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Modeling and Simulation of Release of Radiation in Flow Blockage Accident for Two Loops PWR

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Abstract: In this study modeling and simulation of release of radiation form two loops PWR has been carried out for flow blockage accident. For this purpose, a MATLAB based program "Source Term Evaluator for Flow Blockage Accident" (STEFBA) has been developed, which uses the core inventory as its primary input. The TMI-2 reactor is considered as the reference plant for this study. For 1100 reactor operation days, the core inventory has been evaluated under the core design constrains at average reactor power. With the instantaneous release of radiation to the containment atmosphere, some reliable results have been achieved for determined states of containment, i.e., Normal, Emergency and Isolation state in this accident situation. Resulting noble gasses, iodine and aerosol source term variation has been evaluated against the containment performance. The dependency of the source term on containment performance parameters has been studied. Typical trend of release of radioactivity has been found corresponding to increase in ventilation exhaust rate values. The dependency of radioisotopes on ventilation exhaust rate has been studied in accident situation. The removal dependency of radio iodine and aerosol has also been studied with boric and caustic acid spray in this study.

Keywords: Flow blockage accident, iodine, noble gasses, PWR, source term, TMI-2

INTRODUCTION

The pursuit of generation of electricity from the nuclear fuel began in 20th century soon after the development of first reactor. Now 435 nuclear power plants are operating in 30 countries worldwide generating net electricity of 368,267 MW (ENS, 2010) and 60 commercial nuclear power plants are under construction expecting to generate net electricity 61,032 MW (WNA, 2012). The safety and reliability of nuclear power plants are good when compare to the other energy resources. The three major nuclear accidents, including Three-Mile Island accident (1979), Chernobyl disaster (1986) and Fukushima Daiichi nuclear disaster (2011) has happened still now. However, Research into safety and reliability improvements is continuing.

The Pressurized Water Reactor (PWR) is the most common type of reactor built in various countries. Since the advance PWR reactors have more safety margins as compare to the conventional PWR. However, there remains the threat of the accident. Radiological consequences and source term remains the major threat from NPPs to the public which influence the design of NPPs (Rosen *et al*., 1979).

Three-Mile Island Accident initiates research on source term evaluation and severe accident management. U.S.NRC in-cooperation with different research agencies to pursue research on source term evaluation and sever e accident management, which resulted in development of computer codes (Mehboob, 2012). For the verification and codes and to understand the fission product behaviour numerous in pile and out of pile, experiments have been conducted in various countries (Mehboob *et al*., 2011).

Besides the U.S.NRC progress in source term evaluation and development of severe accident and Source Term Code Packages (STCPs), Researcher used these models and codes with different techniques to calculate the source term. Lee and Yu-Chih (2008) have studied the source term of Westinghouse 3-loop PWR with MAAP 4.0.4 code. Gera jee *et al*. (2010) has estimated the source term of Indian Pressurized Heavy Water Reactor (IPHWR). He adopted multi physics methodology using ORIGEN 2 for core inventory, SCDAP/RELAP 5 for thermal hydrolysis, ASTAC for FP transpiration, CONTAIN for containment source term and COSYMA for dose calculations. This methodology is applied for Reactor Inlet Heater (RIH) and Reactor Outlet Heater (ROH) break. Nasir *et al*. (2010) have evaluated

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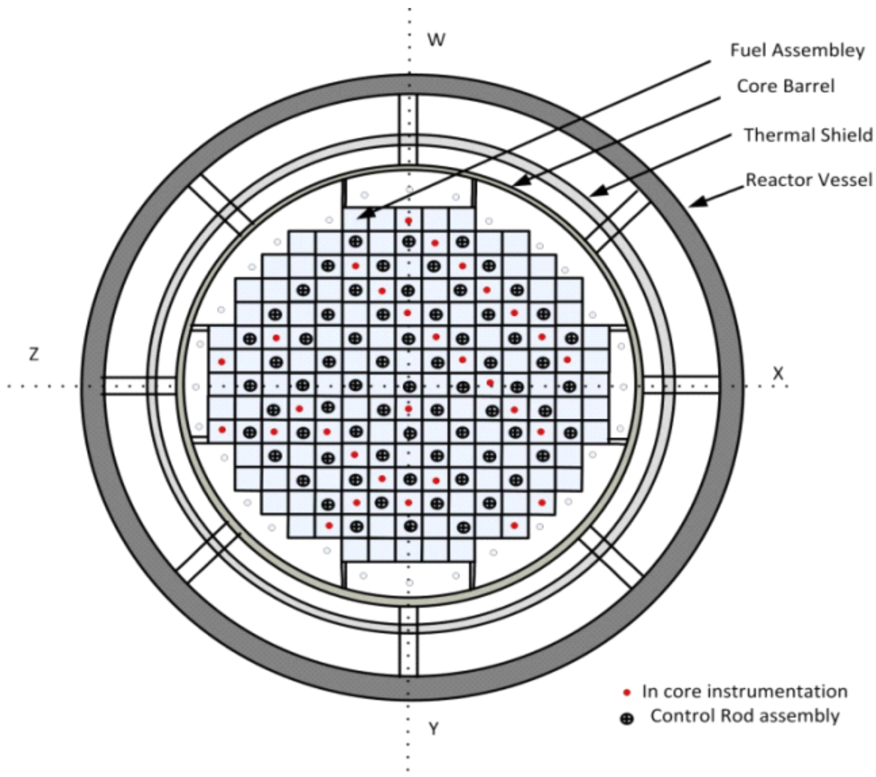


Fig. 1: Cross section view of TMI 2 core design

the source term for Pakistan Research Reactor 1(PARR-1) using the coupled code technique. Recently (Mehboob *et al*., 2012) have evaluated and verified the iodine sauce term for Three Mile Island reactor accident with self-made computer code.

In this study, a MATLAB based code (STEFBA) has been developed for the source term evaluation of 900 MWe power rector. The source term has been evaluated for flow blockage accident assuming the instantaneous release of radiation into containment atmosphere. The dependency of radioisotopes on containment safety parameters has also studied in this work.

Description of reactor: In this study Three-Mile Island unit two reactor is considered for this case study. The TMI-2 reactor was conventional PWR with 905 MWe power and 2720 MWt thermal power. The reactor core was designed with 177 fuel assemblies in 15×15 matrices with 69 control rod assemblies. The core power was controlled with 1104 Ag-In-Cd (Ag 80%, In 15%, Cd 5%) control rods. The primary loop has two Once Through Stream Generators (OTSG) and four coolant pumps. Henry (2007). The cross section view of TMI-2 core is shown in Fig. 1.

The Three-Mile Island unit 2 containment is large dry containment “can design” designed by Babcock and Wilcox Company (B&W). The containment free volume is 57600 m^3; the free surface area is almost 34374 m^2 (Henries and Postoma, 1987) and dome height is 46.7 m having a capacity to sustain 379 kPa (55 psi) pressure (Henries and Postoma, 1987; Thandani, 1993). The Containment Filtered Venting System (CFVS) normal exhaust rate is 50970 m^3/h while the emergency ventilation is 1699 m^3/h (FSAR, 2012; Henries and Postoma, 1983). The large dry containment is leak tight design with the leakage of 1% of the containment free volume per day (24 m^3/h). High Efficiency Particulate Air filters (HEPA) in conjunction with activated charcoal filter and heaters are employed in the ventilation and containment purge system. The HEPA filters have the 99.9% efficiency to stop the aerosols size 0.3 μm where the activated charcoal has 90.0% efficiency for elemental iodine. The containment schematic design is shown in Fig. 2.

Mathematical modeling: The release of radiation depends upon the accident scenario and accident type core inventory and feeding and breeding in coolant. The core inventory depends upon the operation schedule, cover

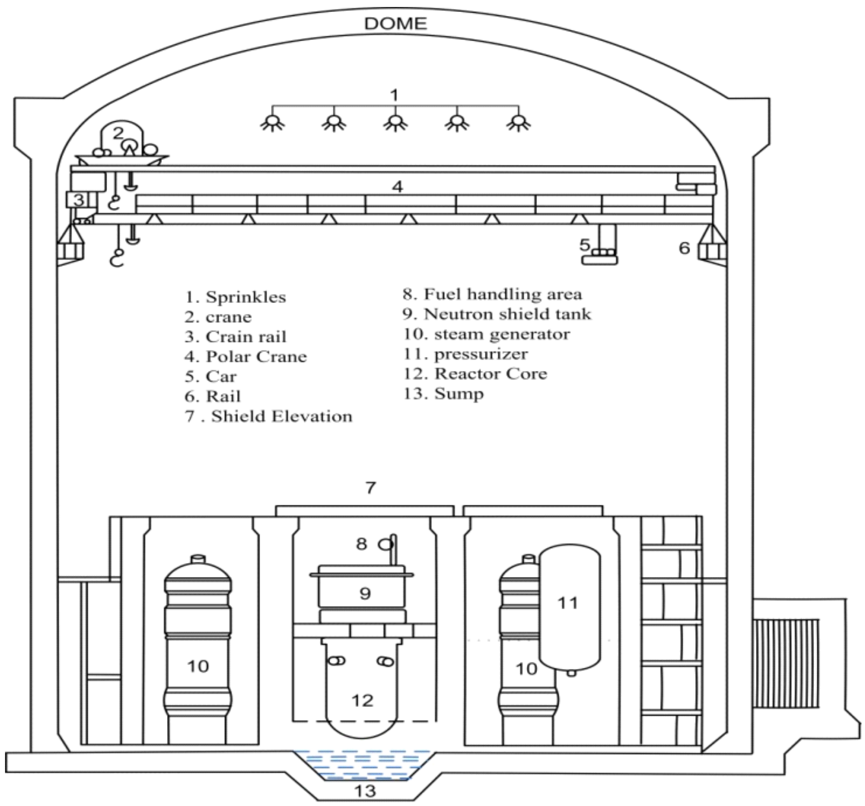


Fig. 2: Schematic diagram of TMI2 containment design

level, refueling time period and fissile contents in the core. The reliable estimation of release of radiation in sight (inside the containment building) and outside the reactor (to the general public) is required (IAEA, 2008). The release of radiations depends upon the number of factors, including core damage, core inventory, accident scenario and, etc. The physical and chemical forms may also influence the release of radiations.

Fission product and core inventory: The fission product generation in the fuel pallets is over the flux Φ is represented in the Eq. (1) (Weston, 2000). The generation of fission products in fuel pallets depends upon the flux profile and fission rate in the core:

$$\frac{dn_j}{dt} = \gamma_j \sum\nolimits_f \phi + \sum_i \left(\lambda^{i\to j} + \sigma^{i\to j}\phi\right) n_i - \left(\lambda^j + \sigma_a^j \phi\right) n_j \quad (1)$$

The first term on the right-hand side is the fission rate in the core at flux Φ; 2nd term is the transmutation of i^{th} isotopes into the j^{th} isotopes and last term is the loss of j^{th} isotopes due to decay and absorption.

where γ_j is the fraction of fission events for j^{th} isotope; Φ is flux (neutrons/cm^2-s). $\sum_f$ ~macroscopic fission cross-section (cm^{-1}); λ^{i-j} decay of i^{th} isotopes to j^{th} isotope (s^{-1}); σ^{i-j} transmutation cross section for i^{th} isotope to j^{th} isotope (cm^{-2}).

Coolant retention and core damage calculation: The coolant retention is the dominant process in case of release of radiation. The absorption of radiations by coolant and plate out are the dominant key phenomenon in case of radiation release to coolant. The core damage propagation depends upon the accident scenario and reactor type. Experience from the NUREG 0722 (NRC, 1981) and WASH 1400 (NRC, 1975). The probability for flow blockage accident for TMI2 reactor is from 4×10^{-5} to 6×10^{-6} per year. For this accident 5% core damage is assumed.

Containment modeling: The last barrier for fission product to escape into environment is containment. If the containment has the large leakage, then there is a probability for all the radioisotopes to release into the environment. The escape of radiation to the environment depends upon the Containment Filtered Ventilation System (CFVS) exhaust rate, leakage rate, deposition, re-suspension, filtration, recirculation and removal via spray. Keeping in view of these factors the containment model is given in Eq. (2) (IAEA, 2008):

$$\frac{dc_i(t)}{dt} = -\lambda c_i(t) - u_t \frac{A}{V} c_i(t) - \frac{RE}{V} c_i(t) - \frac{L_r}{V} c_i(t) + r\frac{A}{V} c_{ri}(t) + P(t) \tag{2}$$

$$\frac{dc_{ri}(t)}{dt} = u_t c_i(t) - rc_{ri}(t) \tag{3}$$

where, $c_i(t)$ is the activity (Curie/m^3); u_t is deposition velocity (cm/s); λ is decay constant (s^{-1}); A is free surface area of the containment (m^2); V is the total volume of the containment (m^3; R is recirculation rate (m^3/h); E is the filtration collection efficiency; L_r is the volumetric leakage rate (s^{-1}); r is re-suspension rate (s^{-1}); $c_{ir}(t)$ is the activity on surfaces of containment (Ci/m^2); and P(t) is the precursor production term (m^3/s).

Consider F_L as the containment exhaust rate and η as the collection efficiency of the exhaust filtration system (HEPA for particle collections and Activated charcoal filets for iodine), the activity of ith radioisotope released to environment at any time t is (($C_i(t)$, (*Curie/s*)):

$$C_i(t) = c_i(t) \times F_L(1-\eta) \tag{4}$$

The total activity release to environment at the time t is given by the following equation:

$$C_i(t) = F_L(1-\eta)\int_o^t c_i(t)dt \tag{5}$$

Now the total activity release to environment is governed by:

$$K_i = F_L(1-\eta)\int_o^\infty c_i(t)dt \tag{6}$$

Numerical modelling and assumptions: The flow blockage accident is selected for the case study. The probability of the flow blockage accidents for TMI2 reactor is from 4×10^{-5} to 6×10^{-6} per year (NRC, 1975). The primary reason for this accident could be the pump failure, heat exchanger blockage, steam generator tube blockage, pump power failure both onside and offside power failure. In extreme case situation 5% core damage is assumed in this case.

The water release fraction for noble gasses is 0.5% (Dadillon and Gesse, 1967) to 1.5% (Main, 1968); 4×10^{-4} % to 5×10^{-3} % (Merchie *et al*., 1968) for iodine. However, for TMI 2 plant, the release fraction of 0.6% for noble gasses; 2×10^{-3} % for iodine; 1×10^{-3} % for Cs and Rb; 2×10^{-3} % Te-Sb; 1×10^{-4} % for Ba-Sr; while 0.3% for La group are assumed (NRC, 1975).

In this study modeling and simulation for flow blockage accident has been carried out by developing a MATLAB code "STEFBA." Some assumptions are used for numeric simulation, which are as follows.

- Instantaneous release of radiation to the containment is assumed.
- Removal of radio isotopes by recirculation is ignored thought out the simulation.
- The precursor production is ignored.
- In isolation state removal by exhaust filtration is ignored.
- All the free surface of the containment is assumed as paint less rough concrete.
- The effective deposition velocity on the concrete is 6.7×10^{-4} cm/s while the iodine deposition velocity on concrete is 5.5×10^{-2} cm/s (ORNL-NSIC-4, 1965). In this study iodine deposition velocity is assumed as 5.1×10^{-2} cm/s.
- The re-suspension rate for iodine is assumed as 2.3×10^{-6} s^{-1}, whereas it varies with surface type and temperature (NRC, 1991).
- The aerosol deposition velocity on the rough surface is assumed as 1.27×10^{-1} cm/s corresponding to 2.5 μm aerosol size (Piskunov, 2009).
- The Re-suspension of aerosols varies from 300×10^{-5} to 0.4×10^{-5} per sec (Papsteafanou, 2008). Here re-suspension value of 1.5×10^{-5} per sec is used for aerosols.
- Re-suspension of iodine and aaerosols are ignored.

RESULTS AND DISCUSSION

The "STEFBA" developed code has been used for the source term evaluation for flow blockage accident. The numerical modelling and assumption have been used for the source term determination. The fission product inventory for 2720MWt reactor (TMI2 reactor) has been evaluated under the core design constrains and listed in Table 1 along with the standard activity data of 3000 MWt reactor.

The standard activity is slightly higher than the calculated activity because the calculated activates are for 2720 MWt rector slightly low thermal power than the typical conventional reactor (3000 MWt). The major contribution to the source term is due to the noble gasses, iodine and Aerosols. In this study source term for noble

Table 1: A compression of fission product inventory of 2720 MWt with typical 3000 MWt reactor

		Core inventories Ci×10^{-8}				Core inventories Ci×10^{-8}	
Nuclide	Decay constant	This study	WASH-1400	Nuclide	Decay constant	This study	WASH-1400
Kr83m	1.05E-04	7.65E-02	---	I135	2.91E-05	1.38E+00	1.50E+00
Kr85	2.05E-09	7.57E-03	5.60E-03	Cs134	1.07E-08	1.60E-01	7.50E-02
Kr85m	4.30E-05	1.49E-01	2.40E-01	Cs136	6.12E-07	4.38E-02	3.00E-02
Kr87	1.51E-04	2.76E-01	4.70E-01	Cs137	7.32E-10	9.41E-02	4.70E-02
Kr88	6.78E-05	3.87E-01	6.80E-01	Cs138	3.59E-04	1.30E+00	-----
Kr89	3.64E-03	4.54E-01	-	Ba140	6.27E-07	1.23E+00	1.60E+00
Kr90	2.15E-02	4.46E-01	-	Sr89	1.59E-07	5.22E-01	9.40E-01
Xe131m	6.74E-07	7.70E-03	-	Sr90	7.54E-10	5.78E-02	3.70E-02
Xe133	1.53E-06	1.47E+00	1.70E+00	Ba140	6.27E-07	1.23E+00	1.60E+00
Xe133m	3.66E-06	4.70E-02	-	Ru103	2.04E-07	1.31E+00	1.10E+00
Xe135	2.12E-05	2.87E-01	3.40E-01	Ru105	4.34E-05	1.00E+00	7.20E-01
Xe135m	7.56E-04	3.05E-01	-	Rh105	5.45E-06	9.03E-01	4.90E-01
Xe137	3.02E-03	1.28E+00	-	Te127	2.06E-05	9.30E-02	5.90E-02
Xe138	8.15E-04	1.16E+00	-	Te127m	7.36E-08	1.26E-02	1.10E-02
I131	9.98E-07	7.60E-01	8.50E-01	Te129	1.66E-04	2.59E-01	3.10E-01
I132	8.37E-05	1.08E+00	1.20E+00	Te129m	2.39E-07	3.87E-02	5.30E-02
I133	9.26E-06	1.47E+00	1.70E+00	Te132	2.46E-06	1.06E+00	1.20E+00
I134	2.20E-04	1.60E+00	1.90E+00	I135	2.91E-05	1.38E+001	.50E+00

Table 2: A comparison of noble gasses source term for flow blockage accident in Normal, Emergency and Isolation

		Source term (Ci)		
Isotopes	Decay constants	Normal 50970 (m^3/h)	Emergency 1699 (m^3/h)	Isolation 24(m^3/h)
Kr83m	1.05E-04	1.59E+03	1.66E+02	2.52E+00
Kr85	2.05E-09	2.25E+02	2.27E+02	1.54E+02
Kr85m	4.30E-05	3.77E+03	7.14E+02	1.20E+01
Kr87	1.51E-04	5.06E+03	4.24E+02	6.31E+00
Kr88	6.78E-05	9.00E+03	1.25E+03	1.97E+01
Kr89	3.64E-03	7.50E+02	2.69E+01	3.81E-01
Kr90	2.15E-02	5.34E+01	1.83E+00	2.58E-02
Xe131m	6.74E-07	2.28E+02	2.13E+02	3.38E+01
Xe13	31.53E-06	4.35E+04	3.71E+04	3.10E+03
Xe133m	3.66E-06	1.38E+03	9.74E+02	4.32E+01
Xe13	52.12E-05	7.86E+03	2.40E+03	4.67E+01
Xe135m	7.56E-04	2.17E+03	9.56E+01	1.37E+00
Xe137	3.02E-03	2.58E+03	9.34E+01	1.32E+00
Xe138	8.15E-04	7.78E+03	3.37E+02	4.81E+00

gasses, iodine and aerosol are determined in confined states (i.e., Normal, Emergency and Isolation) of the containment.

Source term for noble gasses: The noble gasses always remain dominant in any accident and contributed to the whole-body dose. Due to inert in nature the noble gasses are rapidly escaped into the containment atmosphere. Since they are less reactive thus cannot be eliminated easily. The corresponding source term for normal emergency and isolation state is listed in Table 2. It has been observed that with the shift of ventilation fan form normal to isolation no significant reduction has been observed. In isolation reduction in some isotopes of noble gasses has been observed to a magnitude. The variation of the noble gasses source term with containment filtered venting exhaust rate is shown in Fig. 3. The shift of exhaust rate from normal to emergency the source term for ^{83m}Kr, ^{87}Kr, ^{89}Kr, ^{90}Kr, ^{133m}Xe, ^{135m}Xe, ^{137}Xe and ^{138}Xe

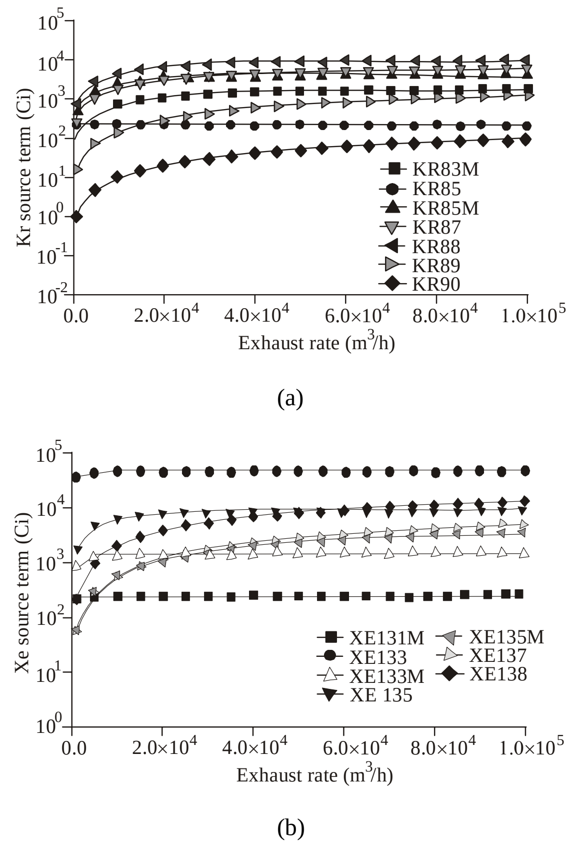


Fig. 3: Noble gasses a (Kr) and b (Xe) source term with variation of the exhaust rate

Table 3: A comparison of iodine source term in normal emergency and isolation state

Isotopes	Decay constant	Source term (Ci)		
		Normal	Emergency	Isolation
I131	9.98E-07	7.51E+00	3.40E+00	6.57E-01
I132	8.37E-05	7.97E+00	9.56E-01	1.48E-01
I133	9.26E-06	1.40E+01	5.00E+0	09.24E-01
I134	2.20E-04	8.31E+00	5.70E-01	8.35E-02
I135	2.91E-05	1.22E+01	2.81E+0	04.79E-01

reduce to significant magnitude but the isotopes ^{85}Kr, ^{85m}Kr, ^{131m}Xe, ^{133}Xe, ^{135}Xe and ^{88}Kr does not show a significant reduction with the change of containment state. Almost, all nuclides of Kr approaches to the saturation at 9000 m^3/h exhaust rate while The Xe isotopes reaches the saturation value at 4000 m^3/h.

Source term for radioiodine: The iodine source term evaluation is important for the licensing and operation of a nuclear power plant. Various isotopes of iodine are radiological hazardous to humans and living beings. The iodine delivers the thyroids and lungs dose. The radioiodine can cause the lungs cancer and breathing diseases.

The iodine source term for normal, emergency and isolation source term is listed in Table 3. With the shift of FCVS exhaust fan from normal to emergency extending the iodine source term is not much affected except ^{134}I. In all situations filters have the efficient role in the removal of iodine but in the isolation state, leakage does not occur through the ventilation system thus the source term in the isolation state is determined without filtration. The variation of the iodine source term with exhaust rate is shown in Fig. 4. A typical trend has been observed with the initial rapid increase with the increase of the ventilation exhaust rate then approaches to saturation. Some isotopes of iodine show independent dependent behaviour towards exhaust rate and approaches to saturation even at lower ventilation exhaust value.

The iodine source term is strongly dependent on the containment spray system. With the variation of boric acid spry flow rate (Fig. 5a) the no significant decrease in source term is observed. While with caustic acid spray (Fig. 5b) the iodine abruptly decreases.

Source term for aerosols: As the result of an accident some fission products form aerosols due to nucleation and agglomentation process. The major contribution to the Aerosol source term is due to ^{127}Te, ^{129}Te, ^{89}Sr, ^{90}Sr, ^{103}Ru, ^{105}Ru, ^{105}Rh, ^{137}Cs, ^{138}Cs and ^{140}Ba. The aerosol delivers the x ray dose and also contribute in the whole-body dose. Most of the aerosols have been found independent of the exhaust rate. The aerosol source term is listed in normal

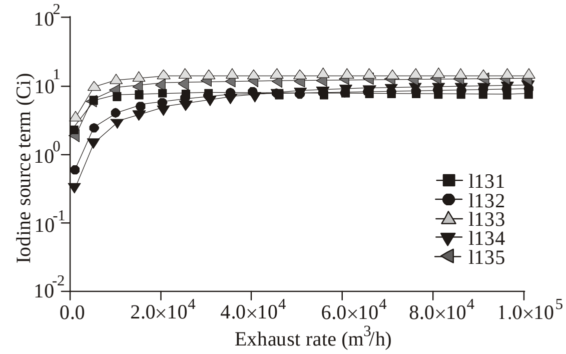
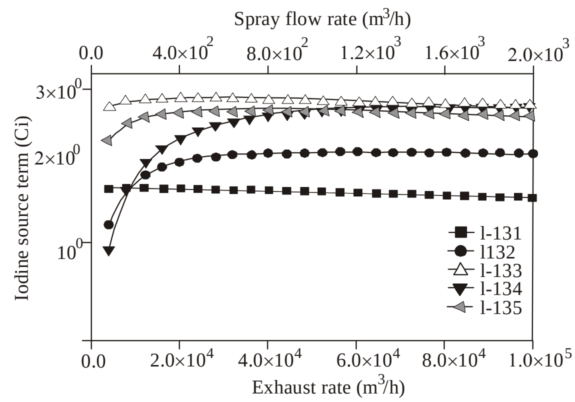
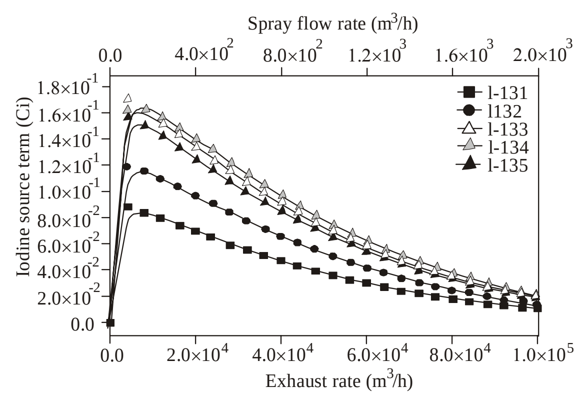


Fig. 4: The variation of the iodine source term with exhaust rate



(a)



(b)

Fig. 5: Variation of the iodine source term with exhaust and spray flow at (a) pH 5.0 and (b) pH 9.5

emergency and isolation state for flow blockage accident is listed in Table 4.

The dominant aerosols are the independent of exhaust rate with the shift of containment state from normal to emergency so no significant decrease on aerosol source term is observed. With the shift of normal to emergency state ^{127}Te reduces to one third value, ^{138}Cs and ^{105}Rh approximately reduce to its half value. Whereas, in the

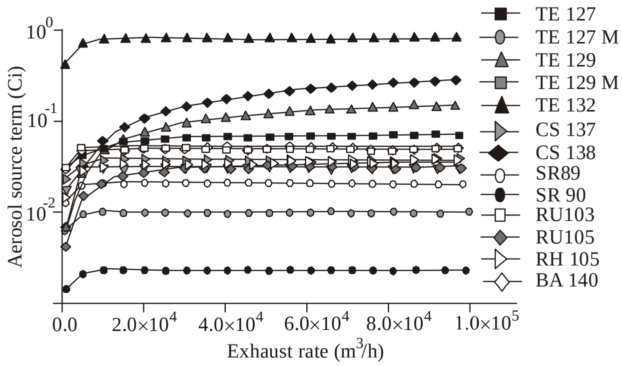


Fig. 6: The variation of the aerosol source term with exhaust rate

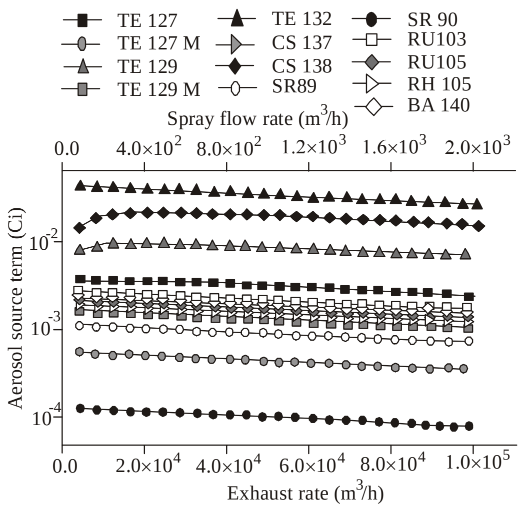


Fig. 7: Variation of aerosol source term with spry flow rate and exhaust rate

Table 4: A comparison of aerosol source term in normal emergency and isolation state

		Source term (Ci)		
Isotopes	Decay constant	Normal	Emergency	Isolation
Cs137	7.32E-10	4.67E-02	3.74E-02	1.05E-01
Cs138	3.59E-04	2.59E-01	1.43E-02	2.07E-02
Te127	2.06E-05	8.50E-02	2.63E-02	5.09E-02
Te127m	7.36E-08	1.25E-02	9.96E-03	2.80E-02
Te129	1.66E-04	1.52E-01	1.21E-02	1.79E-02
Te129m	2.39E-07	3.83E-02	3.02E-02	8.45E-02
Te132	2.46E-06	1.04E+00	7.10E-01	1.88E+00
Ba140	6.27E-07	6.08E-02	4.67E-02	1.29E-01
Sr89	1.59E-07	2.59E-02	2.05E-02	5.74E-02
Sr90	7.54E-10	2.87E-03	2.30E-03	6.46E-03
Ru103	2.04E-07	6.49E-02	5.13E-02	1.43E-01
Ru105	4.34E-05	4.21E-02	7.93E-03	1.33E-02
Rh105	5.45E-06	4.38E-02	2.51E-02	6.20E-02

isolation state, the aerosol source term for ^{129}Te, ^{132}Te, ^{89}Sr, ^{90}Sr observed with twice in magnitude than in the normal emergency state. In all situations HEPA filters have the effective role in reducing the environmental release. The variation and dependency of aerosols is shown in Fig. 6.

Most of the aerosol became independent of exhaust rate at exhaust rate value 1.0×10^4 m^3/h except ^{129}Te, ^{138}Cs and ^{105}Ru. The aerosol particle ^{105}Ru reach to its saturation after 1.0×10^4 m^3/h value of the exhaust rate.

The Aerosol source term is independent of containment Filtered Ventilating exhaust rate (Fig. 6), while the aerosol source term has slight dependency on spray flow rate (Fig. 6). Nearly all the aerosol particles reduce linearly after the flow rate 3.0×10^2 m^3/h value. Figure 7 shows the slight dependency of aerosol on containment spray flow rate.

CONCLUSION

In this study, a computer code "STEFBA" source term evaluator for flow blockage accident is developed for modelling and simulation the release of radiation in the flow blockage accident. The mathematical modelling has been carried out by containment performance and corresponding to flow blockage accident. Experience from NUREG 0772 and WASH 1400, (ORNL-NSIC-4, 1965; NRC, 1991), IAEA safety report no 53 computational modelling has been carried out with plant PSA Study. Some results have been achieved for flow blockage accident and followings can be concluded from the results.

- Typical trend of release of radiation has been for flow blockage accident has been observed, initially rising abruptly then extends towards saturation with exhaust rate value.
- Noble gasses show the dominant behavior in all situations, the ^{88}Kr has been observed to be independent of containment state and ventilation exhaust rate.

- Noble gasses source term found to be independent of exhaust rate and with the shift of exhaust rate, no significant reduction in noble gasses source term has been achieved.
- ^{135m}Xe, ^{137}Xe, ^{138}Xe, ^{90}Kr and ^{89}Kr show the slight dependency on filtered ventilation exhaust rate.
- The isotopes of iodine reached to saturation at ventilation exhaust rate 2.0×10^4 m^3/h. the iodine is dependent on flow rate. Dependency of the iodine source term on spray flow rate with caustic acid is much more than the spray flow with boric acid.
- All the aerosols have been observed independent of the exhaust rete except ^{129}Te and ^{138}Cs.
- The Aerosols have been observed with slight dependency on containment spray flow rate.

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