Uncertainty in Source Term Assessment Analysis for Thorium Fuelled Reactor under Postulated Accident

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INTRODUCTION

Source Term determination under normal operating and accident conditions is one of the important safety aspect of nuclear reactors as “Source Term” strength directly influence the plant and public risk. The general information is used for licensing the reactor and accident management purposes. The source term of a nuclear reactor depends on several design aspects of the reactor. The fuel composition and enrichment, burnup and specific power decide the initial core inventory of fission products and actinides. The release of fission products from the fuel to the coolant is governed by the accident type where fuel temperature plays an important role. Further several mechanism of retention of fission products in coolant system piping and containment surface and leakage paths governs the release to environment.

Study has been carried out to assess the fission products and their activities release to atmosphere during different accidental conditions for Advanced Heavy Water Reactor (AHWR), a thorium fuelled and natural circulation driven reactor. As Source Term estimation depends on the reactor type, its layout, the study encounters several uncertainties in the process of fission product release and transportation post accident. Various uncertainties in Source Term estimation for a reactor depend upon several factors like:

1. Uncertainties in equilibrium incore inventory arises due to calculation based upon fuel average enrichment, average burnup and average exposure time instead of three rings actual fuel composition and burnup and interaction cross-section library for AHWR specific fuel;

2. Uncertainties in radionuclide release from fuel arises due to non-availability of suitable diffusion coefficient model, and lack of fuel grain properties like grain sizes, shape, and grain boundary sweeping behaviour information;

3. Uncertainties in radionuclide transportation (retention and release) within reactor main heat transport (MHT) system arises due to various reasons like: lack of released radionuclides nature (aerosol, or molecular) information and their chemical composition; lack of information about generated radionuclide aerosol properties like shape, size distribution, density and actual injection rate; due to lack of thermal hydraulic parameters like pressure, temperature, flow rate and environmental conditions; simplification in simulation of complex geometry like fuel bundle, end fitting and steam drum regions.

4. Uncertainties in radionuclide transport within reactor containment mainly arises due to uncertainties in the radionuclide chemical compositions, chemical form of iodine (aerosol to elemental ratio), steam and reducing atmosphere with hydrogen, plate out and partitioning factors as well as on the leakage rates from containment under design pressure and beyond design pressure.

The uncertainties in the physical model describing the process and applicability of models toward the reactor systems are addressed in this paper. Along with that recommendations of the regulatory boards (international and national) towards resolving these uncertainties are also being discussed.

As an example, a severe accident case of AHWR has been discussed in the paper without any mitigation measures. A large break LOCA with un-availability of all heat sinks is being considered which may yield maximum fission product release. As fission product release to environment through leakage paths (major contributor) depends upon the extent of pressurization in the containment, hence the containment thermal-hydraulic conditions, amount of combustible gas (hydrogen) presence and released fission product decay power are the major contributors to the containment pressurization. The aim of selection of the mentioned scenario is to maximize the contributor towards pressurization and hence maximize the source term.

RESULTS AND DISCUSSIONS

Multiple physics multi-step calculation methodology mentioned in Gokhale et al. [1] is adopted for estimation of AHWR Source Term. Based upon the average fuel composition of heavy metal per tone (4.3% of
$^{235}\text{U} + 17.477\%$ of $^{238}\text{U} + 78.22\%$ of $^{90}\text{Th}$ and average burnup of 30,000 MWD/t and 980 MW reactor thermal powers; equilibrium incore inventory, activity and decay power is calculated using ORIGEN-2 code (Allen G. Croft [2]). The total equilibrium incore inventory and decay heat are shown in Fig. 1 and Fig. 2 respectively.

The predicted fuel transient temperature just exceeds the fuel failure criteria limit of 800°C. Using in house developed code “FPREL_EMP” FP release form fuel is calculated based upon prevailing fuel temperature and environmental conditions. It consists of instantaneous gap inventory release followed by diffusion controlled release from fuel matrix. Fig. 3 shows the fuel average temperature and volatile FPs release into the MHT.

Once fuel fails, FPs released into reactor Main Heat Transport (MHT) system. These released fission products undergo different chemical species formation which flow in the coolant channels and piping of MHT system. Due course of time, it gets retained (except Xe and Kr) within the long and colder sections of MHT piping and non-retained part gets released into reactor containment through break path. The transport of fission products into MHT are dictated mainly by MHT geometry, and system prevailing thermal hydraulic conditions like pressure, temperature, and flow rate and wall temperature conditions. Code ASTEC (Accident Source Term Analysis Code) is used for FPs transport and retention within MHT system (Cousin et al. [3]) and ASTEC specific nodalisation is shown in Fig. 4. An order of ~94% and ~90% are found for Iodine and Cesium respectively and retention mechanisms are mainly due to thermophoresis and bend impaction physical processes.

Within reactor containment, Iodine and Cesium are mainly retained due to plate out in the leak paths and water trapping. Iodine is mainly present in form of CsI due to reducing environment from hydrogen generation. A combination of plate out and water trapping from steam environment results in a 40% retention within containment.
Based on the actual observations the leakage rate tests conducted for the containments of PHWRs, a conservative value of 0.3% containment volume/hr at a peak pressure of 0.261 MPa has been assumed for analysis of leakage of F.Ps from Primary Containment (PC) to Secondary Containment (SC). A 10 % of primary containment to secondary containment leak rate has been assumed to be directly released to the environment. A 0.6 % of total containment volume (Primary Containment volume + Secondary Containment volume) per hour at 200 mm of water column is considered to be released directly to the environment from secondary containment.

The estimated source term for environment for important radionuclide isotopes as percentage to the core inventory are listed in the Table 1.

<table>
<thead>
<tr>
<th>Species</th>
<th>Inventory released into MHT</th>
<th>Inventory released into containment</th>
<th>Inventory released into environment (Kg)</th>
</tr>
</thead>
<tbody>
<tr>
<td>I\textsuperscript{131}</td>
<td>0.19%</td>
<td>0.01%</td>
<td>0.005%</td>
</tr>
<tr>
<td>Cs\textsuperscript{137}</td>
<td>0.59%</td>
<td>0.035%</td>
<td>0.03%</td>
</tr>
<tr>
<td>Kr\textsuperscript{85}</td>
<td>0.35%</td>
<td>0.35%</td>
<td>0.35%</td>
</tr>
<tr>
<td>Xe\textsuperscript{133}</td>
<td>0.36%</td>
<td>0.36%</td>
<td>0.36%</td>
</tr>
</tbody>
</table>

CONCLUSIONS

The paper addresses different uncertainties in the physical processes involved from generation of fission product to its release to atmosphere during an accident case. As a case study a large break LOCA with unavailability of ECCS is presented with realistic approach which shows an extremely low percentage of Fission Products release into atmosphere. The order of retention in MHT and containment are very high, however the uncertainty of this retention process are not well quantified as of today.

REFERENCES