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Comprehensive Review of Source Term Analysis and Experimental Programs

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Abstract: Source term evaluation is important in legislation for the licensing of a plant. So a comprehensive review of historical progress in source term analysis and source term experimental programs have been presented for better understanding of core melt progression and fission product behavior under severe accident. The discussions mainly focus on the advancement in source term and comparison between source term experimental programs. For In-vessel and Ex-vessel source term estimation, simulated and design based experiments had been performed. Most of the experiments have been performed to understand the behavior and analysis of radionuclides during different accident consequences. Data obtained from these tests have been used for validation of Computer based codes. This study gives a comprehensive review of the source term and the source term experimental programs designed for investing the source term and fission product release behavior.

Keywords: Fission products, severe accidents, source term, source term code package, source term experimenltal programs

INTRODUCTION

Probability of Severe Accidents (SAs) in the Advance Power Reactor (APR) is extremely small. However, SAs may happen due to extraordinary equipment failure or human errors. SAs are identified from the Probability Safety Assessment (PSA) of plant. SAs can lead to the failure of barriers for Fission Products (FPs) to release into the environment to contribute the source term. Source term refers as the determination of possible release of radiations and radionuclides from the Reactor Pressure Vessel (RPV) (Frutos *et al*., 2000). SAs also include the release of radioactive materials with combustible gasses (IAEA, 2003). The release fraction of FPs depends upon the accident sequences. So, it is important to evaluate the FPs inventory and radiological source term for risk assessment under severe conditions. Evaluation of the source term is essential for the safety assessment not only for the test type reactors but also for power plants.

Efforts for source term evaluation have been made in the past. The first estimation of radioactivity released from the core documented in (WASH-740, 1957) which, only described the speculative radiation hazards. Later, TID-14844 ascertains the radiological source term, which set forth for the licensing and sighting of nuclear power plants. This ascertained source term is used as the reference information and guideline set forth in title "10 codes of federal regulation part 100" (DiNunno *et al*., 1962). The first complete documentation on risk assessment and source term evaluation has been documented in WASH-1400 (NRC, 1975). This report describes the risk quantification of the source term determined within the framework of Probabilistic Risk Assessment (PRA).

Besides these efforts researches has used these developed codes with different techniques for source term determination and severe accident management. Shoaib Raza and Iqbal (2005) and Waqar *et al*. (2009) have carried out the research for source term evaluation of Pakistan Research Reactor-1 (PARR-1) and Miniature Neutron Source Reactor (MNSR) using ORIGEN2 and WIMSD4 computer Program respectively. Mirza *et al*. (2010) have studied and compare the source term of MNSRs for HEU and LEU UO_2 with the ORIGEN2 and WIMSD4 computer programs. Recently, (Mirza *et al*., 2010) have carried out the studies for the source term analysis of upgraded LEU (PARR-1) Material Test Reactor (MTR) for startup, Loss of Coolant Accident (LOCA) and fuel loading accidents. Lewis *et al*. (1996) have developed and applied an analytical model for diffusivity of short-lived FPs and validate the model from out of pile experiments. For the NPP source term under SAs, (Lee and Yu-Chih, 2008) have studied the source term of Washington 3 loops NPP based on NUREG-1465, TID-1484 and WASH-1400 using MAAP4 computer code. He developed a methodology for source term estimation. Huang *et al*. (2010) have carried out the investigation on the In-vessel source term of Chinese 600 MWt PWR reactor under Station Blackout (SBO), Loss of Feed Water (LOFW) and Large Break LOCA (LBLOCA) accidents. For Indian Pressurized Heavy Weater Reactor (PHWR), source term has been estimated by Chatter jee

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et al. (2010). He adopted multi physics methodology using ORIGEN2 for FPs inventory, SCDAP/RELAP5 for thermal hydrolysis, ASTAC for fission product transpiration, CONTAIN for containment source term and COSYMA for dose calculations. This methodology is applied for the Reactor Inlet Heater (RIH) and Reactor Outlet Heater (ROH) break. Maximum leakage has been observed for 160 and 75% break, respectively. Most recently, the source term under loss of the Residual Heat Removal (RHR) system in 1000 MWt 3-loop PWR reference reactor has carried out by Ammirabile *et al*. (2011). He assumed the complete removal of the upper head of RPV, introducing the rapid air ignition. Version 2.0 of integrated code ASTAC has been used in the present study.

In view of above selected research it is easy to say that from the last few decays to since now, substainal developments and progress have been made for understanding the nature of severe accidents in NPPs and in research reactors as well. For this, particular interest is being able to predict the FPs release and transport after the core damage (Yang and Khatib-Rahbar, 1987). In the postulated severe accident, fuel experiences the variety of conditions. During the core damage, FPs releases from core and carried out in different forms into the containment. In order to assess the potential of environmental release, the nature and concentration of fission products should be known. The FP source term estimation has been investigated for more than 20 years for safety assessments. For this purpose, benchmark experiments have been performed and simulated.

This study presents the overview historical prospective of source term and the Source Term Experimental Programs (STEPs) conducted to understand the core degradation phenomenon. The aim of this study is focused on the better understanding of source term evaluation and fission product release phenomenon under sever core damage propagations.

PROSPECTIVE OF SOURCE TERM

From the development of NPPs, "source term" has been recognized as a very important factor influencing both the design of certain safety equipment and safety evaluations, including risk assessment (Rosen *et al*., 1985). The quantity, compositions, forms and modes of release of reactivity during some accident scenario constitute the source term of that scenario. Realizable source term required for estimation of release in inside and outside of the containment building (IAEA, 2008). The Source term has always been recognized as the important factor inflicting both the design and safety equipment's performance. The source term evaluation has the significant weightage for the licensing of a NPP.

TID-14844 source term: Wash-740 (1957) the first report form (U.S.NRC, 2000) only explains the radiological hazards. The source term evaluated in this report was much more than the today calculated source term. In 1962 the source term determined in TID 14844 (DiNunno *et al*., 1962) has set forth for the reactor sighting as set forth in title "10 code of federal regulation part 100" (10 CFR 100). This technical document explains the release associated with core inventory with 100% of noble gasses, 50% of iodine and 1% of solid fission products. The 50% of iodine assumed as the airborne release and the rest of iodine supposed to be absorbed in containment during release. For leak tight containment 0.1% of leakage has been assumed in this technical document.

Assumptions of regulatory guide 1.3 (U.S.AEC, 1974a, b) and 1.4 (U.S.AEC, 1974a, b) has been derived from this report for Loss of coolant accident. The basic assumptions of these regulatory guides are 25% release of iodine of which 91% release as an elemental form, 5% as particulate form and 4% as organic iodine, respectively. However, the regulatory guide 1.195 assumes the core inventory release fraction for noble gasses 1.0 and 0.5 for iodine, respectively. These assumptions have a significant role in reactor design and safety equipment's installation.

WASH 1400 source term: The risk assessment of U.S. nuclear power Plants have been documented in Wash 1400 (NRC, 1965). WASH 1400 source term not only used as the reference guide line for reactor sighting but it also for determining the Reactor Emergency Zone (REZ), Genetic issues, unresolved safety issues and other regulatory Analysis. This report mainly focused on the:

- Accident sequence and their probabilities
- Source term
- Description dispersion of radiation to the environment
- Property and health assessment

The 155 key accidents involving core degradation has been explained in WASH 1400 on fault tree and even tree basis. Dominant key accidents of PWR and their release categories are enlisted in Table V 3-14 in WASH-1400 appendix V (NRC, 1975). The source term for the key accident in PWR is listed in Table 1. Wash 1400 categorized the released fission products into 7 groups depending upon their physical and chemical properties. The radionuclide grouping of wash 1400 is listed in Table 2.

The release mechanisms considered in WASH-1400 are:

- Gaps release
- Core melts release

Table 1: A compression of total core release fraction

Group	Gap release fraction	Meltdown release fracti	Vaporization release fraction	Steam explosion release fraction
Noble gases	0.030	0.087	0.100	0.90
Iodine	0.017	0.883	0.100	0.90
Cesium and rubidium	0.050	0.760	0.190	
Tellurium, antimony	1×10^{-4}	0.150	0.850	0.60
Alkaline earth1	$\times10^{-6}$	0.100	0.010	
Volatile oxide (Ru)		0.030	0.050	0.90
Nonvolatile oxide (La)		0.003	0.010	

Table 2: A compression of radionuclide grouping for source terms

Group	Member elements
[c] **WASH-1400**	
Noble gas	Kr-85, Kr-85 m, Kr-87, Kr-88, Xe-133,
Iodine	I-131, I-132, I-133, I-134, I-135
Cesium and rubidium	Cs-134, Cs-136, Cs-137, Rb-86
Tellurium and antimony	Te-127, Te-127 m,Te-129,Te-129 m,Te-131 m, Te-132, Sb-127, Sb-129
Alkaline earth	Sr-89, Sr-90, Sr-91, Ba-140
Volatile oxide (Ru)	Co-58, Co-60, Mo-99, Tc-99 m, Ru-3,
Nonvolatile oxide (La)	Y-90, Y-91, Zr-95, Zr-97, Nb-95, La-140, Ce-141, Ce-143, Ce-144, Pr-143, Nd-147, Np-239, Pu-240, Pu-241, Am-241, Cm-242, Cm-244
[a]**Source term code package**	
1	Xe, Kr
2	I, Br
3	Cs, Rb
4	Te, Sb, Se
5	Sr
6	Ru, Rh, Pd, Mo, Tc
7	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y
8	Ce, Pu, Np
9	Ba
[b]**NUREG-1465 source term**	
Noble gas	Xe, Kr
Halogens	I, Br
Alkali metals	Cs, Rb
Barium and strontium	Ba, Sr
Noble metals	Ru, Rh, Pd, Mo, Tc, Co
Lanthanides	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y
Tellurium group	Te, Sb, Se
Cerium group	Ce, Pu, Np

a: Lee and Yu-Chih (2008); b: NRC (1995); c: NRC (1975)

- Vaporization release
- Steam explosion release

The first three components release occurs according to the accident, but the fourth depends upon the prediction of the steam explosion. The total core release fraction of these seven groups of above mentioned released mechanism is enlisted in Table 3.

Source Term Code Package (STCP) source term: The TMI-2 accidents in 1979, peruse the research on source term analysis, severe accident management, development of severe accident codes and Source Term Code Packages (STCP). In mid of 19th century (U.S.NRC, 2000) incorporated with research laboratories, i.e., Battelle Columbus Laboratory (BCL), Argonne National Laboratories (ANL), Oak Ridge National Laboratory (ORNL), Sandia National Laboratories (SNL) and the American Nuclear Society (ANS), pursued their research on source term evaluation and severe accident management. As a result series of technical reports has been published, concerning to possible accidents in commercial NPPs and research reactors (Mehboob *et al*., 2012). These combine effort results in development of effective computer codes and source term packages. These codes used for the probability risk assignment for U.S nuclear power plants and results had been published in series of reports. The WASH-1400 radiological sequences were peer review and summarized in NUREG-0772 (NRC, 1981). U.S.NRC (2000) funded Battelle Columbus Laboratory (BCL) for development and modification of computer codes. The Battelle suites of code's application to five U.S. operating plants have been published in BML-2104 (Gieseke *et al*., 1983). The peer review of BML 2104 based on source term reassessment has been published in NUREG 0956 (NRC, 1986). Based on this reassessed source term five U.S. nuclear power plants have been analyzed for risk assessment and results have been presented in (NUREG/CR-4624, 1986). The study of NUREG /CR-4624 has been used to refine the risk assessment and accident source term determination. The results of this study for U.S. nuclear power plants have been documented in NUREG 1150 (NRC, 1990).

Table 3: Wash-1400 accidental source terms for PWR

Release categories	Probability /year	Release time (h)	Release duration (h)	Warning time (h)	Fraction of core inventory release							
					Xe-Kr	I	Org. I	Cs-Rb	Te-Sb	Ba-Sr	Ru	Lu
PWR 1	9×10^{-7}	2.5	0.5	1.0	0.9	0.7	6×10^{-3}	0.4	0.4	0.05	0.4	3×10^{-3}
PWR 2	8×10^{-6}	2.5	0.5	1.0	0.9	0.7	7×10^{-3}	0.5	0.3	0.06	0.02	4×10^{-3}
PWR 3	4×10^{-6}	5.0	1.5	2.0	0.8	0.2	6×10^{-3}	0.2	0.3	0.02	0.03	3×10^{-3}
PWR 4	5×10^{-7}	2.0	3.0	2.0	0.6	0.09	2×10^{-3}	0.04	0.03	5×10^{-3}	3×10^{-3}	3×10^{-3}
PWR 5	7×10^{-7}	2.0	4.0	1.0	0.3	0.03	2×10^{-3}	8×10^{-4}	5×10^{-3}	1×10^{-3}	6×10^{-4}	3×10^{-3}
PWR 6	6×10^{-6}	12.0	10.0	1.0	0.3	8×10^{-4}	2×10^{-3}	8×10^{-4}	1×10^{-3}	9×10^{-5}	7×10^{-5}	3×10^{-3}
PWR 7	4×10^{-5}	10.0	10.0	1.0	6×10^{-3}	2×10^{-5}	2×10^{-5}	1×10^{-5}	2×10^{-5}	1×10^{-6}	1×10^{-6}	3×10^{-3}
PWR 8	4×10^{-5}	0.5	0.5	N/A	2×10^{-3}	1×10^{-4}	5×10^{-6}	5×10^{-4}	1×10^{-6}	1×10^{-8}	0	0
PWR 9	4×10^{-4}	0.5	0.5	N/A	3×10^{-6}	1×10^{-7}	7×10^{-9}	6×10^{-7}	1×10^{-9}	1×10^{-11}	0	0

Table 4: Fission product release fraction for PWR into the containment (NRC, 1995)

	Gap release	Early in-vessel	Ex-vessel	Late in-vessel
Duration (H)	0.50	1.3000	2.0	10.0
Nobel gasses	0.05	0.9500	0.0	0
Halogens	0.05	0.35	0.25	0.1
Alkali metals	0.05	0.25	0.35	0.1
Tellurium group	0	0.05	0.25	0.005
Barium, strontium	0	0.02	0.10	0
Nobel metals	0	0.0025	0.0025	0
Cerium group	0	0.0005	0.0050	0
Lanthanides	0	0.0002	0.0050	0

The development of STCP and severe accident codes were the important mile stone for source term determination and severe accident management. The STCP radionuclide has been grouped into nine basic groups introducing in-vessel and ex-vessel groups enlisted in Table 2. However, some codes like MAAP4.0 uses 12 groups for the source term determination. In STCP release mechanisms are:

- Gap release
- In-vessel release from the fuel pellets during heat up and meltdown
- Ex-vessel release from the corium pool due to Molten Core Concrete Interactions (MCCI)

The amount of radionuclide released to the containment atmosphere depends upon the airborne particles (Aerosols). In STCP calculations 100% iodine assumes to release and reacts with Cs to form CsI and reaming as CsOH (NRC, 1981, 1990).

NUREG-1465 revised source term: In 1995, the review of NUREG 1150 results the revised source term which is documented in NUREG-1465 (NRC, 1995). The NUREG-1465 studies both the results of STCP and uncertainty analysis, which indicates that there is a little difference between Br and Sr. Therefore, Br and Sr are grouped together. The elements such as Co, Cm and Am, which were not considered in WASH-1400 and STCP source term, are considered in revised source term. The radionuclide grouping for the revised source term is listed in Table 2: The time-dependent release fraction determined in NUREG 1465 is listed in Table 4.

The In-vessel releases together with the gap release in NUREG-1465 specify the Accidental source term determine in Regulatory Guide 1.183 (NRC, 2000). The revised source term is revolutionary for the design of future LWRs and safety equipment's.

The chemical and physical characteristics of radionuclide also influence the source term accidental consequences (Mirza *et al*., 2010). Source term can be defined in two types; one for the licensing and operation of NPP, e.g., specified in TID-14844 NUREG-1465. Second type is risk assessment of NPP, e.g., specified in WASH-1400. A comparison of radionuclides for WASH-1400, NUREG-1465 and source term code package is enlisted in Table 2.

In comparison, WASH-1400 and NUREG-1465, categorized the radionuclides into eight groups but in source term code packages e.g., MAAP4, MALCORE 1.8.5, etc., radionuclides are categorized into 9 groups for source term estimation.

According to NUREG 1465 FPs releases are categorized into 5 phases, including:

- Coolant activity release
- Gap Activity release
- Early in-Vessel Release
- Ex-Vessel Release
- Late in-Vessel Release

Coolant activity release phase starts from the coolant leakage up to the first fuel rod failure. During this phase reactivity absorbs in coolant and releases to the containment. Gap activity commences from the cladding failure which gives the possibility to escape the radionuclide and FPs which are trapped between clad and pallet's gap. This phase ends when the fuel pellet bulk temperature has risen sufficiently that significant amounts of FPs can't be retained in the fuel. Early In-vessel phase started when the temperature of core is high enough that it does not maintain the geometry of fuel, cladding and structure material. This phase tends to release volatile FPs together with less volatile FPs. This phase ends with the failure of the lower head of RPV allowing the debris to fall onto the bottom concrete. After the Reactor Pressure Vessel (RPV) breach, Ex-vessel phase started, which causes the emission due to the interaction with concrete. The late In-vessel release phase commences at vessel breach and proceeds simultaneously with the occurrence of the Ex-vessel phase (NRC, 1995).

FISSION PRODUCTS RELEASE

Core degradation and melting: As a result of a severe accident the core degradation and relocation occur. The important physical, chemical and melting interaction with temperature which results in formation of liquid phase during severe core damage in LWR is shown in Fig. 1.

Depending upon the severe accident sequences the important physical and chemical material behavior in PWR includes:

- Melting of Ag-In-Cd alloy (1073 K). The chemical interaction between zircaloy guided tube and fuel rod material at temperature 1720 K.
- ~1020-1370 K possibility of bursting of cladding depending upon the pressure.

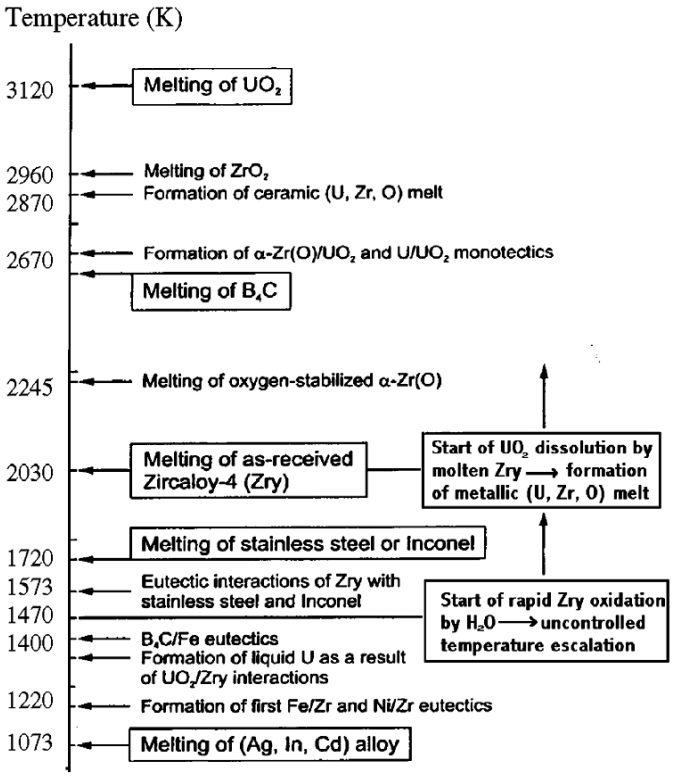

Fig. 1: Severe accident melting and chemical interaction temperatures (Hofmann, 1999)

- ~1370 K interaction of UO_2/Zr due to liquefaction of fuel.
- The temperature rapidly accelerates due to the in Zr oxidation with steam at ~1370 K.
- Eutectic interaction of Zr, stainless steel and Inconel (grid space) with UO_2 at ~1573 K with melting of stainless steel and Inconel's at temperature ~1720 K.
- Melting of as received Zircolay (~2030 K) and melting of oxygen stabilized α-Zr (O) phase (~2245 K).
- Reduction in UO_2 fuel due to interaction with molten zircaloy started at 2030 K resulting in the formation of (U, Zr)O_{2-x} precipitates.
- Relocation of liquid and solid material in different parts of the core (>2030 K).
- Melting of ZrO_2 at temperature 2960 K and melting of UO_2 at temperature 3120 K forming ceramics (Hofmann, 1999; Hofmann *et al*., 1989).

As the consequences of these temperatures phenomena the core melts and relocation propagates with temperature. Generally, core melting initiates with the melting of Ag-Ln-Cd control rod at temperature ~1073 K. The interaction of between stainless steel and zircaloy liquid phase started in temperature range ~1420-1570 K. The liquefied zircaloy comes into contact with the sounding fuel rods and results in relocation and flow blockage. This flow blockage could accelerate the temperature rising and temperature above 1470 K rapid steam oxidation of Zircaloy also contributes in temperature escalation, yielding peak temperature up to 2270 K. The temperature range between ~2030-2270 K, when remaining zircaloy melts, the solid UO_2 starts liquefying and chemically dissolved. The relocation of metallic and ceramics and solicitation enhance the flow blockage result in the extension in core damage. With the melting of fuel and cladding form temperature ranging 2870-3120 K a complete meltdown of core occurs.

Regarding to code damage propagation, the fission products are escape form molten fuel. This release propagation is studied as in-vessel and Ex-vessel release.

In-vessel relese mecheimesms: During reactor operation, the FPs invented at the intragranular and intergranular spaces and diffuses to clad to pellet gaps. During core damage and severe thermal transient s failure of the sheath may occur (~1073-1220 K). FPs located in fuel to sheath gap rapidly released into the Reactor Coolant System (RCS). However, the volatile halogens react with the fuel, steam and other FPs, so they delayed in such inventories.

In oxides fuel, fission products are categorized as volatile FPs (Kr, Xe, Br, I), less volatile FPs, metallic precipitates (oxides) (Mo, Tc, Ru, Rh, Pd, Ag, Cd, In, Sn,

Sb, Te) and oxides dissolved in the fuel (Sr, Zr, Nb, Y, La, Ce, Pr, Nd, Pm, Sm). At the low temperatures <1273 K diffusion of FPs in the fuel matrix is Athermal (independent of temperature) and enhanced with fission (Turnbull *et al*., 1982), but experimentally it has been observed that diffusivity of some FPs like Xe, Kr and I are similar and are affected with the thermal shock (Wise, 1985; Lewis, 1987). When fuel comes into content with coolant, Uranium to Oxygen (U/O) ratio result in enhancement of diffusion of Fps. The chemical state of FPs depends upon the oxygen potential which could change the rate of hydrogen production and affect the FPs oxidation as well. The diffusion of volatile fission products also has been analyzed in annealing experiments (Prussin *et al*., 1988). Moreover, fission gas, cesium and iodine predicted to release from the irradiated fuel at the higher temperatures in gap inventories. Cs and I have been found approximately 2.5 times greater than Xe in high irradiated fuel (1000-4000 Mwd/Mt) (Lorenz *et al*., 1978). Experimental studies of Te, I, Xe, Ru and Mo release in hyper-stoichiometric sintered fuel specimens has been studied under irradiation conditions (Mansouri and Olander, 1998). In this case significant grain growth occurs due to the enhanced uranium ion diffusivity in UO_{2-x}, which supplies the irregular fission products to the grain boundaries. The kinetics release of the volatile fission products (Xe, Kr, Cs, I, Te, Sb) can be described by a limiting process of solid-state diffusion through the UO_2 fuel matrix (Osborne and Lorenz, 1992; Lewis *et al*., 1996; Lewis *et al*., 1998). However, for non-volatile fission products (e.g., Sr, Y, Zr, Nb, Mo, Tc, Ru, Rh, Pd, Ba, La, Ce, Pr, Nd), their escape from free surfaces into the gas phase may be difficult (Prussin *et al*., 1988).

In order to understand the gap inventories B.J. Lewis proposed a model for halogen gap inventories and applied to experimental results from the Clark River Nuclear Laboratories. Experimental test details are described in reference 19 (Iglesias *et al*., 1990).

During heat up, as soon as the cladding exceeds the temperature of 1173 K, failure of the cladding is expected and gaseous fission products contained in the gap are swept to the coolant channels. The fission products released on a large scale with fuel temperatures more than 2300 K, initially the highly volatile ones (such as Xe, Kr, Cs, I, Te) than less volatile ones (such as Ba, Sr, Ru) (Frutos *et al*., 2000).

At temperatures >2273 K some metallic fission products may also be invented with the interaction of coating in coated particle fuels (Katscher *et al*., 1990). In oxidizing environments at temperatures 2100 K or higher it has been observed that volatilization of diverse species like Mo, Y, La and Ba show the same activation energy for release. Nature of the species and nature of solid state in the UO_2 matrix does not show a significant effect, also in CRL Experiments significant weight loss has been reported in UO_2. This phenomena is proportional to temperature as the temperatures increase the weight loss is observed to increase. These two observations indicate a volatilization of uranium mass, which results in a release of FPs from the fuel matrix. If the cladding is severely oxidized, then UO_2 reacts with the oxidizing environment. Under such condition, the UO_2 will incorporate with excess oxygen to form hyperstoichiometric UO_{2+x} or converted to a higher oxide. During oxidation, iodine and fission gasses releases in irregular bubbles and diffusion rate of FPs also increases. The metallic radioisotope like Cs and Sr reacts violently with structure materials which have relevance significance on the source term (Iglesias *et al*., 1999).

The control rods material releases are the major source of aerosols in In-vessel source term. In Pressurized Water Reactors (PWRs), the volatile cadmium vaporizes initially, followed by less volatile silver from Ag-In-Cd control rods. Indium vaporization largely depends upon steam concentration, with the formation of volatile InOH at high temperatures. However, the InOH is stable only at high temperatures, but at low temperatures it decomposes to form low volatile ln_2O_3. Cadmium aerosol formation, e.g., CdO or Cd $(OH)_2$ is also possible, influenced by the presence of steam.

For FPs release analysis, an appropriate model has been used as incorporate to temperature and burn up. Some possible choices of models are FASTGRASS, SCDAP, VICTORIA, MAAP, MELCOR, ICARE, ELSA and KESS or a simplified "Booth-type" model. An alternative simplified model is used such as RelVol, which is based on Boot type kinetics.

In a number of models, there is an option that FPs Te and Sb (as well as the Sn content in Zr) retained in unoxidized Zr and subsequently released only after about 90 to 95% of the (fuel region) Zr has been oxidized. In the number of studies, it has shown that the activity of tin (with which Te and Sb are associated) only significant when the activity of zirconium is relatively low (IAEA-TECDOC-1127, 1999).

Ex-vessel release mechanisms: Ex-vessel source term or source term to containment is the magnitude, physical, chemical form, timing of the release of FPs and aerosols from core materials and concrete to the primary containment atmosphere from both In and Ex-vessel sources, or it is the airborne radioactivity. In order to describe the Ex-vessel source term one must describe the significant release of FPs especially the non-radioactive aerosols predicted to occur pursuant to the Molten Core Concrete Interaction (MCCI) after the corium has melted through the (RPV) and fallen into the reactor cavity. This release of FP mainly derived from gas and steam, which results from the concrete decomposition. Other Ex-vessel sources are related to the transient phenomenon, which occurs at about the vessel breached time. The Ex-vessel release dominates by the release associated with the

interactions of reactor core debris with the structural concrete in the reactor cavity (IAEA-TECDOC-1127, 1999).

These interactions can have the impact in two ways: first radioactive FPs release in the form of aerosols and second production of combustible gasses (H_2, CO) and non combustible gasses (H_2O, CO_2). Former contribute to the FP source term at melting temperature >2000 K while the later contributes the containment loading even when the molten structure is partially solidified <2000 K. During these interactions, radionuclide's can be released in the form of aerosols along with very large amounts of non-radioactive aerosols. The duration of Ex-vessel release is taken to be the period from the vessel lower head failure and the time 95% of Cs in the core debris has been released to the containment atmosphere. The reaction between molten corium and concrete without water is sufficiently calculated using codes such as CORCON (NRC) or WECHSL (FZK and IPSN), which have been validated using the SURC (SNL), BETA (FZK) and ACE experimental programs. Codes to calculate the FPs and aerosol release associated with MCCI, such as VANESA or CHEMSAGE have been validated by the ACE experiments (IAEA-TECDOC-1127, 1999; Corradini, 1990; Leonard *et al*., 2007).

Leonard *et al*. (2007) compared the MALCORE results of late phase and Ex-vessel release with NUREG-1465. Over 95% of the initial core inventory of tellurium and molybdenum are released from fuel during the in-vessel phase of a SA. As a result, a very little amount of these species is left for release from fuel (directly to containment) during the ex-vessel phase. In contrast, the NUREG-1465 prescription reflects a much lower in-vessel release fraction (presumably reflecting fewer complete releases from fuel) and thus a greater quantity available for Ex-vessel release. The NUREG-1465 (NRC, 1995) release fraction for lanthanum is nearly a factor of 10 higher than the largest MELCOR calculated release fraction and a factor of 100 higher than the median MELCOR value. In contrast, the release fraction for cerium is within a factor of 2 of that of MELCOR (Leonard *et al*., 2007).

SOURCE TERM EXPERIMENTAL PROGRAMS

Numerous in-pile and out-of-pile research programs have actively been performed for core degradation and fission product release from the degraded structure material and molten debris. These experimental programs have been conducted in different organization, e.g., ORNL experiments, CAE experiments, JAEA experiments, AECL-CRL experiments. These programs provide evidence of release characteristic of some observable fission products. In these core degradation experiments various size and geometry of bundles were used. Some bundles are shown in Fig. 2. The bundle varies from 4 to 100 with rod length 0.15 to 4 m. These experiments have been carried out with fuel irradiation and without fuel irradiation. Different heating methods have been applied for fuel melting like; internal electric heating, furnace heating, fission heating, annealing heating and decay heating. This section is some major in-pile and out of pile experiments have been discussed.

In-pile experiments: The in-pile experiments include: the Source Term Experiments Project (STEP), the Phebus Fission Product (FP) and the Severe Fuel Damage (SFD) tests; the Annular Core Research Reactor (ACRR) Source Term (ST) tests; the ACRR Damaged Fuel (DF) relocation experiment; the Power Burst Facility (PBF) Severe Fuel Damage (SFD) tests, the Full-Length High Temperature (FLHT) tests and the Loss-of-Fluid Test facility (LOFT) Fission Product (FP) test. In-pile experiment provides the data and understandings of core degradation and FPs behavior. Structural material and control rod materials (Ag-In-Cd and B_4C) has been used

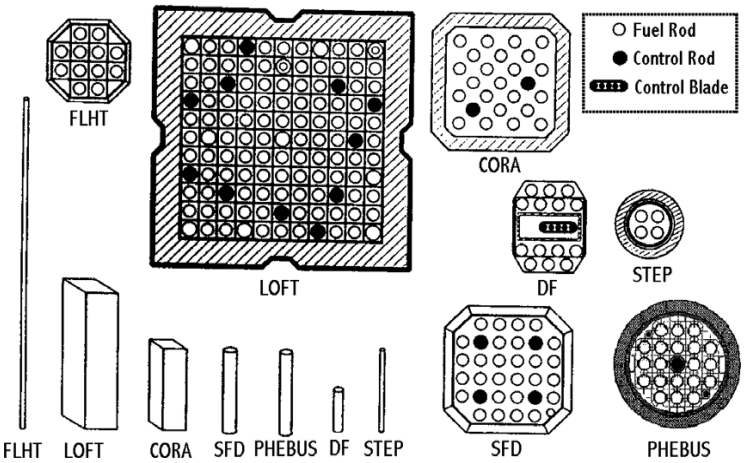

Fig. 2: Bundle configuration used in different tests (Hobbins *et al*., 1991)

in in-pile experiments to understand their effect in core degradation. STEP precisely designed to focus on the FPs behavior and aerosols chemistry. The SFD test examines the bundle behavior during degradation and melts down. The DF test examines the oxidation and coolant effect on core damage. The FLHT determines hydrogen generation and oxidation in rods. LOFT-FP2 as the largest test to investigate the fission product behavior and reflooding condition in severe accident conditions. The PHEBUS FPs examines fission product's behavior via simulation of core and containment in severe accident condition (Kudo *et al*., 2006). Here the discussion on the source term and fission product experiments is focused on.

Phebus FP: The most important project to continuing studies on FPs behavior during an accident is PHÉBUS-FP project. It was carried out by an international consortium at the Cadarache Centre in France. These experiments involved bundles of one meter long rods of irradiated fuel heated neurotically in steam through the point of fuel liquefaction and relocation (Birchley *et al*., 2005). The PHEBUS FP program is the centerpiece of a world-wide co-operation investigating on severe accidents of Light-Water Reactors (LWRs). Program partners are the Institute de Protection et de Suˆrete´ Nucle´aire (IPSN), Electricite´ De France (EDF), the European Commission (EC), the Nuclear Regulatory Commission (NRC, USA), CANDU Owners Group (COG, Canada), Nuclear Power Engineering Corporation (NUPEC, Japan), Japan Atomic Energy Research Institute (JAERI, Japan) and the Korea Atomic Energy Research Institute (KAERI, South Korea), the Swiss Federal Nuclear Safety Inspectorate (HSK) and the Paul Scherrer Institute (PSI, Switzerland) have decided to join the Program (Leonard *et al*., 2007; Schwarz *et al*., 1999).

Fresh and Irradiated fuel was used in PHEBUS test to investigate FPs behavior. PHEBUS FP investigates cladding oxidation, hydrogen production and material interactions from the early phase of up to the late phase of core material melt progression and pool formation. In parallel, chain of FP behavior was investigated. Six PHEPUS FPs tests has been conducted, the first two FP tests FPT-0 and FPT-1 were designed to investigate the difference between fresh and irradiated fuel rod degradation. FPT-0 test analyses shows that the FPs and aerosols release rates are correlated to the temperature level, but also to the bundle degradation process. The FPT-4 was performed to investigate low volatile FPs from UO_2/ZrO_2. Concluding from PHEBUS test results in the FPs and controls rod material can be grouped into four categories (Schwarz *et al*., 1999; Lewis *et al*., 2008).

- Large release (>50%) of the volatiles ones e.g., I, Te, Cs (noble gases) and Cd.
- Significant release (<50%) of structural and control rod materials (Ag, In, Fe).
- Low release (few per cent) of low-volatiles (Ba-La, Ru).
- Very low releases (<1%) of U.

Table 5: A summary of PHEBUS FP tests (Schwarz *et al*., 1999; Lewis *et al*., 2008)

Number	Test date	Objectives	Observations	Comments
FPT-0	Dec. 1993	Core degradation and FPs release from fresh fuel melting in the Oxiding environment	Maximum volatile Fps/ aersols release. twenty percent fuel melting, FPs retention in primary circuit and non condensing steam generator. In containment aerosol's sump, PH = 5	1st test with fresh fuel (50% liquified, >80% volatile fission products)
FPT-1	July 1996	Same as FPT-0 but with pre irradiated fuel	Same as FP-0 but high humidity in containment	Irradiated fuel, 20% liquefied fuel, 70-80% volatile FPs
FPT-2	1999	same as FPT-1 but with boric acid injection in poor steam condition	Maximum, volatile FPs/aerosols in presence of boric acid, in primary circuit same aerosols were observed as in FPT-1, sump and late recirculation spray	same as FP-1 But PH = 9
FPT-3	2004	Open test repetition of an earlier test, test BWR conditions (B_4C), high pressure and high burn-up fuel		Stainless steel clad and B_4C rod
FPT-4	1999	Debris bed to molten pool. Low volatile PFs release	Fuel degradation into molten pool, release and transportation of low volatile fission products (vessel was not used)	Sami volatile FPs and actinides from UO_2/ZrO_2
FPT-5	2002	Air ignition test	Fuel degradation under height oxidizing conditions, FPs retention in primary circuit and in non condensing steam generator, in primary circuit same aerosols were observed as in FPT-2	

Six experiments have been performed since 1999 to 2004. The objectives of these programs were to investigate the severe accident, fission product behavior and transportation in RCS and containment. These experiments provide the information on:

- Core melts progression, hydrogen generation and oxidation.
- Release of fission products from overheated and liquefied fuel.
- Aerosol and iodine deposition.
- Pool boiling and containment spray effect on the source term (von der Hardt and Tattegrain, 1992).

The summaries of these test observations are described in Table 5. Except FP-4 the entire tests were performed at 21 PWR bundles with 0.8 m active length.

STEP program: STEP test program was one from the pioneer programs conducted to estimate the PWR source term. Series of four experiments (STEP 1-4) were performed at the Argonne National Laboratory (ANL) in the Transient Reactor Test Facility (TREAT) for the assessment of the radiological source term in the light-water reactor under SAs contributing to risk assessment. Belgonucleair BR-3 was used as the source reactor for fuel preparation. The objective of these test series was to investigate the radiological source term in different accidental conditions. These tests were performed without the absorption (control rods) and structure material (Inconel's) in steam oxidation conditions. In STEP-1, 2 flow path resistances were found due to CsOH and other dissolved materials. (40-60%) Cs and 40% Tin contents were observed in STEP-1 but 3%; Tn in STEP-2. Mo was 2-10%; Tl, I, Ru and Zr was 10-15%. Selenium, Ag and Cd were 2%; Sr and Br were found few hundreds to some tenths percent in STEP-1&2.

Only Cs, Ru and I were found in STEP-3. Aerosols in STEP 3 & 4 were only iron and silicon with structure materials. The brief summaries of these tests are shows in Table 6 (OECD/NEA, 2001; Herceg *et al*., 1988).

LOFT: The LOFT test facility simulated the 4-loop PWR with core of the 55 MWt, primary coolant system and Emergency Core Coolant System (ECCS). In the beginning from 1976 to 1978 LOCA test was conducted. The LOFT LP-FP-2 was conducted at Idaho National Laboratory on loss of Fluid Test (LOFT) facility on July 9 1985. LOFT LP-FP-2 test was the last test designed to simulate thermal hydraulics and core uncover conditions. This experiment simulated an accident in which FPs are released both form fuel to clad gap and from the fuel matrix itself. This test was performed at 11×11 assemblies for under laying the study of severe fuel damage and FPs release. In LOFT LP FP 2 tests the cladding temperature were observed greater than 1400 or 1500 K. The LOFT LP FP-2 test demonstrates the oxidation of zircaloy escalates rapidly at temperature 1400 K (Vietti-Cook, 2009; NEA / CSNI / R, 2010) .

Power bust facility-SFD/ACRR-ST: Four tests were conducted between 1982 to 1985 at the Idaho National Engineering Laboratory (INEL) on power bust facility. The objectives of these tests were to investigate the fuel damage frequency, FPs release, transportation, hydrogen generation and to determine the behavior of irradiated and fresh fuel rods under severe fuel damage. The SFD test

Table 6: Brief summary of CORA test (Ott, 1997)

Test no.	Date	Objective	Results
CORA-16	24 Nov 1988	Effect of B_4C-stainless steel observer material, height heat up rate (1ºC/s)	Interaction between B_4C and stainless steel at 1200ºC distribution of the melt leads to liquefaction of Zircaloy fuel rod cladding and dissolution of UO_2 pellets. Relocation of melt to lower part of the bundle in absorber and fuel rod region
CORA-17	29 June 1989	Additional damage progression during quenching, high heat-up rate (1ºC/s)	Analogous behavior as in the PWR test, CORA-12 fragmentation preliminary strong temperature increase before final cool down combined with a strong increase in H_2 productions
CORA-18	21 June 1990	Damage initiation in a larger bundle (48 rods instead of 18), high heat-up rate (1ºC/s)	same general behavior as in the small bundles
CORA-28	25 Feb 1992	Effect of pre-oxidation, high heat-up rate (1ºC/s)	Different material behavior in upper and lower parts of the bundle In the pre-oxidized upper bundle, temperature escalation is delayed and reduced Pre-oxidation reduces materials interaction Reduced melt formation and relocation. More of the B_4C absorber remains than in previous tests
CORA-31	25 July 1991	Effect of slow initial heat-up (0.3ºC/s)	Same general damage behavior as in the bundles heated at 1˚C s^{-1} Increased materials interaction Less relocation of melt Axial temperature profile shifted up by 100 mm
CORA-33	1 Oct 1992	Effect of 'dry' core atmosphere (i.e., minimal steaming) with slow initial heat-up (0.3ºC/s)	No temperature escalation as a result of limited steam input. Melt relocated to much lower elevations than in previous tests, as a result of the more prototypic axial temperature profile

Table 7: A summery of STEP tests (OECD/NEA, 2001; Herceg *et al*., 1988)

Test no.	Accident type	Pressure (MPa)	Test type	Comments
Step-1	LOCA with ECCS failure	0.32	PWR	Ninty percent cladding oxidation, hydrogen generation. Major FPs releases, max temp 2900 K
Step-2	Failure of HPECS and LTDHRS	0.14-1.24	BWR	Seventy percent cladding oxidation with major FPs release, max temp 2700 K
Step-3	Feed water failure with SBO	8	PWR	Fourty percent hydrogen generation. Max temp 2200 K
Step-4	Ssame as step-3 but with Ag-In-Cd rod	8	PWR	Tthirty percent hydrogen generation, max temp 2200 K

LOCA: loss of coolant accident; ECCS: emergency core cooling system; HPECS: high pressure emergency cooling system; LTDHRS: long term decay heat removal system; SBO: station black out

examines the fuel behavior, hydrogen generation and transportation of fission products.

Annular core research reactor source term ACRR-ST experiment was performed at U.S. Nuclear Regulatory Commission between 1986 and 1989. In-pile heating method was adopted for the understanding of FPs release phenomenon and interactions to improve the modeling for reactor safety. The objective of the source term program was to categorize the FPs in LWRs, core debris formation and ceramic's particles (UO_2-ZrO_2). Two ACRR-ST tests were conducted and focused on the FPs release rate. The experiments address the basic mechanism of the blocking accident and as occurred in TMI2. In spite of this, some more important information was also obtained during the test like fuel forming, penetration of molten zircaloy into the fuel cracks (OECD/NEA, 2001).

Out of pile experimental facilities: The out-of-pile experiments include: the CORA, QUENCH and CODEX integral programs and annealing experiments conducted at the Oak Ridge National Laboratories (ORNL) (i.e., the Horizontal Induction (HI) and Vertical Induction (VI) test series), Commissariat à l'Energie Atomique (CEA) (i.e., the HEVA and VERCORS test series) and the Verified Experiments of radionuclide's Gas/Aerosol release (VEGA) program at the Japan Atomic Energy Agency (JAEA). In CORA test, electric heating method is adopted to study the temperature behavior of core melt progression. The annealing experiments HI VI and HEVE and VELCORE were designed to investigate the fission products release form spend fuel under hydrogen steam and air condition. The VEGA test investigates the fission product release and behavior in PWR, BWR and Mixed Oxide fuel (MOX) at the high temperatures at pressure range between 0.1 to 1.0 Mpa (Kudo *et al*., 2006). Some of them focused on the ST are described in this part of study.

CORA: The CORA facility at Kernforschungszentrum Karlsruhe (KfK) was especially designed for the Severe Fuel Damage (SFD) experiment's core degradations phenomena's and the facility simulates the decay heat of the fuel rods by electrical heating using central tungsten heaters. CORA-33 experiment is very important to understand the BWR core degradation because it is probably the only experiment to be conducted to understand the core dry out degradation. Before, all BWR fuel damage tests were conducted with wet core conditions and the dry core condition was neglected experimentally. Transition-phase models have been tested in the specific analyses of the CORA BWR experiments and have been implemented at ORNL in the SCDAP/RELAP5 code for the BWR canister and control blade structures. The electrically heated CORA tests (0.2 to 1 MPa) are well to understand the fission product release (Hobbins *et al*., 1991). In high pressure CORA-9 test full length clad rapture was observed while the flowering behavior of fuel was similar to the other low pressure tests (Hofmann *et al*., 1997). But results of these test showed that the ballooning of cladding is enhanced at low pressure. Six BWR experiments have been conducted in the CORA facility since 1988 to until now. The fundamental studies of CORA test are core damage propagation, melt through and reflooding condition (Ott, 1997). The Brief summaries of the tests are shown in Table 7.

VERCORE test: The French Nuclear Protection and Safety Institute (IPSN) launched the HEVA-VERCORS program in 1983, in Collaboration with Electricite´ De France (EDF). VELCORE program is the extension of HEVA program with higher fuel temperature (2600 K). Aim of this program was:

- To improve and validate the models used for the FPs transportation during SAs.
- To create the data bank computing the FPs behavior in the primary circuit.

The VERCORS RT-6 test was performed at height burn up fuel (60 GWdt/U) from the Gravelines nuclear power plant having a burn up of approximately ~55 MWd/tone. VELCORE 1 was performed at 2150 K with mixed hydrogen and steam. VELCORE 2 was performed at the temperature 1070 to 1770 K to quantify the fission gas and low volatile fission products. The other four experiments were performed at the higher temperatures. The RT-6 test was run under oxidizing conditions. This data is used to develop release rate parameters for diffusion release models (Ducros *et al*., 2001). The summary to VELCORE fission product test is shown in the Table 8.

Table 8: A summary of velcore tests (Ducros *et al*., 2001; Akihide *et al*., 2002; Pontillon and Gerard, 2010)

Test	Date	Test condition	Comments
VELCORE 1	11-1989	Max temp:2030 K mixed H_2O and hydrogen	Same level as HEVA test, FPs release fraction is form 20 to 40%, respectively
VELCORE 2	06-1990	Max temp 2150 K, same as VLECOR 1	Fifteen percent release fraction, same as VELCORE 1 but have more oxidation condition
VELCORE 3	04-1992	Max temp 2570 K	
VELCORE 4	06-1993	Max temp 2570 K, hydrogen	Fourty two release fraction, under oxidation Mo was found 47%. (45 and 80%) of rhodium and barium released
VELCORE 5	11-1993	Max temp 2570 K, steam	Ninty two percent release fraction, Mo 92% released under oxidation, 20 and 55% rhodium and barium, respectively
VELCORE 6	09-1994	Max temp 2620 K mixed H_2O and hydrogen	Seventy nine percent release fraction conducted at higher burn up but no significant effect is found on low volatile Fps

Table 9: A summary of HI and VI tests (Osborne *et al*., 1984; Osborne *et al*., 1991)

			Fraction of inventories (%)				
Test no.	Temperature (ºC)	Specimen reactor	Kr-85	Cs-137	I-129	Sb-125	Ag-110m
HI-1	1400		3.13	1.75	2.04	0.018	0
HI-2	1700		51.8	50.5	53.0	1.55	3.13
HI-3	2000	robinson	59.3	58.5	35.4	0.001	0.015
HI-4	1850	peach bottom	31.3	31.7	24.7	0.009	0.097
HI-5	1700	oconee	19.8	20.8	22.9	0.315	18.07
VI-1	1750-2020	oconee	57	65	33	28	33
VI-2	2030	BR3	>31	63	70	68	8.9
VI-3	1730-2430	BR3	100	100	C	99	ND
VI-4	2165	BR3	85	95	68	3.9	ND
VI-5	1730-2450	BR3	100	100	C	18	ND
VI-6	2035	BR3	C	C	C	C	C

ND: Not detected; C: Not known

VEGA: Verified Experiments on Gas/Aerosols (VEGA) release were initiated at the Japan Atomic Energy Agency (JAEA). Ten heat up VEGA programs on radionuclide release from irradiated fuel under severe accident conditions were performed from 1999 to 2004 under inert and steam atmospheres, including the highest pressure or temperature conditions. VEGA facility can be pressurized up to 10.0 MPa. This program was conducted to study the FPs behavior from the MOX fuel under high pressure and temperature. Akihide Hidaka has enlisted the VELCORE 1-6 test results and compared the release rate between PWR, BWR and MOX fuels (Akihide, 2001; Akihide *et al.*, 2002).

HI and VI tests: For accurate information about the FPs release under severe accident in PWR (U.S.NRC, 2000) conducted Horizontal Inductance (HI) and Vertical Inductance (VI) tests at ORNL. The fuel samples of (100-200 g) 15-20 cm long with zircaloy clad is used with irradiation form typical LWRs burn up. These tests were conducted under the atmospheric conditions of temperature 1700-2700 K in the induced furnace. The mains differences between HI and VI tests are:

- The HI are horizontal while the VI were oriented vertically.
- The fuel Burn up in VI is higher than the HI tests.
- VI tests were conducted at higher temperature (2300-2700 K).

The prime objectives of these tests were:

- Release rate determination chemical form of FPs.
- To compare the date with other test results.
- To study the microstructure of fuel and cladding.
- To require data of FPs release from high burnup fuel under SAs in PWRs.

Five HI tests and six VI tests have been conducted, until now. The summary and results of these tests are listed in Table 9. The measurements made in these tests includes:

- Test sample versus time
- Collection of condensed vapors
- Measurement of fission gas
- Radiation detection and measures

From these measurements, the similar release of noble gasses, I and Cs has been observed however a difference in transportation has been observed. Cs vapors were found in hydrogen conditions but the aerosols in steam conditions. Te and Sb had the same release rate but delayed due to the oxidation of Zircaloy (Osborne and Lorenz, 1992).

In all HI test significant release of FPs Kr, I and Cs occurs and more variable fraction of Mo, Ru, Cd, Sb, Ce and Eu were also detected. In addition, some structure material and impurities like Zr, Sn, Mg, Ca S, Cl, Pb and

Bi were also observed. In all tests, I was detected at temperature range 673-873 K. From these test, it was clear that FPs e.g., Cs, I, Kr, may be released within temperature of 1973-2273 K.

In VI tests, highly irradiated zircaloy clad UO_2 fuel was tested at 2000-2700 K in both steam and hydrogen atmospheres. Similar behavior of I-134 and Kr-85 was observed in both conditions. One hundred percent release was observed at 2700 K. In test VI-3 and VI-5 Sb release was 99% in steam but 18% in hydrogen condition, whereas Eu was barely detected in steam atmosphere but 57%, release was found in hydrogen condition. These test shows that Volatile FPs releases at the high temperatures, but a significant difference was observed in the release fraction of Sr, MO Ru, Sb and Eu at different atmospheres (Osborne *et al*., 1984; Osborne *et al*., 1991).

COMPARISON OF EXPERIMENTAL PROGRAM

Various physical and chemical processes were studied in these tests. The results obtained at different pressures only depend upon the accident scenario and melt progression as evidenced from tests (i.e., Phebus FPT-0, FPT-1, PBF SFD, LOFT FP-2 and Cora test) (Hobbins *et al*., 1991). The gap release inventory was measured in PBF-SFD and Phebus FP test. While the absorber material (control rod) degradation and melting have been observed in Out of pile experiments. The Ag-In-Cd alloy rod melts at temperature range between 1070 to 1120 K. These experiments results the chemical stability of molten rods with inconstant to stainless steel at 1420 (Petti, 1989; Hofmann, 1999). As from the CORA test, the molten material can contact to zircaloy guided tube and chemically dissolve in it Hofmann (1999). In Phoebus FPT-0 and FPT-1 the control rod failure occurs at 1390 and 1620 K respectively as consistence with low pressure scenarios.

Metallic melt blockage has been observed in the early in-pile experiments (PBF-SFD, LOFT FP-2 and DF tests) and in out of pile CORA tests. From these tests, the freezing temperature of melt range from 1070 K for Ag-In-Cd alloy rod, 1220 K for Zr-Fe eutectic, 1230 K for the Zr-Ni for elemental silver and 1460 K for Zr-Ag eutectic (Hobbins *et al*., 1991). The blockages formed in the different experiments were similar to the blockage formed in TMI2 accident (Wright *et al*., 1992). In comparison the materials Zr (20-40% wt), Ag (10-50 wt%), U (<15 wt%), O<10 wt%) and stainless steel shows the interaction with Zircaloy cladding with molten Ag-In-Cd alloy rod. Hanniet and Repetto). This indicates the control rods play an important role in debris formation with early melting under accident conditions. The hydrogen generation is observed in Phebus FPT-0 and FPT-1. The zircaloy oxidation and hydrogen generation observed in PBF-ST and LOFT FP-2. For instant 45% core melt was measured in TMI2 accident while (wolf *et al*., 1994) 15-18% in PBF-SFD test and 15% in LOFT FP-2 teat (Hobbins *et al*., 1991), While the phebus PFT-0 and FPT-1 are more severe in which 20 and 50%, liquefaction of fuel is observed (Hanniet and Repetto, Bonnin *et al*., 1997). The observed fuel damage on FPT-0 is more consistences with the fuel damaged at TMI2 accident. The FPT-0 test has the average composition of U (62 wt%), Zr (22 wt%) and O (14 wt%) (Hanniet and Repetto). The upper debris was observed in SFD-ST test (Knipe *et al*., 1986) as was formed in TMI2 accident.

CONCLUSION

This study describes the brief research work on the source term and historical of advancement of the source term based on the previous invented phenomenon. Some important source term experimental programs are discussed in short, which results are used for the validation and verification of computer-based programs up to now. The results of these tests are better to understand the release behavior of fission products in different atmospheric conditions and helpful to mitigate the severe accident release.

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