Long Term Effects and Numerical Simulation of Radiolytic Gas, Non-Condensable Gas and Boron Transport For Small Modular Light Water Reactors

By

S. Lu, C. Thurston, S. I. Haider, A. Barrett, J. Staudenmeier, P. Yarsky, S. Bajorek
U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001

Abstract

In the past decade, nuclear industries and governments worldwide have developed interests in designing and deploying small modular reactors (SMRs) as viable energy source options to reduce carbon dioxide emissions and help resolve climate change issues. According to the International Atomic Energy Agency (IAEA), there are approximately 70 small modular reactor designs under investigation in 17 countries. These new SMR designs are at different stages of research, development, licensing, and commercialization. The collaborations coordinated by IAEA and driven by major industrial countries have put several SMR designs as the front runners in this phase of technology commercialization.

Some of these SMR designs under development have evolved from large light water reactor (LWR) designs and use light water as both a coolant and neutron moderator with passive gravity driven systems for normal operation and accident mitigation. The design goals of these passive safety systems are to maintain reactor core cooling for at least 72 hours without any on-site or off-site power supply or operator intervention for a broad range of hypothetical accident scenarios including loss-of-coolant accidents (LOCA) and station blackout. These SMR designs normally have significantly higher coolant inventory/reactor power ratios than that of conventional large LWRs and are expected to keep the reactor core covered under a two-phase water level, or, at a minimum, preclude the fuel from experiencing prolonged heat-up for at least 72 hours. The reliance of such passive SMR designs on gravity driven buoyancy flow and natural circulation for their long term Emergency Core Cooling System (ECCS) operation makes it possible to eliminate the need for higher cost, active pumping systems, simplify the containment design, and reduce the initial capital investment. The gravity driven ECCS designs can be reliable because of their reliance on inherent features and natural phenomena. However, the gravity driven buoyancy flow and natural circulation change the system mass and energy distribution during the long term cooling period after the initial transient. Depending on the design, the systems may become sensitive to some physical phenomena, such as (1) radiolytic gas generation and migration, (2) non-condensable gas effects, and (3) boric acid transport if used in the primary circuit to control reactivity. These phenomena were not considered to be significant concerns for most current LWR designs utilizing active ECCS, so they were not explicitly modeled in detail historically. Detailed evaluation of the accumulated effects of these phenomena may become necessary for passive LWR SMRs designs as part of design basis analyses.

In this paper, the authors summarized the information on these three phenomena and identified their potential safety implications for passive LWR SMRs. The state-of-the-art computer simulation tools commonly used by both the industry and regulatory agencies are discussed for their applications to evaluate the accumulated effects of these phenomena, including the challenges, limitations of these computer codes and future development needs.

Introduction

Small modular reactors (SMRs) with power outputs between 10 megawatts electric (MWe) and 300 MWe have been gaining attention across the world since 2010 because of their potential simple and passive safety features, modularity, significantly reduced construction cost per MWe capacity, and promising fast deployment schedule (Ref. [1,2]). Approximately 70 small modular reactor (SMR) designs are under investigation in 17 countries. All of them are at different stages of research, development, regulatory review and commercialization. With goals of commercial operations of at least some of these new SMRs by 2030, reactor vendors are accelerating the development of these new design concepts. Among these SMRs under development, several design concepts evolved from the large light water power reactors (LWR) currently in operation. The most significant improvement by these LWR SMRs is the use of gravity driven natural circulation during normal operation and passive Emergency Core Cooling System (ECCS) during accidents or transients. Because of the use of a passive ECCS and the relatively large amount of water inventory inside a SMR reactor vessel, the reactor is expected to be free of fuel cladding heat up during a loss of coolant accident (LOCA) for at least 72 hours without operator intervention or safety grade power supplies. The gravity driven ECCS designs can be reliable because of their reliance on inherent features and natural phenomena. However, the gravity driven buoyancy flow and natural circulation change the system mass and energy distribution during the long term cooling period, e.g, 72 hours after the initial transient. Depending on the design, these kinds of passive LWR SMRs may become sensitive to three well known physical phenomena (1) radiolytic gas generation and migration, (2) non-condensable gas effects on condensation heat transfer and (3) boric acid transport if used in the primary circuit to control the reactivity. These phenomena and their safety consequences have been well handled by current operating LWR technologies. However, the accumulative distribution and transport of these species may cause certain new safety implications which need to be appropriately accounted for to ensure the safe operation of these LWR SMRs.

This paper provides a highlight of the available public domain information regarding the radiolytic gas generation and migration, non-condensable gas effects, and boric acid transport. Based on the existing public domain information, the potential safety implications to LWR SMRs are identified. The numerical analysis capabilities in support of evaluating these phenomena are briefly reviewed and the future development needs are discussed.

Radiolytic Gas Generation and Transport

Radiolysis of water occurs during both normal operation and accidents and involves the decomposition of water molecules by ionizing radiation causing a molecular break sequence into hydrogen peroxide, hydrogen radicals, and other assorted oxygen compounds (Ref. [3]). The rate of hydrogen and oxygen generation is controlled by three factors: (1) decay heat energy or fission power (Ref. [4]) (2) fraction of energy absorbed by the water, and (3) effective rate of hydrogen oxygen production per unit of energy absorbed by the water, generally expressed as the product "G" value.

Traditionally, major concerns regarding hydrogen generation were related to static or dynamic pressure loads from combustion in the containment that could initiate a release of radioactivity to the environment and potential damage to safety-related equipment (Ref.[5]). In recent LWR SMR designs, passive systems and configurations are being employed to (1) limit LOCA mass and energy releases (2) cool and depressurize the reactor and (3) limit and reduce the containment pressurization. However, these newer designs, because of their passive nature, must be analyzed for significantly longer periods of time, up to 72 hours or beyond. These extended analysis times were not considered nor needed for the traditional pumped ECCS injection systems. Therefore, many new phenomena related to the effects of long-term reactor cooling and generation of radiolytic gases become important and must now be considered.

In a PWR closed system, due to recombination, the system eventually attains equilibrium with respect to radiolytic decomposition during normal full power operations. The concentration of gaseous products at equilibrium is a function of the reactor power, water pH and temperature, and the concentration approaches zero, with net production of only very small quantities of hydrogen dissolved in borated water. In a BWR open water/gas system, the product species are being continuously removed via steam through the steam line and then inventory is replenished by new feedwater. In conditions where water is boiling vigorously, H_2 and O_2 could be produced in stoichiometric portions based on applicable “G” values for pure water exposed to primarily gamma radiation. As a result, operating BWR plants have extensive off gas systems to process and control release of radioactive effluents. Additionally, these systems have been used to measure volumetric radiolytic gas production rate of hydrogen and oxygen in a proportion that is found to correlate with reactor power level. However, it is not known how well these measurements taken with traditional jet-pumped forced recirculation plants and whether they are applicable to the newer BWR plant designs that are based on reactor vessel chimney with natural circulation flow. Natural circulation designs with lower recirculation rates, require a larger percentage of new feedwater, as compared to the forced recirculation designs.

Most of the proposed new passive LWR SMR ECCS designs heavily depend on the efficiency of steam condensation to cool the reactor and/or containment for the evaluation of design basis events including accidents and anticipated operational occurrences (AOOs). The new passive ECCS and containment designs, with consideration of extended periods of operation, rely on condensing the steam generated during events to cool and depressurize the reactor system and containment. These designs must consider the generation of radiolytic gases under steady state normal operation and post-accident decay heat, and how they may propagate to the condensing surfaces used and their relative effect on the overall condensation rate. The initial presence and migration of any additional non-condensable gases in the containment atmosphere should also be considered with respect to impact on condensation heat removal capability during the event.

Depending on the geometry selected for the condensation surfaces, the propagation and accumulation of radiolytic gases can potentially collect in tubes or cold surfaces, impeding the condensation of steam and potentially significantly degrading the heat transfer efficiency of the ECCS component. Radiolytic gases are generally transported with steam, and after condensation, are left behind and collected around the condensation surfaces. If the ECCS is unable to efficiently remove heat from the reactor due to the accumulation of non-condensable gases, inadequate core cooling, containment pressurization and a host of other problems can quickly occur.

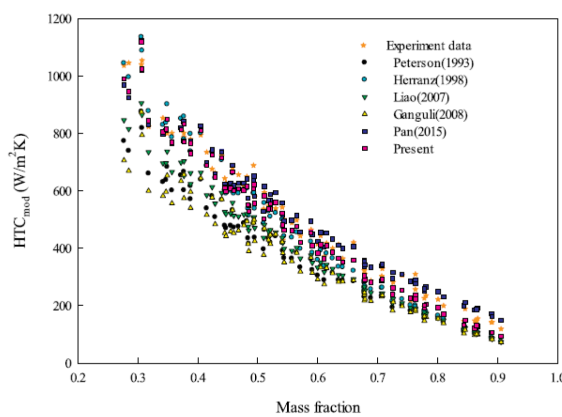
Transient radiolytic gas accumulation leading to combustion of hydrogen is also a major safety concern since high pressures generated could breach containment or damage other important safety-related equipment resulting in release of radioactivity. The requirements for combustible gas control are well established in U.S Code 10 CFR 50.44 as long as the source and propagation of radiolytic hydrogen and oxygen generation and transport in steam are adequately modeled. Adequate measures as prescribed should be taken to mitigate or account for the accumulation of non-condensable gases to avoid combustible limits. The hazards of hydrogen buildup leading to deflagration and detonation has been realized worldwide as results of the TMI-2 and Fukushima accidents. The generation of hydrogen from radiolysis can have the same detrimental effect if not adequately considered and controlled.

Therefore, it is imperative that the designs of these new passive systems adequately account for operational and post-accident generation of radiolytic gases, their initial presences, and their transport during AOOs and accidents.

Non-Condensable Gas Source and Transport

In a gaseous mixture, condensation of one component vapor may occur in the presence of other noncondensable gas components. Vapor condensation in the presence of noncondensable gases (NCG) is of great interest to several engineering, industrial, and environmental applications. It is well established that the presence of NCG leads to a reduction in vapor condensation and heat transfer. Sparrow et al. (Ref.[6]) explained the mechanism by which NCG reduces condensation. They showed that the vapor drawn to the condensing surface entrains with it the NCG that is incapable of crossing the gas-condensate interface, and accumulates near the condensing film and, thus, suppresses the vapor condensation. Hereunder the relevant details of the physical phenomenon are provided.

Considering the component gases of the mixture to be independent, condensation of one component vapor will occur if the condensing surface temperature is below the saturation temperature of the pure vapor at its partial pressure in the mixture, i.e., the dew point. During film condensation, condensation occurs at the interface of a liquid film on the wall. Due to the condensation process at the gas-condensate interface, there is a bulk movement of the gaseous mixture toward the wall, as if there were suction at the interface. As only the vapor is condensed, the NCG concentration is higher at the condensing film interface than its value in the ambient. This, in turn, reduces the partial pressure and saturation temperature of the vapor at the interface below the ambient values. At equilibrium, the NCG concentration at the interface is dictated by the balance between the NCG mass transfer away from the interface due to diffusion and/or convection and the vapor mass transfer from the bulk to the interface to sustain the condensation. The resulting accumulation of NCG and the depression of temperature at the interface reduces the condensation heat transfer rate across the liquid film below what would result for pure vapor under the same conditions. The experimental database summarized on the following figure from Ref.[7] clearly demonstrates the reduction in condensation heat transfer coefficient with the increase of NCG mass fraction in the gas mixture.



Ref. [7] Figure 1(b) Variation of the condensation heat transfer coefficient with non-condensable gas mass fraction.

In nuclear applications, vapor-NCG mixtures exist in the reactor containment and condensers following an accident and the reactor design may account for their active or passive removal. The NCG mixtures in nuclear applications typically include radiolytic gases, nitrogen, and air. Condensation with NCG occurring in the containment and reactor systems plays a key role in designing the heat removal systems in LWR SMRs. The adverse effects of NCG on condensation and saturation temperature depression can be minimized with good condenser design practices that include proper management and venting of the NCG. The degree of adverse impact of NCG on heat transfer also depends on the dominant flow regime, among other factors. Free convection flows are less efficient in sweeping the NCG away from the gas-condensate interface that leads to a significant reduction in condenser performance, while the NCG effect is less severe in forced convection flows that dampen the formation of large NCG concentrations at the interface. Some condensers are designed to perform in the presence of NCG.

A theoretical analysis of condensation from vapor-NCG mixtures using a stagnant film model was first presented by Colburn and Hougen (Ref.[8]). In that analysis, the overall thermal resistance between the condenser tube wall and the bulk vapor-NCG mixture is the sum of the thermal resistances of the condensate film and the NCG boundary layer. The heat transfer through the gas boundary layer includes the sensible and latent parts. The latent heat transfer is evaluated by using a stagnant film model combined with the heat and mass transfer analogy. These calculations involve the convergence of the unknown temperature and NCG mole fraction at the interface through matching the condensate mass flux and the heat transfer through the condensate film. Although Colburn and Hougen did not derive an expression for the NCG boundary layer conductance, they suggested that the mass transfer coefficient in the stagnant film model could be converted to a heat transfer coefficient by using the Clausius-Clapeyron equation, for saturated mixtures. Using this principle of heat and mass transfer analogy, Peterson (Ref. [9]) proposed the diffusion layer model to calculate the latent heat transfer. The theoretical basis underlying the heat and mass transfer analogy is that the conservation equations for mass, momentum, and energy have similar mathematical forms.

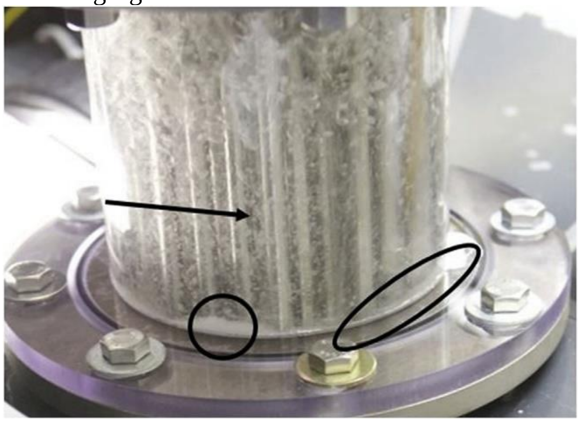
Both the stagnant film model (Ref.[7]) and diffusion layer model (Ref. [9]) neglect the longitudinal flow acceleration that can be accounted for by numerically solving the boundary layer equations for the gas and liquid phases (Ref.[9]). Neglecting the longitudinal flow acceleration makes it possible to obtain an analytical solution (Ref. [6,7]) that is simple to use in engineering applications and provides results not far from the boundary layer analysis. The stagnant film model and the diffusion layer model were originally formulated on a molar basis. Liao and Vierow (Ref. [8]) developed a generalized diffusion layer model on a mass basis that accounts for the effect of variable mixture molecular weight across the diffusion layer and fog formation effects on sensible heat. When compared with a wide-ranging experimental database, the generalized model outperforms the one developed by Peterson et al. (Ref. [7]). Under certain limiting conditions, the generalized model reduces to the one developed by Peterson et al..

Boric Acid Transport Phenomenon

Some LWR SMR designs continue the use of boric acid as a chemical shim. Boric acid, like all other water-soluble substances, can experience precipitation, dilution, volatilization, and deposition processes. The transport of boric acid throughout the primary reactor system or containment during an AOO or accident may potentially cause precipitation in the reactor core region and degrade the heat transfer due to the blockages caused by the precipitated boric acid. The resulting dilution of boric acid concentration in the reactor core region may also introduce excessive reactivity, which, if not controlled properly, could cause the reactor returning to power or experiencing uncontrollable power excursions.

Boric acid is volatile during the boiling process of its water solution (Ref. [17]). Although the volatilization rate is relatively small and the boric acid concentration in the steam is of the order of $10^{-3} \sim 10^{-2}$ of the liquid solution concentration, the accumulative removal of the boric acid from the water solution over a long period of time could significantly affect the reactor core boric acid concentration distribution. The volatized boric acid is carried away from the boiling boric acid solution. It then either deposits on the solid surface along the steam flowing path or redissolves into the solution during steam condensation. The boric acid deposition process can remove the available mass of boric acid from the primary circulation system, which would affect the long term boric acid distribution in the active core region. The boric acid deposits in the primary system could impact the operability of the reactor vessel internals, including, but not limited to control rod drive systems, instrumentation, and measurement devices.

After the Three Mile Island Unit 2 accident in 1979, extensive efforts have been made world wide by nuclear industries, research institutes, and universities to study the boric acid transport phenomena which have significant safety implications to PWR designs. Some of the latest studies on boric acid precipitation can be found in Ref. [11]. The crystallization and deposition of boric acid were clearly visible during the experiment as shown in the circled areas in the following figure.

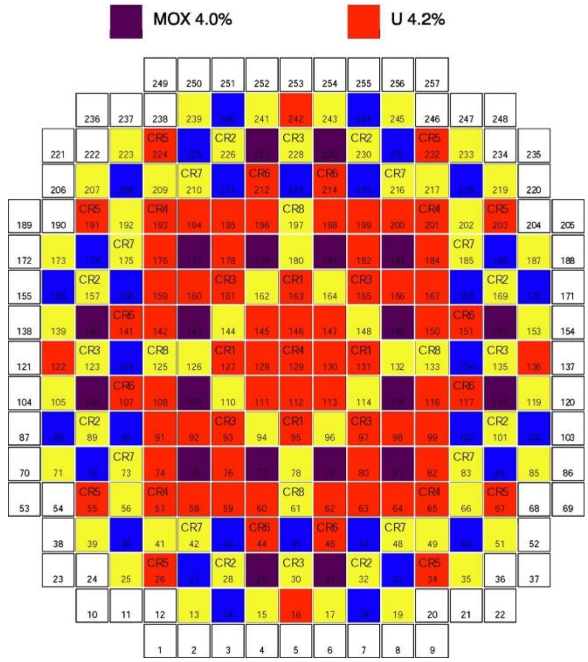


Ref. [11] Fig. 9. Experimental results (end of the test).

The boric acid solubility limit is dependent on the temperature and pressure of the aqueous solution. The lower the temperature, the less the solubility. A literature review of the operating and transfer conditions examined by Ref. [12] include temperatures between 13 ºC (McLeskey, 2008) and 45 ºC (Fondeur, 2007); and concentrations from 0 to 3000 ppm in nitric acid as well as exposure of small amounts of entrained boric acid in the organic phase to the sodium hydroxide caustic wash stream.

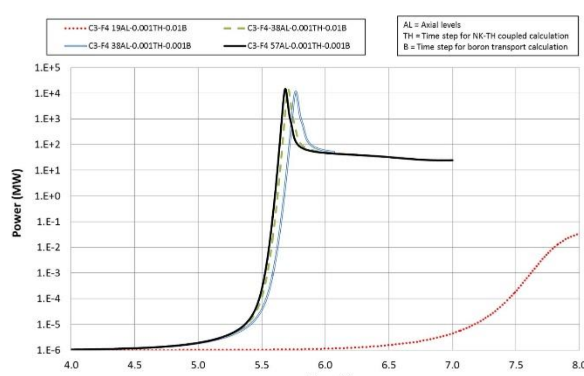
During the 72 hours of post LOCA long term cooling, the core region of a SMR PWR may experience a reduction in solubility due to the continuous decline of the system temperature and pressure. Once the boric acid solution concentration exceeds the solubility limit, white boric acid crystal precipitate in the solution and could accumulate in the core region and cause fuel flow path blockages. Therefore, the core region of the boron concentration of a SMR PWR should be evaluated to show no precipitation is possible during the long term cooling phase of the post LOCA transient.

Boron dilution could happen in a PWR core due to boric acid loss from the fluid region, potentially caused by entrainment, volatilization, and mixing with in-coming pure or highly diluted water. As studied by Argonne National Laboratory in 1995 Ref. [14], the most significant boron dilution event for operating PWRs was found to be the restart of reactor coolant pumps which quickly transport the diluted condensate trapped in the primary system pump loop seals into the core. The sudden core boron dilution could cause recriticality and a return to power. G. Jimenez et al. analyzed a PWR boron dilution transient using the coupled neutronics and thermal-hydraulics codes DYN3D/FLOCAL. Fig.2 of Ref. [15] showed the reactor core fuel and control rod loading.



Ref. [15] Figure. 2. Core configuration and distribution of control rod groups.

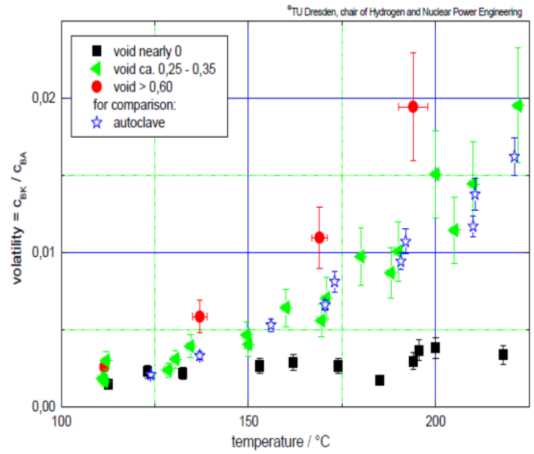
Assuming 18 m^3 of diluted water slug being injected into the reactor core, the coupled computer codes predicted a power increase by 10^{10} times within 5.75 seconds as shown in Figure 4 of Ref. [15]. The power peak was numerically predicted to be stabilized by the negative reactivity feedback due to the Doppler effect. In reality, if a PWR core experiences a 10^{10} power increase starting from the remaining subcritical fission power due to the decay of delay neutron precursors, the reactor core fuel enthalpy and thermal limits could exceed the allowable values and the reactor core could experience severe damage in extreme circumstances.



Ref. [15] Fig. 4. Slug 1 fission power evolution.

A natural circulation PWR, without a pump, however, does not have a mechanism for such a sudden transport of a fresh water slug into the core. If the coolant in the reactor lower plenum or downcomer is diluted, e.g, as a result of evaporation and condensation, this could lead to a slow dilution of the coolant in the core region, depending on mixing at the core inlet. The reactor could reach re-criticality if there is a semi-continuous flow of diluted coolant from the downcomer and lower plenum to the core. The additional fission power could escalate the vaporization and dilution causing either flow instability or sudden level changes due to depressurization, which could lead to a sudden in-flux of diluted coolant into the core and cause the similar power surge as shown in Figure 4 of Reference [15]. Therefore, reactor vendors and regulatory agencies should pay attention to the transport mechanism of boric acid during reactor normal operation, AOOs, design basis accidents, and post accidents mitigation phases. Proper boric acid mixing in the reactor lower plenum and downcomer regions can help to avoid boron dilution induced re-criticality and the potential for power excursions.

The volatilization of boric acid can slowly remove the boric acid from the reactor core region. The accumulated removal over a long period of time can be significant. The volatilization happens at the interface between the vapor and the fluid. The volatilization rate is highly dependent on the vapor generation process in the core region. The flow regimes, the local liquid boric acid concentrations, the vapor bubble rising history, and the system parameters such as pressure and temperature influence the volatilization process. Therefore, there has not been an universal correlation developed to bound all scenarios. S. Bohlke et al., conducted a series of boron volatilization tests and data analysis using a test facility simulating the BWR core boiling processes during a BWR Anticipated Transients Without Scram event with an injection from the standby liquid control system Ref. [13]. As shown in Figure 3 of Ref. [13], the measured boron volatility increases with higher temperature and void fraction. The autoclave data collected by S. Bohlke et al showed much lower volatility than they measured from their test rig with a void fraction greater than 60%. Other autoclave data showed higher volatility for the given temperature Ref. [17]. These different research papers showed that the volatility is normally small within the range of 0.25 ~ 2% and is sensitive to the local fluid conditions at the test rigs. The temperature, local void fraction, and flow regime affect the volatility value. Even at the same test rig, the measured volatility values vary when local void fraction or flow velocity changes. Therefore, if the accumulative loss of boric acid from the core fluid region due to volatility needs to be evaluated for certain reactor design configurations, considerations should be given to the actual core configuration and system parameters.



Ref. [13] Fig. 4. Volatility of boron out of boiling pentaborate solution with a defined boron content (cboron = 1.3 g/L) at various void fractions–measurements with BORAN (BWR ATWS Tests)

Following volatilization, boric acid mixes with the steam and may either deposit on the solid surface or redissolve in the water during the steam condensation process. The gradual accumulation of deposits on the solid surface may affect the operability of some reactor vessel internals, valves or measurement devices. These specific situations need to be evaluated on a case-by-case basis to determine the safety implications if there are any.

Numerical Simulations of Radiolytic Gas, Non-Condensable Gas, and Boric Acid Transport

Radiolytic gas, non-condensable gas, and boric acid in the primary system of light water SMRs are normally carried by either liquid water or vapor. Because of their very low concentrations for most of the scenarios, these species do not normally impact the transport of their carrying media except for the case inside a containment full of non-condensable gas during normal operations. Therefore, the movement of these species is largely determined by their media transport, i.e., liquid and vapor flow. The numerical simulation of liquid and vapor flow in a LWR has been the primary focus of safety analysis code development during the past 70 years. These thermal-hydraulic safety analysis codes are used to model the primary and often the secondary sides of nuclear plants and are called upon to simulate the liquid and vapor flow in a wide range of accident scenarios. The codes currently in use include but not limited to TRAC [18], RELAP [19,28], COBRA/TRAC [20], TRACE [21], TRACG [24], CATHARE [25], ATHLET [23] and MARS [25] for primary systems and GOTHIC [27] and MELCOR [26] for containment analyses. Both GOTHIC and MELCOR codes have been developed to model multiphase, multicomponent fluid flow for performing both containment design basis accident (DBA) analyses, severe accident analyses and equipment qualification analyses.

The general approach for including the non-condensable component consists of assuming that all the non-condensable component present in the vapor-gas mixture moves with the same velocity and has the same temperature as the vapor phase. Based on these assumptions, an additional mass conservation equation is typically used for the total non-condensable component including radiolytic gas in the vapor/gas phase

$$\frac{\partial}{\partial t}(\alpha_g \rho_g X_n) + \frac{1}{A}\frac{\partial}{\partial x}(\alpha_g \rho_g X_n v_g A) = 0$$

(Equation 3.1-39 of Ref. [28])

X_n is the total non-condensable mass fraction in the vapor/gas phase

$$X_n = \frac{\sum_{i=1}^{N} M_{ni}}{\sum_{i=1}^{N} M_{ni} + M_s} = \frac{M_n}{M_n + M_s}$$

(Equation 3.1-40 of Ref. [28])

M_{ni} is the mass of i-th non-condensable gas; M_n is the total mass of non-condensable gas in the vapor/gas phase; M_s is the mass of vapor in the vapor/gas phase; N is the number of non-condensable. The thermal properties of the vapor/gas phase (subscript g) are mixture properties of the vapor/non-condensable mixture. The static quality, X, is likewise defined as the mass fraction based on the mass of the vapor/gas phase.

Similar to the typical non-condensable gas mass conservation approach, an Eulerian boron tracking model is normally used in these system codes to simulate the transport of a dissolved component (solute) in the liquid phase (solvent). The solution is assumed to be sufficiently dilute that the following assumptions are generally made:

- Liquid (solvent) properties are not altered by the presence of the solute.
- Solute is transported only in the liquid phase (solvent) and at the velocity of the liquid phase (solvent).
- Energy transported by the solute is negligible.
- Inertia of the solute is negligible.

Based on these assumptions, the typical mass conservation for boric acid in the liquid can be the following:

$$\frac{\partial \rho_b}{\partial t} + \frac{1}{A}\frac{\partial(\rho_b v_f A)}{\partial x} = 0$$

(Equation 3.1-51 of Ref. [28])

where the spatial boron density, ρ_b, is defined as

$$\rho_b = \alpha_f \rho_f C_b = \rho_m (1 - X) C_b$$

(Equation 3.1-52 of Ref. [28])

C_b is the concentration of boron, X is the static quality and A is the 1-D flow area, ρ_f is the liquid density.

The inclusions of non-condensable gas and boric acid in the mass conservation equations for the two-phase models are typical among the system codes. Finite-volume and finite difference schemes are then applied to solve these two-phase flow conservation equations. Most system analysis codes model relatively large spatial regions and provide only averaged behavior of conditions within those regions. This is primarily what distinguishes nuclear systems codes from codes that perform Computational Fluid Dynamics (CFD, eg, FLUENT code Ref.[29]). A nuclear system code attempts to determine the distribution of mass and energy throughout an entire hydraulic network using large computational nodes, where the highly detailed nodalization can be achieved by CFD to analyze the localized turbulence, vortexes and thermal mixing.

Systems thermal-hydraulic codes are subject to significant uncertainties because of their dependence on large spatial regions and the use of semi-empirical models and correlations to simulate the average behavior of conditions within those large regions. Most of them have been developed to model fast and short system transients, such as a loss of coolant accident (LOCA). Therefore, they have greater prediction uncertainties to deal with gravity or buoyancy driven flow conditions which are important to passive LWR SMR technology. The CFD codes, on the other hand, can simulate boundary-layer mixing and turbulence phenomena. Both CFD codes and system codes are subject to large uncertainties when modeling two-phase flow in a buoyancy force dominant fluid field, where all models for flow regime interfacial phenomena and flow pattern transitions are empirical. Quite often, models and correlations for two-phase flow are derived from experiments with small diameter pipes which do not readily scale to the large flow region in a reactor vessel internal fluid field. Thermal stratification in large pools depends on turbulence and temperature gradients to mix fluids. System codes do not generally have the capability to model the turbulent mixing and thermal stratification due to the lack of detailed nodalization or local phenomenon modeling capability.

Both system codes and CFD codes need to be further developed if bounding analysis approaches are unavailable and analytical tools are desired to replace expensive case-by-case design specific tests. Multifield conservation equations need to be introduced to track different kind of non-condensable gas species. In this way, Hydrogen, one of the radiolytic gas species, can be tracked in a stratified flow regime. In order to track boric acid and predict its localized concentration in a reactor core and primary system, coarse mesh multi-phase CFDs or sub-channel analysis capabilities can be considered.

Conclusions

Radiolytic gas generation, the transport of non-condensable gas and boric acid can impact a light water SMR during normal operation, AOOs and accidents because (1) the gradual accumulation of radiolytic gas may degrade condensation and reach the combustion limit, if not controlled, (2) The presence of non-condensable gas reduces the condensation heat transfer in the reactor system and containment and (3) The boric acid can precipitate in a PWR core to cause undesirable reactor fuel cladding heat-up, and dilute causing re-criticality with the subsequent potential power excursions, if not mitigated. Therefore, proper evaluation of these accumulated effects may become necessary for passive LWR SMR designs. Although significant studies have been done in the past 70 years to develop system codes and computational fluid dynamic codes which have some capabilities to simulate these phenomena, improvements on multi-species tracking and multiphase flow simulation with finer nodalization are necessary to avoid expensive design specific tests.

Disclaimer

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