Science and Global Security, 16:87–114, 2008

ISSN: 0892-9882 print / 1547-7800 online
DOI: 10.1080/08929880802565115

Compromising Safety: Design Choices and Severe Accident Possibilities in India's Prototype Fast Breeder Reactor

Ashwin Kumar[1] and M. V. Ramana[2]

[1]Department of Engineering and Public Policy, Carnegie Mellon University, Pittsburgh, PA, USA
[2]Centre for Interdisciplinary Studies in Environment and Development, Bangalore, India

This article explores the safety capabilities of the 500 MWe Prototype Fast Breeder Reactor that is under construction in India, and which is to be the first of several similar reactors that are proposed to be built over the next few decades, to withstand severe accidents. Such accidents could potentially breach the reactor containment and disperse radioactivity to the environment. The potential for such accidents results from the reactor core not being in its most reactive configuration; further, when there is a loss of the coolant, the reactivity increases rather then decreasing as in the case of water-cooled reactors. The analysis demonstrates that the official safety assessments are based on assumptions about the course of accidents that are not justifiable empirically and the safety features incorporated in the current design are not adequate to deal with the range of accidents that are possible.

INTRODUCTION

India plans a major expansion of nuclear energy based on fast breeder reactors (FBR).[1] The Indian Department of Atomic Energy (DAE) has been committed to this program for a long time and continues to pursue it even though many other countries have suspended their fast breeder programs. India is constructing an

Received 26 December 2007; accepted 14 October 2008.
This article has benefited from guidance and comments from many people, in particular, Frank von Hippel, Tom Cochran, Hal Feiveson, and one anonymous reviewer. All errors are the authors' responsibility. Alex Glaser and Ahnde Lin provided much appreciated assistance and useful references. The authors thank the Ploughshares Fund for financial support.
Address correspondence to Ashwin Kumar, Department of Engineering and Public Policy, Carnegie Mellon University, Baker Hall 129, Pittsburgh, PA 15213, USA. E-mail: ashwink@cmu.edu

industrial-scale Prototype Fast Breeder Reactor (PFBR), which will be the first of several breeder reactors in India.

Fast reactor programs in many countries have been suspended due to safety concerns (e.g., the SNR-300 reactor in Kalkar, Germany). Fast reactors have the potential for core disruptive accidents, in which large energies can be explosively generated. This article looks at the energetics of a core disruptive accident in the PFBR design from what is known in the open literature and the capabilities of the physical barriers of the design. The study finds that the PFBR is not designed to protect against a severe Core Disruptive Accident (CDA), and the DAE makes favorable assumptions that it has not justified. Even slight variations in their assumptions could have consequences far worse than acknowledged by the DAE and could overwhelm the PFBR containment. Additionally, many uncertainties are omitted from the DAE's published studies. These omissions are reason to doubt the safety of the Prototype Fast Breeder Reactor design.

The article begins with an overview of the Indian breeder program and the characteristics of the PFBR. Features of fast neutron reactors impacting safety are then highlighted, especially in the scenarios of severe accidents that involve melting and potential relocation of the fuel. It then discusses energy releases, and how these releases may affect reactor structures, in particular the overpressure on the containment. It concludes by assessing elements of the design of PFBR that compromise safety.

INDIA'S FAST REACTOR PROGRAM AND THE PROTOTYPE FAST BREEDER REACTOR

The DAE's three-phase nuclear energy strategy envisions use of both India's limited uranium reserves and much larger thorium reserves.[2] The first stage involves using uranium fuel in pressurized heavy water reactors, followed by reprocessing the irradiated spent fuel to extract plutonium. In the second stage the plutonium is used in the nuclear cores of FBRs. The nuclear cores of the FBRs could be surrounded by a "blanket" of either (depleted) uranium or thorium to produce more plutonium or uranium-233, respectively. Plans for the third stage largely focus on breeder reactors using uranium-233 in their cores and thorium in their blankets.

The only FBR that has been commissioned in India is the Fast Breeder Test Reactor (FBTR). The FBTR has suffered a series of accidents that prevented it from operating for long periods.[3] It took 15 years before the FBTR operated for more than 50 days at full power.[4] In the first 20 years of operation it operated for only 36,000 hours, resulting in an availability factor of approximately 20%.[5]

Even before the FBTR came on line in 1985, the DAE started making plans for a larger PFBR. In 1983, the DAE requested financial support from the government.[6] The first expenditures on the PFBR started in 1987–88.[7] In 1990, it was reported that the government had "recently approved the reactor's preliminary design and has awarded construction permits" and that the reactor would be on line by 2000.[8] But there were lengthy delays and construction of the reactor finally began in October 2004; the reactor is now expected to be commissioned in 2010.[9] The PFBR will be the first of the many breeder reactors that the DAE envisions building. By mid-century, the DAE has projected that it would install 262,500 megawatts (MW) in oxide and metallic fueled breeders.[10]

The PFBR has a power rating of 1,250 megawatts thermal (MWt) and 500 megawatts electric (MWe). It uses MOX fuel (a mixture of plutonium and uranium oxides) in the core and depleted uranium oxide in the blanket regions. The core has two enrichment zones, an inner one consisting of 85 fuel assemblies with a plutonium fraction of 21%, and an outer zone consisting of 96 assemblies with a plutonium fraction of 28%. The radial blanket consists of 120 assemblies, surrounded by a steel neutron reflector. There are 12 control and safety rods.

The DAE has set up two organizations to develop and construct breeder reactors, the Indira Gandhi Centre for Atomic Research (IGCAR) and the construction company, BHAVINI. In addition, the Bhabha Atomic Research Centre (BARC) has been involved in breeder research. This article uses the umbrella term "DAE" to refer to all three organizations.[11]

FAST REACTOR CHARACTERISTICS

The behavior of fast reactors is quite different in some respects from water-moderated thermal reactors, with implications for safety. The main differences are their neutron dynamics and properties of the coolant (liquid sodium in the PFBR). The same initiating events occurring in both thermal and fast reactors could produce very different outcomes.

A particular concern with fast reactors is that they are susceptible to large and explosive energy releases and dispersal of radioactivity following a core meltdown, or a CDA. CDAs have been the distinguishing concern in safety studies of fast reactors. The potential for a CDA results from the core not being in its most reactive configuration. If accident conditions cause the fuel bundles to melt and rearrange, reactivity could increase. This typically does not occur in a thermal reactor because moderation of neutrons is necessary to sustain a reaction. The core in thermal reactors is usually designed so that the fuel is in its optimal configuration and reactivity decreases when it is rearranged.

The progression of a CDA is classified into two phases. The first phase includes rearrangement of the core with increasing reactivity and increased fission energy production. This is accompanied by increased internal pressure due to coolant or fuel vaporization, ultimately leading to the second phase: explosive disassembly with rapid expansion of the fuel and subsequent termination of the chain reaction.

Another potential reason to be concerned about CDAs in fast reactors is that they could have a positive void coefficient. In such reactors, if the coolant heats up and becomes less dense, forms bubbles, or is expelled from the core, reactivity increases. This increase results from a slight moderating effect by the coolant, which slows the neutrons. The magnitude of the void coefficient is a measure of the feedback and tends to increase with core size.[12] The core design adopted for the PFBR has a value of $4.3.[13] A dollar worth of reactivity is an increase in reactivity equal to β, the delayed neutron fraction.[14] In thermal reactors, by contrast, faster neutrons typically lead to fewer fissions and thereby reduced reactivity.[15]

In isolation, the positive sodium void coefficient will lead to a self-reinforcing cycle in which the reactivity increase leads to further heating of the core. In the absence of control rod action, this will lead to very high fuel temperatures and possibly melting whenever a transient disturbance increases the coolant temperature. However, there are other feedbacks that tend to have the opposite effect.

Two negative feedback effects that are prompt are the axial expansion of the fuel, which tends to increase neutron leakage and reduce reactivity, and Doppler absorption, resulting in higher neutron capture as the temperature increases.[16] In addition, there are reactivity feedbacks due to the expansion of the core, its structural supports, and the control rod system. But these take time and are more difficult to quantify; some might be positive depending on the configuration. The extents to which these feedback effects stabilize a transient depend on various design details.

The neutronic behavior of plutonium-based fast reactors has an impact on safety because of the smaller fraction of delayed neutrons compared to uranium-based reactors. This implies that for even small increases in reactivity, the reactor could become prompt critical, wherein an exponentially growing chain reaction is maintained by prompt neutrons thereby making reactor stabilization through movements of control rods difficult.

Other safety concerns result from the choice of sodium as a coolant. Chosen mainly for its thermal properties (high boiling point and thermal conductivity, which make low pressure operation possible) and compatibility with cladding, its disadvantage is that it reacts violently with air and water.[17] These reactions are exothermic; additional failure modes must be considered in sodium-cooled reactors. These include the possibility of sodium–water reactions arising from leaks in the steam generator, sodium leaks from piping and resulting reactions with air, and sodium ejection from the primary vessel in a severe accident and the resulting pressurization and corrosion effects on the concrete containment building. Therefore, in addition to accidents involving large energy releases, FBRs are also prone to a host of less severe accidents, especially involving

Table 1: Maximum CDA work energy calculations for FBR systems.*

	Year critical	Power (MWth)	Approximate maximum CDA work energy (MJ)	CDA/Power ratio
Fermi	1963	200	2000	10
EBR-II	1964	65	600	9.2
SEFOR	1969	20	100	5
PFR	1974	600	600–1000	1–1.7
FFTF	1980	400	150–350	0.4–0.9
SNR-300	1983 (anticipated)	760	150–370	0.2–0.5
PFBR	2010	1200	100	0.083

*Calculations based in part on data from Alan E. Waltar and Albert B. Reynolds, *Fast Breeder Reactors.*

sodium, which have consequences for safety and economics.[18] This is not discussed further in this article.

CDA ENERGY RELEASES

Due to the potential magnitude of energy release, much of the research on safety in fast neutron breeder reactors has focused on core disassembly accidents. The first calculation of the energy released by core disassembly was carried out by H. A. Bethe and J. H. Tait.[19] Since then CDA studies have been conducted for nearly all of the FBRs constructed or proposed in the United States and Western Europe. Due to the role of Doppler feedback (neglected by Bethe and Tait),[20] improved treatment of fuel vapor effects, and the employment of a mechanistic mode of accident propagation, the majority of subsequent studies have produced energy estimates lower than the early Bethe Tait analysis (see Table 1). Despite the trend, the ratio of energy release to thermal power has not fallen below a value of 0.2 for any reactor.

The absolute numbers given in Table 1 are not particularly significant. "Because of the sensitive relationship between CDA energy release calculations and licensing considerations," it is argued, "it is difficult to obtain precise numbers."[21] Other analyses have estimated different values for CDA energy release. In the case of the Clinch River Breeder Reactor (CRBR), for example, estimates of the CDA (work) energy release have ranged from 350 MJ to 1200 MJ (implying ratios of 0.36 to 1.23 for work energy to power).[22]

Clearly the estimated work energy as a fraction of the nominal power for the PFBR is far lower than other fast reactors. There are no new reasons to expect a lower ratio in the PFBR than, say, for the SNR-300. Indeed, the sodium void coefficient is higher in the PFBR.

If one were to use a ratio of 0.2 for CDA energy to power, the lowest among all ratios, a PFBR CDA could lead to the release of 240 MJ. Using a figure of 1, more representative of the more cautious estimates, would result in a CDA energy release of 1200 MJ for the PFBR. The following sections sketch out a method by which one can estimate the CDA energy release.

Reactivity Insertion Rate

The energy releases from core collapse are dependent on the reactivity insertion rate. This is the rate at which the fuel rearrangement increases ("inserts") the reactivity of the reactor core. Once the reactivity insertion rate is specified, there are standard computer codes that calculate the total mechanical energy release, although even these have underlying uncertainties, especially in the modeling of the conversion of heat energy into mechanical energy.

During core collapse, an increase in reactivity may be influenced by several factors. These include rearrangement of fuel into a more compact geometry, such as when fuel in the upper half collapses to fill any voided coolant space below; changes in neutronic worth of the fuel due to changed relative positions, in particular when molten fuel flows to central regions after fuel pin failure; and by changing reactivity worths of control rods due to fuel collapse. If the reactor has a positive void coefficient, then the expulsion of coolant from central regions of the core, often accelerated by molten fuel coolant interaction, also contributes to reactivity increase.

These factors usually occur in combination, and a complete calculation of the dynamics of the process involves analysis of the coupled neutronics-thermal-hydraulics of the neutronically active regions of the core. In practice, CDA analyses have frequently used different computer codes for different stages.

More generally, it has been argued that due to the complexity of the reactor core during an accident and once large parts of the core melt, the detailed modeling of severe accidents using "mechanistic models" is extremely difficult and laden with uncertainties.[23] Therefore it has been "conventional to take a second approach and postulate a mechanistic series of events leading to an initial condition, accompanied by a pessimistic assumption of another event leading to a large ramp (i.e., reactivity insertion) rate."[24]

The specification of an initial reactivity insertion rate has been a central feature of accident studies of other fast reactors. A standard figure has been 100 dollars per second ($/s), illustrated by the sample exercise in the manual for the VENUS II code, a standard computer program used to calculate CDA energy releases.[25] For the CRBR, estimates have ranged from 100 $/s,[26] to 200 $/s.[27] The latter figure corresponds to reactor compaction of fuel to fill all the void spaces.[28]

A suggested rule of thumb is that a collapse velocity in cm/s is equivalent to the corresponding insertion rate in $/s. As presented in Appendix 1, a simplified calculation of the PFBR collapse lasts for 280 microseconds. Even assuming that the collapse is terminated earlier, that is, at 100 microseconds, the final fuel collapse velocity is approximately 1 meter per second, resulting in an insertion rate of approximately 100 $/s.

There is thus ample reason and precedent to use an insertion rate of 100 $/s as a benchmark for disassembly calculations, with the caveat that it is not an upper bound. As shown through a simplified and illustrative calculation for the PBFR (Appendix 1) that takes into account the reduced leakage arising from fuel collapse and the decreased reactivity worths of control rods, the resulting reactivity increase rates could be in excess of 100$/s.

In contrast, the DAE's study of a CDA in the PFBR uses an internally developed code called PREDIS.[29] Based on PREDIS, the DAE claims that the reactivity insertion rate cannot exceed 50 $/s. The PFBR containment design is meant to withstand an energy release of 100 MJ, which corresponds to an insertion rate of 65 $/s. But these relatively low reactivity insertion rates, and calculated energy releases, are the result of assuming only limited amounts of core involvement in disassembly. According to the DAE, at the end of the pre-disassembly phase, the molten fuel fraction is only 53%.[30] Barely half of the core can contribute to the reactivity increase. The collapse of the whole reactor core would lead to much larger reactivity insertion rates and energy releases. The assumption that there is a limited amount of core participation is not representative of a severe accident.

The results of the PREDIS code, and the consequent assumptions about extent of core participation and reactivity insertion rates, are questionable because there are several omissions in the DAE's pre-disassembly scenario (discussed in detail in Appendix 2 and Appendix 3). Broadly classified, these omissions relate to the neglect of fuel failure modes other than melting of fuel, neglect of safety constraints such as limiting coolant temperatures in the accident analysis, and ignoring uncertainties in thermophysical properties and reactivity feedback coefficients, especially that from axial expansion due to burn-up. These omissions are difficult to understand because the DAE's own studies reveal the importance of some of these factors. For example, its studies of protected transients suggest the possibility of cladding failure, but the DAE has ignored the potentially destabilizing feedback due to cladding relocation or fission product induced fuel coolant interactions (FCI).

These omissions render the detailed results of the calculation of accident propagation in the pre-disassembly stage questionable, and suggest that the reactivity insertion rate used in the DAE's disassembly calculation is not representative of a severe accident.

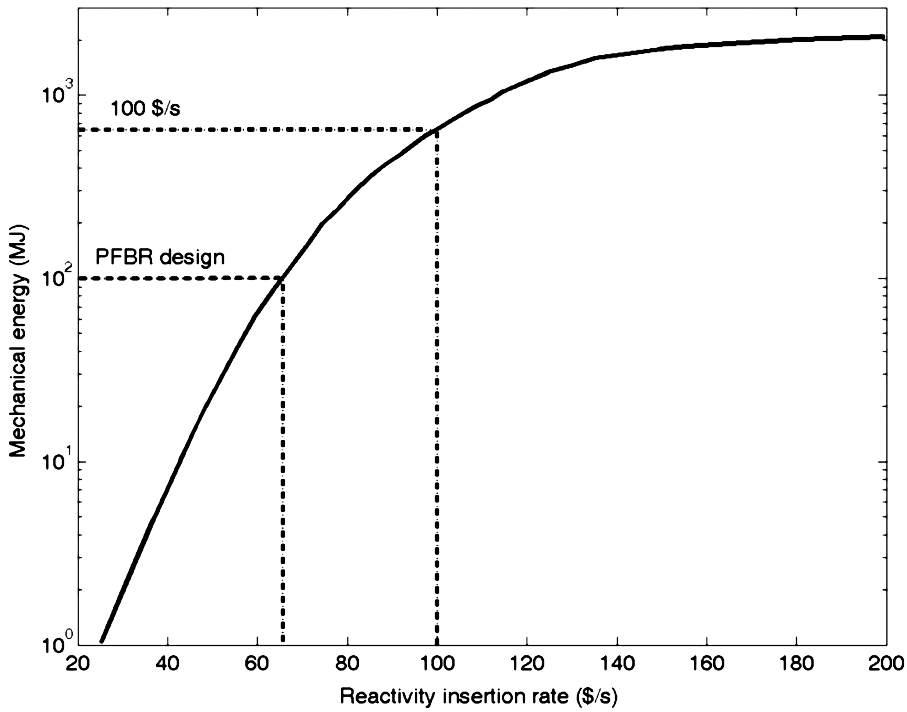

Figure 1: DAE's calculations of mechanical energy release for different reactivity addition rates.

Energy Release

Once the reactivity insertion is specified, standard codes, such as VENUS II, can be used to calculate disassembly dynamics and energy release as a function of reactivity insertion rate. Developed at the Argonne National Laboratory, VENUS II is a two-dimensional code that models the coupled neutronics and hydrodynamics processes in the molten fuel to calculate the space-dependent time histories of fuel temperatures, core material pressures, and core material motions during disassembly.[31] The VENUS II code only addresses the disassembly phase and does not calculate "what portion of the nuclear energy deposited during the excursion can ultimately be converted into work done on the containment."[32]

The DAE also uses the VENUS II code and their analysis of the PFBR is shown in Figure 1.[33] The mechanical energy release in Figure 1 "is calculated by assuming isentropic expansion of the fuel."[34] Using the DAE's figures for thermal and mechanical energy releases, the efficiency of conversion is approximately 1%.[35] The next section describes why this estimate may be low when modeling a severe accident.

Using a reactivity insertion rate of 65 $/s, the DAE estimates that the maximum work energy release in a CDA in the PFBR is only 100 MJ. This is an artifact of two assumptions: assumption of a low reactivity insertion, and a low thermal to work energy conversion efficiency of 1%. For a reactivity insertion rate of 100 $/s that should be considered a minimum benchmark for safety analyses, the energy release from a CDA is 650 MJ, if one assumes the same thermal to work energy conversion efficiency as the DAE.

Energy Conversion Efficiency

As the fuel vapor generated in a core disassembly expands, it transfers energy to its surroundings. At any given temperature, sodium has a higher vapor pressure than the vaporized fuel and will transfer energy more efficiently. Although high work energies of approximately 30% of heat energy are possible with thermodynamic considerations alone,[36] in the short time scales of the expansion, heat transfer is small and therefore fuel-coolant mixing and hydrodynamics must be considered.

There is very limited understanding of heat transfer in accident situations involving hundreds of kilograms of fuel. All experimental data pertain to comparatively smaller systems. At the lower end, tests performed on a few fuel pins at a time (less than a kg of core melt) at the Transient Reactor Test (TREAT) facility at Idaho National Engineering Laboratory suggest mechanical energy releases of a fraction of a percent.[37] But, at the higher end, tests at the U.K.'s Winfrith facility with core melt amounts of up to 25 kg suggest releases of approximately 4%.[38] In reactor safety calculations in France for the PHENIX and in the United States for the Fast Flux Test Facility (FFTF), efficiencies of 5–10% have been used.[39]

Assuming just 1% for the conversion of heat into mechanical energy therefore seems inadequate for safety evaluations. Higher conversion factors would imply higher mechanical energy releases and thus higher overpressures and greater damage.

POTENTIAL REACTOR DAMAGE DUE TO A CDA

The large quantity of energy released during a CDA will result in significant damage to the reactor. In order to reduce the likelihood of accidents leading to large environmental radioactivity releases, reactor designs typically rely on a "defense in depth" approach wherein multiple barriers have to be breached. In the case of most reactors, including fast reactors, these barriers include the fuel cladding, the primary vessel, and the outer containment building.

Because of high temperatures and the large amounts of fuel melting, CDAs result in significant cladding failure. There are other causes for cladding failure described in Appendix 2. Therefore, the first barrier is not significant in a CDA.

During the explosive disassembly phase of the CDA, the primary vessel or the containment could be breached in two ways; damage from the high-pressure shock wave caused by rapid expansion of fuel or coolant vapor, or as a result of impact from a “missile” such as a “slug” of coolant that is accelerated upward by the expanding vapor. The reactor vessel is designed so that the strain in the vessel head holds down bolts; the radial strain in the vessel also absorbs significant amounts of work energy.[40] Nevertheless, if the resulting force is large enough, it could dislodge the top cover of the pressure vessel or rupture the vessel; according to the DAE,[41] this is unlikely in the PFBR up to 1200 MJ of energy.[42] Even if neither of these events occurs, leaks could develop in the vessel and sodium could be ejected into the containment.

There are two ways that sodium can be ejected. First, a vertical pathway is cleared, allowing for unobstructed ejection of sodium into the containment; this could happen if a component becomes completely dislodged from the vessel. The Intermediate Heat Exchanger (IHX) is a likely weak link because its structural integrity has been assured only up to 200 MJ.[43] Second, the ejection could occur in a radial direction if bolts are strained and seals, if any, at the base of the bolt are broken, and the sodium escapes through the gap between the bolt and the vessel. According to the DAE,[44] a CDA with a 100 MJ energy release results in plastic elongation of the hold down bolts of components such as rotatable plugs, control plug, IHX, primary sodium pump, and the decay heat exchanger, by 0.5 to 1 mm.[45]

A direct vertical flow-path for the sodium could lead to a high-velocity spray that burns rapidly in the containment air. In contrast, a radial leak through gaps is likely to lead to formation of a sodium pool, which burns more gradually (i.e., in minutes).[46] In both cases, the sodium burns in the containment atmosphere leading to elevated temperatures and pressure. If these pressures are large enough, then the outer containment itself could be breached. Conversely, if the containment must ensure safety, it should be designed to withstand the pressures generated by the burning of ejected sodium in a CDA.

The PFBR design includes a rectangular, single, non-vented containment building made of reinforced concrete[47] designed to withstand 25 kPa of overpressure.[48]

Containment Overpressure From Sodium Burning

As discussed previously, a CDA could lead to sodium ejection into the containment building. The effects of sodium ejection on the containment are calculated in two steps. First, the effect of the energy released[49] (W) in the CDA on the amount of sodium leaked (Q) into the containment is determined by calculating scaling relationships among the important physical quantities followed by an analysis of the thermodynamic effects of sodium burning.

The high pressures generated during the vapor expansion create pathways similar to the ones described previously for sodium leaks to the containment building. The amount of sodium leaked, Q, is a product of the area A of the pathway, the velocity V of ejection, and the duration T of the pressure-wave produced during the CDA. Thus,

$$Q = V^{*}A^{*}T \tag{1}$$

The velocity V is related to the pressure P within the reactor vessel via Bernoulli's equation. Thus, V varies with pressure as $\sqrt{P}$.[50] The area of the leakage paths A is assumed to vary with pressure as P^{γ}. In a spray fire, where some direct paths are opened and the sodium is ejected in a vertical direction, A will be, to first approximation, independent of pressure. Thus, the value of γ will be zero. In a pool fire, where the leakage is proportional to the strain on the bolt, $\gamma = 1$.

The overpressure P at the upper wall of the reactor vessel is assumed to depend on CDA energy W as W^{α}. The resulting impulse, defined as $I = P \times T$, is assumed to be related to the energy as W^{β}. Experimental results for contained explosions in water suggest values of $\alpha = 0.50$ and $\beta = 0.66$.[51] Combining these relationships, the volume of sodium Q that leaks depends on the energy W as W^{n}, where $n = \alpha\,(\gamma - 0.5) + \beta$. For spray and pool fires, this leads to values of n of 0.41 and 0.91, respectively.[52]

Overpressure Generated by Spray Fires

In a spray fire, containment heating occurs rapidly within a few seconds and heat loss cannot mitigate the rise in pressure. For the range of sodium expulsion masses considered here, oxygen concentration is not a limiting factor.[53] Therefore, it is assumed that containment overpressure increases in proportion to the amount of sodium sprayed. The latter is proportional to the nth power of W, the energy released in the accident, with $n = 0.41$ as calculated previously. Thus, the ratio of overpressures produced at two different energies in the case of a spray fire scales as the ratio of the energies to the nth power.

The DAE estimates that the maximum credible energy release in a CDA is 100 MJ. It then calculates that such a CDA leading to sodium leakage into the containment, which burns in a spray fire, will result in a containment overpressure of 20 kPa.[54] This figure is used as the baseline to calculate the overpressure generated in a spray fire as a function of the energy released in a CDA in the manner described previously.

Overpressure Generated by Pool Fires

In a pool fire, the sodium burns in a flame sheet above the pool of molten sodium. Approximately 9.5 MJ of heat is produced for every kilogram of sodium that burns to produce monoxide.[55] The fraction of this heat that goes into heating the containment volume depends on the temperatures of the pool and the flame. A pool temperature T_{p} of 900 K and a flame temperature T_{f} of 1230 K[56] are used in this analysis. The burning of sodium is modeled at an assumed

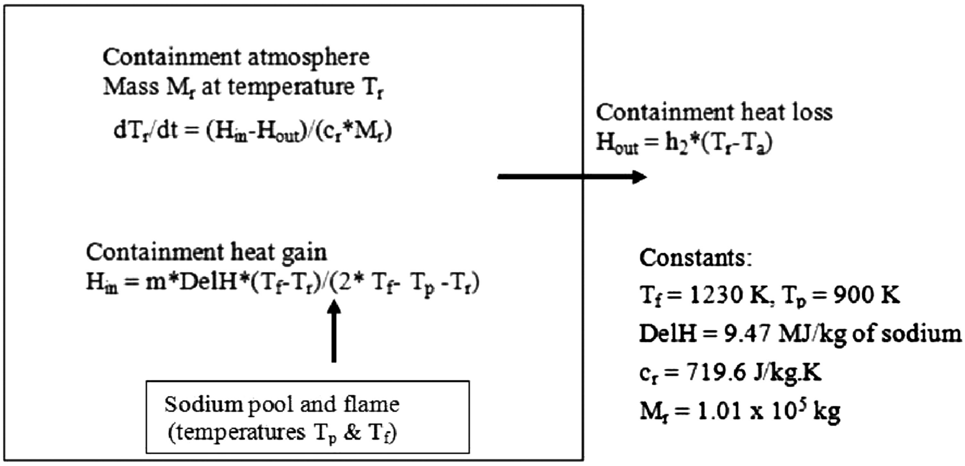

Figure 2: Calculation of containment overpressure in a pool fire.

rate (in kg/s) given by $m = kA_p$, where A_p is the surface area of the pool and k depends on the temperature of the pool. Empirical estimates of the coefficient k lie between 0.005 and 0.015 kg/(m^{2}sec), depending on the temperature of the pool.[57] An average value of 0.01 kg/(m^{2}sec), corresponding to a pool temperature of 900 K, is used in the calculations.[58] In the DAE's studies of a pool fire, 350 kg of sodium is consumed in approximately 15 minutes to produce an overpressure of 9 KPa on the containment.[59] From this, the area of the 350 kg pool is calculated to be approximately 40 m^2. On this basis it is possible to calculate the rate of heat production from the pool fire.

Impact on the Containment Atmosphere

It is assumed that the heat transferred to the containment atmosphere from the flame instantaneously increases the temperature of the containment atmosphere by a spatially uniform amount.[60] The containment walls are assumed to be instantaneously in temperature equilibrium with the containment air. The containment walls, in turn, lose heat to the environment by convection. To model this, it is assumed that heat is lost from the entire surface of the containment building to an outer environment temperature of 303 K. The heat transfer coefficient h_2 for this calculation[61] depends on the wall temperature, with a maximum value of approximately 39.5 kJ/K*s (Figure 2).

Using these values gives us a maximum containment pressure of 9 kPa when 350 kg of sodium burns as a pool fire, according to the DAE's estimate. The DAE multiplies its estimate by 1.3 to account for uncertainties in its computer models. Our analysis does not use any multiplicative factors.

For larger quantities of sodium, increases in the area of the pool fire are calculated with the assumption that the thickness is constant if the area of

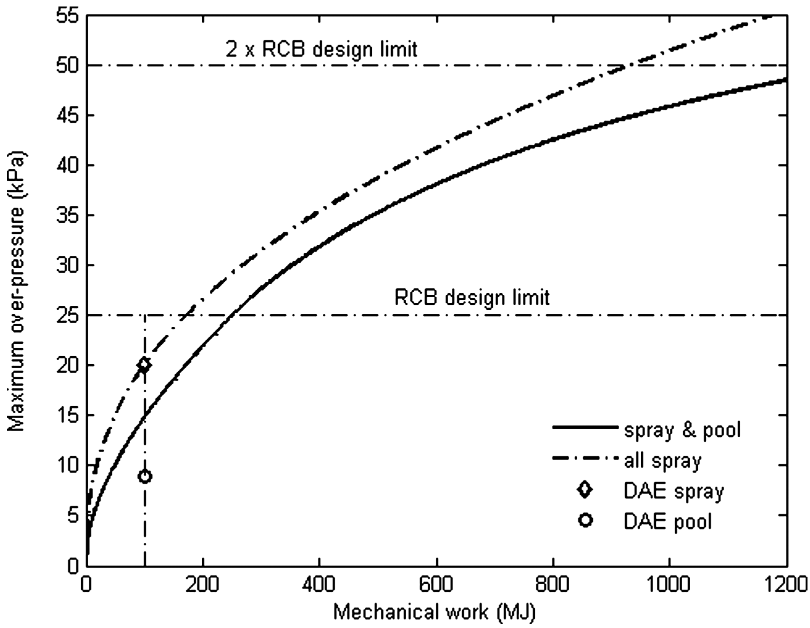

Figure 3: Containment overpressure as a function of mechanical work energy from a core disruptive accident.

the pool is less than 50 m^2 (the maximum available area on the vessel head for the sodium to occupy).[62] The model described previously is used to calculate the corresponding temperature and pressure increases.

To determine the overpressure on the reactor containment building (RCB) in a CDA, two possibilities are considered. First, that half the ejected sodium burns as a spray fire, while the rest burns as a pool fire. Second, more severe in terms of containment loading, is that all the sodium burns as a spray fire. Figure 3 illustrates the possible effects on secondary containment overpressure for a range of mechanical energy releases. A CDA that releases more than 200 MJ, a value that the DAE itself has considered in some of its studies,[63] could lead to overpressures in excess of the design value. Pressures twice the design value of 25 kPa can be exceeded if the mechanical energies exceed 900 MJ. Further, if the pressure vessel ruptures during a CDA, a much larger amount of sodium would rapidly enter the secondary containment, leading to even greater overpressures.[64]

There is not enough information available in the public domain regarding the PFBR's containment to calculate what might happen when it is subjected to this level of pressure, in particular whether the containment will maintain its integrity. Conservative safety studies typically assume that containments that are stressed much beyond their design pressure limits would fail; an example is the U.S. Nuclear Regulatory Commission's Reactor Safety Study.[65] However,

Table 2: Containment design specifications of demonstration fast reactors.*

Name	Thermal power (MWth)	Sodium void coefficient ($)	Volume V (m^3)	Pressure P (kPa)	V*P/E (kNm/MWth)
Phenix	563	—	31000	40	2.20×10^3
PFR	650	2.6	74000	5	0.57×10^3
CRBRP	975	2.29	170000	170	29.6×10^3
SNR-300	762	2.9	323000	24	10.2×10^3
MONJU	714	—	130000	30	5.46×10^3
PFBR	1250	4.3	87000	25	1.74×10^3

*Calculations based in part on data from IAEA, "Fast Reactor Database: 2006 Update."

it has been suggested that the ultimate level of pressure that would initiate a failure will be much higher, "assuming that good quality control practices assure that the construction matches the requirements."[66]

There have been questions about quality control in other reactors built by the DAE, including the quality of the containment buildings. In 1994, the inner containment dome of one of the units of the Kaiga atomic power station collapsed during construction due to faulty design.[67] This faulty design resulted from "a major alteration at site of the approved construction design."[68] Another cause was lack of adequate quality controls: According to DAE officials, "while inputs such as cement and steel had been tested for quality that was not the case with the concrete blocks as a whole."[69] These events do not inspire confidence that the PFBR's containment's could withstand high pressures.

Another question to explore is whether the containment has been designed with adequate safety margins. Answering this requires a comparison of its design with the containments of other demonstration reactors (see Table 2). The maximum design overpressure of the PFBR is low compared to most of these reactors, especially if the size of the reactor as measured by its thermal power is considered. Further, if the ratio V*P/E, a measure of the containment's capacity to withstand accidents is taken into account, the PFBR design performs worse than all other reactors except the Prototype Fast Reactor.[70] The difference appears more acute when considering the higher positive sodium void coefficient of the PFBR in comparison to other reactors.

Containments for light water reactors routinely have design pressures above 200 kPa.[71] The design for 700 MW pressurized heavy water reactors that the DAE is planning to construct includes a containment designed to withstand up to 156 kPa.[72] Therefore, it is possible to design containments to withstand much higher pressures. The DAE could have chosen a containment design that could withstand higher presures—and given the uncertainties, that would have been the safer choice.

DESIGN CHOICES

Despite the general concern about reactors with large and positive reactivity coefficients, the DAE has also chosen to design the PFBR with a sodium void coefficient of $4.3.[73] This coefficient could be reduced by designing heterogeneous or modular cores, which enhance leakage of neutrons. The strength of the sodium void coefficient could affect both the extent and rate of accident propagation, and by reducing it one could reduce the likelihood of large parts of the core participating in a CDA. Despite trade-offs, such as higher temperature gradients in the coolant and larger fissile material needs, a heterogeneous core was chosen for the CRBR,[74] and was considered for the Russian BN-1600 reactor.[75]

The DAE has defended its choice of a core with a large sodium void coefficient based on two arguments. First, and more ironically, it argues that the emphasis on not having a positive sodium void coefficient is mistaken because a "partial and selective voiding of the core with an overall negative sodium void reactivity effect can still lead to dangerous situations."[76] This is puzzling because a similar situation in a core with an overall positive sodium void coefficient is more dangerous. A related (also puzzling) argument is that the reactivity addition due to sodium voiding is small compared to reactivity due to fuel rearrangement. This misses the importance of the sodium void coefficient in the early stages of a CDA, with larger void coefficient making the CDA more likely.[77]

The second argument has been that despite the positive sodium void coefficient, the PFBR is safe because the energy released during the most severe accident can be contained. But as demonstrated the PFBR's containment may not maintain its integrity during a severe accident.

Finally, the DAE has argued that since the effect on safety is minimal, imposing the economic cost of a higher fissile material inventory is not justified.[78] But the DAE does not seem to have carried out a comparative study of the energy releases in a CDA in homogeneous and heterogeneous cores, with the latter having a negative or smaller sodium void coefficient. Therefore, regardless of value judgments about how much one should invest in safety, there is no basis to the claim that the effect on safety is minimal.

CONCLUSIONS

It has been long understood that fast reactors have unique problems posed by core rearrangements. These safety concerns became more prominent after the Chernobyl accident, when a positive coolant void coefficient contributed to the meltdown.[79] Although on the one hand arguing that these safety concerns have been considered in the PFBR design, the DAE has also argued that safety concerns are completely misplaced in the first place. Thus, a DAE official argued that the fast reactor community "ought to assert themselves and destroy the sodium void phobia ... the necessity of a dome on the top of the reactor vessel and the core catchers needs to be challenged . . . after all, if the reactor can be designed to be inherently safe or if the probability of failure of the shutdown function can be brought down to 1e-8 per demand, why invest more funds for safety features."[80] This conviction that the reactor is inherently safe comes in the way of reliable safety studies and is manifested partly through assumptions that cannot really be considered "reasonably worst case."

At the same time, there is concern about the possibility of a CDA at breeder reactors, necessitating the incorporation of at least some safety features. The DAE has tried to resolve this contradiction between general perception and its own convictions by adding minimal and inadequate safety features. Although the PFBR design does include a core catcher and secondary containment building, the containment has a low design pressure, and the core catcher is designed to retain debris during a meltdown of only 7 of the 181 subassemblies in the core.[81]

The DAE has maintained that "the capital cost of FBRs will remain the most important hurdle" to rapid deployment of breeder reactors.[82] Even with the current PFBR design, the electricity that it produces will be more expensive than electricity from other reactors.[83] The economic imperative, therefore, might argue for designs that are less safe. For example, the DAE's choice of containment design is directly linked to cost reduction efforts made in the 1990s.[84] The DAE has also emphasized that "minimizing capital cost" was one of the design objectives for the PFBR as it "would be the head of a series of at least a few reactors."[85]

In addition to the problems with the design, safety also depends on organizational practices and here the DAE's record is poor. Poor quality control practices during construction have already been identified. Other problems that have surfaced in the past at DAE facilities include inoperative safety systems, neglect of necessary maintenance activities, and other precautions, and repeated occurrences of accident initiators despite efforts to control these. All of this is compounded by the lack of independence of the regulatory agency, the Atomic Energy Regulatory Board, and the DAE's ability to ignore its recommendations.[86]

The PFBR is the first reactor of its kind in India, and as the name "Prototype" indicates, it is to be the basis for future reactors. The DAE has forecast that it would build 262.5 GW of fast breeder reactors by 2052.[87] Shortcomings of the PFBR's design create serious concerns regarding the safety of these breeder reactors.

The current study suggests several ways of making the reactor safer or reducing the potential impact of severe accidents. First, the design pressure of the containment should be much higher with an associated increase in containment size. Values for both should be selected on the basis of complete and comprehensive severe accident studies that use "reasonably worst case" assumptions; for example, a reactivity insertion of 100 $/s and a thermal to work energy conversion efficiency of at least 5%. These would reduce the chances of a massive radioactive release in the event of a severe reactor accident. A more basic change is to construct reactors that have a much smaller or negative sodium void coefficient. Despite economic consequences, this would be much safer. Similarly, although it would imply greater fueling costs, safety would be enhanced if the burn-up of the fuel was reduced. Given the potentially catastrophic impacts of a severe reactor accident at the PFBR, which would be especially destructive in a densely populated country like India, it is imperative that the DAE discontinue construction of breeder reactors without adequate safety margins.

NOTES AND REFERENCES

1. Fast breeder reactors are thus termed because they are based on energetic (fast) neutrons and because they produce (breed) more fissile material than they consume.

2. H. J. Bhabha and N. B. Prasad, "A Study of the Contribution of Atomic Energy to a Power Programme in India" (paper presented at the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1958): 89–101, R. Chidambaram and C. Ganguly, "Plutonium and Thorium in the Indian Nuclear Programme," *Current Science* 70(1) (1996): 21–35. While uranium can be used to fuel reactors, thorium cannot fission at low energies and so cannot fuel a reactor directly. But it can be converted to the isotope uranium-233 through neutron absorption followed by radioactive decay. Uranium-233, in turn, is fissile and can fuel reactors.

3. K. V. Suresh Kumar et al., "Fast Breeder Test Reactor. 15 Years of Operating Experience" (paper presented at the Technical Meeting on Operational and Decommissioning Experience with Fast Reactors, Cadarache, 11–15 March 2002): 15–27.

4. R. Prasad, "India: FBTR Passes 53-Day Continuous Operation Test," *The Hindu*, 22 March 2001.

5. DAE, "Atomic Energy in India: A Perspective" (Department of Atomic Energy, Government of India, 2006).

6. Praful Bidwai, "The Fast Breeder Reactor: DAE's Strange Nuclear Priorities," *The Times of India*, 31 August 1983.

7. DAE, "Performance Budget 1990–91" (Mumbai: Department of Atomic Energy, 1991).

8. Mark Hibbs, "India's New Breeder Will be on Line by 2000, Iyengar Says," *Nucleonics Week* 31(42) (1990): 11.

9. T. S. Subramanian, "A Milestone at Kalpakkam," *Frontline*, 19 November 2004.

10. R. B. Grover and Subash Chandra, "Scenario for Growth of Electricity in India," *Energy Policy* 34(17) (2006): 2834–2847; M. R. Srinivasan, R. B. Grover, and S. A. Bhardwaj, "Nuclear Power in India: Winds of Change," *Economic and Political Weekly* XL(49) (2005): 5183–5188.

11. Another organization, the Atomic Energy Regulatory Board (AERB), has apparently reviewed some of the safety features of the PFBR, but the reviews are not publicly available.

12. A reduction of coolant density has three effects. The reduced coolant absorbs fewer neutrons, the mean energy of neutrons is higher, and there is more leakage. In a fast reactor, higher neutron energy results in more Pu-239 fissions and therefore the first two effects increase reactivity. Leakage effects are important only near the periphery of the core, and therefore become less important as a whole as the volume of the core increases.

13. IAEA, "Fast Reactor Database: 2006 Update" (Vienna: International Atomic Energy Agency, 2006).

14. This is a natural scale because a reactivity increase of that magnitude would allow the reactor to be critical on only prompt neutrons; accordingly very rapid rates of power increase would be possible. The resultant timescales would be too short for control rods to be inserted and stabilize the reactor.

15. It is possible to obtain a positive void coefficient even when thermal neutrons are responsible for most of the fission. This happens when the coolant plays an important role in neutron absorption, but does not contribute much to reducing neutron energies to the thermal range. This is true to some extent in Pressurized Heavy Water Reactors, but is more prominent in the graphite moderated RBMK design, as exhibited dramatically in the 1986 Chernobyl accident.

16. The Doppler effect results from increased absorption of neutrons by fertile uranium-238 nuclei. When the fuel in a reactor becomes hotter, the average speed of the nuclei increases, thereby generating a wider range of relative neutron speeds or energies. The bulk of the fuel is uranium-238, which resonantly absorbs neutrons at specific energies. As the random motion of the uranium-238 nuclei increases, the probability of neutrons having an energy corresponding to one of these resonant energies increases. This increases the number of neutrons absorbed by uranium-238 nuclei, reducing the number of neutrons available to cause fission, reducing the reactivity. As the ratio of uranium-238 to fissile nuclei increases, the Doppler coefficient becomes increasingly negative. See T. D. Beynon, "The Nuclear Physics of Fast Reactors," *Reports on Progress in Physics* 37 (1974): 951–1034, at 1029.

17. There are also advantages of using a sodium coolant. First, because operations at low pressures are possible, a breach in the coolant pipes would not by itself lead to boiling of the remaining coolant. Second, high thermal conductivity allows for more reliable decay heat removal under accident conditions.

18. The Russian BN-600 reactor has had numerous sodium leaks and fires. V. N. Ivanenko and V. A. Zybin, "Fast Reactor Sodium Systems Operation Experience and Leak-before-Break Criterion" (paper presented at the Technical Committee Meeting on Evaluation of Radioactive Materials Release and Sodium Fires in Fast Reactors, IWGFR—92, O-arai, Ibaraki (Japan), 11–14 November 1996): 255–269; N. N. Oshkanov, M. V. Bakanov, and O. A. Potapov, "Experience in Operating the BN-600 Unit at the Belyi Yar Nuclear Power Plant," *Atomic Energy* 96(5) (2004): 315–319.

19. Hans A. Bethe and J. H. Tate, "An Estimate of the Order of Magnitude of the Explosion When the Core of a Fast Reactor Collapses" (United Kingdom Atomic Energy Agency, 1956).

20. There was a reason to ignore these feedbacks. Early reactors were small and had high enrichments. Because the Doppler absorption is primarily from fertile nuclei, the corresponding reactivity feedback effects were small.

21. *Ibid*.

22. Richard Wilson, "Physics of Liquid Metal Fast Breeder Reactor Safety," *Reviews of Modern Physics* 49(4) (1977): 893–924.

23. C. R. Bell, "Multiphase, Multicomponent Hydrodynamics in HCDA Analysis: Present Status and Future Trends," *Nuclear Engineering and Design* 68 (1981): 91–99. The uncertainties arise from significant core motions, large-scale fluid dynamics, and variable neutronic states, and a variety of possible flow regimes, especially when many of the core components have failed or melted. In addition, feedbacks are often nonlinear, such as the relationship between liquid temperatures and vapor pressures, or between material motions and reactivity effects.

24. Wilson, *op. cit.*, 911.

25. J. F. Jackson and R. B. Nicholson, "Venus-II: An LMFBR Disassembly Program" (Argonne, IL: Applied Physics Division, Argonne National Laboratory, 1972), 88–96.

26. KAERI, "Review of Core Disruptive Accident Analysis for Liquid-Metal Cooled Fast Reactors" (Korea Atomic Energy Research Institute, 1997).

27. T. G. Theofanous and C. R. Bell, "Assessment of CRBR Core Disruptive Accident Energetics" (Los Alamos National Laboratory, 1984).

28. V. Badham and C. K. Chan, "A Look at Alternative Core-Disruptive Accidents in LMFBRs—Part II: Neutronic and Fuel Element Behavior," *Nuclear Engineering and Design* 55 (1979): 1–7. The authors estimate that coherent core collapse leads to a reactivity insertion of around 0.3, or approximately $75, and that this occurs in around 0.35 seconds.

29. Om Pal Singh and R. Harish, "Energetics of Core Disruptive Accident for Different Fuels for a Medium Sized Fast Reactor," *Annals of Nuclear Energy* 29 (2002): 673–683.

30. *Ibid.*, 678.

31. J. F. Jackson and R. B. Nicholson, "Venus-II: An LMFBR Disassembly Program," NEA, *Venus-2, Reactor Kinetics with Feedback, 2-D LMFBR Disassembly Excursions* (Nuclear Energy Agency, 1980 [cited 21 October 2008]), http://www.nea.fr/abs/html/nesc0511.html.

32. Jackson and Nicholson, "Venus-II: An LMFBR Disassembly Program," 8.

33. This is redrawn from a figure in Singh and Harish, *op. cit*.

34. *Ibid.*, 679.

35. Table 6 of *Ibid*. shows 2130 MJ thermal energy and 23 MJ mechanical energy for an insertion rate of 50 $/s.

36. Georges Berthoud, "Vapor Explosions," *Annual Review of Fluid Mechanics* 32 (2000): 573–611.

37. IAEA, "Fast Reactor Fuel Failures and Steam Generator Leaks: Transient and Accident Analysis Approaches" (Vienna: International Atomic Energy Agency, 1996).

38. Berthoud, *op. cit.*, 594.

39. Wilson, *op. cit.*, 915.

40. Thomas B. Cochran, *The Liquid Metal Fast Breeder Reactor: An Environmental and Economic Critique* (Washington, D.C.: Resources for the Future, Inc., 1974), 183.

41. P. Chellapandi et al., "Analysis for Mechanical Consequences of a Core Disruptive Accident in Prototype Fast Breeder Reactor" (paper presented at the 17th International Conference on Structural Mechanics in Reactor Technology (SMiRT 17), Prague (Czech Republic), 17–22 August 2003): 1–8.

42. One concern about that claim is that the analysis of the reactor vessel's integrity does not incorporate the possibility of corrosion.

43. S. M. Lee, "Status of Fast Reactor Development in India: April 2001–March 2002" (paper presented at the Technical Working Group on Fast Reactors, Karlsruhe, Germany, 22–26 April 2002): 333–385, 343–344.

44. Chellapandi et al., *op. cit*.

45. It is assumed that the elongation increases linearly for higher energies.

46. E. L. Gluekler and T. C. Huang, "Response of Secondary Containment to Presence of Sodium and Hydrogen," *Nuclear Engineering and Design* 55 (1979): 283–291. A pool fire can also occur in the primary vessel if the cover gas is lost and air takes its place, as has been assumed in some analyses, for example, KAERI, "Preliminary Safety Analysis for Key Design Features of Kalimer" (Korea Atomic Energy Research Institute, 2000).

47. IGCAR, *Design of Prototype Fast Breeder Reactor* (Indira Gandhi Centre for Atomic Research, 2003 [cited 10 March 2006]), www.igcar.ernet.in/broucher/design.pdf.

48. S. C. Chetal et al., "The Design of the Prototype Fast Breeder Reactor," *Nuclear Engineering and Design* 236 (2006): 852–860.

49. Energy here refers to the mechanical energy, which is a small fraction (a few percent) of the total heat energy released.

50. It is assumed that the density of sodium in the reactor vessel remains constant, and the pressure in the containment building throughout the leakage remains much smaller than the pressure inside the reactor vessel, while the velocity is much higher.

51. L. V. Krishnan and D. D. Garg, "Scaling Laws for Contained Explosions," *Nuclear Engineering and Design* 56 (1980): 405–412.

52. It is not clear what the leakage areas are in the DAE analysis. A spray fire requires unobstructed paths to the containment, which is more likely when the reactor vessel has suffered greater damage. Therefore, the leakage area for spray fires is likely to be higher. The calculations here do not require the magnitude of these areas, only their possible dependence on CDA energy.

53. The containment atmosphere has a volume of 87000 m^3; IAEA, "Fast Reactor Database: 2006 Update." At 30^{o}C and atmospheric pressure, around 24,000 kg of oxygen is present in this volume. Stoichiometrically, to burn 23 g of sodium, 32 g of oxygen are needed. Hence, the oxygen in the containment is sufficient to burn over 17 tons of sodium.

54. In older analyses, other values for the overpressure have been given. One paper presented at a conference in 1998 mentioned that "preliminary analysis indicate(s) a pressure rise of $\sim$10 kPa resulting from the sodium spray fire inside the containment." S. B. Bhoje et al., "Impact of LMFBR Operating Experience on PFBR Design" (paper presented at the Technical Committee Meeting on Unusual Occurrences During LMFBR Operation, Vienna, 1998): 180–196. Another paper quotes an overpressure of 30 kPa resulting from the burning of 1.5 tons of sodium. S. B. Bhoje, "Status of Fast Reactor Development in India: April 1996–March 1997" (paper presented at the 30th Meeting of the International Working Group on Fast Reactors, Beijing, 14–17 May 1997): 79–109, 93.

55. During formation of sodium monoxide, 871 KJ is released for combustion of 4 moles (or 23*4 grams) of sodium. This yields the aforementioned figure. Formation of peroxide results in a higher energy release per unit mass.

56. The authors of this paper have chosen the maximum temperature difference between the pool and the flame of around 330 K. See Akira Yamaguchi and Yuji Tajima, "Numerical Investigations of Mass and Heat Transfer in Sodium Pool Combustion," *Numerical Heat Transfer, Part A* 41 (2002): 697–709. This gives the smallest temperature

and pressure loading on the containment because as the pool temperature increases and becomes closer to the flame temperature, the fraction of the heat produced that is transferred to the containment atmosphere increases.

57. *Ibid*.

58. This lies within the range of burning rates that have been found in the FAUNA test facility. W. Cherdron and S. Jordan, "Thermodynamic Consequences of Sodium Leaks and Fires in Reactor Containments" (paper presented at the Specialists' Meeting on Sodium Fires, Obninsk, USSR, 1988): 57–77.

59. Chellapandi et al., *op. cit*. The burning time (15 minutes) is estimated by measuring from the plot given in the DAE paper, and is an upper bound taking into account measurement uncertainties.

60. Inhomogeneity could lead to local hot spots in the containment and a higher chance of failure.

61. This coefficient has been calculated by assuming heat transfer by natural convection, and that the height of the containment building is 1.5 times both the length and the width.

62. This is obtained by subtracting the area occupied by the intermediate heat exchanger, rotating plugs, and sodium pump from the area of the vessel head.

63. P. Chellapandi, S. Jalaldeen, and S. B. Bhoje, "Assessment of Structural Integrity of PFBR Reactor Assembly under HCDA" (paper presented at the National Symposium on Safety of Nuclear Power Plants and Other Facilities, Bhabha Atomic Research Centre, Trombay, 11–13 March 1992): III.30; G. Vaidyanathan et al., "Safety Considerations in the Design of PFBR" (paper presented at the Conceptual Designs of Advanced Fast Reactors: Technical Committee Meeting, Kalpakkam (India), 3–6 October 1995): 159–166.

64. The sodium inventory in the primary circuits is around 1,200 tons. Of this, a significant fraction could escape into the containment atmosphere during a CDA if the pressure vessel is ruptured. However, the amount of sodium that can be burnt is limited by the oxygen in the containment, which can sustain only up to 17 tons of sodium. But even this is much larger than the sodium leakage calculated from the scaling relationship, which is 3,360 kg of sodium leak for a CDA with 1,200 MJ mechanical energy. If one assumes that of the sodium entering the containment building from a ruptured primary vessel, about 15 tons were to burn as a spray fire, then the overpressure on the containment can exceed 150 kPa. Similarly high overpressures have been calculated in some accident scenarios for the much larger CRBRP containment. See Gluekler and Huang, *op. cit*., 287.

65. NRC, "Reactor Safety Study (RSS), Study Director: N. V. Rasmussen" (Washington, D.C.: U.S. Nuclear Regulatory Commission, 1975).

66. APS Study Group, "Radionuclide Release from Severe Accidents at Nuclear Power Plants," *Reviews of Modern Physics* 57(3) (1985): S1–S154, at S92.

67. A. S. Pannerselvan, "Close to a Critical Mess," *Outlook*, 8 November 1999.

68. M. Madan Mohan, "Kaiga Questions: A Gaping Hole in Safety Standards," *Frontline*, 17 June 1994, 84–85.

69. *Ibid*.

70. The numerator is a product of two measures of the ability of the containment, its design pressure, and its (large) volume, to withstand an accident. The choice of the denominator stems from the expectation that the energy that would potentially be released during an accident would be very roughly proportional to the power rating of

the reactor. A different metric that has also been used is just the ratio of the containment volume to the thermal power; Ed Lyman, "Can Nuclear Plants Be Safer?," *Bulletin of the Atomic Scientists* (September/October 2008), 34–37, but this does not take into account the design pressure.

71. APS Study Group, *op. cit.*, S94.

72. S. A. Bhardwaj, "The Future 700 MWe Pressurized Heavy Water Reactor," *Nuclear Engineering and Design* 236 (2006): 861–871.

73. IAEA, "Fast Reactor Database: 2006 Update." This is the maximum coolant void coefficient and includes only regions with a positive void coolant reactivity worth.

74. Waltar and Reynolds, *op. cit.*

75. M. F. Troyanov et al., "The Present Status of Fast Breeder Reactors in the USSR," *Philosophical Transactions of the Royal Society of London, Series A, Mathematical and Physical Sciences* 331(1619) (1990): 313–321.

76. S. R. Paranjpe, "Report on the Specialists' Meeting on Passive and Active Safety Features of Liquid-Metal Fast Breeder Reactors Organized by the International Atomic Energy Agency at Oarai Engineering Centre of Power Reactor and Nuclear Development Corporation, Japan, November 5–7, 1991," *Nuclear Safety* 33(4) (1992): 506–513.

77. Moreover, the contribution of rapid core voiding to the reactivity of the core in a CDA cannot be neglected, as was clear in a previous section.

78. Paranjpe, *op. cit*.

79. Christoph Hohenemser, "The Accident at Chernobyl: Health and Environmental Consequences and the Implications for Risk Management," *Annual Review of Energy* 13 (1988): 383–428.

80. Paranjpe, *op. cit*.

81. Chetal et al., *op. cit*.

82. S. R. Paranjpe, "An Update on Indian Fast Breeder Programme" (paper presented at the International Conference on Fast Reactors and Related Fuel Cycles, Kyoto, October 28–November 1 1991): 1.4-1–1.4-9.

83. M. V. Ramana and J. Y. Suchitra, "The Many Phases of Nuclear Insecurity," in *India's Energy Security*, ed. Ligia Noronha and Anant Sudarshan (New York: Routledge, forthcoming (2009)); J. Y. Suchitra and M. V. Ramana, "The Costs of Power: Plutonium and the Economics of India's Prototype Fast Breeder Reactor" (in preparation).

84. S. B. Bhoje, "Cost Competitiveness of Breeder Reactor," *Indira Gandhi Centre for Atomic Research Newsletter*, October 2001.

85. S. B. Bhoje, "Prototype Fast Breeder Reactor," *Nu-Power* 16(1–2) (2002): 1–5.

86. For example, the PFBR has a design pressure of only 25 kPa although the AERB recommended that the design pressure of the containment building "should not be less than 30 kPa." AERB, "Annual Report 2002–2003" (Mumbai: Atomic Energy Regulatory Board, 2003), 12.

87. Grover and Chandra, *op. cit*.

88. John Graham, *Fast Reactor Safety* (New York: Academic Press, 1971), 175–176, Wirtz, *op. cit.*, 138.

89. Wirtz, *op. cit*.

90. IAEA, "Fast Reactor Database: 2006 Update," 24.

91. Waltar and Reynolds, *op. cit*.

92. T. Saito and K. Suzuki, “Role of Fission Products in Whole Core Accidents” (paper presented at the Specialists’ Meeting on Role of Fission Products in Whole Core Accidents, AERE Harwell (United Kingdom), 28 June–1 July 1977 1977): 41–55.

93. M. Eriksson et al., “Inherent Safety of Fuels for Accelerator-Driven Systems,” *Nuclear Technology* 151 (1995): 314–333.

94. R. Lallement, A. Tuzov, and K. Q. Bagley, “Fast Breedor Reactor Fuel Performances,” *Philosophical Transactions of the Royal Society of London, Series A, Mathematical and Physical Sciences* 331(1619) (1990): 343–354, Saito and Suzuki, *op. cit*.

95. Venkatachari Jagannathan, *Powerful Leap: IGCAR Press Release* (2003 [cited 8 August 2004]), http://www.igcar.ernet.in/press_releases/press6a.htm.

96. IGCAR, *IGCAR Milestones* (Indira Gandhi Centre for Atomic Research, 2005 [cited 24 February 2007]).

97. Chetal et al., *op. cit*.

98. Saito and Suzuki, *op. cit*.

99. Waltar and Reynolds, *op. cit*., 558.

100. S. R. Paranjpe, Om Pal Singh, and R. Harish, “Influence of a Positive Sodium Void Coefficient of Reactivity on the Consequences of Transient Overpower and Loss-of-Flow Accidents in a Medium-Sized Fast Reactor,” *Annals of Nuclear Energy* 19(7) (1992): 369–375; S. R. Paranjpe, Om Pal Singh, and R. Harish, ”Sodium Void Coefficient and Fast Reactor Safety” (paper presented at the International Conference on Fast Reactors and Related Fuel Cycles, Kyoto, 28 October–1 November 1991): 13.6-1–13.6-9.

101. Koji Maeda, Kozo Katsuyama, and Takeo Asaga, “Fission Gas Release in FBR Mox Irradiated to High Burnup,” *Journal of Nuclear Materials* 346 (2005): 244–252.

102. Eriksson et al., *op. cit*.

103. IAEA, “Fast Reactor Fuel Failures and Steam Generator Leaks: Transient and Accident Analysis Approaches.”

104. T. Aoyama et al., “Operational Experience and Upgrading Program of the Experimental Fast Reactor Joyo” (paper presented at the Technical Meeting on Operational and Decommissioning Experience with Fast Reactors, Cadarache (France), 11–15 March 2002): 244–249.

105. Takashi Kawakita, Sadanori Aoi, and Takeshi Hojuyama, ”Study on Large FBR Core Optimization to Enhance Core Safety” (paper presented at the Passive and Active Safety Features of LMFRs: Meeting of the Technical Working Group on Fast Reactors, O-arai, Japan, 5–7 November 1991).

106. Juan J. Carbajo et al., “A Review of the Thermophysical Properties of Mox and UO_2 Fuels,” *Journal of Nuclear Materials* 299 (2001): 181–198.

107. According to the DAE the intrinsic density of the fuel is 90% of its theoretical density, which corresponds to 10% porosity volume.

108. Baldev Raj et al., “Development of Fuels and Structural Materials for Fast Breeder Reactors,” *Sadhana* 27(5) (2002): 527–558.

109. S. R. Paranjpe, “Core (PFBR) Safety Characteristics” (paper presented at the Passive and Active Safety Features of LMFRs: Meeting of the Technical Working Group on Fast Reactors, O-arai, Japan, 5–7 November 1991): 176–190, Om Pal Singh et al., ”Energetics of a Hypothetical Core Disruptive Accident for Different Fuels for a Medium Sized Fast Reactor” (paper presented at the International Conference on Fast Reactors and Related Fuel Cycles: Current Status and Innovations Leading to Promising Plants FR’91, Kyoto (Japan), 1991): P3.3-1–P3.3-10.

110. See P. R. Vasudeva Rao et al., "Oxygen Potential and Thermal Conductivity of (U, Pu) Mixed Oxides," *Journal of Nuclear Materials* 348 (2006): 329–334.

111. Carbajo et al., *op. cit*.

112. Paranjpe, Singh, and Harish, *op. cit*.

113. The assumption is that loss of coolant flow occurs only due to a failure of power supply to the coolant pumps and this loss does not occur instantaneously but is characterized by a flow halving time constant.

114. Paranjpe, Singh, and Harish, *op. cit*.

115. The boiling point of sodium is between 960°C and 1050°C, depending on the pressure.

116. Paranjpe, Singh, and Harish, *op. cit*., 373.

APPENDIX 1: REACTIVITY INSERTION RATE FROM A SIMPLE MODEL

When the core collapses, reactivity is affected by the change in neutron leakage, the changing reactivity worth distribution of the fuel, and any simultaneous coolant voiding that occurs. If the fuel is assumed to collapse axially to fill the void spaces beneath, a rough estimate of reactivity insertion can be calculated by considering only the leakage. The simple model that follows describes how this might be approximately calculated. The neutron multiplication factor of a cylindrical core with height H and radius R, taking into account leakage effects, is given by the standard formula:[88]

$$k_{eff} = \frac{k_\infty}{1 + L^2\left(\left[\frac{2.405}{R}\right]^2 + \left[\frac{\pi}{H}\right]^2\right)} \tag{A1-1}$$

where L is the diffusion length (which, following Wirtz,[89] is assumed to be 20 cm for the initial core) and k_∞ is the neutron multiplication of an infinite reactor. The term in the denominator represents the leakage from a cylindrical core. The initial height of the core is 100 cm and the equivalent diameter of the core is 197 cm.[90] The coolant fraction is 0.41 and so it is assumed that the core collapses to 60% of its original height while maintaining the same radius. The change in core density affects the diffusion length, which varies inversely with the atom density.[91]

In the simplified model, only the height of the core varies, and so L is proportional to the height. The initial reactivity of the core is taken from the DAE's calculation showing that as the coolant starts boiling at the end of the predisassembly phase the reactivity of the PFBR core is \$.96. Assuming further that the composition of the core does not change significantly during the collapse so that the infinite neutron multiplication k_∞ is unchanged, and assuming collapse driven by gravitation, the new reactivity after collapse and from this the reactivity insertion rate, can be calculated. The inserted reactivity is 0.094;

with a delayed neutron fraction of 0.0034, this is equal to $27.6. The collapse occurs in 0.28 seconds and therefore the insertion rate is 97 $/s (these estimates are approximated).

To this, the effect of changing control rod worth must be added. During normal operation, each of the control rods is inserted to a worth of approximately 250 pcm, or approximately $75. The 12 control rods therefore have a combined worth of $9 when the reactor is operating at steady state. Because the collapse length of 40 cm is larger than the 25 cm for which the control rods are inserted, the reactivity insertion from the changing relative position of the control rods and the core is $9. The insertion rate is approximately 32 $/s. In addition, there is reactivity insertion from displacement of coolant. Even though this simple calculation cannot be taken to be a definitive estimate of the reactivity insertion rate, it does illustrate that with whole-core collapse, reactivity insertion at a rate higher than 100 $/s is possible.

APPENDIX 2: WEAKNESSES IN THE DAE'S CALCULATION OF THE PRE-DISASSEMBLY PHASE

There are problems with the DAE's analysis of accidents, which might affect its conclusions about how much of a core will participate in a CDA. There are at least three ways through which larger fractions of the core could participate in a CDA. The first is if fuel assemblies that are assumed to stay intact actually fail and add reactivity. The second is if assumed reactivity feedback effects are wrong. Third, differences in thermophysical parameters could alter the onset of fuel melting. Increased core participation would typically imply a higher reactivity insertion rate and a greater energy release during a CDA.

Fuel and Cladding Failure

In the DAE's CDA study, fuel failure occurs exclusively due to melting of the fuel. It ignores the possibility of cladding failure at high temperatures, even before the fuel melts. This is inadequate because fuel pins in FBRs are subjected to a hostile environment, which includes high temperatures, high neutron fluences, molten sodium, and fission products.[92] Two of the mechanisms that cause rapid fuel failure are corrosive thinning of cladding and creep rupture.[93]

The possibility of failure depends on both the burn-up and the rate of the transient. Increased burn-up increases the energy stored in fission products, decreases the fuel cladding gap, and increases the likelihood of corrosion by fission products.[94] Therefore the temperature at which fuel pins fail reduces rapidly with burn-up.

As a way of lowering the cost of electricity from the PFBR, the DAE has emphasized reducing the fuel requirements by increasing the average burn-up of the core.[95] Over the years, the DAE has highlighted the high burn-up to which fuel has been exposed in the FBTR.[96] The DAE claims that "initial peak fuel burnup" for the PFBR will be "limited to 100 GWd/t" but that in the long run the targeted burnup is 200 GWd/t.[97]

For a loss of flow (LOF) scenario, the cladding temperatures at which fuel pins fail are estimated to come down from over 1,200°C at a burn-up of 10 GWd/t to less than 900°C for a burn-up of approximately 70 GWd/t.[98] Thus, one would expect failure temperatures well below 900°C at the PFBR's design burn-up. In comparison, safety studies of the FFTF suggest a maximum cladding temperature of approximately 870°C.[99]

These temperature limits are exceeded even in the DAE's studies of protected loss of flow accidents where the reactor shutdown systems are assumed to work.[100] In the DAE's simulation of protected loss of flow with a flow-halving time of 2 seconds, the maximum cladding temperature goes up to 1,284°C within 6 seconds, with an apparently high rate of increase. It is therefore likely that the cladding temperature would exceed safe limits described earlier and there would be significant fuel pin failure—a factor not included in the DAE's safety studies.

Cladding failure has two effects. First, gaseous and volatile fission products could be ejected into the coolant and contribute to local boiling; this effect also increases with burn-up.[101] In turn, this could result in an increase in reactivity. Second, because the cladding contributes to reducing the neutron energy, its removal from the core would have a similar effect as the loss of coolant. Simulations of a 700 MWt lead-bismuth cooled reactor suggested a reactivity insertion of approximately 0.03 occurs when all the cladding is removed, and this is close to the reactivity insertion from coolant removal calculated in the same study.[102] Therefore, neglect of these cladding failure modes implies that various reactivity insertions and potentially destabilizing feedbacks have been ignored.

Burn-up Effects on Reactivity Feedback Coefficients

Although the projected or targeted burn-up for the fuel to be used in the PFBR has increased over the years, the DAE has not taken these increases into account in its safety assessments in general, and its estimates of reactivity coefficients in particular. The burn-up affects reactivity coefficients in several ways, including through the build-up of fission products, the changing position of the control rods, the changing volumes of fuel and structural materials, and the feedback due to axial expansion of the fuel. The last factor is because the rapid expansion of fuel in response to temperature increases requires a gap between the fuel and cladding.

Transient tests at the French CABRI facility of fuel irradiated in the PHENIX reactor showed that fresh fuel and fuel with open gaps showed the highest axial expansion.[103] In the JOYO reactor operated in Japan, the measured power coefficient decreased with burn-up; in going from an average burnup of 22 GWd/tU to 35 GWd/tU, the absolute value of the coefficient at the beginning of the cycle reduces by approximately 45%.[104] The explanation centered on a decrease in fuel expansion due to thermal restructuring of the fuel at high burn-up.

These uncertainties have led some analyses to even omit axial expansion altogether from consideration in transient analysis.[105] Because it is comparable in magnitude to the Doppler coefficient for the PFBR, a reduced fuel axial expansion coefficient could make fast transients more severe. The DAE assumes that axial expansion operates in the initial stages of an accident before fuel melting occurs.

Burn-up Effects on Thermophysical Properties

Both the thermal conductivity and melting point of fuel decrease with burn-up. Thermal conductivity is also affected by the porosity of the fuel, its stoichiometry, temperature, and some correlations from data are described in the literature.[106] These correlations can be used to calculate the thermal conductivity, which, at a porosity of 10% and average fuel temperature of 1290°C, work out to be is 1.9, 1.60, and 1.45 W/m/C at burn-up of 0, 5, and 10 atomic percent.[107] The last value corresponds to a burn-up of approximately 88 GWd/t, smaller than the initial target burn-up.[108] Rather than using a smaller value of thermal conductivity depending on the burn-up, the DAE uses a constant value of 2 W/m/C.[109] Its own measurements do not either cover temperatures above 1,200°C or consider the effects of burn-up.[110] As for the effect of burn-up on melting point of MOX fuel, a decrease of 5°C/at percent is suggested.[111] Therefore, high burn-up could result in both higher fuel temperatures and lower temperature margins that are available before the fuel melts.

APPENDIX 3: POTENTIAL FOR CDA INITIATION

As suggested earlier, one reason for the problems with the DAE's analysis might be its confidence that a CDA would never occur. Such confidence may have been buttressed by its finding that in case there is a transient, the reactor shutdown logic requirements are met before the coolant temperature rises by 100°C.[112] The DAE takes this to be evidence that safety has been ensured. However, this analysis is problematic.

Although the DAE's analysis does indicate that the reactivity starts coming down, the problem is that the spatially maximum coolant temperature continues to rise even after the reactor is shutdown. For the case of the flow halving time being 10 seconds,[113] the maximum coolant temperature reaches 900°C in less than a minute, the period for which results are presented, and all evidence at this time (rising cladding temperature, high core power at 50% of nominal, rising power to flow ratio) suggests that the coolant temperature would rise further.[114] For the more drastic case of a flow halving time of 2 seconds, the maximum coolant temperature reaches over 900°C in less than 6 seconds. Once again, all indicators suggest that the temperature will continue to increase. Therefore, even though the control rod logic acts to shut down the reactor, it cannot shut down the reactor fast enough to maintain the coolant temperature at safe levels.[115]

This is where a positive void coefficient might be disruptive and prevent the reactivity from falling quickly enough. If boiling were to occur, void formation and associated reactivity increases would render the DAE analysis incomplete. The DAE analysis acknowledges that boiling occurs in the oxide core for even a much larger flow halving time constant of 100 seconds, but stops short of assessing its effects.[116]