Consequence Analysis of Core Meltdown Accidents in Liquid Metal Fast Reactor

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Abstract

Core disruptive accidents have been investigated at Korea Atomic Energy Research Institute(KAERI) as part of work to demonstrate the inherent and ultimate safety of the conceptual design of the Korea Advanced Liquid Metal Reactor(KALIMER), a 150 Mwe pool-type sodium cooled prototype fast reactor that uses U-Pu-Zr metallic fuel. In this study, a simple method was developed using a modified Bethe-Tait method to simulate the kinetics and hydraulic behavior of a homogeneous spherical core over the period of the super-prompt critical power excursion induced by the ramp reactivity insertion.

Calculations of energy release during excursions in the sodium-voided core of the KALIMER were subsequently performed using the method for various reactivity insertion rates up to 100 $/s, which has been widely considered to be the upper limit of ramp rates due to fuel compaction. Benchmark calculations were made to compare with the results of more detailed analysis for core meltdown energetics of the oxide fuelled fast reactor. A set of parametric studies was also performed to investigate the sensitivity of the results on the various thermodynamics and reactor parameters.

Introduction

In early safety studies of small uranium metal reactors like EBR-II [1] and the Fermi Reactor[2], a sequence of super-prompt critical accident caused by fuel slumping in the sodium voided core, which is eventually terminated by disassembly of the core, was assumed to set the upper-bound design limits of containment systems. The analytic method used in the evaluation of this type of super-prompt critical core disruptive accident(CDA) in a fast reactor was originally developed by Bethe and Tait [3], and further elaborated by Jankus [4].

Various studies, mostly for the oxide-fuelled cores, since have indicated that such a rate of reactivity insertion with coherent slumping of the whole core would be impossible. Moreover, molten fuel would move down through the lower structure, spreading widely into the lower plenum. It was shown that the most severe classes of events that have the potential to develop into core disruptive accidents are the unprotected transient overpower(UTOP) and unprotected loss of flow(ULOF), which are extremely unlikely to occur and to be arrested should they occur[5]. It was estimated that, depending upon the coherence of fuel slumping, a few tens of dollars per second of reactivity insertion rate would be theoretically possible in the steady-state core when fuel slumping starts. In this study, the upper limit of reactivity insertion rate was set at 100$/s to test the structural strength of the reactor system.

In an effort to evaluate the inherent safety of a conceptual design of the KALIMER for core

meltdown accidents, a simple analytic method has been developed based on the Bethe-Tait method. Modifications were made to the original method mainly in the use of a more realistic equation of state of the fuel as well as the inclusion of the Doppler reactivity effect. The equations of state of the pressure-energy density relationship were derived for the saturated-vapor as well as the solid liquid of metallic uranium fuel, and implemented into the formulations of the disassembly reactivity. Mathematical formulations and a computer code called SCHAMBETA[6] were developed in a relevant form to utilize the improved equations of state as well as to consider the Doppler effects. Calculations of the energy release during excursions in the sodium-voided core of the KALIMER were subsequently performed using the SCHAMBETA code for various reactivity insertion rates up to 100 $/s.

To test the accuracy of calculations with the simple method developed, a number of calculations were carried out and compared with a more detailed analysis results given in the work by Hicks and Menzies for oxide fuelled fast reactor[7]. Our method results in conservative estimate of the core energy density relative to those by Hicks and Menzies. Various parametric studies was also performed to investigate the sensitivity of the results on the equation of state for pressure and energy, and other thermodynamics and reactor parameters. A scoping code like SCHAMBETA proved very useful for sensitivity studies of various parameters of uncertainties. Sensitivity studies are in need particularly for the fast reactor core loaded with metal fuel, for which our experience and knowledge are limited relative to the oxide-fuelled core.

Basic Approach

It is assumed that the power excursion begins with the reactor prompt critical at time zero and the energy density generated during the excursion is governed by the reactor kinetics equation with no delayed neutrons and the source,

$$\frac{d^2Q}{dt^2} = \frac{k-1-\beta}{\lambda}\frac{dQ}{dt} \qquad (1)$$

where $Q(t)$ is the time dependence of the energy generation density $E(\vec{r},t) = N(\vec{r})Q(t)$. $N(\vec{r})$ is the normalized spatial power distribution. The other quantities in Eq.(1) is expressed in standard notation; k for multiplication constant, λ for prompt neutron lifetime, and β delayed neutron fraction.

The neutron multiplication constant as a function of time may be expressed in the form

$$k(t) = k_0 + k_I(t) + k_d(t) + k_D(t) \qquad (2)$$

where k_0 is the initial multiplication constant, $k_I(t)$ is the reactivity insertion responsible for initiating the excursion, $k_d(t)$ is the reactivity feedback resulting from material displacement during disassembly process, and $k_D(t)$ is the feedback from Doppler effect. The initial multiplication constant at prompt critical is by definition $k_0 = 1+\beta$. Initial energy content $Q(0)$, initial power level $\dot{Q}(0)$, and $k(0)$ are the initial conditions to be specified for a set of the coupled equations in the above to have a unique solution. Starting with the initial conditions, the coupled equations (1) and (2) can be numerically solved by iteration[8,9].

Reactivity Insertion Rate

The rate of reactivity insertion initiating the excursion is assumed constant and $k_I(t)$ may be written as; $k_I(t) = [dk/dt]t = \alpha t$. In the case that a ramp insertion of reactivity initiates the accident, an equivalent step insertion is frequently used in

the Bethe-Tait type of analysis. For the purpose of determining the equivalent step insertion, it is convenient to divide the power excursion into two phases. During the first phase, reactivity is added at an assumed constant rate and power rises until time t_1, when the total energy generated becomes sufficiently large to produce pressures that bring about significant material movement. Once the core begins to disassemble it goes very rapidly, and it is found that one can safely neglect any further addition of reactivity afterward. It is assumed in our study that t_1 comes when the fuel boiling occurs at the peak power location of the core. An asymptotic representation of the time t_1 may be obtained by solving Eq.(1) without reactivity feedback. The result is given as[8,9],

$$t_1 \approx \sqrt{\frac{\lambda}{\alpha}}\sqrt{\ln X + \ln(\ln X)} \tag{3}$$

where

$$X \equiv \frac{\alpha Q^2(t_1)}{\lambda}[\dot{Q}(0)]^{-2} \tag{4}$$

Total reactivity inserted by the ramp prior to the large pressure is then given by

$$k_1(t_1) = \alpha t_1 = \sqrt{\alpha\lambda}\sqrt{\ln X + \ln(\ln X)} \tag{5}$$

Initial multiplication constant is then defined as; $k(0) = k_0 + k_1(t_1) = 1 + \beta + \alpha t_1$. Since the net reactivity is initially at its maximum and reduced with negative reactivity feedback from the Doppler effect and /or core disassembly during the excursion, $k_1(t_1)$ is termed k_{max} in the following for clarity as well as for convenience.

Disassembly Reactivity Feedback

Applying the first-order perturbation theory to the one group diffusion equation, we obtain an expression for the second derivative of the disassembly reactivity in time for a spherical reactor,

$$\ddot{k}_d = \frac{48q^2F/\rho_c}{4\pi\Sigma_{tr}\nu\Sigma_f b^7[1-(6q/5)+(3q^2/7)]}\int pdV \tag{6}$$

where Σ_{tr} , $\nu\Sigma_f$ are the transport and fission cross-sections, b is the core radius, F is fraction of fission in the core, ρ_c and p are density and pressure of the core, respectively. It was assumed that the flux can be approximated by a parabola in the core, $\Phi = 1 - q(r^2/b^2)$. Thus $\ddot{k}_d$ is proportional to the pressure integrated over the volume of the core. The pressure-energy relations for the core during the power excursion are among the key parameters to be provided for the core disassembly process.

At the initiation of the super-prompt critical excursion, liquid uranium is assumed interspersed with void spaces left in the core when the coolant is expelled. As the temperature rises, the voids are filled with the expanded liquid producing saturated vapor pressure. If the liquid reaches the threshold energy to fill the voids completely, the pressure begins to rise rapidly thereupon. In this context, therefore, equations of state of pressure-energy density relationship are derived for the saturated-vapor as well as the single-phase liquid of metallic uranium fuel. In this study, vapor pressure is defined as a fourth-order polynomial of

energy density ; $p = \sum_{i=0}^{4} B_i E^i$, while a Bethe-Tait form of linear threshold equation of state is employed for the single-phase liquid region..

Benchmark Analysis of Oxide-Fuelled Core

A series of simulations was performed for the cases studied by Hicks and Menzies, as a means of checking the extent of the accuracy or conservatism of our method, particularly the assumption of step reactivity insertion equivalent to ramp rate. Hicks and Menzies investigated various aspects of the course of events during and following a hypothetical meltdown accident in a sodium-cooled PuO_2 /UO_2 fuelled fast reactor. An extensive set of density equations of state for temperature and energy density as well as pressure and energy density was developed for the fuel assumed to be UO_2. The Doppler constant was estimated to be –0.24 %.

In our calculations of energy release, information available in the report by Hicks and Menzies or typical values of oxide-fuelled core were assumed for the reactor parameters. As for the pressure-energy density relationship at constant volume, the one developed by Hicks and Menzies for the saturated vapor plus the single-phase region with the specific reduced volume equivalent to the core density of the KALIMER was curve-fitted to a fourth-order polynomial series.

The results of these calculations are summarized for several reactivity insertion rates in Table 1, which compares the peak values of the energy density at the core center with those given by Hicks and Menzies. It may be noted from the fourth column of the table that our method, using the asymptotic values of t_1 and k_{max} given in Equations(3), (4), and (5), consistently predicts higher values of energy release(about two times on average) than those of Hicks and Menzies.

Table 1. Comparison of Energy Densities of Oxide-Fuelled Core

Ramp Rate ($/s)	Doppler Constant (α_D)	Energy Density at Core Center(KJ/g)		
		Hicks & Menzies	Our Methods	
			Asymptotic k_{max}	Adjusted k_{max}
75	0.0	3.52	7.25	3.47
	-0.001	1.81	4.10	1.63
	-0.002	1.71	2.42	1.57
150	0.0	4.11	8.82	3.96
	-0.001	1.99	5.30	1.76
	-0.002	1.90	3.55	1.70

The trend of overestimates of our method for energy release mostly comes from the conservatism put into estimating the amounts of step reactivity equivalent to the ramp rates. It was observed in the course of our calculations that there exist significant differences between the asymptotic values of t_1, time of fuel boiling at the center of the core obtained as per Equation (3), and the actual values of time of boiling, t_b , which resulted from our analyses of the excursions. The values of t_1 are much larger than those of t_b, about 50 times larger depending on the ramp rates and Doppler constants assumed for the study. Being converted into the inserted reactivity by Eq. (5), this gives rise to overestimation of k_{max} , which comes to drive power excursions much more severely in our calculations. The values of t_1 were

subsequently adjusted so that the resulting values of t_b come close to t_1 for each case of excursions. The results are listed in the last column of Table 1. As shown, the results are in good agreement with those of Hicks and Menzies, being within about 10 % in the case of ramp rates of 75 and 150 $/s, which are in the range of our design-basis ramp rates. Such agreement appears fairly remarkable, considering the uncertainties involved in these kinds of hypothetical accidents, including the high-temperature material properties, equations of state and reactor parameters, among others. Even the asymptotic method can be of use for a conservative estimate of the accident energetics in such a scoping study.

Analysis of Core Meltdown Accident in KALIMER

The KALIMER core system is designed to generate 392MWt of power. The reference core utilizes a heterogeneous core configuration with driver fuel and internal blanket zones alternately loaded in the radial direction[10]. Details of the configuration of the KALIMER core and the reactor parameters used in this study for the base cases are described in the paper presented last year in this conference[8].

The core is assumed to be initially in molten state. Initial energy content of the core, Q_0, is therefore taken to be 0.25 KJ/g, the internal energy needed to heat uranium from room temperature to the melting point(1,400 K). The boiling temperature of the core is set at around 4,500 K and the corresponding energy Q_b at 0.8 KJ/g. The specific heat of metallic fuel is assumed to be close to 0.2 J/g-K just above the melting point and assumed to stay constant beyond.

A vapor pressure equation for uranium is given by Raugh and Thorn [11],

$$\log p = 5.702 - \left(\frac{23{,}300}{T}\right) \qquad (7)$$

where pressure is in atmosphere and temperature in K. We need an expression relating pressure to energy rather than to temperature. Assuming 0.1J/g-K as a reference value of the specific heat of the vaporized uranium core, the pressure-temperature relation was converted to that of pressure and energy density, which was then curve-fitted to a fourth-order polynomial, $p = \sum_{i=0}^{4} B_i E^i$,

where $B_0=1.297\times10^3$, $B_1= -6.018 \times 10^3$, $B_2=10.495 \times 10^3$,
$B_3= -8.182\times10^3$, $B_4 =2.416\times10^3$ (8)

Pressure is measured in MPa and the liquid energy in kJ/g (See Figure 1).

Meanwhile, for the single-phase liquid region, an equation of state is developed in a linear threshold type. The use is made of the equation-of-state data calculated by Brout[12] for the uranium density of 10.0 g/cm^3,.which is close to the density of the sodium-voided core of the KALIMER. The result of our fitting is

$$p = 11{,}000(E - 1.10) \qquad (9)$$

where the pressure is measured in MPa and the liquid energy is in kJ/g (See Figure 1).

The fuel temperature (Doppler) coefficients are evaluated for sodium-flooded/voided cases. It is estimated to vary as $0.11T^{-1.49}$ for the sodium-voided case, whereas it varies as $0.10T^{-1.43}$ in the case of the sodium-flooded core. The Doppler coefficient does not show any substantial change with burnup. Taking into account some uncertainty in the correlation for Doppler coefficients, –0.002 is taken as the best-estimate value of the Doppler constant for subsequent analyses, for the sake of conservatism [13]..

Results of the reference case are listed in Table 2 for the peak values of energy generation density, temperature and pressure for reactivity insertion rates in the range of 10 to 100 dollars per second, with three different values of Doppler constants considered; α_D = 0(no Doppler effect), –0.001 and –0.002. With the Doppler constant of -0.002, which is set to be the reference value for the KALIMER in this study, the power excursions are terminated even before the core reaches the assumed energy density of the boiling point (0.8KJ/g) for reactivity insertion rates up to 50 $/s. And reactor would shutdown without any significant pressure rise or energy release. For the reactivity insertion rate of 100$/s, the energy density at the peak location of the core goes over the boiling point but stays around the threshold value of the solid liquid region (1.10 KJ/g). Only the peak spot of the core would boil, however, whereas most area of the core would be in the pre-boiling liquid state. As the fuel vapor generated at the peak spot of the core fill some of the voids left out of sodium coolant, the pressure gradually rises, while the power continues to be in decline under the influence of Doppler feedback. The core dispersion would be then with the fuel of low energy density driven by relatively low pressure.

Table 2. Calculated Results of Energy, Temperature and Pressure

Ramp Rate ($/s)	Doppler Constant (α_D)	Peak Values at Core Center		
		Energy Density(KJ/g)	Temperature (K)	Pressure (bar)
10	0.0	1.46	7,440	4,030
	-0.001	0.68	3,560	38
	-0.002	0.48	2,570	0
20	0.0	1.58	8,030	52,400
	-0.001	0.91	4,720	70
	-0.002	0.58	3,040	0
50	0.0	1.79	9,090	75,700
	-0.001	1.32	6,770	1,700
	-0.002	0.80	4,160	0
100	0.0	2.01	10,220	100,700
	-0.001	1.58	8,030	5,250
	-0.002	1.12	5,770	360

Another set of calculations was repeated with the amounts of the maximum step reactivity k_{max} adjusted as in the benchmark analysis with an oxide-fuelled core. The result was that all the power excursions studied were terminated before the

core got to its boiling point, given the reference value of Doppler constant. It may also be noted that the accidents are terminated without any significant energy yield even with a lower value of the Doppler constant (eg., α_D = –0.001).

Sensitivity Study

Equation of State

The saturated vapor pressure of uranium, as given by Eq.(7), has been shown to provide the vapor pressure in reasonable accuracy from the melting point to the critical point. The framework of methods in our study, however, requires an expression relating pressure to energy rather than to temperature. The specific heat of uranium is not well known in the high temperature region, particularly so above the vaporization temperature. Under the circumstances, the reference value of specific heat constant of uranium is set to be 0.1 J/g-K at the vapor region, in line with the works by Brout and Nicholson[12]. Eq.(8) lists the coefficients of the polynomial for the reference value of the specific heat. Sensitive studies are carried out assuming 0.2 J/g-K as the specific heat, which is expected to yield higher pressure as well as energy release. The resulting coefficients for the pressure as function of energy, curve-fitted to a fourth-order power series are $p = \sum_{i=0}^{4} B_i E^i$;

where B_0 = -87.25 , B_1 = 104.7 , B_2 = -1.530 , B_3 = -36.25, B_4 = 11.07 (10)
where the pressure is measured in MPa and the liquid energy in KJ/g.

As for the single-phase region, another equation of state is generated based on the Brout's data for the density of 7.4 g/cm^3 , to see the effect of higher value of threshold energy and slower pressure on the energy yield. The equation is

$$p = 5{,}940(E - 1.44) \quad (11)$$

where the pressure is measured in MPa and the liquid energy in KJ/g.
Two sets of the pressure-energy relationship for each phase then makes four cases of sensitivity calculation in this study, as illustrated in Figure1;

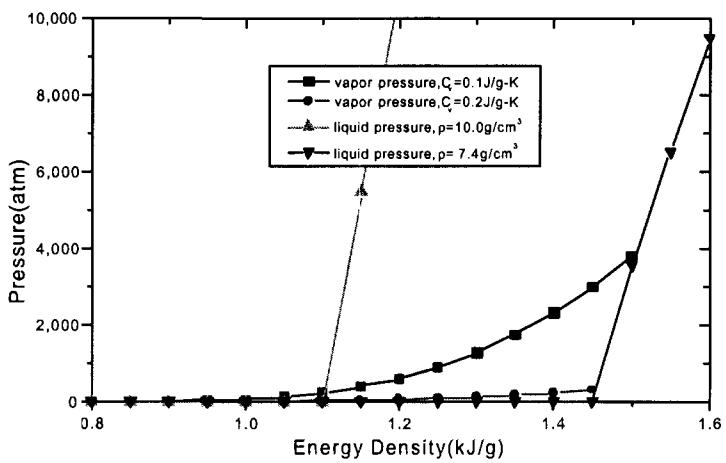

Figure1. Comparison of Equations of State

1) Case 1(Reference Case) : Eq.(8) for saturated vapor(C_v = 0.1 J/g-K),
   Eq.(9) for single-phase liquid (ρ = 10.0g/cm^3),
2) Case 2 : Eq.(10) for saturated vapor(C_v = 0.2 J/g-K),
   Eq.(9) for single-phase liquid (ρ = 10.0g/cm^3)
3) Case 3 : Eq.(8) for saturated vapor(C_v = 0.1 J/g-K),
   Eq.(11) for single-phase liquid (ρ = 7.4g/cm^3)
4) Case 4 : Eq.(10) for saturated vapor(C_v = 0.2 J/g-K),
   Eq.(11) for single-phase liquid (ρ = 7.4g/cm^3)

Results of the energy densities at the peak spot of the core are listed in Table 3 for each of the four cases described in the above. Calculations were made for reactivity insertion rates of 50 and 100 dollars per second with various Doppler constants for each case of the sensitivity study. There are essentially no differences in the results between the first two cases, meaning that the results are insensitive to the particular equation of state for the saturated vapor pressure. For Cases 3 and 4, where a higher threshold energy and slower rate of pressure rise were chosen, some differences exist due to the longer period of time to reach the threshold energy, but these are insignificant.

Meanwhile, we can see by comparing Cases 1 and 3 (,and Cases 2 and 4 as well) that the results are rather sensitive to the choice of linear threshold equation of state for the liquid. The differences are more pronounced with a smaller value of Doppler constant, reaching as much as 35 % for the cases of no Doppler feedback. It was observed that the value of threshold energy affects the results more than the gradient of the linear curve. For the Doppler constant of –0002, however, the results essentially remain the same upon changing the linear threshold equation, simply because the core is not heated up much above the threshold energy of 1.10 kJ/g. An observation to be drawn from this study is then that the results of energy release are not sensitive to the equation of state for vapor pressure. The threshold energy of the single-phase liquid of uranium affects the results but only when Doppler effects are rather small. For the KALIMER core, in which the Doppler constant is in the range of –0.002, equation of state for pressure should not be critical to such a scoping analysis as this study.

Table 3. Calculation of Energy Densities for Various Equations of State

Ramp Rate($/s)	Doppler Constant	Peak Energy Densities(kJ/g)			
		Case 1(Ref.)	Case 2	Case 3	Case 4
50	0.0	1.79	1.79	2.30	2.41
	-0.001	1.32	1.34	1.38	1.53
	-0.002	0.80	0.80	0.80	0.80
100	0.0	2.01	2.02	2.67	2.74
	-0.001	1.58	1.58	1.85	1.98
	-0.002	1.12	1.14	1.12	1.14

Specific Heat of the Metal Fuel

We have seen that the Doppler reactivity feedback effect plays a crucial role in determining the core behavior during the accidents. One of the parameter of importance for the effect is the specific heat of the metal fuel. There has been considerable disagreement about its value for uranium at a high temperature above its melting point. Some measurement or argument about the value of the specific heat of uranium has been made just above the melting point, the values ranging from 0.1 to 0.2 J/g-K[12]. More recent measurement indicates toward close to 0.2J/g-K[14,15]. There exist, however, large uncertainties about the behavior of the specific heat far above the melting point. A parametric study was performed, therefore, to look into the sensitivity of our calculation with three values of the specific heat at the melting temperature and beyond. Table 4 lists the peak-spot energy densities of the core calculated for each of the three values of specific heat.. In summary, effect of the specific heat of the fuel on the Doppler reactivity feedback would not be that important as long as it stays in the range of 0.15 to 0.2 J/g-K above the melting point

Table 4. Energy Densities with Various Specific Heat Constants

Ramp Rate ($/s)	Doppler Constant (α_D)	Peak Energy at Core Center (kJ/g)		
		C_v = 0.10 (J/g-K)	C_v =0.15 (J/g-K)	C_v =0.20 (J/g-K)
50	0.0	1.71	1.79	1.79
	-0.001	0.64	1.18	1.32
	-0.002	0.44	0.66	0.80
100	0.0	2.01	2.01	2.01
	-0.001	0.80	1.48	1.58
	-0.002	0.53	0.90	1.12

Reactor Parameters

There are a number of reactor parameters known to potentially influence the energy yield, which include the prompt neutron lifetime, and the power distribution in the core, and neutron cross sections, among others. In this scoping study, the power distribution in the assumed spherical core is represented by the power-shape factor q in the normalized distribution; $N(r) = 1 - q(r^2/b^2)$, where b is the core radius. A parametric study was performed with three values of q, 0.6, 0.4 and 0.2 in the increasing order of flattening, for various reactivity insertion rates and Doppler constants. When the value of q was changed from the reference value of 0.6 to 0.2, the peak energy density is increased by about 10% from the reference value, for the case of ramp rate of 100$/s and Doppler constant –0.002. If the Doppler effect is not considered, the amount of increase reached 20 % or so for the same case. The effect of the core flattening is shown to become smaller for the lower ramp rates. It is that the peak energy density turned out not so much sensitive to the power flattening as expected.

Parametric study was also performed for three values of prompt neutron

lifetime, 1×10^{-7}s, 2.65×10^{-7}s(design value) and 5×10^{-7} s. It was noted that, in the case of no Doppler effect, the energy release decreases with increase in neutron lifetime. With the Doppler effect considered, however, the energy density increases with increasing neutron lifetime. This is because, with a large Doppler effect, the amount of reactivity put into the core gets larger with a long neutron lifetime for a given ramp rate. With the increase of the neutron lifetime to about two times the reference value, the energy yield increases about 20% for the reference case of 100$/s reactivity insertion and the Doppler constant of –0.002. The extent of the increase gets smaller for lower ramp rates and Doppler constants. It was also shown in our parametric study that the results of energy yield are not sensitive to neutron cross sections of the core.

Conclusions

Analysis of the behavior of the sodium-voided core of the KALIMER during super prompt-critical excursions was performed for various reactivity insertion rates up to100 $/s, using the simple method developed in this study. The results show that there exist significant influences of Doppler effect on the power excursions in the metallic core of the KALIMER. For the best-estimate value of - 0.2 % (at the melting temperature) for the KALIMER, the power excursion was terminated without an energetic disassembly even for the extremely large reactivity insertion rates of 100$/s,

Benchmark calculations showed that our method predicts core energy density to be about two times higher than that of more detailed analyses by Hicks and Menzies on the oxide fueled core of fast reactor. It was found that the trend of our method to overestimate energy release mostly comes from the conservatism put into estimating the amounts of step reactivity equivalent to the ramp rates. With a parametric adjustment of the maximum reactivity inserted into the core, the results came to be in good agreement with those of Hicks and Menzies being within about 10 % in the range of our design-basis ramp rates. The current scoping method should be useful for first-time conservative estimate of core disruptive accident energetics. However, simulating the ramp insertion of reactivity may well improve the accuracy of the results.

Finally, sensitivity studies were performed to look into the influences of various parameters on the consequences of the power excursions. Parameters investigated in this study include equations of state for pressure and energy, specific heat of uranium, and such reactor parameters as neutron lifetime, power distribution and neutron cross sections. It turned out that the results of energy release were insensitive to the equation of state for vapor pressure. The threshold energy of the single-phase liquid of uranium affects the results to some degree but only when Doppler effects are rather small. For the KALIMER core, in which the Doppler constant is in the range of –0.002, equations of state for pressure should not be critical to a scoping analysis like this one. On the other hand, the specific heat of the fuel may significantly affect the consequence via changing the Doppler reactivity feedback effect if its value is out of the range between about 0.15 and 0.2 J/g-K at the melting point and beyond. The influence of the reactor parameters was not significant in terms of the peak energy density of the core as long as they remain within a reasonable range of the design value.

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References

1. L.J.Koch et al, "Experimental Breeder Reactor II(EBR-II), Hazards Summary Report," ANL-5719, 1957
2. Enrico Fermi Atomic Power Plant, "Technical Information and Hazards Summary Report," Part B, USAEC File N0. NP-11526, Oak Ridge Natioal Laboratory,1961
3. H.A.Bethe, J.H. Tate, " An Estimate of the Order of Magnitude of the Explosion when the Core of a Fast Reactor Collapses", UKAEA-RHM(56)/113., 1956
4. V.J. Jankus, "A Theoretical Study of Destructive Nuclear Bursts in Fast Power Reactors," ANL-6512.,1962
5. T.G.Theofanus, C.R.Bell," An Assessment of CRBR Core Disruptive Accident Energetics,"NUREG/CR-3224,1984
6. S.D.Suk," Development of the SCHAMBETA Code for Scoping Analysis of HCDA, ", KAERI/TR-1581/2000, KAERI Technical Report, 2000
7. E.P..Hicks, D.C.Menzies," Theoretical Studies on the Fast Reactor Maximum Accident," *Proceedings of a Conference on Safety, Fuels, and Core Design in Large Fast Power Reactors*, ANL-7120, 654-670, 1965
8. S.D.Suk, D.Hahn, "Scoping Analysis of LMFR Hypothetical Core Disruptive Accidents Using Modified Bethe-Tait Methods, " *ICONE-9*, Baltimore, Md, 2000
9. H.H.Hummel , D. Okrent, " *Reactivity Coefficients in Large Fast Reactors*," *Am. Nucl. Soc.*, Hinsdale, Il.,US, 1970
10. E.G. Rauh, R.J. Thorn, " The Vapor Pressure of Uranium," *J. of Chem.Phys.*, 22, 1954
11. D.Hahn et al., "KALIMER Preliminary Conceptual Design Report," KAERI/TR-1636/2000, 2000
12. R.B.Nicholson, "Methods for Determining the Energy Release in Hypothetical Fast-Reactor Meltdown Accidents", *Nucl. Sci. and Eng.*, 18, 207-219 ., 1964. Also see Author's Ph.D. Thesis, *U. of Michigan*, Ann Arbor, 1963
13. Y.I Kim., "KALIMER Equilibrium Core Conceptual Design and Analysis," LMR/CD120-ER-01,KAERI, 1999
14. H.P.Stephens, "Determination of the Enthalpy of Liquid Copper and Uranium with a Liquid Argon Calorimeter," *High Temp.Sci.* 6,156, 1974
15. W.L.Thayer and J.L.Robbins ," Enthalpy and Heat Capacity of Liquid Plutonium and Uranium",.*Proc. 5th Int. Conf. on Plutonium and Other Actinides,* Karlsruhe, 1975