Study of Iodine Spiking Phenomena in Pressurized Water Reactor in KOREA

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Abstract. When defects occur in the fuel rods of nuclear power plants, the cladding materials no longer serve as a protection between the gap atmosphere of the internal fuel material and the primary coolant. At this point, a leak path in the form of a crack appears and the RCS coolant can thereby enter the gap between the fuel rod and the clad. The fission products in the gap can escape and enter the reactor coolant system (RCS). Various insight has provided a better understanding of the physical processes of activity release during the reactor's operation at steady-state. In a full-power operating state, only a small fraction of the fission-product iodine is released into the RCS.
Most of the released iodine is present as a liquid soluble deposit state on the fuel surface or the inner surface of the cladding. When the temperature of the gap between the pellet and the cladding drops, the clad temperature can drop below the coolant saturation temperature during reactor shutdown. The water that enters the rod remains in the liquid phase and is deposited on the cladding surface. In this study, the behavior of iodine is estimated and explained in detail. Furthermore, the input materials for analysis are generated by a Monte-Carlo simulation through the statistical methods. The estimation method includes the sensitivity results, the correlation with an NPP's thermal power, iodine diffusion mechanism, and so on. From these results, the conservatism of NRC's methodology is seen. Various results from checking the detailed iodine-behavior are introduced. In this study, the iodine spiking factor is proportional to the NPP power and escape rate. Finally, the radiation dose is calculated and the calculated value is proportional to the spiking factor.

KEYWORDS: Iodine-spike, Spiking Phenomena, Radiation Dose, NRC.

1 INTRODUCTION

In a defected nuclear reactor fuel rod, the cladding no longer provides a separate wall between the internal rod gap and the primary coolant. A leak path then exists and the coolant thus can enter the rod and fission products can escape into the reactor coolant system (RCS). Some investigations have been introduced to obtain a better understanding of the physical processes of activity release during the reactor's operation at steady-state power [1-5]. In the case of full-power operation, a small fraction of the fission-product iodine in a defected rod is released into the RCS constantly. Most of the iodine available for release is present as a liquid-water soluble deposit on the UO 2 fuel surface or the inner surface of the cladding. If the temperature in the pellet-to-clad gap drops below that of coolant saturation, as during a reactor shutdown, the water that has entered the rod remains in the liquid phase and leaches these deposits. The dissolved iodine can then migrate along the water-filled gap to the defect site, resulting in an increased release to the RCS [1-10].
This accelerated release leads to the so-called "iodine spike' [1-5]. An activity contribution to the spike can also result from coolant depressurization and temperature transients during shutdown.
The iodine-spiking phenomenon is an important consideration in safety analysis [2, 3]. For instance, in a pressurized water reactor (PWR), the high pressure coolant is circulated through heat exchanger tubes in the steam generators, which represent a large fraction of the RCS boundary.
The rupture of a tube will result in a reactor trip, with enhanced release of iodine into the coolant and a direct path for release to the environment [2-5]. A steam generator tube rupture (SGTR) accident has in fact been designated as a design basis accident for PWRs [1].
This paper describes the development of a physical model that may be used for those type of events in which activity release due to the iodine-spiking phenomenon can occur. The theoretical basis is based on B.J. Lewis et al.'s study [9]. Many equations is used from the results of B.J. Lewis et al.'s study [9]. Previous treatments have generally ignored the time-dependent behavior for the rate of release of iodine

from a defective fuel rod into the coolant [2, 5]. In another approach, the release rate was modelled as a simple impulse function where mass conservation was ignored; i.e., the escape rate constant for steady-state operation was simply multiplied by a 'spiking factor' that was proportional to the fractional change in power (or pressure) [1-4, 8]. As conceded in this latter study, a more realistic time-response function was needed because the spike was generally predicted too early. In contrast, the present work accounts for enhanced-diffusional release during reactor shutdown, and includes any forced-convective release that may result from temperature and pressure transients that are associated with the shutdown event.

The model has been benchmarked against a database of reactor trips from PWR operational experience [1, 5, 8].

The ultimate purpose in this paper is to evaluate the effective dose and the sensitivity of its effect. From these results, a deep understanding can be extracted. The OPR1000 type nuclear power plant in Korea was selected as a reference plant. In this paper, a new methodology and an iodine spiking analysis are introduced.

2 MODEL DEVELOPMENT

2.1 Steady-state release

Some fission products are generated and diffuse into the UO_2 fuel matrix. A small fraction of these products have volatility. The volatile products are partially released into the fuel-to-clad gap during steady-state reactor operation. In defected fuel rods, these products are mixed with the steam in the fuel-rod gap and migrate toward the defect site. At this time the products are released rapidly into the primary coolant. The release rate pattern from the defected fuel rod to the coolant can be written as given below. The pattern is generally introduced as a first-order form of the kinetic model. This model have been understood as the small gap transfer phenomena and the equation is below [2-5, 9]:

$$\mathrm{R_c} = \left(\frac{\mathrm{r}}{\lambda+\nu}\right)\mathrm{R_f} \tag{1}$$

where $\mathrm{R_c}$ = the release rate from the gap into the coolant (atom/s), $\mathrm{R_f}$ = the release rate from the fuel into the gap (atom/s), ν= gap escape rate constant (s^{-1}), and λ = radioactive decay constant (s^{-1}). In the condition of the defected PWR fuel rods, R_f is predominant because of a diffusion mechanism [6,11]. Additionally, a small contribution from fission recoil is considered. The escape rate ν depends on the defect location and the defect size. Generally, the escape rate is used to explain the holdup phenomena due to physical transport along the gap to the defect site [5]. Moreover, as demonstrated in a number of in-reactor studies, this parameter also accounts for chemical holdup effects in the gap: i.e., it is particularly sensitive to the defect characteristics, owing to the relative quantity of water-to-steam around the detect site, resulting in a localized release of iodine [2, 5, 8, 9].

During steady-state conditions, the following mass balance for the fission product inventory $\mathrm{N_c}$ (atoms) in the RCS applies:

$$\frac{dN_c}{dt} = R_c - \lambda_N - LN_c = 0 \tag{2}$$

where the rate of release into the coolant (R_c) is balanced by various losses. These losses include radioactive decay (the term containing A), as well as coolant purification, leakage, and fission-product deposition, as described by an overall loss-rate constant:

$$\mathrm{L} = \frac{\mathrm{F\varepsilon}}{\mathrm{M}} + \frac{\mathrm{L_r}}{\mathrm{M}} + \alpha \tag{3}$$

where F is the cleanup system flow rate (kg/s), ε is the efficiency of the cleanup system (for iodine), M is the mass of water in the primary system (kg), L_r is the leak rate from the primary system (kg/s), and α is the deposition rate constant (for iodine) (s^{-1}). Typically, the rate constants for coolant leakage and deposition are much smaller than that for coolant purification.

Using equations (1) and (2), the total steady-state coolant activity resulting from x (defected fuel rods) is given by:

$$\lambda N_c = x\left(\frac{\lambda}{\lambda+L}\right)\left(\frac{v}{\lambda+v}\right)R_f \quad (4)$$

where the parameters R_f and v correspond to an 'average' defective rod.
Here the release rate into the RCS is proportional to the fission-products gap inventory N_g [1, 2, 5, 8, 9]:

$$R_c = \nu N_g \quad (5)$$

Using equations (l) and (5), the total gap activity for the x(defected fuel rods) is:

$$\lambda N_g = x\left(\frac{v}{\lambda+v}\right)R_f \quad (6)$$

Equation (6) can be used to calculate the iodine activity in the fuel-to-clad gap that is available for release during reactor shutdown, if release due to fuel cracking is negligible.
Sweep gas experiments with fuel rods operating at a relatively high linear power of 55 kW/m have shown that the gap inventory is only increased by ~ 15% as a result of fuel cracking effects upon shutdown for the long-lived isotope ^{133}Xe [23]. This process will therefore be negligible for PWR rods at the much lower average linear heat rating of 22 kW/m. Using equations (4) and (6), the ratio of the gap activity to coolant activity is independent of the number of defected rods.

$$\frac{\lambda N_g}{\lambda N_c} = \frac{\lambda+L}{\nu} \quad (7)$$

Equations (5) and (7) physically explain that a small defect size will result in a lower coolant activity. Otherwise, in the condition of a greater stored gap activity, the relation is unclear. Given the measured (simulated) steady-state coolant activity, the gap activity can be predicted without the information of the power and burnup of the defected rod.

2.2 Release of Reactor Shutdown Condition

2.2.1 Kinetic models for diffusion

During reactor shutdown, the coolant that has entered the rod remains in the liquid phase and dissolves the iodine that is deposited on the internal rod surfaces. In the absence of any temperature or pressure fluctuations, this iodine leaching process can be described by either a diffusion or first-order kinetic process. In both representations, the release rate R (atom/s) from the defective rod into the RCS is given by the following time-dependent relation [2, 9]:

$$R(t) = kN_{g0}\exp[-(A + k)t] \quad (8)$$

where N_{g0} is the initial iodine inventory in the fuel-to-clad gap (atom) and k is the escape rate in the fuel-to-clad gap (s^{-t}).
The parameter k is again dependent on the transport path length (i.e., defect location), and explains the delay due to the iodine's physical migration through the gap.
An additional consideration is that a chemical trapping process may appear, similar to the iodine chemical reaction in the gap with the cladding, fuel or other fission products.
In this work, it is assumed that the inventory in the gap (N_{g0}) is fixed at shutdown. At this time, the inventory could be depleted. This assumption is based on that the cracking phenomena of fuel are not predominant in the case of the release process during the shutdown condition. In addition, any thermal diffusion is negligible because the fuel temperature is rapidly reduced in the shutdown condition (ex.

under which the RCS temperature immediately falls below the saturation point). Thus, the diffusion and transfer model is valid in conditions of normal and off-normal operations.

2.2.2 *Convection modeling*

Iodine spiking can also result from coolant depressurization and temperature transients. When the RCS pressure is reduced or the RCS temperature is increased, and non-condensable gases that are trapped in the plenum at the top of the rod can expand, thereby forcing iodine-rich water out of the rod and into the coolant. The non-condensable gases in the plenum include the stable and radioactive isotopes of xenon and krypton, which are generated in the fission process, and hydrogen that is produced by coolant radiolysis and oxidation of the uranium and Zircaloy cladding materials.
On the other hand, if a defect is located at the top of the rod, the gases can escape from the plenum.
The rod will then entirely fill with water as the steam condenses upon shutdown. With a temperature or pressure change in the RCS, the fluid density in the gap will also change, resulting in possible expulsion of iodine-rich water.
As a consequence of gas expansion in the plenum, or water expansion in the rod, a forced-convective release will result until a pressure or temperature equalization is achieved. The release rate expression for this transport process is given as below [2, 5, 8, 9]:

$$R(t) = k_0 N_{g0} \exp[-(\lambda + f^{-1} k_0)t] \quad (9)$$

where

$$k_0 = \frac{\Delta_P(0)h^2}{12\mu l^2} \quad (10)$$

and h is the fuel-to-clad gap thickness (m), l is the fuel-stack length (m), μ is the fluid viscosity in the fuel-to-clad gap (kg/m · s), and $\Delta_P(0)$ is the pressure differential between the coolant and internal rod atmosphere at the beginning of the time step (Pa). The parameter f depends on the axial location of the defect.
If the plenum is gas-filled (bottom-end defect):

$$f = \frac{\Delta P(0)}{P_c}\frac{\xi v_p}{v_{gap}} \quad (11)$$

or, if the rod is entirely filled with water (top-end defect):

$$f = \frac{\Delta P(0)}{\beta} \quad (12)$$

where β is the volume of gas in the plenum (V_{plenum}) (m^3), V_{gap} is the fuel-to-clad gap volume (m^3), V_{plenum} = plenum volume
(m^3), f is the volumetric fraction of gas in the plenum, $P_{coolant}$ is the coolant pressure (Pa), and /3 = fluid expansion coefficient (Pa).
The fluid expansion coefficient β can be estimated from the following:

$$\beta = \frac{\Delta P(0)\rho}{\Delta\rho} \quad (13)$$

where Δρ is the change in density over the time step. The fluid density ρ is relatively independent of pressure. Thus, for typical shutdown transients, ρ (in kg/m^3) can be represented by a polynomial correlation in temperature T(°C) over the temperature range of 50 to 300°C, based on standard steam table values. In this work, this correlation equation coefficients are generated by a Monte-Carlo simulation for the normal distribution of the temperature range and the fluid density range:

$$\rho = 900.1 + 0.1277T - 4.01 \times 10^{-3}T^2 \quad (14)$$

Similarly, the fluid viscosity μ(kg/m · s) is independent of pressure and can be calculated over the same temperature range from the correlation in T (°C):

$$\mu = (8.77\times 10^{-4}) - (9.56 \times 10^{-6}T) + (5.00 \times 10^{-8}T^2) - (1.23 \times 10^{-10}T^3) + (1.24 \times 10^{-13} T^4) \qquad (15)$$

Consequently, the fraction f in equation (11) is much larger than that in equation (12) for the pressure and temperature during shutdown after the transient condition.

2.2.3 Mass balance equations in RCS

The diffusion release model as $R_{diffusion}$ (t) (equation (8)), the convection release model as $R_{convection}$ (t) (equation (9)) and the coolant iodine inventory model can be written by the mass balance equation in the RCS (compare with Equation (2))[2, 8, 9]:

$$\frac{dN_c}{dt} = R_c(t) - (\lambda + L) N_c \qquad (16)$$

where $R_c(t) = R_{diffusion}(t) + R_{convection}(t)$.

Equation (16) is very conservative from the assumption that both transport processes of the diffusion and the convection are considered. In the initial condition of $N_c(t = 0) = N_{c0}$, equation (16) is written as equation (17). The initial condition is the starting point of release modeling as given below:

$$N_c(t) = \left[N_{c0} + N_{g0}\left\{\left(\frac{k_0}{L-\frac{k_0}{f}}\right)[e^{\left(L-\frac{k_0}{f}\right)} - 1] + \left(\frac{k}{L-k}\right)[e^{(L-k)t} - 1]\right\}\right]e^{-(\lambda+L)t} \qquad (17)$$

Generally speaking, in the condition of $\frac{k_0}{f} >> L$, equation (17) is reduced into

$$N_c(t) = \left[N_{c0} + N_{g0}\left\{f[1 - e^{-\left(\frac{k_0}{f}\right)t}] + \left(\frac{k}{k-L}\right)[1 - e^{(L-k)t}]\right\}\right]e^{-(\lambda+L)t} \qquad (18)$$

Similarly, considering the inventory, the mass balance equation of the fuel-rod gap is given below:

$$\frac{dN_g}{dt} = -R_c(t) - \lambda N_g \qquad (19)$$

Equation (19) considers inventory losses only because of the release to the coolant and the radioactive decay. In this work, any effect of iodine deposition on the surface of the internal fuel-clad is conservatively ignored.
However, this effect is not assumed to be significant because the sediment rate (k) can be much greater than the corresponding deposition rate. Thus, the solution of Equation (19) is given below:

$$N_g(t) = N_{g0}\left\{e^{-kt} - f[1 - e^{-\left(\frac{k_0}{f}\right)t}]\right\}e^{-\lambda t} \qquad (20)$$

where N_{g0} is the gap inventory of iodine at the beginning of the time step. Immediately after shutdown, N_{g0} can be calculated from the measured coolant inventory N_{c0} with the use of equation (7). The exponential term $e^{-\left(\frac{k_0}{f}\right)t}$ in Equation (20) rapidly converged to a zero value in the given time step.
Hence, it can be seen that even if no diffusion were to occur (i.e., k = 0), the existing gap inventory would still be completely depleted in the given time step when f is unity.

Iodine-rich water will be expelled from the rod as a result of a forced-convective release. The volume of water displaced in the plenum will be occupied by the expanding gas. This process will continue at each time step.
The volume of water ΔV displaced in a given time step is given as follows:

$$\Delta V = A_{gap}\int v(t)\,dt = fv_{gap}[1-\exp(-l/t)] \approx fv_{gap} \quad (21)$$

where A_{gap} is the cross-sectional area of the gap and t (= f / k_o) is the characteristic time of pressure equalization, which is on the order of one second at 12 MPa.

The bulk-flow velocity v(t) has been used in the derivation of equation (21).

Thus, at each time step for equation (11), the volume of gas in the plenum ($V_g = \xi V_p$) will be increased by the amount ΔV.
The bulk velocity in the fuel-to-clad gap (averaged over a given time step) can be calculated for the present transient as given below:

$$\bar{v} = \frac{\int_0^t v(t)\,dt}{\int_0^t dt} = \frac{\gamma t^*}{t}\,[1-\exp(-t/t^*)] \quad (22)$$

where $\gamma = k_o l$ and $t^* = 12\mu l^2 V_g/(h^2 P_c V_{gap})$ is the characteristic time constant of pressure equalization in the plenum. Here, assuming the nominal values and when $\Delta_P(0) = 1.4 \times 10^5$ Pa in equation (10), the bulk velocity is evaluated as 4.1×10^{-3} m/s.

Calculation of Reynold's number ($Re = 2\pi\bar{v}\rho/\mu = 5.2$, where ρ is the fluid density = 7.2×10^2 kg/m ^{3} evaluated in Eq. (14)) indicates the presence of laminar flow, as assumed in the derivation of the pressure-differential release model.
The SRP guidelines can be quantified for an individual plant. It is assumed that a reactor trip occurs instantaneously from 100% of full power at time zero, and that the coolant cleanup system does not operate for time greater than zero (importantly, only taken for radioactive decay). The SRP guideline stipulates that an accident-initiated spike, a release rate of 500 times the corresponding value at equilibrium must be used; the equilibrium release rate R_{c0} (atoms/s) is based on a coolant concentration level of $C_{c0} = 1000\mu Ci/kg$, i.e., from Equation (2) [2, 9]:

$$R_{c0} = \frac{\lambda + L_0}{\lambda}\; M C_{c0}\varepsilon \quad (23)$$

where λ = decay constant (s^{-1}), L_0 = steady-state coolant cleanup rate constant (s^{-1}) , M= RCS mass (kg), and ε = conversion factor (= 3.7×10^4 Bq /μCi). A typical value of L_0 is 2×10^{-5} s^{-1} [20]. The transient release rate is therefore taken to be a constant such that $Rc = 500R_{co}$. The time-dependent coolant activity concentration for the spike event, $C_c(t)$ (in μCi/kg), follows from the solution of the mass-balance equation (see Equation (16)), where L is now equal to zero:

$$N_c(t) = N_{c0}\, e^{-\lambda t} + \frac{R_c}{\lambda}\,(1 - e^{-\lambda t}) \quad (24)$$

or equivalently,

$$C_c(t) = C_{c0}\left\{e^{-\lambda t} + 500\left(\frac{\lambda + L_0}{\lambda}\right)[1 - e^{-\lambda t}]\right\} \quad (25)$$

The SRP model prediction of Eq. (25) is shown in Fig.2 for ^{131} I. This calculation can be compared to the model prediction of Section 2.2. The gap inventory can be conservatively estimated from the assumed SRP value of $C_{c0} = 1000\mu Ci/kg$ and an average value of $\nu = 9.1 \times 10^{-7}$ s^{-1} in Table 2. Thus, using the DOSE-SGTR program, as the gap inventory is depleted, the predicted coolant activity levels, in contrast to the SRP model, approach a relatively constant level because of a depleted gap inventory and little radioactive decay in the coolant (assuming no coolant cleanup). A more realistic estimate can be further obtained with the DOSE-SGTR model using an average value of the steady-state coolant concentration for the thirteen cases in Table 1 (initial condition , 140 μCi/kg). Thus, at two hours, it can be seen that the SRP analysis is overly conservative by a factor of 45.

2.3 DOSE-SGTR Code Implementation

The iodine spike model is introduced in section 2.2.3. In other words, the mass balance equation has been developed as the analytical solutions for the iodine spike model. The mass balance equation includes the transport differential equations for diffusion and convection modeling. In addition, it has been implemented into a computer code named DOSE-SGTR [10].

The DOSE-SGTR program provides the utility to determine the response of a pressurized water reactor to a steam generator tube rupture (SGTR). The iodine concentration in the reactor coolant system is rapidly increased due to iodine spike, primary flashing, primary bypass fractions and iodine partitioning in the secondary coolant. Otherwise, experimental and analytical investigations have recently been completed and these assumptions were tested. The calculation mechanism is based on the analytical results of Equations (10), (11), (18), (20), and (21). This code maintains an appropriate mass balance of iodine in the fuel-to-clad gap and the reactor coolant system.

The initial gap inventory is calculated from the measured steady-state coolant activity in accordance with equation (7). This program implicitly assumes the detector's location being on the bottom of the fuel rod. This assumption is very conservative because the release fraction f is much larger than 1.

3 MODEL VALIDATION

The iodine-spiking model was validated with data collected from thirteen items of information of various PWRs. Table 1 gives the information of the calculation methodology such as Monte-Carlo simulation values, FSAR averaged values, and so on.

Table 1 : PWR Analysis parameters used for modeling input materials

Item	Sampling NPPs (1 ~ 10)	Korean WH (KOREA1)	OPR-1000 (KOREA2)	APR-1400 (KOREA3)
Escape rate	Monte-Carlo simulation	FSAR averaged Value	FSAR averaged Value	FSAR averaged Value
Sediment rate	Monte-Carlo simulation	The averaged value From Sampling NPPs	The averaged value From Sampling NPPs	The averaged value From Sampling NPPs
Diffusion modeling	Equation 17	Equation 17	Equation 17	Equation 17
Convection modeling	Equation 19	Equation 19	Equation 19	Equation 19
Mass balance equation	Equation 22	Equation 22	Equation 22	Equation 22

The input data needed as a function of time for the DOSE-SGTR application included the following: the reactor power, coolant pressure, coolant temperature, and the coolant cleanup rate (or escape rate). This information, as well as the measured coolant activity for ^{131}I, is shown for several typical cases in Figure 1. In Table 1, the coolant temperature data were estimated as 40℃ below the coolant saturation temperature (based on the 10 Monte-Carlo simulations and three FSAR averaged values).

The cleanup rate constant (L) was obtained from the slope of a semi-log plot of the coolant activity versus time for those periods that were away from any virtual transient events of power, pressure or temperature.

The zero time was chosen to correspond to the time where the reactor's is 40% or less (during the shutdown state). This value would guarantee an average fuel rod power of less than 10 kW/m; i.e., in this case, the temperature of the fuel surface would be below that of coolant saturation so that only water would be present in the gap [10].

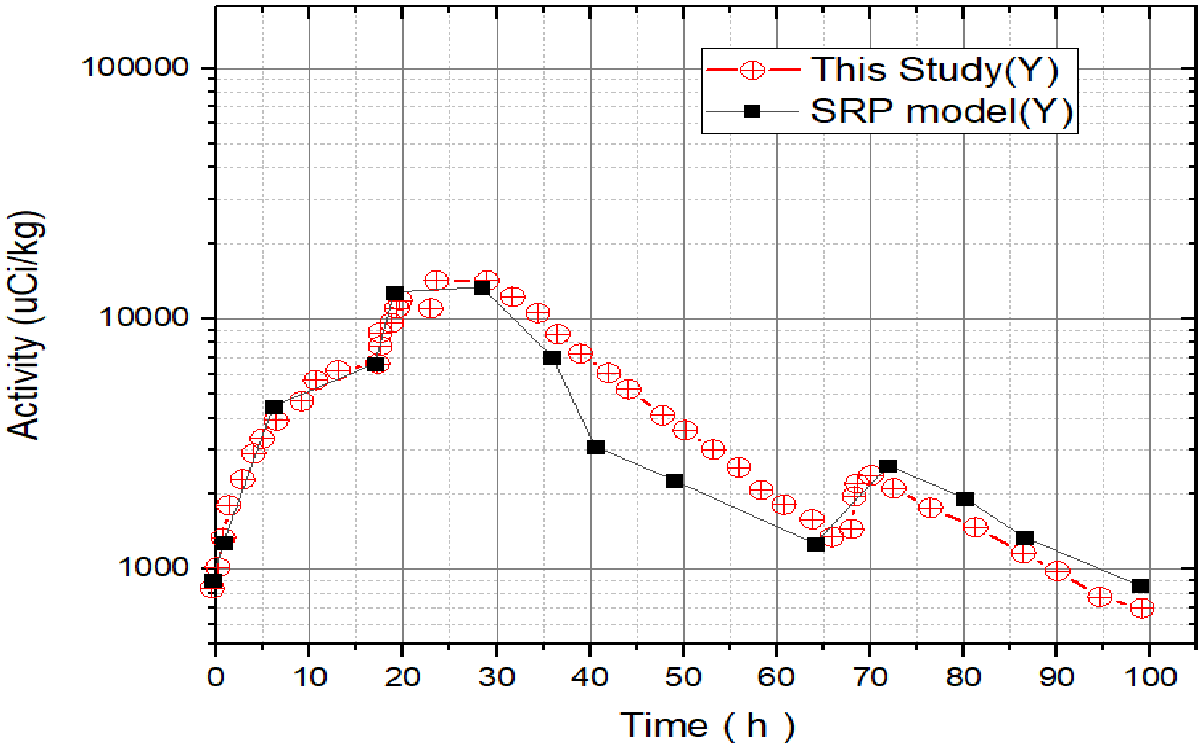
In this calculation, a defected position at the bottom end of the fuel rod was assumed.

The calculation's key parameters are the plenum volume (V_{plenum}), the fuel-to-clad gap volume (V_{gap}), and the fuel stack length (l).

The fitting parameters of the model included u, k, and s c. The parameter v provides an estimate of the

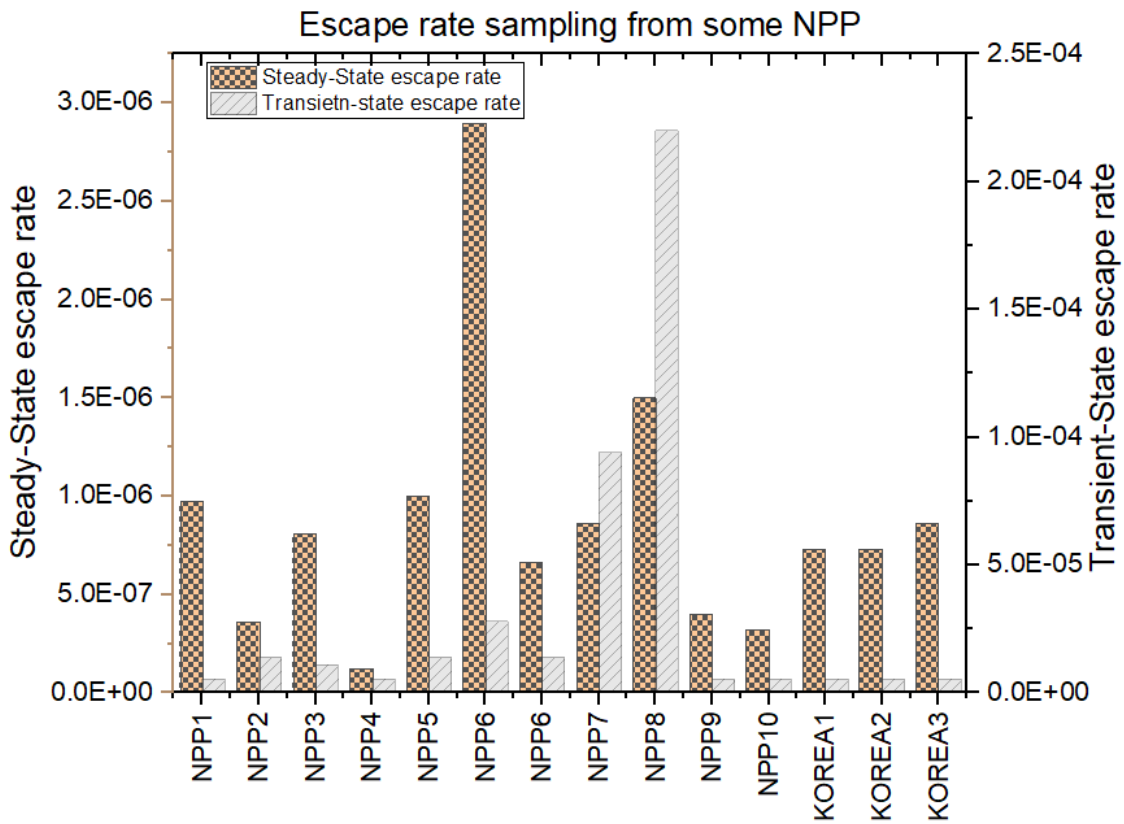
iodine inventory in the gap that is available for release, given the measured value of the steady-state inventory in the coolant (see equation (7)).
The parameter k describes the rate of diffusional transport of the water-soluble iodine in the gap during the reactor shutdown.
The parameter determines the quantity of iodine-enriched water in the transient condition that can be expelled by the expanding plenum gas with pressure or temperature transients. The simulated and predicted values of the ^{131}I in coolant activity concentration is estimated as a function of time in Figure 1.
The analysis results are provided in Figure 1. In general, this study model is in good agreement with the simulated results from the average values NPP1 ~ NPP10. This study is based on the application of Monte-Carlo simulation of some equation's parameters. The SRP model is delineated by equations (11, 18, 20, 21). In this work, the parameters of the equations are simulated by Monte-Carlo modeling for each parameter assuming a normal distribution. Figure 1 shows that this work model is in good agreement with the SRP model (equations (11, 18, 20, 21)).

Figure 1 : Comparison of SRP and this work for diffusion/convection model predictions ($^{131}1$ coolant activity concentration under a condition of a 100gpm leak rate at SGTR/MSLB)



In addition, Figure 2 shows the simulated escape rates. NPP1 ~ NPP10 are simulated. NPP6 is the maximum case in the condition of a steady-state value of 2.7e-6 s^{-1}. NPP8 is the maximum case in the condition of a transient-state value of 2.25e-4 s^{-1}. In the final escape rate results, Figure 2 shows that the transient case is more conservative than the steady-state case. In this study, the weighted average value of NPP1 ~ NPP10 is compared with KOREA1, KOREA2, and KOREA3. The difference between them is within 2%.

Figure 2 : Escape rate from Monte-Carlo simulation (NPP6 : maximum in steady-state, NPP8: maximum in transient-state)



4 SGTR/MSLB TRANSIENT ANALYSIS

4.1 Comparison with SRP methodology

Generally, the physical model derived in section 2 can be used to evaluate the standard review plan (SRP) guideline of the NRC. In this work, a steam generator tube rupture (SGTR) accident is assumed to occur with coincident iodine spike. A typical assumption for a main steam line break (MSLB) is used in FSAR, assuming a leak rate of less than 100 gpm.

For this calculation, the pressure and the temperature histories of FSAR are employed, where a time-step size range of 2 min ~10 min was chosen to show the trend in detail. In accordance with the assumptions of the model, the coolant temperature is always below the saturation temperature for the MSLB. The pressure transient in Figure 3 rapidly appeared compared to the actual condition during the normal shutdown procedure.

In Figure 3, the spiking-focused phenomena are introduced considering the power history, RCS pressure, and RCS temperature. The circled-transparent plot line in Figure 3 is ^{131}I accumulated concentration prediction. At the point of one hour after the event, a large fluctuation appeared in the RCS pressure. Due to this large change, during 1 hour at 30minutes after reactor trip, the ^{131}I concentration slowly accumulated. In addition, along with the RCS pressure drop, the accumulation velocity of ^{131}I concentration is reduced. At the starting point of the remarkable RCS pressure drop, the accumulation velocity of ^{131}I concentration decreased. At the ending point of the RCS pressure drop, the accumulation velocity converged to a zero value in the region of three hours later.

In Figure 4, for this work comparing with SRP model, the accumulated iodine concentration behavior is shown. For simple comparison, this modeling condition considers only the iodine accumulated concentration without isotope decay, clean up system, iodine remove process.

Figure 3: Comparison of SRP and diffusion/convection model predictions of the 131 1 coolant activity concentration for a SGTR/MSLB sequence (assumption of leakage rate of 100 gpm). The RCS pressure and temperature histories for the event are also shown.

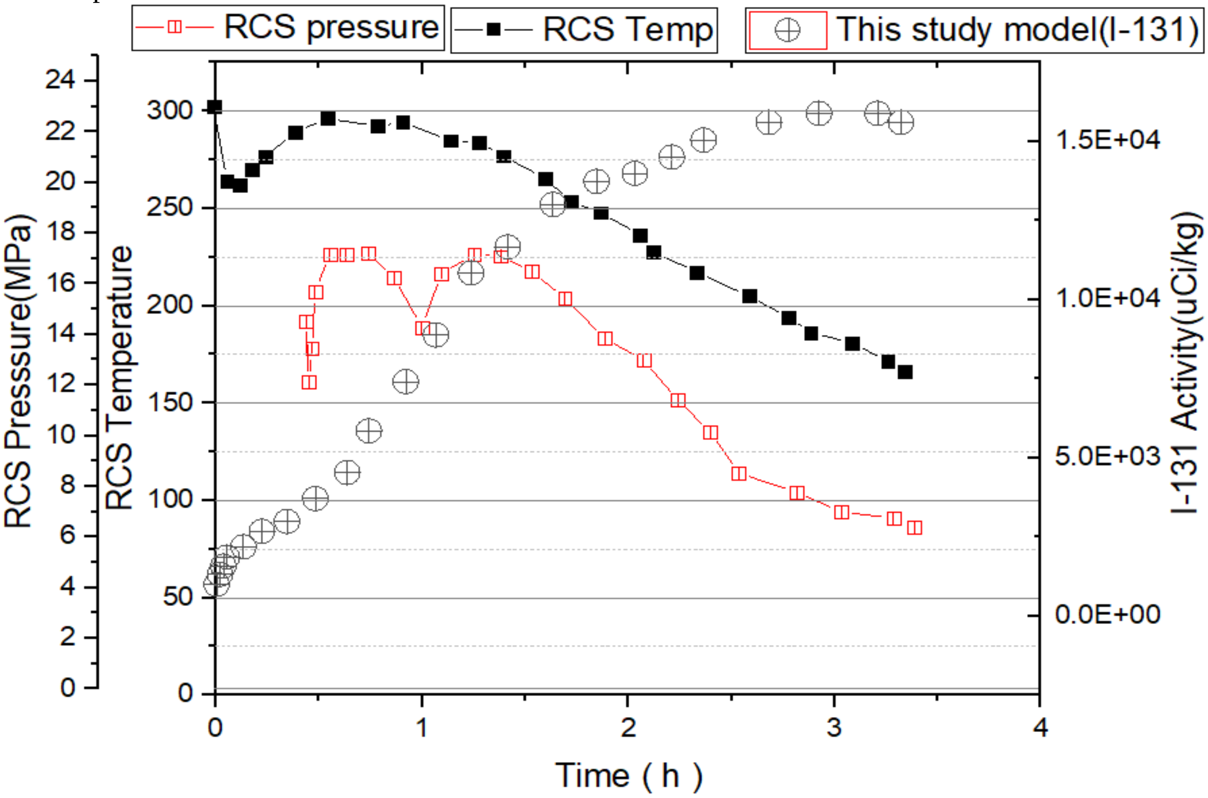
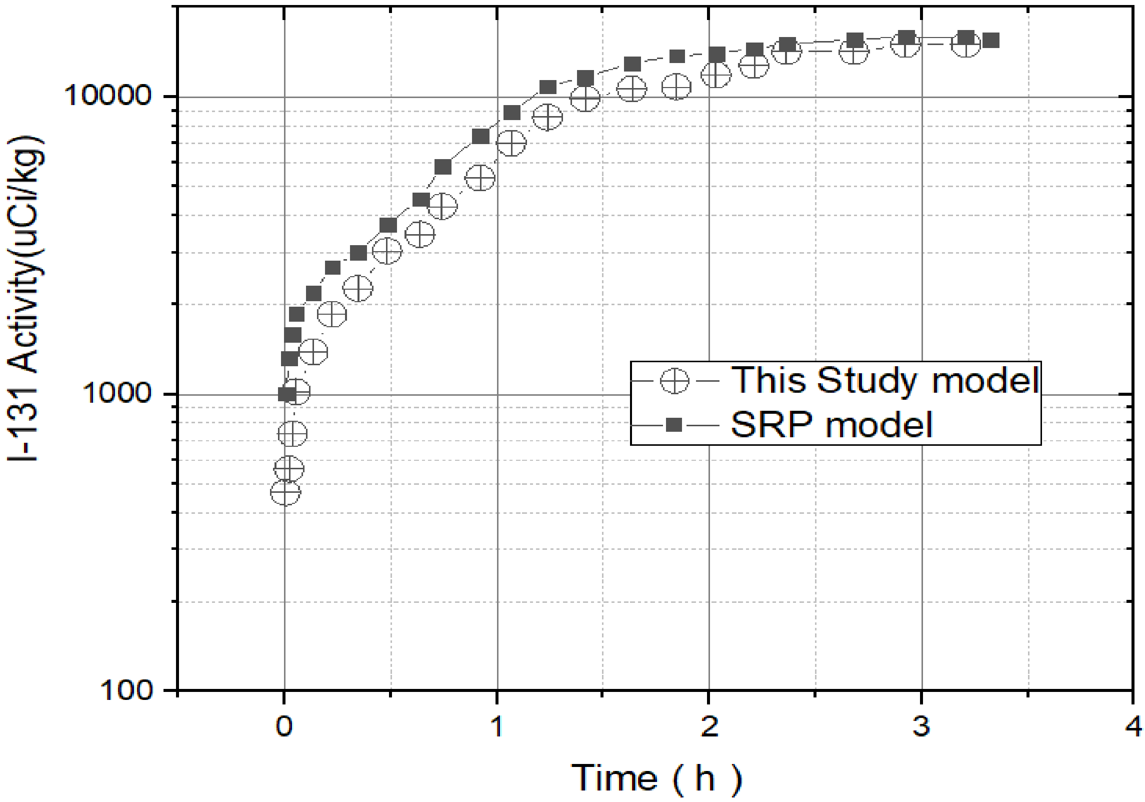


Figure 4: ^{131}I accumulated behavior expectation between this work and NRC SRP model.



4.2 Comparison with simulated SGTR events

The DOSE-SGTR model can be validated against the observed iodine-spiking behavior in the simulated SGTR events at FSAR of KOREA1, 2, 3.

The activity in the fuel-to-clad gap can be estimated from the measured activity in the coolant using Equation (7), and a coolant cleanup rate constant of $L_o = 1.7 \times 10^{-5}\ s^{-1}$, for the steady-state situation.

In accordance with the modeling in section 2.3, L is assumed to be zero after the reactor trip. In addition, the average escape rate coefficients in Figure 2 were used for the DOSE-SGTR calculation.

A comparison of the DOSE-SGTR predictions with the measured coolant activities is presented in Figure 5 along with the RCS pressure and temperature histories.

The DOSE-SGTR prediction is in good agreement with the simulated ^{131}I concentration in the case of the accumulated concentration prediction of DOSE-SGTR.

However, later in the transient, it over predicts the simulated data as a result of the conservative approach taken where any losses in the system were neglected.

In this case, any losses may arise either due to leakage of primary coolant to the secondary side or a dilution of the RCS by safety injection. And the coolant cleanup system is normally turned off after reactor trip. The DOSE-SGTR model over predicts the simulated data in some cases.

From Figure 5, the delay in the iodine spike following reactor shutdown may reduce the transport of iodine. For instance, the main coolant pumps were turned off about two minutes after reactor trip, whereas the pump in the unaffected loop restarted about several hours after the starting point of trip, which coincides with the time when the iodine activity is first observed to remarkably increase. The initial drop in iodine activity may result from the effect of coolant leakage, dilution or coolant cleanup, which is conservatively ignored in the DOSE-SGTR simulation.

In this study, the power history and the concentrations of ^{131}I and ^{135}I are introduced in Figure 6.

In Figure 6, along the power history, iodine spiking factor is calculated. Here, the reverse iodine spiking factor is defined as the reciprocal number of the iodine spiking factor.

From Figure 6, we see that the iodine spiking factor is proportional to the reactor power in ^{131}I and ^{135}I.

Figure 5 : ^{131}I accumulated activity of KOREA 1, 2, 3 as average concentration.

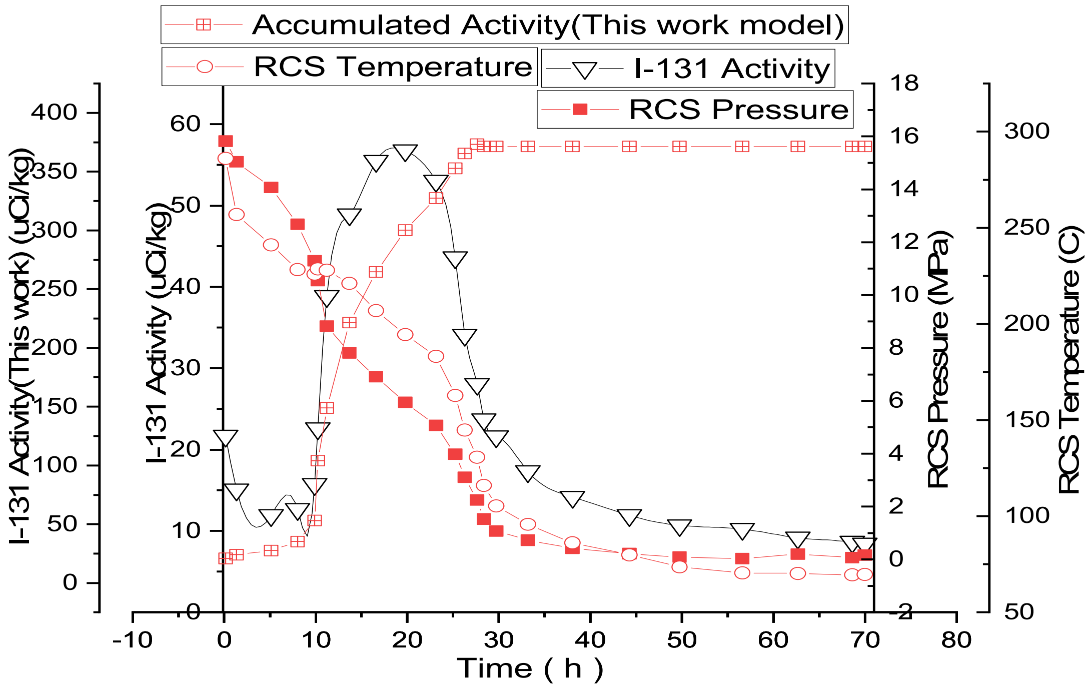
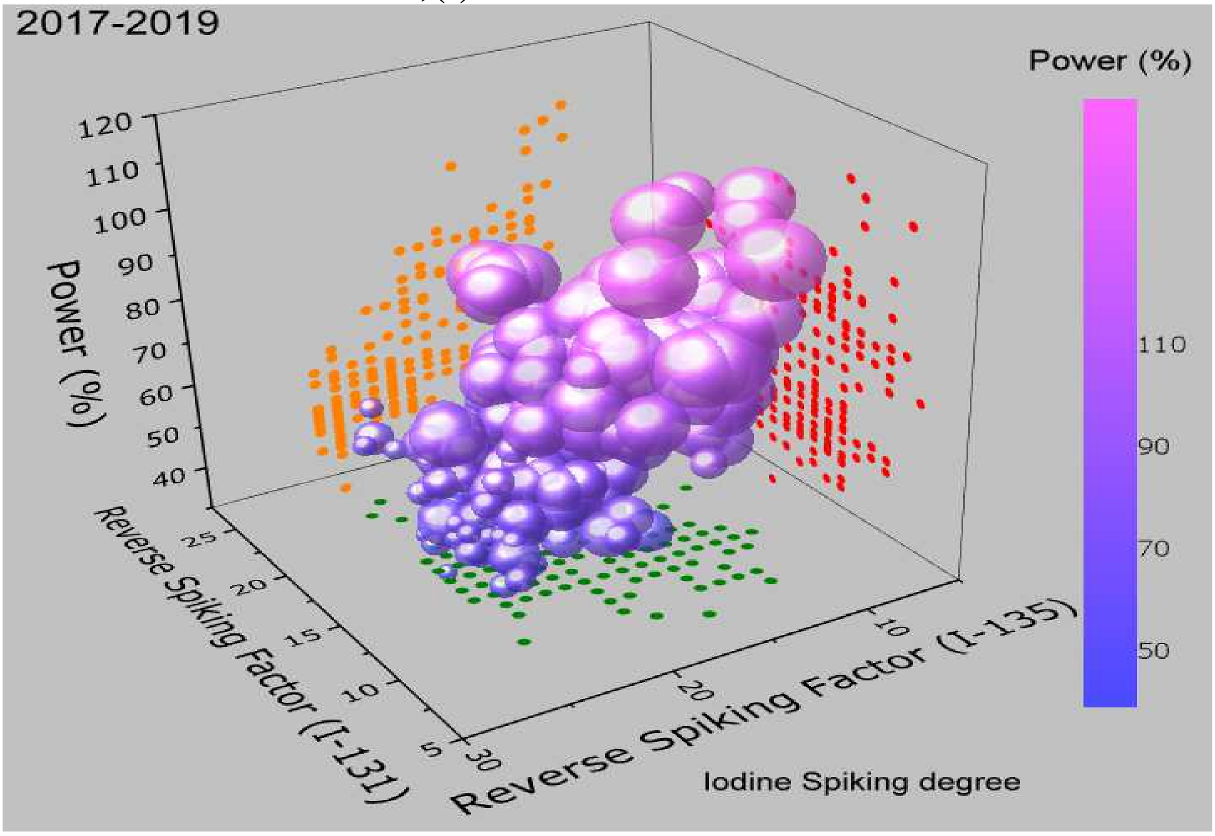


Figure 6 : The prediction of DOSE-SGTR ^{131}I and ^{135}I spike activity for the (a) SGTR of Monte-Carlo simulated results in NPP01~NPP10, (b) SGTR of FSAR in KOREA1~KOREA3.



In comparison, the SRP prediction shown by the box symbol line in Figure 4 can be directly applied and, as expected, provides significantly good matching with this study model under the same conditions. The accumulated iodine concentration prediction is very similar to the pattern of the transparent box plot in Figure 5. Here, Figure 4(section 4.1) and Figure5(section 4.2) are in good agreement in the trend of iodine concentration accumulation pattern.

In conclusion, a best-estimate analysis with the DOSE-SGTR model appears to represent or accurately predict the simulated spike event for the two SGTR cases, in accordance with the methodology of Section 2.3.

5 CONCLUSIONS

1). A physically-based model has been developed by the iodine-spiking phenomenon for PWR fuel upon reactor shutdown. The modeling methodology is based on the total mass balance of iodine in both the fuel-to-clad gap and the reactor coolant system (RCS). The transport modeling of iodine in the gap and its release modeling to the coolant include diffusion and convection equations.

In addition, the model considers the convective release for a pressure differential between the interior of the rod and the bulk coolant, which can result from variable pressure and temperature conditions in the RCS during the shutdown event.

2). The model has been successfully validated in reference to the DOSE-SGTR model and SRP model in the condition of iodine-spiking events during normal PWR operation.

3). Monte-Carlo simulation carried out in polynomial term's coefficient of the equations using the assumption of normal distribution. And the results is inserted to the equation's of B.J. Lewis et al study.

4). The model has been used to evaluate the standard review plan (SRP) methodology for an iodine spike initiated by a steam generator tube rupture and a main steam line break sequence and assumptions.

5).This work's results are expected to show the same pattern as the case of the SRP methodology, which ignores any mass conservation in the gap and assumes a constant release rate.

6).In contrast, the DOSE-SGTR model maintains a mass balance in both the gap and coolant, where

there is only a finite supply of iodine in the gap and it is continually being depleted.
7). The concentration accumulation velocity exponentially decreases in the case of decreasing the release rate in comparison to a constant release rate in the SRP model.

This work is focused on physical phenomena in accordance with physical transport processes of diffusion and convection for understanding iodine spiking phenomena.

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