are briefly listed in Table 1. The time sequence of fuel rod failures and fraction of fuel rods failed is generated stochastically. If t_j is a random time for release of fraction $\varepsilon(t_j)$ and it obeys a probability distribution $g(t)$, then the intensity function is

$$\lambda(t) = \frac{g(t)}{G(t)}, \tag{1}$$

where $G(t)$ is the cumulative distribution given as

$$G(t) = \exp\left(-\int_0^t \lambda(s)\mathrm{d}s\right). \tag{2}$$

In, this work, the probability of accepting a move of fuel failure at t_k after t_j is calculated using Monte Carlo method based rejection technique [13]. In this method, a random ratio, q is generated as

$$q = \frac{g(t_k)}{g(t_j)}. \tag{3}$$

If $q < 1$, then a random number (η) based on probability distribution $g(t)$ is generated and if $q > \eta$, then time t_k is accepted as failure event time [14]. Otherwise, the procedure is repeated.

The fission product inventory in the fuel has been modeled by extending the standard equations [11]. The rate of change of number of ith radionuclide in fuel region for the whole core, ($N_{F,i}$), is

$$\frac{\mathrm{d}N_{F,i}}{\mathrm{d}t} = FY_iP + \sum_{j=1}^{i-1} f_{ij}\lambda_j N_{F,j} - (\lambda_i + v_i + \sigma_i\phi)N_{F,i}; \quad i = 1,2,\ldots,4, \tag{4}$$

where

$F \equiv$ average fission rate (fissions/W s),
$Y_i \equiv$ fission yield of the ith radionuclide,
$P \equiv$ rReactor core thermal power (W),
$\lambda_i \equiv$ decay constant of the ith radionuclide (s^{-1}),
$v_i \equiv$ first-order escape rate coefficient for the ith radionuclide (s^{-1}),
$\sigma_i \equiv$ microscopic absorption cross section for the ith radionuclide (cm^2).

The rate of change of ith radionuclide in the gap region for the entire core, ($N_{G,i}$), is

$$\frac{\mathrm{d}N_{G,i}}{\mathrm{d}t} = v_iN_{F,i} + \sum_{j=1}^{i-1} f_{ij}\lambda_j N_{G,j} - (\lambda_i + \varepsilon_i + \sigma_i\phi)N_{G,i}; \quad i = 1,2,\ldots,4, \tag{5}$$

where $f_{ij} \equiv \mathrm{I} \to j$ Branching ratio and $\phi \equiv$ neutron flux (neutrons per cm^2 per s).

The rate of change of ith radionuclide in the coolant region ($N_{C,i}$) is

$$\frac{\mathrm{d}N_{C,i}}{\mathrm{d}t} = \varepsilon_iN_{G,i} + \sum_{j=1}^{i-1} f_{ij}\lambda_j N_{C,j} - \left(\lambda_i + \frac{Q}{W}\eta_i + \beta + \tau\sigma_i\phi + \frac{L}{W}\right)N_{C,i}; \quad i = 1,2,\ldots,4, \tag{6}$$

Table 1
The design parameters of a typical PWR [16]

Parameter	Value
Specific power (MW_{th}/kg U)	33
Power density (MW_{th}/m^3)	102
Core height (m)	4.17
Core diameter (m)	3.37
No. of fuel assemblies	194
Rods per assembly	264
Fuel type	UO_2
Clad type	Zircoloy
Lattice pitch (mm)	12.6
Fuel rod outer diameter (mm)	9.5
Average enrichment (w/o)	3.0
Flow rate (Mg/s)	18.3
Linear heat rating (kW/m)	17.5
Coolant pressure (MPa)	15.5
Inlet coolant temperature (°C)	293
Outlet coolant temperature (°C)	329

where

$\varepsilon_I \equiv$ first-order release rate coefficient from fuel clad gap to reactor coolant for i'th type of radionuclide atoms (s^{-1}),
$\tau \equiv$ (primary coolant residence time in core)/(total primary circuit time),
$\beta \equiv$ first-order cumulative removal rate coefficient for ith type of radionuclide by boron removal system (BRS) (s^{-1}),
$L \equiv$ coolant leakage rate (g/s),
$W \equiv$ total coolant mass (g),
$\eta_i \equiv$ resin purification efficiency for ith radionuclide,
$Q \equiv$ let-down flow rate (g/s).

The value of 'i' refers to the position of isotope in the chain considered. The values of parameters used are listed as Table 2. If the ε_0 is initial burst release rate from a punctured fuel rod, ξ is decay constant for the escape rate and t_0 is the time of initiation of rod puncturing, then the escape rate coefficient from gap to coolant, ε is given as

$$\varepsilon = D_0\varepsilon_0 e^{-\xi(t-t_0)} + D\varepsilon_0, \quad (7)$$

where D is the number of failed fuel rods and D_0 also refers to a rod failed but at a time has value numerically equal to one, i.e., the first term of this equation is relevant to the rod that just fails, and after sufficient time is passed the burst effect settles down to the steady state value expressed as second term of Eq. (7), and first term will again contribute when next rod failure takes place, an so on. The values of ε_I, ξ and$D$, used in simulations are shown in Table 3,

Table 3
Values of various operational parameters used in these simulations

Parameter	Value
L (g/s)	2.3
Q (g/s)	3000.0
β (s^{-1})	1.0×10^{-5}
W (g)	1.072×10^{9}
V (cm^3)	1.485×10^{9}
τ	0.056
P_0 (MW_{th})	998
F (fission/W s)	3.03×10^{10}
D	60
ε_0 (s^{-1})	1.0×10^{-8}
ξ (s^{-1})	7.2×10^{-5}

Table 2
Data for various fission product gases and associated decay products

Isotope (location in Chain)	$T_{1/2}$[a]	η	Y[b] (atoms/100-fissions)	v (s^{-1})	σ (b)[b]	Branching ratios, f_{ij}				
						$1\to2$	$1\to3$	$2\to3$	$2\to4$	$3\to4$
Kr-85M(1,I)	4.48 h	0.4538	1.3E−2	6.5E−8	0.0	1.0	0.0			
Kr-85(2,J)	10.752 y	3.958E−5	1.3E−2	6.5E−8	1.84E−1			1.0		
Kr-87(3,K)	1.272 h	0.7454	2.56E−2	6.5E−8	5.518E+1	1.0				
Kr-88(1,I)	2.84 h	0.5674	3.55E−2	6.5E−8	0.0	1.0	0.0			
Kr-89(1,I)	3.15 m	0.986	4.63E−2	6.5E−8	0.0	1.0	0.0			
Rb-89(2,J)	15.4 m	0.9	2.05E−3	1.3E−8	0.0			1.0	0.0	
Sr-89(3,K)	50.7 d	0.986	1.75E−4	1.0E−11	5.264E−2					1.0
Sr-90(1,I)	28.79 y	0.9	7.58E−2	1.0E−11	8.739E−2	1.0	0.0			
Y-90(2,J)	2.671 d	0.9	8.97E−8	1.6E−12	5.93E−1			1.0	0.0	
Sb-129(1,I)	4.36 h	0.9	5.43E−3	1.0E−11	0.0	0.166	0.834			
Te-129M(2,J)	33.6 d	0.9	1.4E−7	1.0E−9	2.90E−1			0.63		
Te-129(3,K)	1.16 h	0.9	5.11E−3	1.0E−9	0.0				0.37	
I-129(4,L)	1.61E7 y	0.99	5.0E−6	1.3E−8	5.225					1.0
Te-131M(1,I)	1.25 d	0.9	1.1E−2	1.0E−9	0.0	0.21	0.79			
Te-131(2,J)	25.0 m	0.9	1.71E−2	1.0E−9	0.0			1.0	0.0	
I-131(3,K)	8.023 d	0.99	3.29E−5	1.3E−8	3.229E−1					0.0109
Xe-131M(4,L)	11.93 d	8.923E−3	4.05E−4	6.5E−8	0.0					
Sb-132M(1,I)	4.1 m	0.9	1.07E−2	1.0E−11	0.0	0.0	1.0			
Sb-132(2,J)	2.8 m	0.9	1.67E−2	1.0E−11	0.0			1.0	0.0	
Te-132(3,K)	3.204 d	0.9	1.54E−2	1.0E−9	4.89E−4					1.0
I-132(4,K)	2.28 h	0.99	2.06E−4	1.3E−8	0.0					
Te-133M	55.4 m	0.9	3.06E−2	1.0E−9	0.0	1.0	0.0			
I-133	20.8 h	0.99	4.9E−2	1.3E−8	0.0			0.0285	0.9715	
Xe-133M	2.19 d	4.663E−2	2.67E−5	6.5E−8	0.0					1.0
Xe-133	5.244 d	2.001E−2	4.9E−02	6.5E−8	2.44E+1					
Te-134	41.8 m	0.9	6.97E−2	1.0E−9		1.0	0.0			
I-134	52.8 m	0.99	7.83E−2	1.3E−8				1.0	0.0	
Te-135(1,I)	19.0 s	0.9	3.11E−2	1.0E−9	0.0	1.0	0.0			
I-135(2,J)	6.57 h	0.9	2.98E−2	1.3E−8	2.119E−3			0.1651	0.8349	
XE-135M(3,K)	15.29 m	0.9098	1.10E−2	6.5E−8	0.0					0.994
XE-135(4,L)	9.14 h	0.2205	6.54E−2	6.5E−8	2.445E+5					

[a] Ref. [13].
[b] Ref. [24].

respectively. For the case of a constant release from fuel failure, the release fraction has been estimated using following:

$$\varepsilon = D\varepsilon_0. \tag{8}$$

3. Calculation methodology

Based on the coupled stochastic and deterministic methods, the computer program FPCART [12] has been extended and implemented in program FPCART-ST (fission-product primary coolant activity in reactor transients-stochastic) in this work. This program is written in Fortran-77 for personal computers and its flow chart is shown as Fig. 2. As a first step, it generates the failed fuel rod event sequence time values and the failed fuel fraction stochastically by sampling employing Weibull-distribution [14] or uniform distribution, depending on user's selection. The acceptance of failure state or rejection is based on algorithm described by Eqs. (1)–(3).

In second step, for each time value of sequence, the FPCART-ST computes group constants employing the LEOPRAD code [15] used as a subroutine. The design data values for a typical PWR as given in Table 1 [16] are employed. The LEOPRAD program is a zero-dimensional unit cell code that employs 54 fast and 172 thermal groups. It uses evaluated nuclear data file (ENDF-IV) as cross section library. The cell-averaged group constants are then passed on to the ODMUG code [17] which is also used as a subroutine in the FPCART-ST program (Fig. 2). The ODMUG program solves one-dimensional multi-group diffusion equation and computes spatial profile of group fluxes in the reactor core. Subsequently, these group fluxes are averaged over the core region and the resulting values are used in the transient governing equations for fission product activity within fuel rod, fuel-clad gap and finally in the primary coolant.

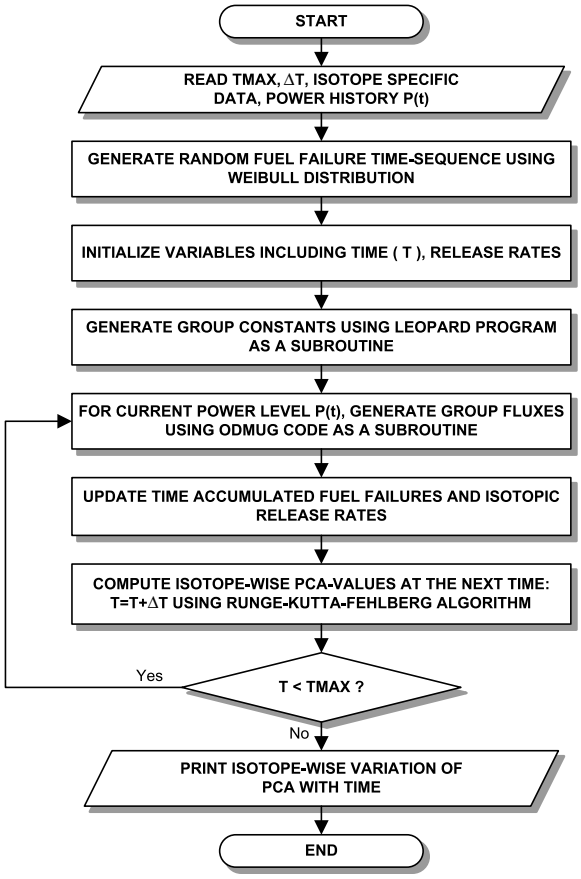

Fig. 2. Flow chart of program FPCART-ST.

The FPCART-ST program then solves the coupled system of ordinary differential equations (Eqs. (4)–(6)) using the fifth-order Runge–Kutta–Fehlberg adaptive method to find the activity values due to various fission products/progeny in the fuel, fuel-clad gap and finally in the primary coolant and gives output at the end of the time interval, over which a loop runs, and the length of this time step is user defined. In this program two options have been provided, namely, burst release and a constant release from a single rod. In the burst release option, the release fraction, ε, have been modeled by an exponentially decaying function as shown in Eq. (7). However, in the second option a constant release from a single rod has been assumed (Eq. 8). The calculations can be performed for steady state as well as for various power transients and for all major isotope chains.

4. Simulation results

In previous studies [12], simulations were done for all major fission products and their progeny covering all major isotopes that contribute significantly to radiological consequences. Thirty-nine different fission products and their progeny have been considered in this study encompassing all dominant isotopes from dose/exposure and radiological consequences perspective. These include various isotopes of iodine, cesium, strontium, xenon, krypton, etc. During normal full power operation of nuclear reactor, these isotopes buildup to their saturation values at various times depending on their characteristic half-lives. The saturation value of the quantity of these radioisotopes depends on the flux levels, the corresponding values of half-lives and their fission yield (Table 4 [18–20]). The saturation values were found in good agreement for various reactors, as shown in Table 4, with reported experimental results at given power levels, purification efficiency and leakages out of the system.

In this work, the Angra-1 Nuclear Power Plant has been considered first. It is a Westinghouse designed, (657 MWe) PWR that was prematurely shut down in cycle 4 due to high activity in the reactor coolant system. Inspections indicated failures in one-sixth of fuel assemblies. The grid-to-rod fretting failures were caused by grid spring force losses [21]. Simulations for the reactor using FPCART-ST code have been carried with 100 h of reactor operation at full power. Then ten punctured fuel rods were introduced to create the same initial fission product inven-

Table 4
Specific activity of various isotopes in primary coolant

Sr. no.	Isotope	Escape rate coefficient from gap to coolant, ε (s^{-1})	This work (μ Ci/g)	Standard ANS-18.1 (μ Ci/g)[a]	Measured (micro-curies/g) of coolant			
					Surry Unit-1[b]	Turkey Point Unit-3[c]	Turkey Point Unit-4[c]	H.B. Robinson unit-2[c]
1	Kr-85M	0.005	1.62E−1	1.6E−1		1.8E−2	1.7E−2	7.94E−3
2	Kr-85	0.005	4.58E−2	4.3E−1	2.01E−1		<8.0E−4	
3	Kr-87	1.0E−3	1.49E−1	1.5E−1	7.86E−4	3.4E−2	2.5E−2	6.74E−3
4	Kr-88	4.5E−3	2.74E−1	2.8E−1		4.7E−2	3.8E−2	9.45E−3
5	Sr-89	2.5E−6	2.27E−5			1.9E−5	8.9E−5	
6	Sr-90	1.0E−7	2.65E−7			2.1E−7	3.3E−7	
7	Te-129M	3.0E−6	2.43E−5			2.7E−5	<3.0E−5	
8	Te-129	2.5E−3	1.10E−2			1.0E−2	<1.0E−2	
9	Te-131M	4.0E−5	1.50E−4			1.5E−4	<2.0E−4	
10	Te-132	7.0E−7	2.93E−5			2.5E−5	3.0E−5	
11	I-131	6.0E−5	4.21E−2	4.5E−2	8.26E−4	1.1E−2	8.0E−3	2.07E−3
12	I-132	4.5E−3	2.08E−1	2.1E−1		1.4E−1	1.9E−2	1.49E−2
13	I-133	7.0E−4	1.50E−1	1.4E−1	4.89E−3	8.8E−2	1.7E−2	8.40E−3
14	I-134	1.5E−2	3.32E−1	3.4E−1	1.17E−3	2.6E−1	2.2E−2	3.03E−2
15	I-135	5.0E−3	2.12E−1	2.6E−1	5.20E−3	1.5E−1	1.9E−2	2.14E−2
16	Xe-131M	0.8	3.94E−2	7.3E−1		<2.0E−3	3.0E−4	
17	Xe-133M	1.5E−3	7.02E−2	7.0E−2		4.3E−3	6.6E−3	3.28E−3
18	Xe-133	3.4E−4	2.60	2.6	1.42E−2	2.0E−1	2.1E−1	1.75E−1
19	Xe-135M	2.0E−2	1.25E−1	1.3E−1		1.1E−1	2.1E−2	
20	Xe-135	3.5E−3	8.20E−1	8.5E−1		1.5E−1	1.2E−1	3.82E−2

[a] ANSI/ANS-18.1-1984, Radioactive source term for normal operation of light water reactors.
[b] WCAP-8253, May 1974, Source term data for westinghouse PWRs.
[c] NUREG/CR-1992, US NRC, Aug 81, In plant source term measurements at four PWRs.

tory in the coolant as was available at the start of fourth cycle in Angra-1 nuclear power plant [21]. The build-up of fission product inventory in fuel depends directly on power level of the plant and for a given power pattern, as shown in Fig. 3, simulations were done using ANGRA-1 nuclear power plant data for the fourth cycle. The computer codes LEOPARD and ODMUG were used as a subroutine in the FPCART-ST code to compute core averaged group fluxes. Typical values of failed fuel fraction, coolant leakage rate, and purification and removal rates as given in Table 2 were employed. The resulting coolant activity for ^{131}I was compared with the measured values during first 22 days of its operation in cycle-4. The results are shown in Fig. 3. The coolant specific activity shows a good agreement with the measured data. When power levels approach to 70% of full power, the overall

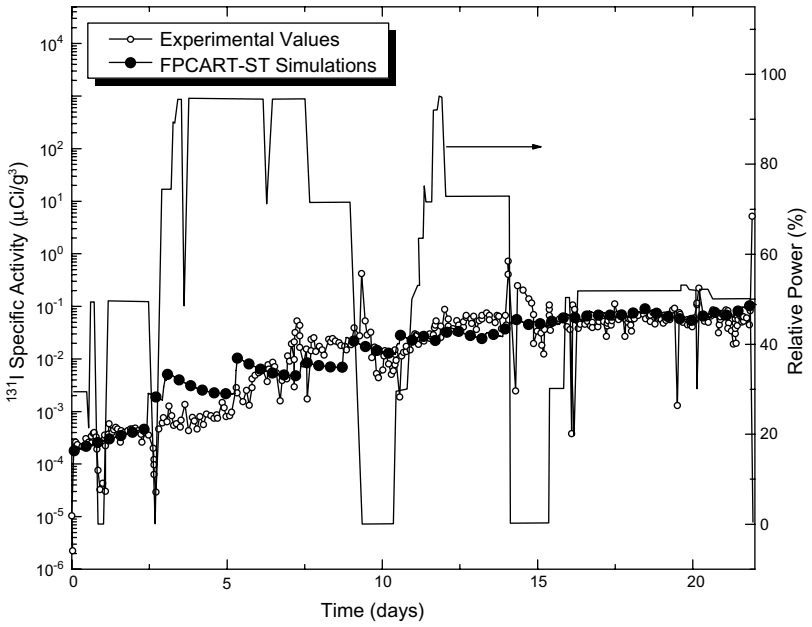

Fig. 3. Comparison of calculated ^{131}I activity in primary coolant with the experimentally measured values at ANGRA-1, Nuclear power plant. The irradiation history for the reactor as relative power is also shown [21].

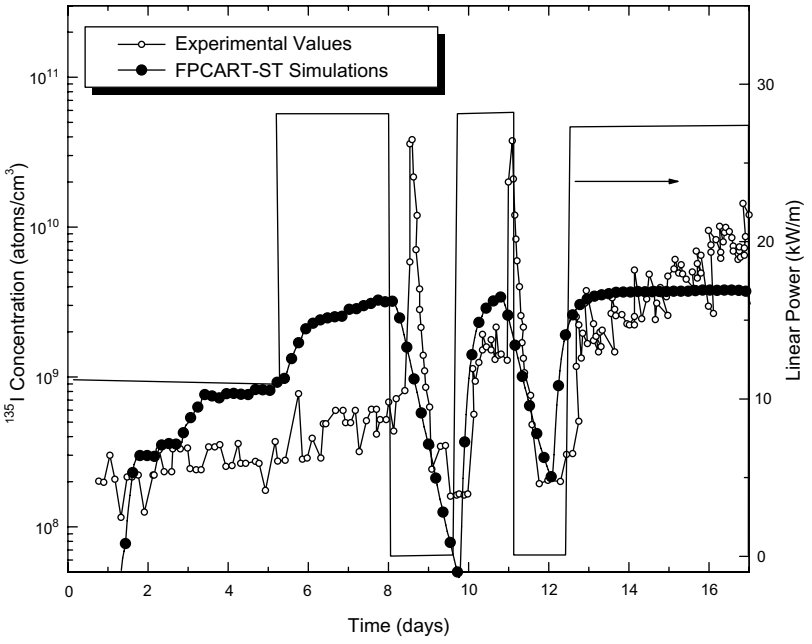

Fig. 4. Comparison of computed values of ^{131}I concentration in primary coolant along with EDITHMOX-1 results [22,23]. The time dependent linear power (kW/m) is also shown.

behavior was successfully simulated. The measure and predicted values of specific activity remained with in 15% of each other as shown in Fig. 3. However, spikes observed in the experimental data remain smeared in prediction of FPCART-ST program.

In second part of the comparison, the Siloe research reactor, France's wide experimental program on irradiation of defective fuel rods was considered. The detailed descriptions of the EDITHMOX experiments are given in OECD/NEA/IAEA IFPE database [22,23]. It was performed in Jet Pompe water loop with long periods of steady state irradiation conditions. These experiments created typical conditions for failed fuel rods of typical PWR under power transients. Release rates of various radioactive nuclides were studied using mixed oxide fuel (7.8% Pu isotopes) under various heat generation rates. These experiments were simulated using FPCART-ST code. The calculations assumed a constant flow rate during power

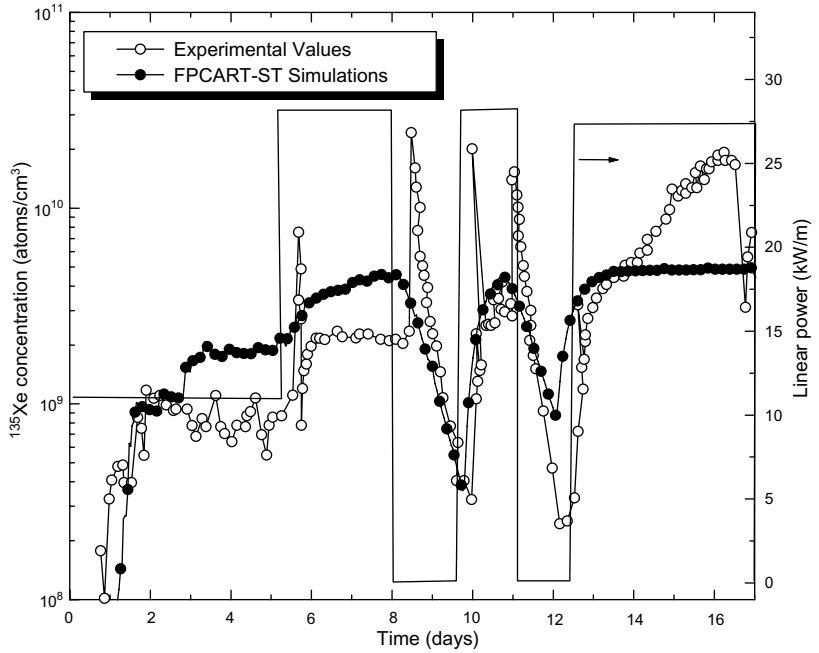

Fig. 5. Simulated values of ^{135}Xe concentration in primary coolant versus experimental value from EDITHMOX-1 experiments [22,23]. The time dependent linear power (kW/m) is also shown.

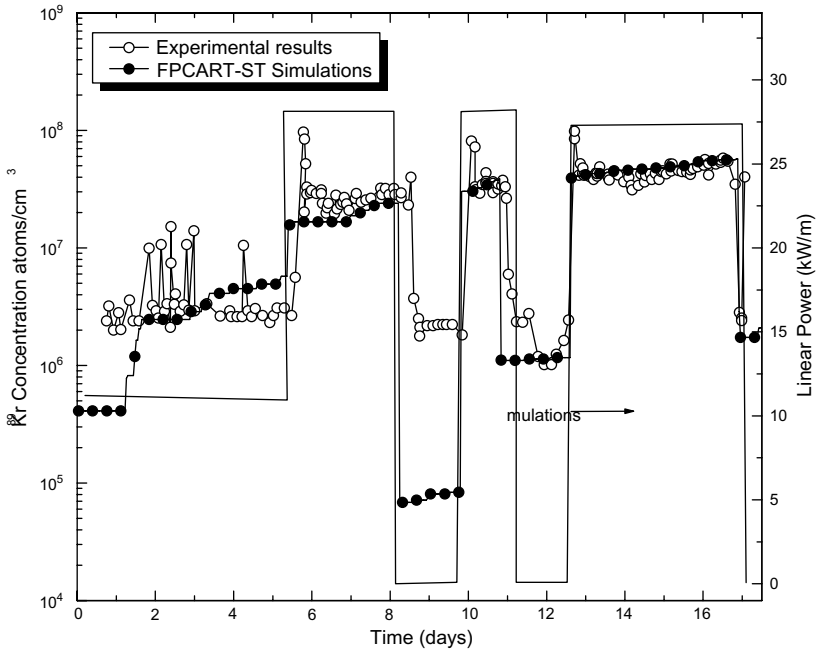

Fig. 6. Comparison of computed values of ^{89}Kr concentration in primary coolant with EDITHMOX-1 experimental results [22,23]. The time dependent linear power (kW/m) is also shown.

transients. Simulation results for ^{135}I and corresponding measured values at Jet Pompe water loop in Siloe reactor are shown in Fig. 4 at various power levels. These results show an overall good agreement with the experimental values. The stochastic model in FPCART-ST seems to explain qualitatively the statistical fluctuations found experimental kinetic data for fission fragment release rate in primary coolant. However, there are two spikes observed experimentally just after the power is reduced to almost zero level. This spiking behavior seems to be linked with power and related feed back to the flow rate that has not been modeled in the FPCART-ST program.

Similarly, FPCART-ST calculations and experimentally measured values for ^{135}Xe concentration in primary coolant for EDITHMOX-1 were also done. Fig. 5 compares the measured values with the predictions of the FPCART-ST program. The figure indicates a close agreement of the predictions with the corresponding experimental data during constant power periods. Here fixed release rate model was adopted along with stochastic time sequence for the fuel rod failures. Finally, the calculational results for ^{89}Kr (relatively short lived) in primary coolant for Jet Pompe water loop were compared with experimentally values as shown in Fig. 6. These results show a good agreement in both values during most part of experimental power changes.

5. Conclusions

- The stochastic failure of fuel rods and consequent release of fission product activity from fuel-clad gap to primary coolant as a burst release and as a constant release rate for PWRs has been studied in this work. A two-stage methodology has been developed for this purpose, and implemented in a Fortran-77 based computer program FPCART-ST.
- Results for kinetics of ^{131}I in ANGRA-1 nuclear power plant predicted by FPCART-ST, with burst release option give a good match with the respective measured values.
- For experimental EDITHMOX-1 studies, using its time dependent heat generation rate, the kinetic predictions of FPCART-ST code with fixed release option were found in good agreement with measured values of ^{135}I, ^{135}Xe, and ^{89}Kr.
- The simulations show strong dependence of primary coolant activity on prevailing power level and removal rates. The primary coolant specific activity generally shows approach towards a higher saturation level at higher power and goes down as power decreases depending on the half life and purification efficiency of system for the particular isotope. It also depends on failed fuel fraction and measured activity can be related to the number of failed fuel rods.
- In these simulations a core averaged value of neutron flux was applied. Similarly, an average defect size in each failed fuel rod was considered.

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