A Criticality Safety Study for the Disposal of Damaged Fuel Debris

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The present work focuses on neutronics analysis for the disposal of damaged fuels from Fukushima nuclear power plants. MCNP calculations were performed for a canister containing fuel debris surrounded by buffer in deep geologic repository in a water-saturated reducing environment at different times after canister emplacement. The damaged fuel debris is modeled as spherical particles in a hexagonal lattice. Four different cases were compared based on various assumptions about moderation and fuel relocation. Based on the numerical results, the key findings include, (a) the calculated neutron multiplication factor (k_{eff}) is sensitively dependent on assumptions related to moderation, (b) the carbon steel canister plays an important role in reducing the likelihood of criticality, (c) the maximum k_{eff} of the canister-buffer system could be achieved after a certain fraction of fissile nuclides has been released from the canister, and (d) under several assumptions, the maximum k_{eff} of the canister-buffer system could be principally determined by the dimension and composition of the canister rather than the initial fuel loading.

I. Introduction

The damaged fuels from Fukushima Daiichi Nuclear Power Station will have to be disposed of in a deep geological repository. For a prospective repository, a criticality safety assessment (CSA) should be performed to ensure that the repository system including the engineered barriers and far-field geological formations remains sub-critical for tens of thousands to millions of years.

Current understanding about the conditions of the damaged fuel is very limited, and the location and design of the repository have not been determined. Therefore, the primary objective of the present paper is to establish a consistent methodology to evaluate the criticality safety for a certain design of the engineered barriers. The methodology could be further improved and utilized to assist the repository system design and criticality safety assessment in the future.

For various repository concepts[1,2], CSA is considered to include three major stages in a chronological order: (1) the stage before package failure, (2) the stage after package failure, while fissile nuclides remain within the engineered barriers, and (3) the stage in which fissile nuclides originated from multiple packages deposit in far-field host rocks. Stage (3) was investigated in our previous work[3]. In the present paper, we focus on stages (1) and (2) by performing neutronics analysis for the engineered barrier region consisting of a single waste package containing damaged fuel debris, failed overpack and the buffer materials.

A brief review on previous works will first be introduced in Section II. In addition to the review of defueling process[4, 5] for the Three Mile Island (TMI) accident, studies on CSA for storage[6] and disposal[2] of spent fuel are also included. Assumptions will be discussed based on the key findings in the literature review. The neutronics analysis in the present work was performed by a Monte Carlo code MCNP[7]. The MCNP model and input will be introduced in Section III. In the present work, four different cases for redistribution of fissile nuclides are compared. The numerical results for the neutron multiplication factor (k_{eff}) for different stages after disposal are shown for each case for different initial canister loadings in Section IV. And discussions on the results will be given in Section V.

II. Background and Assumptions

Defining the model for neutronics calculations plays a central role in CSAs, where conservative assumptions are usually made to cope with various uncertainties and to simplify the model. To define our present work, we briefly reviewed relevant previous studies, especially focusing on the assumptions for the neutronics models.

II.A. Background

Criticality simulations for damaged fuels from a severe accident were previously carried out in the defueling completion report[4] after the TMI accident. During the defueling process, intact and partly intact fuel assemblies were first removed. And then, about 6000kg of re-solidified fuel and structural materials were cut and removed. Remaining damaged fuels in small particles, called fuel debris, were also recovered during defueling.

To avoid criticality accidents when handling fuel debris, Safe Fuel Mass Limit (SFML) was conservatively defined as the critical mass of a water-reflected spherical system filled with fuel particles in a hexagonal lattice and moderated by water. The pitch distance between fuels particles is chosen to optimize the moderation by maximizing the infinite neutron multiplication factor (k_{inf}). Only the depletion of fissile nuclides was taken into account as burnup credit, and no neutron poisons from fission products or structural materials were included. The radius of the spherical fuel particle was calculated to give the same volume of a fuel pellet. According to the defueling report, this assumption was made for the following reasons: (a) decreasing the radius while keeping the same total fuel mass will decrease the k_{eff}, and (b) larger particles from re-solidified fuels contain significant amount of structural materials and have much lower reactivity. The fuel debris was assumed to have uniform averaged burnup because of the mixing during the accident and defueling. All the assumptions above were applied to calculate the SFML, and were confirmed by actual measurements later on.

In a CSA for spent fuel storage casks[6], various cases of failures for fuel assemblies and fuel pins are constructed and compared. Among all the cases considered, the highest increase in k_{eff} was observed when the storage cask was assumed to be filled with fuel particle lattice and water. Similar to the assumption made for the SFML, although highly conservative from the neutronics point of view, the lattice configuration assumes that fuel particles are not contacting each other and are "floating" in water, which is unphysical and highly unlikely to occur during storage period even after final disposal.

Neutron absorbing materials in the disposal canister, such as iron in carbon steel, could significantly enhance the long-term criticality safety after final disposal. As was shown by the criticality safety study for United Kingdom's High Level Waste (HLW) disposal project[2], the k_{eff} of the canister is significantly decreased when the corroded carbon-steel canister is mixed with fissile materials.

II.B. Assumptions

Based on the aforementioned literature review, the assumptions for the present work have been set as described below, and will be discussed in detail in Section III.

The repository is assumed to be in a water-saturated reducing environment. The neutronics model consists of a canister containing fuel debris from Fukushima Unit 1 reactor and the buffer surrounding the canister. Because there is no current design for the disposal system for the damaged fuels, the composition and dimension of the canister and buffer are assumed based on the design for spent fuel disposal[8]. Because the canister for damaged fuels' disposal might not need to fit the length of a fuel assembly, the canister inner height is arbitrarily assumed as 1 meter, and all other parameter values are assumed to be identical to the values in Ref. 8.

The damaged fuel is assumed to be disposed of after 50 years of cooling. The fuel composition after the accident was calculated by burnup code ORIGEN, which was reported in Ref. [9]. Gaseous, soluble, and volatile neutron absorbing nuclides in the fission products (such as Xe and Cs) might have been separated from the fuel and released during and after the accident[10]. Therefore, in this study, only physically and chemically stable, and strongly neutron absorbing nuclides in fission products are considered, which include Gd, Nd, Sm, Rh, and Eu isotopes.

The present work considers six nominal time steps for neutronics analysis: the emplacement time (t=0), the canister failure time (t= T_f), and four steps during the dissolution of debris particles (t= T_f+0.2T_l, t= T_f+0.4T_l, t= T_f+0.6T_l, and t= T_f+0.8T_l). At t=0, the canister only contains fuel debris. The failure time (T_f) of the carbon steel canister is assumed to be 1000 years[8]. After canister failure, water fills the canister, and the canister is modeled as a porous medium with porosity of 0.3.

The fuel is assumed to be released from the canister at a constant rate during the leach time (T_l). The fission products are assumed to be released congruently with the damaged fuel dissolution, which is conservative because the canister will contain more neutron poison, if the release is limited by solubility. The time scale of the leaching of the damaged fuels (mainly UO_2) in a reducing environment is assumed to be much longer than the half-lives of Pu-239 (24100 years) and Pu-240 (6560 years), and be much shorter than the half-lives of U-235 (704 million years) and U-238 (4.5 billion years). Therefore, after a fraction of leach time, almost all Pu-239 and Pu-240 will be decayed to U-235 and U-236. For this reason, we assumed composition in the four leaching steps to be the same as that of the damaged fuels after 200,000 years. The neutronics analysis is hence decoupled from the actual leach time, as long as the assumptions for the time scale are valid.

The geometry of neutronics model at different time steps have been built based on our literature review and will be discussed in details in the next section. The hexagonal lattice of spherical fuel particles is assumed. The pitch distance between particles is assumed to be either (1) make particles contact each other or (2) make the particles lattice fully fill the canister. In the leaching steps, the released materials from the damaged fuel particles is assumed to be either (a) removed from the canister-buffer system, or (b) be homogeneously mixed with the corroded canister. Combinations of the above variations makes four cases: case 1a, case 1b, case 2a, and case 2b. For simplicity, in later discussions, we use phrase "case (1)" to represent both case 1a and case 1b, "case (2)" to represent case 2a and case 2b, "case (a)" to represent case 1a and case 2a, and "case (b)" to represent case 1b and case 2b.

III. Method and Input

III.A. MCNP Model

The schematic layout of the MCNP model is shown in Fig. 1. The canister-buffer system has been determined mainly based on the design from Ref. 8, which consists of a carbon-steel canister (inner radius r, inner height h and thickness θ_1 surrounded by buffer[11] (70% Kunigel V1 Bentonite + 15% Silica sand No. 3 + 15% Silica Sand No. 5, thickness θ_2). The canister is filled with spherical fuel particles (radius R) in a hexagonal lattice (with pitch distance D), up to height H. The unit cell of the lattice is shown in the right bottom of Fig. 1.

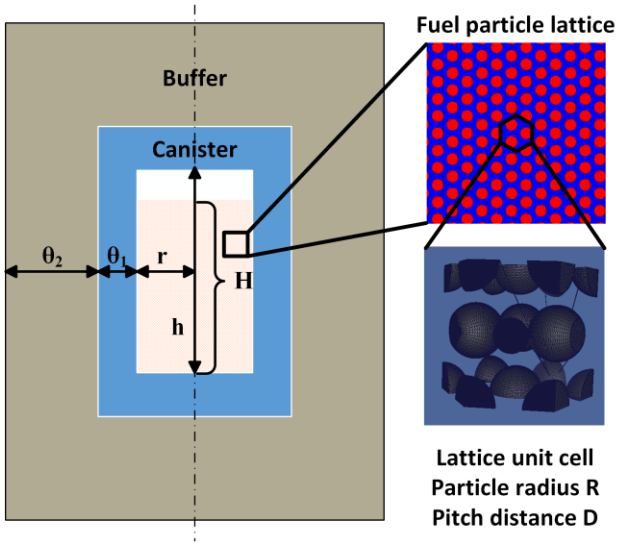

Fig. 1. Schematic layout of the MCNP model

The dimensions and compositions of the canister and buffer are shown in TABLEs I and II. The inner height of the canister is assumed to be 100 cm. The radius of the fuel particle is calculated to match the volume of a fuel pellet. The canister is initially contains M_0 [kg] of damaged fuel. After canister failure, the mass of remaining damaged fuel in the canister $M(t)$ can be written as

$$M(t) = M_0\left(1 - \frac{t-T_f}{T_l}\right), T_f \geq t \geq T_f + T_l. \quad (1)$$

The fraction of volume taken by the fuel lattice f for a hexagonal lattice with pitch distance D and fuel radius R can be written as

$$f = \frac{8\pi}{3\sqrt{2}}\left(\frac{R}{D}\right)^3, D \geq 2R. \quad (2)$$

From the mass balance in a canister with inner radius r and height H filled by fuel lattice, the following formula can be written:

$$M(t) = \pi r^2 H(t) f \rho, T_f \geq t \geq T_f + T_l, \quad (3)$$

where ρ is the density of the damaged fuel, assumed to be the density of UO_2. Note that the height of the lattice is a function of time.

TABLE I. System dimensions

System dimension	Values
Canister inner radius r [cm][8]	27.95
Canister inner height h [cm]†	100.00
Canister thickness θ_1 [cm][8]	14.00
Buffer thickness θ_2 [cm][8]	70.00
Fuel particle radius R [cm]†	0.42

† Values assumed in the paper.

TABLE II. Canister and buffer compositions

Canister composition: Carbon steel[12]	Weight fraction [%]
Fe	99.5
C	0.5
Density [g/cm^3]	7.82
Buffer composition[11]: 70% Bentonite + 30% Silica sand	Weight fraction [%]
SiO_2	78.157
TiO_2	0.138
Al_2O_3	11.033
Fe_2O_3	1.368
MgO	1.449
CaO	1.533
Na_2O	1.429
K_2O	0.358
MnO	0.035
P_2O_3	0.021
SO_3	0.448
H_2O	4.033
Dry density [g/cm^3]	1.60
Porosity	0.38

By substituting (1) and (2) in equation (3), the relation between *H*, *D*, and *t* can be given as

$$\frac{8\pi^2}{3\sqrt{2}} r^2 H(t) \left(\frac{R}{D(t)}\right)^3 \rho = M_0 \left(1 - \frac{t - T_f}{T_L}\right), \quad T_f \geq t \geq T_f + T_l \qquad (4)$$

Note that the pitch distance *D* of the lattice is also dependent on time.

In order to determine *D* and *H* for a given time *t*, further assumptions are needed. Among four cases defined in the last paragraph of Section II, the pitch distance *D* can either make particles contact each other (in cases 1a and 1b), which means $D(t)=2R$=constant, or make the particles lattice fully fill the canister (in cases 2a and 2b) which means $H(t)= h$ =constant. Therefore, for case 1a and case 1b, $D=2R$. With this applied, eqn. (4) can be modified as

$$H(t) = M_0 \left(1 - \frac{t - T_f}{T_L}\right) / \left(\rho \frac{\pi^2}{3\sqrt{2}} r^2\right), T_f \geq t \geq T_f + T_l. \qquad (5)$$

And for case2a and case2b, $H=h$, and thus

$$D(t) = \left\{ \frac{8\pi^2}{3\sqrt{2}} r^2 h \frac{\rho R^3}{\left[M_0 \left(1 - \frac{t - T_f}{T_L}\right)\right]} \right\}^{\frac{1}{3}}, \quad T_f \geq t \geq T_f + T_l. \qquad (6)$$

The difference between case 1a and case 1b or case 2a and case 2b is that, in the case (a) the released fuel is assumed to be removed from the system, while in the case (b) the released fuel is assumed to be homogeneously mixed with the corroded canister, which has a porosity of 0.3. By definition, there is no difference between case (a) and case (b) in $t=0$ and $t= T_f$.

III. B. Fuel Composition

The composition of the damaged fuels at the time of the accident was calculated by the burnup code ORIGEN in Ref. 9. Table III shows the actinide and fission product included in the damaged fuel in the Fukushima Unit 1 reactor after 50 years of cooling. For actinides, only nuclides with more than one kilogram in the core are shown. For fission products, the relative importance as neutron absorber was ranked by their thermal neutron absorption cross section[13] times the total number of atoms in the reactor core. As mentioned in Section II, gaseous, soluble, and volatile elements were screened out, and the top five remaining elements are included. From previous discussions, the compositions in the last four time steps are assumed to be the damaged fuels after 200,000 years decay, which is shown in Table IV. Comparing with data shown in Table III, most of the plutonium decayed; while the fission product poisons composition remain the same, because they are all stable nuclides.

With the fuel composition, the dimension of the canister-buffer system, and the fuel lattice parameters calculated by eqns. (5) or (6), MCNP criticality calculations have been made for different four cases with initial loadings of 500kg, 1000kg, and 1500kg. Six nominal time steps are considered, at t=0, the canister is filled with fuel particles (no water is included) defined by Table III. The canister has zero porosity initially. At the canister failure time T_f, the canister suddenly becomes porous with 30% porosity, and water fills the canister. The fuel composition after 1000 year decay was calculated based on the data shown in Table III. At the subsequent four leaching steps, Table IV data are used. In addition, the hypothetical cases in which the canister is filled with water at time zero were also calculated.

TABLE III. Actinide and Fission products compositions of Fukushima Unit 1 after 50 years decay

Actinides	Mass [kg]	Fission product poisons	Mass [kg]
U-234	2.79E+00	Gd-155	3.59E-01
U-235	1.11E+03	Gd-157	4.59E-03
U-236	2.42E+02	Nd-145	4.58E+01
U-238	6.53E+04	Nd-147	3.76E+01
Np-237	2.33E+01	Sm-147	1.63E+01
Pu-238	5.43E+00	Sm-149	1.68E-01
Pu-239	3.07E+02	Sm-150	1.44E+01
Pu-240	1.05E+02	Sm-152	6.01E+00
Pu-241	5.28E+00	Rh-103	2.46E+01
Pu-242	2.02E+01	Eu-151	2.27E-01
Am-241	5.45E+01	Eu-153	5.87E+00
Am-243	3.41E+00		
Total	6.72E+04	Total	1.51E+02

TABLE IV. Actinide and Fission products compositions of Fukushima Unit 1 after 200,000 years decay

Actinides	Mass [kg]	Fission product poisons	Mass [kg]
U-233	3.36E+00	Gd-155	3.59E-01
U-234	6.16E+00	Gd-157	4.59E-03
U-235	1.42E+03	Nd-145	4.58E+01
U-236	3.44E+02	Nd-147	3.76E+01
U-238	6.53E+04	Sm-147	1.63E+01
Np-237	7.70E+01	Sm-149	1.68E-01
Pu-242	1.41E+01	Sm-150	1.44E+01
		Sm-152	6.01E+00
		Rh-103	2.46E+01
		Eu-151	2.27E-01
		Eu-153	5.87E+00
Total	6.72E+04	Total	1.51E+02

IV. Results

The numerical results are shown in Fig. 2, where the neutron multiplication factor k_{eff} is plotted against the nominal time steps for various combinations of cases and initial loadings. Note that the time axis only represents the order of the time steps and does not represent the actual time. The failure time (1000 years) should be several orders of magnitude smaller than the leach time. In all three plots in Fig. 2, the solid points represent case (1), or cases assuming that fuel particles are in close contact each other, whereas the hollow points represent case (2), or cases assuming that the fuel lattice fills the canister. Red points represent case (a), assuming all fissile nuclide released from the canister are removed from the system, and blue points represent case (b), assuming released nuclides are homogeneously mixed with the corroded canister. The initial loadings of 500kg, 1000kg, and 1500kg are labeled by squares, circles, and triangles, respectively. The green points in three figures in Fig. 2 represent cases assuming the canister is filled with water at time zero.

At time zero, if the canister is filled with water, the k_{eff} is significantly greater than no water cases, among all results, because the reactivity will be greatly increased by introducing moderation in water. For the same reason, there is a significant increase in k_{eff} observed in all cases from time zero to canister failure time, when water is assumed to fill the void spaces in the system.

After t= T_f, the damaged fuels in the canister will be released from the canister. Also between t= T_f and t= T_f + 0.2T_l, most of the plutonium isotopes decay into uranium isotopes. In the following time steps, the isotopic composition of the damaged fuels is assumed to remain the same, while the mass in the canister is decreased. However, only with the 500kg initial loading (the top figure), decrease in k_{eff} with time for all cases is observed. In case 2a, case 2b for 1000kg and 1500kg initial loadings, respectively, the k_{eff} once increases, and then decreases, as an increasing fraction of the fuel is released from the canister. This observation will be discussed and explained in Section V. In addition, the maximum k_{eff} values for M_0=1000kg and M_0=1500kg are nearly equal to each other, which indicates that if the canister and buffer are in the same configurations, which are designed to realize sub-critical right after disposal, the initial loading might not be a sensitive factor to determine the magnitude of k_{eff} of the system.

Comparing cases (1) (solid points) and (2) (hollow points), in all time steps with water in canister, case (2) gives much higher k_{eff} than case (1). Case (2), in which fuel particles are assumed to "float" in the canister, might be considered unphysical but more conservative. This result is consistent with our findings in the literature survey.

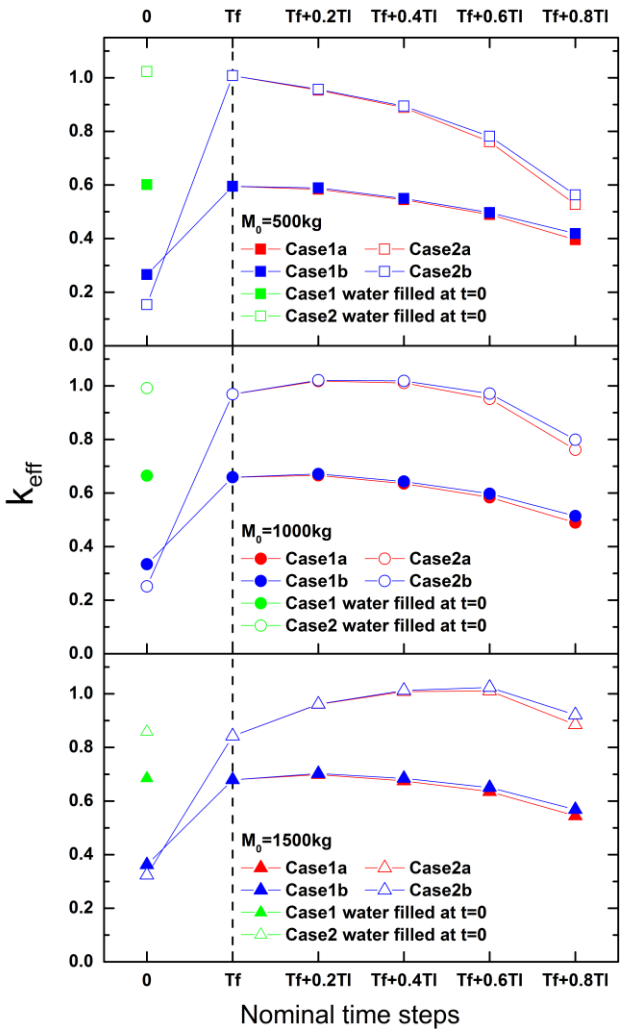

Fig. 2. Calculated k_{eff} for various cases versus time. Solid points represent case (1), hollow points represent case (2). Red points represent case (a), and blue points represent case (b). Squares, circles, and triangles represent initial loading of 500kg, 1000kg, and 1500kg, respectively.

Another important finding is that, after a fraction of damaged fuels is released, by mixing all the released fissile nuclide with the corroded porous carbon-steel canister, case (b) results are only slightly higher than results from case (a), and be very different from the canister at T_f, which have the same amount of total fissile material. This result shows that, within the assumed conditions, although the fissile materials released from the canister are assumed to be conservatively retained within the canister-buffer system, the influence on criticality can be very small.

The above findings from numerical results will be discussed in details in the next section.

V. Discussions

V. A. Influence of Leakage and Moderation

Generally speaking, nuclear criticality is influenced by neutron leakage and moderation[14]. Among the parameters defined in the present work, neutron leakage is related with the mass of fuels in the canister $M(t)$. The greater mass in canister, the lower the neutron leakage, and the higher the k_{eff}. The moderation, or neutron slowing down, is determined by the amount of water relative to the amount of the fuel in the system, and can be represented by the D/R ratio. With fuel particle radius R assumed constant, the larger pitch distance D, the more neutrons are moderated or thermalized.

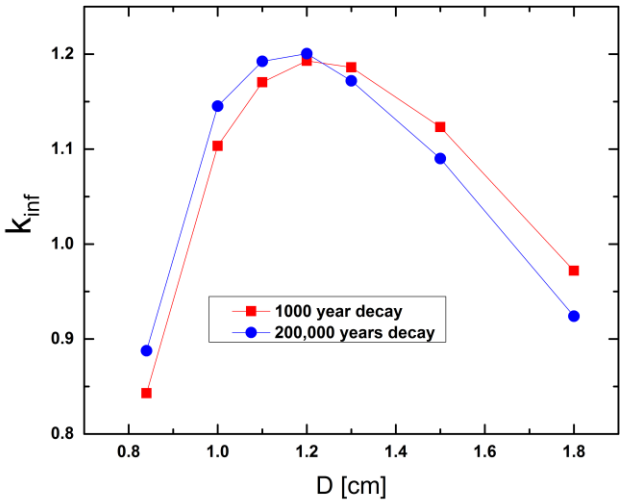

Fig. 3. Neutron multiplication factors of infinite fuel particle lattice (k_{inf}) for different pitch distance D.

The dependence of criticality on moderation can be understood by calculating the neutron multiplication factor k_{inf} of infinite fuel particle lattice for different pitch distance D. The results for damaged fuels after different decay time are plotted on Fig. 3. Note that the pitch distance must be greater than or equal to the diameter of the fuel particle (in the present work $2R$=0.84). Both curves shows drastic changes in k_{inf} as a function of D; the maximum is found at around D=1.2 cm, and the minimum (within the range of calculations) is found at D=0.84 cm. The system is called under-moderated or over-moderated when D is lower or higher than the value for maximum k_{inf} respectively. By definition, case (1) assumes fuel particles are closely packed, in which D=0.84 remains constant at its minimum for different time steps. While in case (2), according to eqn. (6), the pitch distance is increased as time increases, and is decreased with greater initial loading M_0, but be always higher than 0.84.

As a result, case (2) will always give higher k_{eff} than case (1). For case (2), the initial loading which gives $D(t=0)$=1.2 can be calculated from eqn. (6) as M_0^*=680.4 kg. If M_0 is smaller than M_0^*, $D(t=0)$ will be greater than 1.2, and the system started under-moderation will be even further under-moderated as time increases. However, if M_0 is greater than M_0^*, the system will reach its optimized moderation after a fraction of the fissile nuclide has been released. This explained the observations for case (2) results when M_0=1000 kg and 1500 kg.

V.B Role of the Carbon-Steel Canister

The canister considered in the present work is made of 14 cm thick carbon steel. The thickness of the canister was primarily determined to delay sufficiently the failure from corrosion, so that at least the major heat-emitting radionuclides, Cs-137 and Sr-90, in the waste canisters will decay out. However, our numerical results indicate that the corroded canister could profoundly enhance the criticality safety. For all initial loadings after canister failure time in Fig. 2, case (a) and case (b) show very close results, which means mixing the fissile nuclide with the corroded canister is almost equivalent to removing those nuclides from the system. A fundamental reason for this result is that the minimum critical mass of a homogeneous mixture between the damaged fuel and corroded canisters or buffer (ranging from several tons to infinity) is much higher than the minimum critical mass of the fuel particle lattice (the critical mass for damaged fuels from Fukushima Unit 1 in water reflected spherical geometry with optimized moderation is around 419 kg). Therefore, as long as a fraction of the damaged fuels remains, the neutronics property of system is principally determined by the materials inside the canister.

Coming back to the three stages introduced in Section I in CSA: (1) the stage before package failure, (2) the stage after package failure, while fissile nuclides remain within the engineered barriers, and (3) the stage in which fissile nuclides originated from multiple packages deposit in far-field host rocks. The boundary between stages (2) and (3) was not well defined. There are conceivable scenarios when fissile nuclides from multiple canisters form a deposition near the canisters while there are fissile nuclides remaining in one or more of these canisters. Naturally, the situation can be much more simplified if the deposition and individual canisters can be decoupled and modeled separately. Whether or not this can be achieved certainly depends on the design of the disposal system. Although further confirmation by numerical results is necessary, the canister seems to play an important role to decouple the domains inside and outside the canister from CSA.

V. C. Criticality Constrains for Canister Design

The engineered barrier, including canister and buffer, needs to be designed to minimize or eliminate the

potential of criticality over hundreds of thousands of years. Sub-criticality with certain safety margin assuming water flooded configuration with geometry fixed at initial condition is the one which is the most commonly used in spent fuel canister design criteria.

Due to the uncertainties on geometry change during material degradation, various conservative assumptions could be applied. For the case of damaged fuels, the geometry of materials is not well known at the time of emplacement. Therefore, in the present work, two different cases have been defined and compared. Case (2) is considered more conservative but unphysical. However for case (1), due to the existence of cladding and structural materials, fuel particles could not be so closely packed, resulting in underestimation of k_{eff} due to optimistic assumption of poor moderation.

If the canister and the initial loading need to be designed to be subcritical according to the case (2) assumptions, several interesting points could be pointed out. First, depending on the initial loading, the maximum k_{eff} value might or might not occur before canister failure, when the canister contains maximum amount of fissile nuclides. Second, the maximum k_{eff} value could be almost independent from the initial loading, when the initial loading exceeds a threshold. The threshold is the mass of fuel when the canister is filled with fuel particles with optimized pitch distance for moderation. This quantity has already been defined as M_0^* in Section V.A, which is dependent on the dimension and material of the canister but independent of the initial mass loading. Even if the initial mass loading is much greater than the threshold, the maximum k_{eff} value will not increase, because (1) the maximum k_{eff} only occurs when the mass inside the canister is reduced to the amount of the threshold (see Section V.A), and (2) the materials released from the canister contribute very little to the criticality (see Section V.B).

From the above discussions, if the canister could be designed to be sub-critical (with necessary safety margin) with initial loading at M_0^*, the system is very likely to be sub-critical with initial mass loading higher than M_0^*. As a result, actual limit on the initial mass loading might not be coming from criticality, but by other constrains such as decay heat emission. The numerical results also indicate that, the criticality control possibly could be achieved by limiting the moderation. For example, adding backfilling materials inside the canister could be a promising option to control criticality by reducing the void space (which will be taken by water after canister failure) in the canister. We will continue developing this preliminary idea in the future works.

VI. CONCLUSION

The present work focuses on neutronics analysis to evaluate the criticality safety for a system consisting of a canister containing fuel debris from Fukushima Unit 1 reactor and the surrounding buffer, in a water-saturated deep geological repository. The composition and dimension of the canister and buffer are assumed according to Ref. 8. Based on literature review, the fuel debris has been modeled as a hexagonal lattice of spherical fuel particles. The pitch distance has been determined either by (1) making particles in contact each other or (2) letting the particles float in a lattice fully filling the canister. And during the leach time, the released materials from the damaged fuels are assumed to be either (a) removed from the system, or (b) be homogeneously mixed with the corroded canister. Combining the variations above, k_{eff} values were calculated by MCNP code for four different cases were compared at different time points for different initial canister loadings.

Based on the numerical results, the following key observations can be made: (a) the calculated neutron multiplication factor (k_{eff}) is sensitively dependent on assumptions related to moderation, (b) the carbon steel canister plays an important role in reducing the potential for criticality, (c) the maximum k_{eff} of the canister-buffer system could be achieved after a fraction of fissile nuclides been released from the canister, and (d) under several assumptions, the maximum k_{eff} of the canister-buffer system could be principally determined by the dimension and composition of the canister, not by the initial fuel loading.

Future works are planned to apply the present approach for damaged fuels from Unit 2 and Unit 3, to consider more modes for release from the canister, such as leaching of the damaged fuels by reducing the radius of each fuel particle, to consider buffer swelling or collapsing due to degradations, and to develop detailed models to connect the models for single canister with models for the deposition from multiple canisters. The dependence on model parameters, such as fuel particle radius, need to be further examined. We will also investigate the option of using backfilling materials to control criticality in the engineered barrier design.

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