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SAFETY

When anyone asks me how I can best describe my experiences of nearly forty years at sea, I merely say, uneventful. Of course, there have been winter gales and storms and fog and the like, but in all my experience, I have never been in any accident of any sort worth speaking about. . . . I never saw a wreck and have never been wrecked, nor was I ever in any predicament that threatened to end in disaster of any sort.

I will say that I cannot imagine any condition which would cause a ship to founder. I cannot conceive of any vital disaster happening to this vessel. Modern shipbuilding has gone beyond that.

—Captain Smith, Commander of the *Titanic*[1]

In its submission to the International Atomic Energy Agency (IAEA) as part of its responsibilities under the 1994 Convention on Nuclear Safety, the DAE stated:

> Safety is accorded overriding priority in all activities.[2] All nuclear facilities are sited, designed, constructed, commissioned and operated in accordance with strict quality and safety standards . . . As a result, India's safety record has been excellent in over 260 reactor years of operation of power reactors and various other applications. (GoI 2007, 3)

Is this assertion true? This chapter tries to evaluate the DAE's claims by examining the historical record of safety performance at the DAE's

1. Quote from Butler 2002 (48).

2. Similarly, in the 2008–09 annual report of the Nuclear Power Corporation, S.K. Jain, chairman of NPCIL and of BHAVINI—the organization responsible for the construction of the PFBR—wrote about the organization's 'culture of an overriding priority to nuclear safety' (NPCIL 2009, 19).

facilities. This record shows that nearly all the nuclear reactors and other facilities associated with the nuclear fuel cycle operated by the DAE have had accidents of varying severity. These have been euphemistically termed 'incidents', primarily to mollify public concern. Between 1998 and 2010, there have been between twenty-one and fifty-four incidents (average thirty-four) every year, according to the Atomic Energy Regulatory Board annual reports (AERB 2001, 2002, 2003, 2004a, 2007, 2008, 2009, 2010, 2011).[3] Although none of these may have resulted in large-scale damage, ignoring them is not justified.[4] An analogy might help. Various symptoms that a patient might experience, such as shortness of breath or frequent chest pain, are typically indicators of a deeper problem, such as clogged arteries, and potential precursors of major heart attacks. In the same way, these small mishaps are also often indicators of potential accidents. Further, the pattern of failures at the DAE's facilities is suggestive of deeper problems, in particular, inadequate institutional priority to safety.

Another lens to examine the priority accorded to safety is the kind of choices made during the design of reactors. In the case of the PFBR, numerous design choices—most probably motivated by cost-cutting—have made the reactor more vulnerable to severe accidents and the impacts of such an accident more catastrophic. A different manifestation of the low priority to safety has been the regulatory structure which is, in effect, not independent of the nuclear establishment. All this is exacerbated by unjustified confidence in the safety of its facilities.

This state of affair is dangerous. Among all electricity-generating technologies, nuclear energy alone is prone to catastrophic accidents with potentially global impact, leaving fallout that may stay hazardous for decades, if not centuries. Poor safety culture increases the likelihood of such accidents. The likelihood of accidents will also increase as the number of nuclear reactors and other facilities go up. Unless there is a significant change in the organization managing the technology, the prospect of hundreds of nuclear reactors around the country does bring with it grave risks, and should be a cause of concern for all.

3. As of July 2012, the 2010–11 annual report with information up to 2010 was the latest available on the AERB website.

4. As physicist Richard Feynman (1986) wrote while discussing the problems of the United States Space Shuttle Programme, 'The fact that this danger did not lead to a catastrophe before is no guarantee that it will not the next time, unless it is completely understood. When playing Russian roulette the fact that the first shot got off safely is little comfort for the next.'

Organization

One reflection of the attitude of the DAE towards safety is its institutional structure. Till 1972, it did not have a separate, identifiable organization or personnel to review safety of its nuclear installations (Gopalakrishnan 2002, 384).[5] In February 1972, the DAE constituted an internal Safety Review Committee (SRC) to oversee safety at DAE facilities (R. Shankar 2008, 18). But this was a purely internal organization and there must have been severe conflicts of priorities.

As early as the 1970s, even before any special safety agency was created, Ashok Parthasarathi, a senior bureaucrat, had suggested that the 'inspection of all nuclear installations from the point of view of health and environmental safety should be administered by a body with a suitable name and located in Department of Science and Technology, as that department had been assigned the national responsibility for ensuring the preservation of environmental quality' (Parthasarathi 2007, 131–32). But this was not accepted by the Atomic Energy Commission.

In 1979, after the Three Mile Island accident in the United States, the DAE constituted a committee to review 'the specific functions and responsibilities of DAE–SRC in order to enable DAE to discharge its obligations under the Atomic Energy Act' (Sundararajan, Parthasarathy, and Sinha 2008, iii). The committee recommended 'the creation of Atomic Energy Regulatory Board by the Atomic Energy Commission with powers to lay down safety standards and assist DAE in framing rules and regulations for enforcing regulatory and safety requirements'. In 1983, the Atomic Energy Regulatory Board (AERB) was set up to oversee and enforce safety in *all* nuclear operations. But in 2000, facilities that were involved, even if peripherally, in the nuclear weapons programme, were excluded from the AERB's purview.

The AERB's effectiveness is constrained by the AEC's choice of institutional structure. Rather than make the AERB independent of the DAE, the AEC made the AERB report to itself. Constitutionally, the Secretary of DAE is the ex officio chairman of the AEC. This allows the DAE to exercise administrative powers over the AERB. Till recently, the chairman of the Nuclear Power Corporation was also a member of the AEC. The AERB's budget comes from the DAE. All these factors place structural limits on the AERB's effectiveness. As common experience would indicate, it is hard to criticize one's boss or force action in ways that he or she does not want.

5. The first reactor, Apsara, did not undergo even a formal safety analysis (Sundararajan, Parthasarathy, and Sinha 2008, 2).

This is illustrated by the AERB's inability to force the DAE to carry out its directives on safety tests in the 1990s.[6]

In 2012, the Comptroller and Auditor General assessed the AERB in detail and found legal and practical hurdles in the way of independent functioning of the organization. Noting the implications of the weakness of regulation in Japan for the accidents at Fukushima, CAG warned that 'failure to have an autonomous and empowered regulator is clearly fraught with grave risks' and recommended that the government 'ensure that the nuclear regulator is empowered and independent' (CAG 2012, 9).

There are other ways in which the DAE has marginalized the AERB. In the case of the Kalpakkam Atomic Reprocessing Plant, AERB approval for construction was sought only in 1994 when 'construction of the plant was already in progress' (Sundararajan, Parthasarathy, and Sinha 2008, 26). What, one wonders, were the odds that AERB would disapprove of the project even if it had found a problem with the design?

This administrative control is compounded by the AERB's lack of technical staff and testing facilities. As Gopalakrishnan, former chairman of the AERB, has observed, '95 per cent of the members of the AERB's evaluation committees are scientists and engineers on the payrolls of the DAE. This dependency is deliberately exploited by the DAE management to influence, directly and indirectly, the AERB's safety evaluations and decisions. The interference has manifested itself in the AERB's toning down the seriousness of safety concerns, agreeing to the postponement of essential repairs to suit the DAE's time schedules, and allowing continued operation of installations when public safety considerations would warrant their immediate shutdown and repair' (Gopalakrishnan 1999a).

Neither does the AERB carry out any monitoring of essential performance metrics such as radiological exposure of workers at DAE facilities or measurement of levels of radionuclides in the vicinity of nuclear facilities. These tasks are entrusted to NPCIL or BARC. This state of affairs prompted the Comptroller and Auditor General to criticize the fact that 'AERB had no direct role in conducting independent assessments and monitoring to ensure radiological protection of workers despite being the nuclear regulator of India' (CAG 2012, vii).

Following the Fukushima accidents, in September 2011, the Indian government introduced the Nuclear Safety Regulatory Authority (NSRA)

6. The NPCIL did not comply with an AERB directive to carry out integrated tests of the Emergency Core Cooling System in Kaiga I and II and RAPS III and IV before they were commissioned (Panneerselvan 1999). Of the 3200 recommendations by the AERB's Safety Review Committee for Operating Plants, the DAE and NPCIL had not complied with 375 (CAG 2012, 42).

Bill in Parliament to create a new organization that would regulate nuclear activities in the country. The NSRA would replace AERB as the regulator of safety. In March 2012, the bill was reportedly approved by the parliamentary committee on science, technology, environment and forests with minor modifications, though two Communist Party of India (Marxist) members in the panel disagreed with its recommendations and wanted a thorough overhaul of the bill (DHNS 2012).

Looking at the content of the bill and the context under which the NSRA has been created, it seems unlikely to create an effective separation between the regulatory authority and the nuclear establishment. Many of the key processes involved in ensuring effective regulation will continue to be controlled by the AEC. The power for crucial steps like the appointment of members is vested with the Central government. But for most purposes, the authority empowered to act on behalf of the Central government is the AEC. The AEC chairman will also be one of the key members of the Council of Nuclear Safety that will set the policies with respect to radiation and nuclear safety that will fall under the purview of the NSRA. Further, as evidenced by Gopalakrishnan's observation, there is little expertise outside the nuclear establishment on technical issues relating to nuclear facilities.

Design

As I write this in September 2012, in Kalpakkam the DAE is constructing its newest and most heralded reactor—the prototype fast breeder reactor, the first commercial-sized reactor of the second phase of the DAE's much-vaunted three-phase nuclear strategy. DAE scientists 'are confident that the PFBR will be safe and will work efficiently' (Raghotam 2008). The PFBR has a history that stretches back more than a quarter-century and has been designed and redesigned by two generations of DAE engineers and scientists. Its design is thus a good test case to examine how the DAE weighs safety against other priorities.

For a reactor that has been so carefully bred, the PFBR has some serious deficiencies: a design that makes it more susceptible to catastrophic accidents and a containment structure that will not be able to contain the radioactivity released in the event of a severe accident (Kumar and Ramana 2008; Kumar and Ramana 2011). While all nuclear reactors are susceptible to catastrophic accidents, fast reactors pose a unique risk. In fast reactors, the core where the bulk of the nuclear reactions take place is not in its most optimal configuration in terms of energy production when operating normally. If the fuel in the core is rearranged for some reason, it could lead to an increase in the reaction rate and an increase in energy production. This, in turn, could lead to further core rearrangement and a potential positive feedback loop. An example of what is meant by core rearrangement is the localized melting of fuel elements, which is a likely possibility if some part of the core becomes overheated. When the fuel melts, the plutonium and the uranium could flow into those regions of the reactor, closer to other fuel elements. As the fissile materials form a more compact mass, fewer of the neutrons produced during fissions would escape. Then, the rate of fission would increase, causing greater amounts of heat to be produced, in turn causing further fuel elements to melt. The result of this feedback process could be what is called a core disruptive accident (CDA), a core meltdown involving a large and explosive energy release and the potential dispersal of radioactive material into the environment.

The first major study of the energy released by a CDA was done by Nobel Prize-winning physicist H.A. Bethe and British scientist J.H. Tait in 1956 (Bethe and Tait 1956).[7] Since then, estimates of the energy released in a CDA and the ability of the reactor to withstand the accident have been a standard part of the safety evaluations of nearly all of the fast breeder reactors constructed or proposed in the United States and Western Europe (Waltar and Reynolds 1981).

The PFBR also has what is called a positive coolant void coefficient or, a positive sodium void coefficient, because sodium is the material used to carry away the heat produced in the core. This is another positive feedback loop. If some of the molten sodium in the core were to heat up and form bubbles of sodium vapour, the rate of fission reactions, and therefore the heat produced by the reactor, would increase.[8] In turn, this would increase the rate of sodium heating up and forming bubbles. The word 'positive' is because the feedback to the sodium that results from the change in sodium properties causes a greater change in the properties. The word 'void' refers to regions where the sodium might have boiled away.

This positive feedback loop is dangerous and, unless the reactor is immediately shut down, all the sodium could boil away. Because the heat produced would not be taken away by the sodium, the fuel elements in the reactor would start heating up and eventually melting. The magnitude of the void coefficient tends to increase with core size.[9] The fast breeder test

7. For the record, Bethe felt that the development of nuclear power was 'a necessity' (Ioffe 2006).

8. The heating up of the sodium might occur as a result of, say, the cessation or slowing down of coolant flow in some part of the reactor, perhaps because of some blockage.

9. 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

reactor, which the DAE has been operating, does not have a positive coolant void coefficient. For the core design that has been adopted for the PFBR, the coolant void coefficient has a value of 4.3 $ ('dollars'), large compared to most commercial breeder reactor designs (IAEA 2006, 48).[10]

Knowing these general facts, my former colleague Ashwin Kumar, an engineer by training, and I looked in more detail at the DAE's studies of potential accidents at the PFBR (Kumar and Ramana 2008). It turns out that the DAE's assertions about the safety of the PFBR are based on the assumption that a CDA will release only 100 megajoules of energy (Chetal et al. 2006, 860). This estimate is much smaller when compared with other fast reactors, especially when the much larger power capacity of the PFBR—and therefore, the larger amount of fissile material used in the reactor—is taken into account. For example, it was estimated that the German SNR-300 reactor, which was designed to produce only 760 megawatts of thermal energy, could produce 370 megajoules in the event of a CDA—much higher than the PFBR estimate. Other fast reactors around the world have similarly higher estimates for how much energy would be produced in such accidents (Waltar and Reynolds 1981, 524).

What Ashwin realized by looking at the DAE's studies was that its estimate is based on two main assumptions: (1) that only part of the core will melt down and contribute to the accident; and (2) that only about 1 per cent of the thermal energy released during the accident would be converted into mechanical energy that could damage the containment building and cause ejection of radioactive materials into the atmosphere (Singh and Harish 2002).

Neither of these assumptions is justifiable. The phenomena that might occur inside the reactor core during a severe accident are very complex (Bell 1981; Wilson 1977), and most studies of breeder accidents assume larger parts of core meltdown in their reference scenarios (Badham and Chan 1979; Jackson and Nicholson 1972; KAERI 1997; Theofanous and Bell 1984). In addition, important omissions of various uncertainties in the parameters used in the DAE's safety studies make its estimate inadequate

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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, hence, become less important as a whole as the volume of the core increases.

10. A 'dollar' is the margin of reactivity provided by delayed neutrons, which are typically released only tens of seconds after the fission event. If the reactivity increases above unity by more than this amount, the rate of power increase is controlled by the much shorter generation time (about 10 microseconds) of prompt neutrons.

(Kumar and Ramana 2011). Likewise, the efficiency of conversion could be much larger than the 1 per cent assumed by the DAE. Tests at the British Winfrith facility suggest energy-conversion efficiencies of about 4 per cent (Berthoud 2000).[11] Thus, a severe CDA could release much more energy than the DAE has assumed in its safety analysis.

Finally, if such an accident were to take place, would the PFBR containment—the most visible structure from the outside of any nuclear plant—be able to withstand the shock? The purpose of a containment building is to act as a barrier to the escape of radioactive materials into the atmosphere. Compared to most other breeder reactors, and light water reactors for that matter, the design of the PFBR's containment is relatively weak, only designed to withstand 25 kilopascals of pressure difference (over-pressure) before failing (IAEA 2006, 224; Chetal et al. 2006, 860). Containments for light water reactors routinely have design pressures above 200 kPa (APS Study Group 1985, S94). The design for the 700-MW pressurized heavy water reactors that the DAE is planning to construct includes a containment that is designed to withstand up to 156 kPa (S.A. Bhardwaj 2006, 871). Thus, it is possible to design containments to withstand much higher pressures. The DAE could have chosen such a containment design—and, given the many uncertainties involved, that would have been the safer choice—but did not.

While the DAE could have built a stronger containment, it would also have been possible to decrease the positive void coefficient and, indeed, to make it negative. This could have been done by, for example, designing the reactor core so that fuel subassemblies were interspersed within the depleted uranium blanket, in what is termed a heterogeneous core. The US Clinch River Breeder Reactor, which was eventually cancelled, was designed with a heterogeneous core (Waltar and Reynolds 1981), and Russia has considered a heterogeneous core for its planned BN-1600 reactor (Troyanov et al. 1990). Such a design would have lowered the likelihood of a CDA.

The DAE has defended its choice of a core with a large sodium void coefficient with 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' (Paranjpe 1992, 512). This is puzzling because a similar situation in a core with an overall positive sodium void coefficient can only be more dangerous. The

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11. These tests are performed on relatively small systems and it is not completely clear whether large reactors such as the PFBR would have larger or smaller efficiencies compared to them.

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 we have seen, the PFBR's containment may not maintain its integrity during severe accidents.

The DAE, while arguing on the one hand that these safety concerns have been taken into account in the PFBR design, has also argued that safety concerns were completely misplaced in the first place. Thus, S.R. Paranjpe, former director of IGCAR, 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' (Paranjpe 1992, 513).[12]

This statement reveals two reasons why the PFBR has been designed the way it has. First, there is the conviction that the reactor is inherently safe. If one starts with such a conviction, then it is unlikely that safety studies will be reliable (Kumar and Ramana 2011). This conviction is manifested partly through assumptions that cannot really be considered 'reasonably worst case'.

The second point implied in the same statement ('why invest more funds') is that cost has been given a higher priority than safety (Paranjpe 1992, 513). Reducing the sodium coolant void coefficient would have increased the fissile material requirement of the reactor by 30–50 per cent—an expensive component of the initial costs. Likewise, a stronger containment building would have cost more. Another example of where cost comes before safety is the Tarapur I and II reactors that were imported from the United States. These suffered regularly from vibrations, but the DAE chose not to make design and other changes to eliminate them 'for economic reasons' (Nanjundeswaran and Sharma 1986, IV-10.8). Now, lowered costs in the case of already-expensive nuclear electricity would be welcome, but not if they come with the increased risk of catastrophic accidents.

12. A similar question was posed by AEC Chairman Raja Ramanna: 'I would like to ask, are we not spending too much money on health and safety? Should we not have a look and find out whether the international standards of safety (in nuclear programmes) are indeed that necessary? . . . Should we follow the international standards blindly? I think we should have courage to look at these standards especially where they are leading to runaway costs' (Ramanna 1973, cited in D. Sharma 1983 [110]).

Experience

The DAE, like other organizations involved in nuclear activities, often verbalizes safety goals. But its performance and decision-making often depart from these public pronouncements. As a way of trying to assess the potential for accidents at Indian nuclear facilities, we turn to three related questions. What has been the experience with accidents, both small and large, at the DAE's facilities? What kind of practices lie beneath the DAE's planning and operations? What has been the DAE's attitude towards nuclear safety?

First, we will discuss some of the major accidents that have occurred in facilities run by the DAE.[13] This is by no means exhaustive and is purely indicative. As argued earlier, the fact that none of these led to catastrophic radioactive release to the environment isn't by itself a source of comfort. Safety theorists have argued cogently that this absence of evidence of 'accidents should never be taken as evidence of the absence of risk . . .' and that '. . . just because an operation has not failed catastrophically in the past does not mean it is immune to such failure in the future' (Wolf 2001, 294).

Narora 1993

The most serious of the accidents at nuclear reactors in India occurred on 31 March 1993. Early that morning, two blades of the turbine of the first unit at Narora broke off due to fatigue. They sliced through other blades, destabilizing the turbine and making it vibrate excessively. The vibrations caused the pipes carrying hydrogen gas that cool the turbine to break, releasing the hydrogen, which soon caught fire. Around the same time, lubricant oil too leaked. The fire spread to the oil and throughout the entire turbine building. Among the systems burnt by the fire were four cables that carried wires and electricity, which led to a general blackout in the plant. One set of cables was the one that supplied power to the secondary cooling systems, and when it got burnt those cooling systems were rendered inoperable. To make things worse, the control room was filled

13. To the extent possible, we have derived these descriptions from documents put out by the DAE and its sister organizations. We have used news and media reports where DAE documents were not available—or as a supplement. We have assumed that these were accurate unless there was some strong reason to not believe them. We have tried not to place too much stock on any one report. Another source of information has been the detailed annual reports entitled 'Operating Experience with Nuclear Power Stations in Member States' that the International Atomic Energy Agency puts out. These reports are based entirely on information that the DAE gives the IAEA—information that is highly unlikely to be given to the citizens of the country directly.

with smoke and the operators were forced to leave it about ten minutes after the blade failure.

Prior to leaving, however, the operators manually actuated the primary shutdown system of the reactor (Koley et al. 2006). Fortunately, the reactor shutdown systems worked and control rods were inserted to stop the chain reaction. The problem then was something that was on display at Fukushima: the reactor went on generating heat because the fuel rods in a reactor accumulate fission products which continue to undergo radioactive decay. While this so-called decay heat is only a small fraction of the power that is produced when the reactor is operating at full power, it doesn't stop being produced when the reactor is shut off. It is also significant enough to cause the fuel to get heated up and melt down if ignored, as happened at Fukushima. Thus, the reactor must continue to be cooled even after shutdown. To do this, the operators had to start up diesel-driven fire pumps and circulate *water meant for fire control* to carry away decay heat (NEI 1993).

In addition, there was still concern that the reactor might become critical again. This happens because some of the fission products can absorb neutrons. Therefore, as their concentration changes, the neutron balances change, posing the risk of re-criticality. Therefore, some operators climbed on to the top of the reactor building, with the aid of battery-operated torches, and manually opened valves to release liquid boron into the core, further absorbing neutrons. Had these workers not acted as they did, it is possible, according to the chairman of the AERB, that there would have been a local core-melt and explosive fuel–coolant interaction (Chanda 1999). The names of those heroic workers have never been made public.

It took seventeen hours from the time the fire started for power to be restored to the reactor and its safety systems. The operators who were forced to leave the control room because of smoke could not re-enter it for close to thirteen hours. An attempt was made to control the plant from the emergency control room. But since there was no power available, the Unit 1 control panel of the emergency control room had no functioning indications. Thus, Narora was almost unique in that operators had no indications about the condition of the reactor and were, in effect, 'flying blind' (Nowlen, Kazarians, and Wyant 2001, A23-2).

The Narora accident has been the DAE's closest approach to a catastrophic accident. More worrisome is the evidence that the accident could have been foreseen and prevented.

First, the failure of the turbine blades was avoidable. In 1989, General Electric Company had informed the turbine manufacturer, BHEL, about a design flaw which had led to cracks in similar turbines used around the world. They recommended design modifications, and the manufacturer had responded by preparing detailed drawings for Nuclear Power Corporation, which operated the Narora reactor. However, NPCIL had not taken any action (Gopalakrishnan 1999a). In addition to GE, BHEL had also recommended that NPCIL replace the blade design before any accident occurred. But NPCIL did not act on this advice until months after the accident (Gopalakrishnan 1999b).

Second, even if the turbine blade did fail, the accident might have been averted if the safety systems had been operating which, presumably, they would have if only their power supply had been encased in separate and fire-resistant ducts. By the time the Narora reactor was commissioned, this was established wisdom in the nuclear design community and had been so ever since a fire at Brown's Ferry plant in the United States in 1975. The 1975 accident resulted in significant changes being mandated at all US nuclear plants (Ramsey and Modarres 1998, 106). Nuclear reactor operators had to implement one or more of a number of possible and redundant—physically and electrically—ways of avoiding fire damage. Other countries also adopted similar measures.

Some in the DAE also appear to have been aware of the necessary measures to reduce the risk of fires. The probabilistic safety assessment for Narora performed in 1989 by DAE analysts, including Anil Kakodkar, who was to become head of the DAE in 2000, points out that the effects of common cause failures (described below) would be reduced if 'physical diversity and fire barriers are provided' (Babar et al. 1989). Another paper by safety analysts in the DAE, albeit well after the Narora accident, acknowledges that after 'the Browns Ferry experience in the United States, fire risk evaluation and hazard analysis has been considered as an essential component of safety evaluation and assurance' (Vinod et al. 2008).

Thus, it would seem that even though some sections of the DAE were aware of the risks of a common cause failure in the event of a fire, organizational leaders ignored important safety practices needed to reduce these risks. Instead, the plant was constructed with four backup power supply cables laid in the same duct, without any fire-resistant material enclosing or separating the cable systems. This set-up was perfect for a single cause—a fire—to render these many safety systems inoperable, and constitutes an excellent example of what is called a common cause failure. Such failures are very difficult to predict and model in probabilistic safety assessments (Ramana 2011a).

Third, the DAE had not taken any serious steps towards fire mitigation despite earlier fire accidents at its own reactors.[14] Not only did the fire have

14. In 1985, an overheated cable joint at RAPS II caused a fire that spread through

precursors, but so did the events that led to the Narora fire. In other words, the factors that contributed to the Narora accident had also been present prior to the accident. This was not something that was discussed in the literature surrounding the Narora accident put out by the DAE. Rather, this realization came about as a result of my colleague Ashwin's looking through the various issues of the IAEA's 'Operating Experience with Nuclear Power Stations in Member States' reports that I had collected over the years.[15]

Let us see the record on two important factors: excessive vibrations in the turbine bearings and oil leaks. In 1981, RAPS I was shut down twice because oil leakage in the turbine building had led to high levels of sparking in the generator exciter (IAEA 1982, 235). After it was restarted, it had to be shut down yet again when it was found that large amounts of oil had leaked from the turbine governing system. Only when the reactor was restarted a third time, in early 1982, were the high vibrations of the turbine bearings noticed and the failure of turbine blades discovered (IAEA 1983, 250). This led to a prolonged shutdown of more than five months; even after this problem had apparently been fixed the reactor had to be shut down once again because of high temperatures in the turbine bearing (IAEA 1983, 250). Again in 1983, high vibrations were noticed in turbine generator bearings and it was revealed that two blades in the second stage of the high-pressure rotor had sheared off at the root (IAEA 1984, 292). In 1985, the first unit of the Madras Atomic Power Station was shut down repeatedly because of high bearing vibrations in the turbine generator (IAEA 1986, 240). RAPS I had to be shut down due to vibrations in the turbine bearing in 1985, 1989 and 1990 (IAEA 1986, 242; 1990, 302; 1991, 298).

Oil leaks have also been common in Indian reactors. In 1988, MAPS II was shut down due to an oil leak from the generator transformer (IAEA 1990, 288). Oil leaked from a turbine bearing in MAPS II in 1989 (IAEA 1990, 300). In 1992, there was an oil leak in the turbine stop valve in MAPS II (IAEA 1993, 288). Additionally, in 1992, there were two separate

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the cable trays and disabled four pumps (IAEA 1986, 244; Gopalakrishnan 1999a). A few years later, in 1991, there were fires in the boiler room of the same unit and the turbo generator oil system of RAPS I (IAEA 1992, 394–96). In 1989, a large spark was observed from slip rings on the exciter end of the turbine in MAPS I; there were also two other fires in the same reactor near the primary heat transport system (IAEA 1990, 298).

15. Unfortunately, this sort of analysis may no longer be possible. Starting in 2005, IAEA does not give information on each outage but aggregates them into various 'outage causes'. Thus, the kind of detail that is needed to perform historical analyses of different kinds of accidents is no longer available in the public domain.

oil leak incidents in the Narora I turbine generator system (IAEA 1993, 289). There was at least one hydrogen gas leak prior to the Narora accident: this happened in 1991 in the generator stator water system of MAPS II (IAEA 1992, 390).

The DAE did not take serious note of these earlier failures. When asked by an interviewer about recurring turbine blade failures at nuclear reactors, AEC Chairman Chidambaram sidestepped the issue by suggesting that 'this kind of failure at Narora has happened for the first time . . . two blades failing', and then offering the non sequitur, 'You must remember that as far as nuclear reactor is concerned, there was no problem at Narora. The reactor worked perfectly according to design' (Chidambaram 1993, 102). By ignoring these early warnings, the DAE set the stage for the Narora failure that led to 'widespread damage to the [turbine generator] set, condenser and caused [a] fire which engulfed the cables, the turbine building and control equipment room' (G. Ghosh 1996, 30). But that's not all. As described later, events such as fires, turbine vibrations, and leaks, have persisted even after the Narora accident, raising questions about what has been learnt from that event.

Kakrapar 1994

One of the actions that the AERB took in the wake of the fire at the Narora plant was to order the DAE to close down other nuclear power stations for inspection of their turbine blades. This may also have prevented an accident at the Kakrapar Atomic Power Station (KAPS). Just behind the turbine building of KAPS is the Moticher Lake; the outlet ducts of the turbine building lead to the lake (Gadekar 1994b). Moticher is an artificial lake and has gates to control the flow of water. However, these gates had not been operated for the most part. As explained by the chief superintendent of KAPS in an interview with *Gujarat Samachar*, mud had 'collected around [the] gates [and] a lot of tall grass had grown' near them; therefore, despite a 'lot of effort the gates of the lake could not be opened' (Gadekar 1994c, 4).

On 15 and 16 June 1994, there were heavy rains in South Gujarat and the water level of the lake began to rise. That resulted in the ducts that were meant to let out water becoming conduits for water to come in. Water began entering the turbine building on the night of 15 June (Gadekar 1994a). There were no arrangements either for sealing cable trenches and valve pits, both of which also allowed water to enter the reactor building. By the morning of 16 June, there was water not only in the turbine building but also in other parts of the reactor complex. The workers in the morning shift had to swim in chest-high water, and the control room was reportedly inaccessible for some time (Gadekar 1994a). No action was taken till eleven in the morning on the 16th, when a site emergency was declared and workers were evacuated.

As mentioned, the gates of the Moticher Lake could not be opened, even after the KAPS management requested help from the district and state authorities. Fortunately, villagers from the area, who were worried about the security of their own homes, made a breach in the embankment of the lake which allowed the water to drain out. It was only on 18 June that a large pump was brought to Kakrapar from Tarapur, and the work of removing the water from the turbine building begun (Gadekar 1994a). During all this time, the KAPS management did not think it fit to report to the Atomic Energy Regulatory Board what had happened; the chairman of the Board found out about it from the newspapers (Gadekar 1994c).

Due to floodwater entry, much of the equipment in the turbine building was submerged (Bajaj and Gore 2006). This included the water pumps used to cool the reactor core. Electrical power from the grid failed, and diesel generators had to be used. And finally, some accounts books, correspondence regarding plant construction, and some (engineering) drawings were destroyed (Gadekar 1994c).

Since the reactor had been shut down for over four months at the time of the flooding, there was no great danger of an accident. Had it been functioning and there had been reason to issue an off-site emergency, the situation would have been desperate. In the words of Surendra and Sanghamitra Gadekar: 'The scene . . . was one of utter devastation. A thousand houses were demolished in Bardoli alone, and many more elsewhere were damaged. Whole sections of roads, railway lines and bridges vanished into oblivion. Trees were laid low and farms turned into ponds. We were caught 100 km from home and had to trudge back through rivers and streams and making long detours' (Gadekar 1994b). There was simply no way that people could have been evacuated on time. When the station director was asked how he would have evacuated people if it had become necessary, he said that he would have resorted to the use of helicopters. On being asked how many helicopters did he have access to, he replied that there were two of them. More than 10,000 people were reported to live within just three kilometres of the plant.

As it happened, the main damage resulting from the rains was that the floodwater breached the solid waste–management facility, lifted canisters of radioactive waste and carried them out into the open (Gadekar 1994c). It is not known exactly how many canisters were swept away. Nor is it clear if they were ever recovered or if any of them released its contents into the waters.

Now, none of this would have come to light but for Manoj Mishra. At the time of the accident, he was employed by NPCIL at Kakrapar and was the president of Kakrapar Unit Kendriya Sachivalaya Hindi Parishad (Gujarat High Court 2007). Mishra wrote a letter to *Gujarat Samachar* about what had happened during the flooding. He was suspended and, after an internal inquiry, was removed from service in March 1996. Since then, Mishra has been supporting himself and his family by taking up various odd jobs, and fighting the nuclear establishment in courts—and losing (Gadekar 2010). The most recent, in 2007, was in the Gujarat High Court (Gujarat High Court 2007). Mishra has appealed to the Supreme Court.

The root cause of the Kakrapar flooding—the low elevation of the site—had been identified earlier, and earth supposedly added (M.R. Srinivasan 2002, 139). Clearly, that had proved ineffective. Moreover, the reactor was not well designed: the former head of the AERB has stated that 'sealing arrangements were not provided to prevent water ingress through cable trenches and valve pits. Similar flooding had occurred twice at RAPS in 1976 and 1982, owing to the very same construction errors as at KAPS' (Gopalakrishnan 1999a).

Of course, the DAE reassures us that this will never recur and lists various corrective measures it has taken.[16] These would have been reassuring, if only they had included anything that was not obvious—especially at a site that had been determined to be at risk of flooding even before construction started. One would imagine that these measures should have been standard practice long before the first nuclear reactor started operating.

Kaiga 1994

As mentioned in chapter 2, on 13 May 1994, the inner containment dome of one of the units of the Kaiga nuclear power plant collapsed during reactor construction. The dome itself had been completed but cabling and other works were being carried out (Havanur 1994).

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16. For example, Bajaj and Gore (2006) state that 'The event was analyzed in depth and corrective measures were taken not only at Kakrapar Atomic Power Station but at other stations and projects also' and list some 'salient recommendations': '(i) All the entry points of pipe tunnels/cable trenches to the basement of all buildings having safety-related equipment are sealed with multiple barriers. (ii) The openings (like manholes, ventilation points) are either raised above design basis flood level, or are sealed. (iii) A number of administrative measures (like maintenance/inspection of flood prevention measures) and review/modifications of flood protection plans/ emergency operating procedures'.

Accidents have a way of revealing things that are normally left unsaid. This was the case with the dome collapse, and a number of officials who wished to remain anonymous talked to the media about what went wrong. They revealed at least three of the underlying reasons. The first was inadequate quality control: according to the DAE officials, it seems that 'while inputs, such as, cement and steel, had been tested for quality, that was not the case with the concrete blocks as a whole' (Mohan 1994, 85). This goes against a basic requirement of nuclear safety, which maintains that 'facilities [have to be] constructed to the highest standards' (NEA 1993, 51). Construction workers have accused the contractors of various malpractices in construction (Havanur 1994). Faulty work practices may also have played a role.[17] The second was faulty design (Panneerselvan 1999). Apparently, some cables that run through the dome for prestressing were too closely spaced in certain zones, particularly near the openings for the steam generator, and this caused excessive loading (Sundararajan, Parthasarathy, and Sinha 2008, 41).

The third was a failure of regulation:

> Senior [NPCIL] civil engineers and the private firms which provide civil engineering designs and construction drawings to the DAE have had a close relationship. In this atmosphere of comradeship, the [NPCIL] engineers did not carry out the necessary quality checks on the designs they received before passing them on to the Kaiga project team. The AERB also did not check this, because it had almost no civil engineering staff with it. Serious design errors went undetected and these eventually led to the failure of the dome (Gopalakrishnan 1999b).

The Kaiga dome collapse is unprecedented in the annals of nuclear energy history. It also points to one of the dangers of relying on redundancy as a safety mechanism. The idea behind constructing a containment dome is that even if all safety mechanisms within the reactor fail and a severe accident occurs, the strong containment building will be capable of withstanding the high pressures that would accompany the accident and hold ('contain') all radioactive substances released from the reactor core during the accident. So, at face value, this makes for greater safety. But, consider this speculative scenario:

> If such a collapse had taken place during operation of the nuclear plant, about 130 tonnes of concrete falling from a height of nearly 30 metres would have

17. Such practices led some years later to a fire involving many cans of paint on the same dome (ToI 1999a).

> damaged the automatic control rod drives that lie below the crown of the dome, disabling them and making the safe shutdown of the reactor difficult. The massive weight of concrete might have led to damage to the nuclear coolant pumps and pipes, resulting in severe loss of coolant. This could have led to nuclear core meltdown and the escape of large amounts of radioactive substances to the environment. (Subbarao 1998)

In this admittedly hypothetical scenario, one safety mechanism's failure would have disabled other safety mechanisms and triggered an accident. Fortunately, at the time of the accident, the reactor had not been fully constructed and the core had not been loaded.[18]

The Kaiga episode was also an example of direct interference in the activities of the AERB. According to AERB's Gopalakrishnan, 'When, as chairman, I appointed an independent expert committee to investigate the containment collapse at Kaiga, the AEC chairman wanted its withdrawal and matters left to the committee formed by the [NPCIL] [Managing Director]. DAE also complained to the [Prime Minister's Office] who tried to force me to back off' (Panneerselvan 1999).

Kalpakkam 2003

As described in chapter 4, the Kalpakkam Atomic Reprocessing Plant produces large amounts of chemical and radioactive wastes. Some of this waste is stored in a part of the facility called the Waste Tank Farm (WTF). On 21 January 2003, some employees were tasked with collecting a sample of low-level waste from it. Unknown to them, a valve had failed, resulting in the release of high-level waste, with much higher levels of radioactivity, into that part of the WTF where they were working. Although the plant was five years old, no radiation monitors or mechanisms to detect valve failure had been installed in that area. Therefore, the workers had no way of knowing that the sample they went in to collect was actually emitting high levels of radiation. The accident was recognized only after a sample collected was taken to a different room and processed. In the meantime, six workers had been exposed to high radiation doses (S. Anand 2003).

Apart from the lack of monitoring mechanisms, what is really a cause for concern is the response of the management, in this case, BARC. Despite a safety committee's recommendation that the plant be shut down, the top management of BARC decided to continue operating the plant. Then, the

18. This is clearly a worst-case scenario and might seem far-fetched. But, as sociologist Lee Clarke argues, there is value in engaging in constructing such scenarios because disasters, even worst-case ones, are a part of life (Clarke 2006). Analysing them can lead to insights.

employees' union, the BARC Facilities Employees Association (BFEA), wrote a letter to the director setting forth ten safety-related demands, including the appointment of a full-time safety officer. The letter also recounted two previous incidents in the past two years when workers had been exposed to high levels of radiation, and how the higher officials had always cited emergency as a reason for the Health Physics department not following safety procedures. Once again, there was no response from the management. Finally, some months later, the union resorted to a strike.[19] The management's response was to transfer some of the key workers involved in the agitation and give notice to others; this had the effect desired by the management and, two days later, all the striking workers rejoined. The BARC director's interpretation was: 'If the place was not safe, they would not have joined back'. Finally, the union leaked information about the radiation exposure to the press.

Once the news had become public, the management grudgingly admitted that this was the 'worst accident in radiation exposure in the history of nuclear India' (S. Anand 2003). But it claimed that the 'incident' had resulted from 'over enthusiasm and error of judgment' on the part of the workers (Venkatesh 2003). The management also tried to blame the workers for not wearing their thermoluminescent dosimeter (TLD) badges (Subramanian 2003). But this has nothing to do with the accident: TLD badges would not have warned the workers about radiation levels until well after the fact.[20]

For its part, the BFEA claimed in essence that the accident was only to be expected, and that because of the unrelenting pace of work at KARP and the 'unsafe practices being forced on the workers', accidents have become a regular feature (PTI 2003).

Practices

One can look at safety even in the absence of major accidents, in the failures that occur during day-to-day work in DAE facilities.

19. There have been previous instances of strikes at BARC, including one by operators in the CIRUS and Dhruva reactors in June 1992 (AERB 1993, 10).

20. These badges measure cumulative exposure over a period of time, and are meant to be periodically submitted to the Health Physics department for assessment. The practice of inadequate methods of detecting high levels of radiation seems to continue. In 2011, workers at the Kakrapar reactor found that the radiation detectors they were given did not record beyond a certain value, whereas they had been asked to work in a zone that had nearly twenty times that level of radiation (Banerji 2011).

Patterns

One feature that points to poor safety practices at DAE facilities is repeated occurrences of similar accidents. An important example is regular leaks and spills of heavy water.[21] Despite a lot of effort—quite understandable because heavy water is expensive and hard to produce—the DAE has not managed to contain them. In 1997, such leaks occurred at the Kakrapar I, MAPS II and Narora II reactors (IAEA 1998, 301–20). The amount of heavy water that leaks can be significant. For example, on 15 April 2000, there was a leak involving seven tonnes of heavy water at the Narora II reactor (AERB 2001, 13). Three years later, on 25 April 2003, there was another heavy water leak at the same reactor, this time involving six tonnes (AERB 2004a, 18). In June and July of 2012, there were two significant heavy water leaks at RAPS, one involving thirty-eight workers being exposed to radioactive tritium (Sundaram 2012; Sebastian 2012a).

The 2003 leak occurred while a device called BARCCIS (Bhabha Atomic Research Centre Channel Inspection System), which is used to inspect coolant tubes in reactors, was in operation. A similar leak had occurred in March 1999 at the Madras Atomic Power Station (described in chapter 8). The Atomic Energy Regulatory Board had undertaken a review of the BARCCIS system and suggested a number of changes in design, operating procedures, and training (AERB 2004a, 18). The occurrence of a similar leak at the Narora reactor despite these changes suggests technical weaknesses in the regulatory board, fundamental flaws in the system, or continued operator errors.

Another set of examples is of the problems that led to the Narora accident. The many instances of problems prior to the accident have already been listed. This trend has continued even after the Narora accident. In late 1993, high vibrations and temperature in both Narora II and RAPS I turbine generator buildings led to their being shut down (IAEA 1994, 333–36). The problems in these reactors persisted into 1994, with Narora II being shut down due to high bearing temperatures, and RAPS I due to turbine bearing vibrations (IAEA 1995, 313–16). In 1995, despite repeated shutdowns supposedly meant to mitigate turbine problems, blades failed in the turbine of Narora II (IAEA 1996, 314).

Even after being restarted following the accident in 1993, Narora I was shut down repeatedly in 1995 because of high vibrations of the turbine generator bearing (IAEA 1996, 312). In 1997, RAPS I had to be repeatedly shut down due to high turbine bearing vibrations (IAEA 1998, 314). In 2000, Kaiga II suffered from repeated turbine vibration problems (IAEA

21. On the resultant risks to health, see chapter 8.

2001, 294). Fires have also occurred repeatedly. In 1996, in Narora II, there was heavy oil smoke from the turbine building (IAEA 1997, 314). That same year, there was an oil fire in the turbine building of Kalpakkam II (IAEA 1997, 310). The following year, smoke was observed in Kalpakkam II, there was a fire in the turbine generator of Kakrapar II, and smoke was seen coming from the insulation of the main steam line of the turbine generator in Kakrapar II (IAEA 1998, 302–08). There was a fire due to an oil leak in Kalpakkam I in 2000 (IAEA 2001, 300). There have also been numerous oil and hydrogen leaks.[22]

Such repeated problems are indicative of the inability of the organization as a whole to learn appropriate lessons or its inability to control the technology and operate it adequately.

Inoperative Safety Systems

A second notable and disturbing feature is the frequent failure of safety devices. These are the mechanisms by which control of the reactor is sought to be maintained under unanticipated circumstances. Therefore, if they do not work as expected, then, it is more than likely that a small event will cascade into a major accident. A related problem is that of safety devices being left in an inoperative state or of periodic maintenance of equipment being neglected.

An example of how minor failures contribute to escalating an accident is the 1993 Narora accident discussed earlier. The accident could have been prevented had the smoke sensors in the power control room at Narora detected the fire immediately. But that did not happen and the fire was detected only when the flames were noticed by plant personnel (Srinivas 1993). A different complication arose three hours and fifty minutes into the accident when the two operating diesel-driven fire water pumps tripped inexplicably (Nowlen, Kazarians, and Wyant 2001, A23-8). Up to now, the cause for the failure has not been identified. A third pump was out of service for maintenance.

Many of these problems continue to recur. In 2005, for example, the AERB found instances of failures in fire detectors at Kakrapar and in the power supply for emergency cooling at MAPS (PTI 2005).

22. Listed below are just those from the period between 1995 and 2000. Operating records reveal repeated oil leaks occurred in Kakrapar II in 1995 (IAEA 1996, 306). In 1997, there were oil leaks in Kalpakkam II and a hydrogen leak in Kakrapar II (IAEA 1998, 304–08). In 1999, there was another hydrogen leak in Kakrapar II, as well as one in Narora II (IAEA 2000, 288–96). In 2000, there were hydrogen leaks in Narora I, Narora II and RAPS III, and oil leaks in RAPS III and Kaiga II (IAEA 2001, 294–312).

Another frequent occurrence was heat transport pumps being made unavailable by grid frequency fluctuations. Quite often when this happens, some pumps are already unavailable for use. For example, RAPS I tripped four times in the year 1980, at the time of power system fluctuations, and at least three of these failures were due to the disabling of primary heat transport pumps (IAEA 1981, IN-3, 3). At least once, some of the pumps were already unavailable when this happened. That year, in TAPS II, generation was restricted for nearly two weeks because only one recirculating pump was in service (IAEA 1981, IN-2, 3). Subsequently that year, the unit had to be shut down for twelve days to attend to the failure of the sole recirculating pump.[23] Such problems have occurred repeatedly throughout the 1980s and 1990s.

Similarly in 2004, MAPS II was shut down for eight days because the two main primary coolant pumps were unavailable (IAEA 2005, 324). After being restarted, the reactor had to be shut down again because the motor bearings of one of the pumps had to be replaced. Another example of inadequate maintenance is the response to the leaks of the end shield of RAPS I throughout much of the 1980s.

Emergency Plans

A different perspective on safety practices comes from looking at the design and operation of emergency plans. Though the DAE has argued that nuclear accidents are not possible in India, such plans are typically justified as 'a measure of abundant caution' by its personnel (Sundararajan 1991, 389). But, the institution has made these detailed emergency plans inaccessible to the public who should be the real audience for these plans.[24] Independent

23. Another recurring problem at this reactor is leaks from a primary feed-water pump recirculation line.

24. The DAE used to offer an official document on its website titled 'Nuclear Emergencies—How to Respond' (DAE N.D.). The DAE first argues at length that it is 'extremely unlikely that the public near a nuclear facility will be exposed to any radiation beyond the permissible limits'. But it suggests that 'if you still feel concerned on hearing any news or rumour about an incident at a nearby nuclear facility, follow these simple guidelines'. Among its list of Dos are: 'Go indoors. Stay inside', 'Close doors and windows', 'Cover all food, water and consume only such covered items', and 'Extend full cooperation to local authorities and obey their instructions completely—be it for taking medication, evacuation, etc.' Its list of Don'ts includes: 'Do not panic', 'Do not believe in rumours passed on by word of mouth from one person to another', and 'Do not disobey any instruction of the District or Civil Defence Authorities who would be doing their best to ensure the safety of yourself, your family and your property'. The DAE appears to have removed this document.

surveys have observed that panchayat chiefs are often not clear about emergency procedures (MAI 1993, 379). Apart from the culture of secrecy and arrogance characteristic of the nuclear establishment, there are two other plausible reasons. First, publicizing such plans would make people realize that nuclear reactors and other fuel cycle facilities are hazardous. Second, examination of these plans by people outside of the bureaucracy would reveal that they will not work.

An example is the emergency plan for the Kakrapar reactors, which was accidentally made public. The plan was to deal with an estimated 273,843 people who lived within sixteen kilometres of the site (KAPS 1989, 81). The Kakrapar power station is located on the banks of the Tapi River, which had one bridge across it in the vicinity of the reactor. The plan required all evacuees to go over this single bridge, a sure-fire recipe for a major traffic bottleneck (Rawat 1998). The plan also absurdly requires people in villages and towns further up the river to come *towards* the reactor first, cross the bridge and then go away from the reactor. Finally, the plan would try to accommodate 40,000 people in a high-school building and all the residents of Mandvi town and two villages in a primary school in the town of Mangrol.[25]

Emergency plans also demonstrate their propensity for failure during trial exercises—the operational part of our examination. During such drills, officials and local inhabitants are supposed to behave as though a real emergency were under way. But nothing functions the way it should. For example, in 2001, the DAE, along with local officials, carried out an off-site emergency evacuation exercise of villages around MAPS in Kalpakkam. Reflecting, perhaps, the lackadaisical attitude that characterizes safety considerations at nuclear power plants, the station director's wireless set refused to function. Instead, it 'produced just a kee-kee sound' (Radhakrishnan 2001). During an emergency drill in Tarapur in 1988, the collector of Thane district was informed of the 'disaster' at 8.15 p.m. According to the emergency plan, the collector is supposed to immediately proceed to the Emergency Control Room. Instead, he reached at 10.50 a.m. the next morning, and it 'seemed' to observers as if he was 'going about the whole thing as if he was attending just another function where he was asked to perform the role of the chief guest' (Shenoy 1988). In 2012, the Comptroller and Auditor General reported that 'Offsite emergency exercises carried out highlighted inadequate emergency preparedness. Further, AERB was not empowered to secure compliance of corrective measures suggested by it' (CAG 2012, viii).

25. The populations of Mandvi and Mangrol, according to the 2001 Census, are about 45,000 and 55,000 respectively.

If emergency plans are so laughably inadequate, why are they even concocted? The best answer comes from sociologist Lee Clarke, who has termed such plans 'fantasy documents', and has argued that these 'are rhetorical instruments that have political utility . . . for organizations and experts' rather than having any actual use in the event of a real emergency (Clarke 1999, 13). Their purpose is mainly to maintain the legitimacy of the nuclear establishment and allow it to stave off opposition from local inhabitants who are rightly concerned about the potential for catastrophic accidents.

'Absurd Confidence'

There is yet another subtle concern with the nuclear establishment's approach to safety. At least from their public persona, the DAE and its attendant organizations are completely confident that the facilities which they build and operate are safe. Following the multiple accidents at Fukushima, the DAE Secretary assured the viewers of the television channel, NDTV, that Indian reactors are 'one hundred percent' safe (Bagla 2011), and that the probability of a nuclear accident is one in infinity, i.e., zero (PTI 2011a).

In fact, the former chairman of the Nuclear Power Corporation has stated that it is 'important' that 'the people (operating the nuclear plant) should be confident about safety' (Subramanian 2000). This suggests that the confident view of safety should not be just a public position—intended to assuage the concerns of the citizenry and the policymakers—but that this confidence should be deeply internalized.

DAE officials routinely exhort employees to be confident of the safety of their operations. An example of this occurred in the aftermath of the 1999 Tokaimura criticality accident in Japan, when newspapers quoted Gopalakrishnan as saying that because 'the degree of automation and cross-checks on safety in our older plants are very minimal . . . one cannot assert at all that an accident like the one which occurred in Japan will not happen in India' (*Tribune* 1999). This cautionary statement about the risks associated with the technology was too much for the DAE. Delivering the Founder's Day Speech at the annual high-profile special event held at the Bhabha Atomic Research Centre, the AEC chairman responded by saying, 'such a statement, made without any scientific basis, was a symptom of the technological diffidence in some persons who considered that as a nation, India was not capable of dealing with high technology . . . I do not think so. And there is no doubt that all of you, who have a spectacular record of achievements, do not think so' (Chidambaram 1999, 4). Elsewhere, when asked about Gopalakrishnan's warning by reporters, the AEC chairman categorically asserted that there 'is no possibility of any nuclear accident in the near or distant future in India. We have 150 safe years of nuclear reactors' (ToI 1999b).

The extent of the misguided belief that accidents are impossible was apparent in how leaders of the nuclear establishment denied the very nature of what happened at Fukushima. The chairman and managing director of NPCIL claimed: 'there is no nuclear accident or incident in Japan's Fukushima plants. It is a well planned emergency preparedness programme which the nuclear operators of the Tokyo Electric Power Company are carrying out to contain the residual heat after the plants had an automatic shutdown following a major earthquake' (PTI 2011d). The AEC chairman said, 'It was not a nuclear accident . . . immediately after the earthquake, the nuclear reactor shut down and nuclear chain reaction stopped' (PTI 2011a). Such utterances reveal the lengths to which the nuclear establishment goes to maintain its ideology of safety in the face of contrary evidence.

This sort of confidence permeates even the way the DAE characterizes its understanding of the world around it. The DAE's Reactor Safety Analysis Group had confidently declared in 1986: 'For coastal sites, flooding may be due to tropical cyclones, tsunamis, seiches and wind waves. In India, tsunamis and seiches do not occur. Hence, cyclones alone have been singled out for detailed study.' (RSAG 1986, 18). This assertion was belied by the December 2004 tsunami. Though that event has triggered some additional safety features at coastal reactors, it does not seem to have resulted in any introspection within the DAE about how other assumptions that underlie their analyses might prove wrong.

There is a close parallel between a) the way the DAE concluded that just because no tsunamis and seiches had hit India till 1986 that they 'do not occur' and b) the way it concluded that since no catastrophic accidents had taken place that its nuclear facilities were safe. The latter simply does not follow from the former. To put it baldly, just because they had not experienced any accident earlier, can one say that the Fukushima Daichi reactors and the Chernobyl reactor were safe before 11 March 2011 and 26 April 1986 respectively? The historical absence of catastrophic accidents cannot be cited as evidence of the absence of *risk* of catastrophic accidents. As in India, 'many nuclear safety related events occur year after year, all over the world, in all types of nuclear plants and in all reactor designs, and that there are very serious events that go either entirely unnoticed by the broader public or remain significantly under-evaluated when it comes to their potential risk' (Kastchiev et al. 2007, 95).[26] There have also been many accidents that did not escalate into major ones purely because of chance or the intervention of human operators rather than any technical safety feature. Such factors cannot be taken for granted.

For an organization operating hazardous technologies, being confident that its facilities are not likely to suffer accidents is not conducive to safety. One of the many paradoxes about safety is that 'if an organization is convinced that it has achieved a safe culture, it almost certainly has not' (Reason 2000, 4). The pertinence of that observation for the DAE has already been borne out in our examination of the DAE's operating practices.

In his *Wealth of Nations*, Adam Smith bitingly comments on the search for silver and gold mines: 'Such in reality is the absurd confidence which almost all men have in their own good fortune, that wherever there is the least probability of success, too great a share of it is apt to go to them of its own accord' (A. Smith 2001, 742). The statement can well apply, mutatis mutandis, to the DAE's view of its success in avoiding nuclear accidents.

Theoretical Perspectives

The DAE's general approach to nuclear safety is typical of how many engineers have thought about the subject, focusing mainly on specific aspects of technology and safety devices. This view has been severely criticized in the scholarly literature following the 1979 Three Mile Island accident, and academics studying safety and safety practitioners have largely moved beyond this framework. In particular, there is widespread recognition that two inherent properties of reactors—'interactive complexity' (subsystems interacting in unexpected ways) and 'tight coupling' (subsystems having rapid impact on each other)—are serious challenges to ensuring that operations at nuclear reactors are free of accidents (Perrow 1984). Interactive complexity pertains to the potential for hidden and unexpected interactions between different parts of the system, and tight coupling refers to the time-dependence of the system, the presence of strictly prescribed steps, and invariant sequences in operation that cannot be changed. These properties were identified by sociologist Charles Perrow, who also coined the term 'normal accidents' to explain that serious accidents appear to be the inevitable consequence of such technologies, regardless of the intent or skill of their designers or operators. Since then, Perrow's work has spurred an enormous range of analyses on a

26. Since the early 1980s, the Nuclear Energy Agency of the OECD and the IAEA have been operating a reporting system wherein about thirty countries report safety-related events every year. Over twenty-five years of operation it has gathered more than 3250 reports (NEA 2006, 9).

variety of different systems (Sagan 2004a). According to Perrow, these are inherent features of nuclear reactors and there is a limit to how far they can be reduced through engineering efforts.

From this perspective, there are significant limitations in the way the DAE thinks about safety. First, because of the complexity, the physical conditions that develop during the operation of a reactor may never be fully comprehended, and the understanding that designers or operators have of the reactor would always be partial. Such comprehension might be particularly challenging during accident situations, especially in real time. Second, because of the complexity and short time scales involved, operator actions may not seem erroneous until an ex post facto analysis has been performed. Third, because system components and phenomena could interact in unanticipated ways, it is not possible to predict all possible failure modes.[27] Whether the reactor is safe against unanticipated failures cannot be predicted in advance. An obvious corollary is that numerical estimates of probabilities of catastrophic accidents are uncertain.[28]

Finally, the Normal Accidents School has advanced a very important criticism of a standard method of enhancing safety used by engineers—redundancy. The problem is that redundancy often, if not always, adds to

27. Analogously, as US Secretary of Defense Donald Rumsfeld put it, 'there are known knowns; there are things we know we know. We also know there are known unknowns; that is to say, we know there are some things we do not know. But there are also unknown unknowns—the ones we don't know we don't know'.

28. This was stated somewhat differently by a committee appointed by the US Nuclear Regulatory Commission which pointed out that it is 'conceptually impossible to be complete in a mathematical sense in the construction of event-trees and fault-trees', the methodology used to calculate probabilities, and this 'inherent limitation means that any calculation using this methodology is always subject to revision and to doubt as to its completeness' (Lewis et al. 1978, ix). This begs the question: Why does the nuclear establishment come up with statements like 'the core damage frequency [for TAPS] is around 7.0E-05/Reactor year', that is, the probability of an accident that damages the core of the reactor is 70 parts per million during each year that the reactor operates (Koley et al. 2006)? Such assessments of probability of accidents are problematic and should not be taken seriously (Ramana 2011a). In addition to the problem of unaccounted-for accident pathways, if one looks at the details underlying such a calculation, one always comes up with unproven assumptions of various kinds. One reason for experts resorting to such detailed technical calculations that are, in practice, unverifiable and incredible, is that they serve to mystify discussions which should ideally involve people who are put at risk from these facilities, and who should be involved in decisions about locating, constructing and operating them.

interactive complexity and produces unanticipated problems (Sagan 1993, 2004b). Thus, systems that are added to increase safety might well end up undermining safety in hidden ways. The classic example of this occurred at the Fermi fast breeder reactor in Lagoona Beach, USA, where pieces of zirconium added to the 'core catcher'—a safety system that is supposed to prevent molten fuel from burning through the reactor vessel—broke off and blocked the entry of liquid sodium into a couple of fuel assemblies. These melted, causing the reactor to shut down (Fuller 1975).

There are also implications for safety from the institutional nature of the DAE. Large bureaucratic organizations exhibit a tendency to downplay the possibility of failures for fear of destroying their reputation and losing their budgets. This is reflected in their not recognizing all possible contingencies and not incorporating adequate safety measures (Rajaraman, Ramana, and Mian 2002). This sense of infallibility is particularly marked in institutions that are characterized by 'expertise' and 'discipline', and further compounded where national security is involved. Political scientist Scott Sagan, in an important and wide-ranging study of several decades of experience with nuclear weapon systems in the United States, points out that in 'total institutions', like a military command, the strong organizational control over members can 'encourage excessive loyalty and secrecy, disdain for outside expertise, and in some cases even cover-ups of safety problems, in order to protect the reputation of the institution' (Sagan 1993, 254). Many, if not all, of these characteristics apply to the DAE.

There is one obvious question that this constellation of hazards—physical and institutional—provokes about nuclear power around the globe: why are there not more accidents? One part of the answer is that, while the nature of the technology implies that there will be failures, there is nothing that determines the rate of failure as such. Therefore, the chances of an accident on any given day may be small but, sooner or later, there will be one. This parallels what happens in a lottery: 'the odds of any given person winning are extremely remote, but the likelihood that *someone* is going to win, sooner or later, is certain' (Chiles 2001, 286).

Another part of the answer has been advanced by the High Reliability Organization (HRO) School, a group of theorists at the University of California, Berkeley. While not dismissing the challenges posed by the structural features identified by Perrow, the HRO theorists tried to explain what allowed some organizations to operate such technological facilities with what they felt was 'an extraordinary level of safety and productive capacity' (La Porte 1996, 60). In other words, their task was to identify the factors that allowed the management of risky technologies with a relative degree of safety. Their search for such explanatory factors primarily focused on intelligent organizational design and good management practices, not technological ones. The HRO Group maintains, though, that they have only uncovered 'conditions that were *necessary* for relatively safe and productive management of technologies' but distance themselves from the suggestion or implication that 'these conditions were *sufficient*' (La Porte and Rochlin 1994, 225).

The ideal organizations of the HRO theorists have formal structures and clear and consistent goals that ensure reliable operations. The common ingredients that contribute towards the safe operation of hazardous technologies include: political elite and organization leaders placing a high priority on safety in design and operations; setting and maintaining safety standards and practices; ensuring a healthy relationship between management and workers; redundancy in technical operations and personnel management; allowing compensation for failures; and continuous organizational learning via systematic gleaning of feedback (Sagan 1993; La Porte 1996; Bigley and Roberts 2001).

Does the DAE meet with these high standards? The history of accidents and operating practices at the DAE's facilities outlined here suggests otherwise.[29] In this chapter, we have illustrated it through various examples, including lack of redundancy in safety (KARP),[30] repeated occurrences of avoidable accidents in its facilities, poor organizational learning from previous failures (Narora), questionable construction and manufacturing quality (Kaiga), and the system elite not being sufficiently interested in safety and not being attentive to concerns raised by lower-level staff (KARP). These factors are compounded by an absence of peer-review mechanisms and independent, external oversight.

The lack of high standards of safety is the reason that one cannot take comfort in the fact that, so far, there have not been any catastrophic accidents in India. At the same time, this does not imply that a catastrophic accident will occur tomorrow or in the next month. As safety theorist James Reason argues, 'Even the most vulnerable systems can evade disaster, at least for a time. Chance does not take sides. It afflicts the deserving and preserves the

29. This is elaborated by Kumar and Ramana (Forthcoming).

30. First, there was no redundancy that would prevent mixing of high-level and low-level waste, or protect workers, in case of valve failure. Valves are known to fail despite measures to improve their reliability. Second, the plant had been operating for five years without any monitors in the region to detect radiation levels. There was no procedure that would alert workers to high radiation levels before they received large doses.

unworthy' (Reason 2000). But, if such an accident were to occur, ex post facto analyses are likely to come to the conclusion that, as with Bhopal and Chernobyl, it was an accident waiting to happen.

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'One in infinity': failing to learn from accidents and implications for nuclear safety in India

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'One in infinity': failing to learn from accidents and implications for nuclear safety in India

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Safe operation of nuclear power facilities requires a culture of learning, but Indian nuclear authorities appear to continuously fail to learn the lessons of accidents including at facilities they operate. This paper examines how nuclear authorities in India responded to the Fukushima accidents and a previous accident at one of India's nuclear power plants, and infers what they seem to have learned from them. By evaluating this experience in light of a wide body of research on factors promoting reliability and safety in organizations managing complex and hazardous systems, it seeks to draw lessons about the prospects for nuclear safety in India.

Keywords: India; nuclear power; learning; nuclear safety; accidents

Introduction

On 14 March 2011, following the commencement of multiple accidents at Japan's Fukushima Daiichi reactors, India's Prime Minister Manmohan Singh offered the assurance that the Department of Atomic Energy (DAE) would review all safety systems at the country's nuclear plants, 'particularly with a view to ensuring that they would be able to withstand the impact of large natural disasters such as tsunamis and earthquakes' (Timmons and Bajaj 2011). But, in the same statement to the houses of Parliament, before any such safety review had been initiated, the Prime Minister also stressed that Indian reactors were safe. As evidence, he pointed to the fact that Indian reactors had survived the Bhuj earthquake in January 2001 and the December 2004 Tsunami (Balaji 2011). From all public accounts, no one in Parliament at the time challenged this line of argument.[1]

Manmohan Singh's assertion that Indian reactors were safe was not unique in India or elsewhere.[2] As we detail in this paper, over the next weeks and months, various high-level nuclear officials dismissed the possibility that anything like the Fukushima accidents could occur in India. Instead, there were claims such as in November 2011, when the Chairman of the Atomic Energy Commission (AEC) stated that the probability of an accident was 'one in infinity' (PTI 2011b).

In parallel, the government prepared to start operations at the Koodankulam Nuclear Power Plant (KKNPP) in southern India and constructing reactors at the Jait-

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apur site in western India. Local residents have opposed both projects, in part due to safety concerns.[3] In the case of KKNPP, some of the safety measures recommended by the regulatory body had not been implemented when fuel loading commenced.

Many safety scholars have emphasized the importance of effective learning within organizations, as part of a wide range of practices that are essential to maintaining safe and reliable operations. We begin with a brief discussion of previous work of researchers studying organizations managing complex high hazard technologies. This is followed by an examination of the Indian nuclear authorities' interpretations of what happened in Fukushima. The paper then analyzes the work of the two main panels set up in India in the aftermath of these accidents, whose objectives were to examine their lessons for nuclear safety in India. We discuss the failure to implement these recommendations for the Koodankulam and Jaitapur projects. This is followed by the case study of a major fire in the Narora reactor in 1993, which also examines the lessons the DAE learned before and following the accident. We conclude with a synthesis of the problems within the DAE when it comes to learning from experience.

Organizational perspectives on safety

Our approach to studying safety in DAE's nuclear facilities is grounded by the fact that organizational culture has external manifestations that can be apparent to the outsider. This is indeed the case of the DAE, as will be described. In the context of nuclear hazards, much has been written about the importance of safety culture, especially in the aftermath of the 1986 Chernobyl accident (Pidgeon 1991). A prominent approach to examining safety via the study of organizational practices has been the work of the High Reliability Organization (HRO) School. HRO researchers tried to explain what allowed some organizations to operate hazardous technologies with what was believed to be 'an extraordinary level of safety and productive capacity' (LaPorte 1996). HRO researchers usually describe 'conditions that were necessary for relatively safe and productive management of technologies' but do not suggest that 'these conditions were sufficient' (LaPorte and Rochlin 1994).

The common features of organizations with a record of relatively safe operation, as identified by HRO researchers, are: the importance of political elites and organizational leaders placing a high priority on safety in design and operations; setting and maintaining safety standards and practices; sophisticated learning from failures; and ensuring a healthy relationship between management and workers (Bigley and Roberts 2001; LaPorte 1996; LaPorte and Consolini 1991; Roberts 1989; Roberts 1990; Sagan 1993).

Several theorists emphasize the importance of learning from failures. Weick, Sutcliffe, and Obstfeld observe that 'the best HROs provide the cognitive infrastructure that enables simultaneous adaptive learning and reliable performance' and 'in non-HROs … people tend to focus on success rather than failure' (Weick, Sutcliffe, and Obstfeld 1999, 81). How organizations interpret events is also key to safety and theorists point out the importance of exhibiting a 'reluctance to simplify interpretations' for an organization's health (Vogus and Welbourne 2003).

Other safety theorists identify similar factors. James Reason calls for an organizational culture where managers are knowledgeable and pay attention to safety in the organization as a whole, showing the ability to learn appropriate lessons from the safety information system and act upon those, and where the relationship between managers and workers encourages the reporting of errors (Reason 1997, 195–196). Further, an organization with good safety culture 'must possess a learning culture – the willingness and the competence to draw the right conclusions from its safety information system' (Reason 1997, 196).

Overview of organizational structure

Before we go on to discuss reactions to the Fukushima accidents, a brief description of the Indian nuclear program and its organization might provide context. India's nuclear program has had a long history and the country's leadership has, almost without exception, supported the development of large nuclear power capacity. Over the decades, the DAE has revealed a series of ambitious goals, but these have never been achieved. Nevertheless, the ability of the DAE to promise such ambitious goals has been one of its key sources of political power (Ramana 2012), which in turn has enabled the organization to evade accountability on a variety of aspects including safety of its operations.

As of March 2013, the installed nuclear capacity in the country is 4780 MW, about 2.26 percent of the country's total electricity generation capacity of 210,951 MW. Most of these are 220 MWe (mega watt electric) Pressurized Heavy Water Reactors (PHWRs) based on the Canadian CANDU design. Seven reactors with a combined capacity of 5300 MW are currently being constructed. The largest aspect of the planned expansion consists of two Russian 1000 MW VVER-1000 reactors being constructed in Koodankulam (Agrawal, Chauhan, and Mishra 2006).

The family of nuclear organizations in India is headed by the AEC. The AEC's role is to formulate policies and programs, while the actual execution of these policies is carried out by the DAE. The DAE has in turn set up a number of associated or subsidiary organizations. Most central of these for our discussion are the Nuclear Power Corporation of India Limited (NPCIL), which is responsible for designing, constructing, and operating nuclear power plants and the Atomic Energy Regulatory Board (AERB), which is in charge of regulating safety for all nuclear facilities in the country.[4] Since its inception, the AERB has reported to the AEC, whose Chairman has always also played the role of the Secretary of the DAE.[5] Further, the AERB receives its funding from the DAE (CAG 2012, 13). Following the Fukushima accidents, there have been widespread expressions of concern about the safety of Indian nuclear facilities, and the AERB's lack of independence.[6] As a result, the Indian government has proposed changing this arrangement.

Nuclear authorities' reactions to Fukushima

The nuclear establishment in India, namely the DAE and its subsidiary organizations, chiefly the NPCIL and the AERB, responded to Fukushima both immediately and in the following months in two main ways. On the one hand, they followed the Prime Minister's directives and set up various review committees.[7] But, in parallel, there were a series of public statements about the Fukushima accidents and its implications, and these are discussed below.

One set of statements had the effect of denying or trivializing what happened at Fukushima. The Chairman and Managing Director of NPCIL set the tone when he claimed:

> there is no nuclear accident or incident in Japan's Fukushima plants. It is a well planned emergency preparedness programme which the nuclear operators of the Tokyo Electric Power Company are carrying out to contain the residual heat after the plants had an automatic shutdown following a major earthquake (PTI 2011a).

This reference to residual heat, produced by the radioactive elements generated during the fission process, did not do justice to the gravity of the situation – as was clear when it melted the cores of the Fukushima reactors. As physicist Robert Socolow described this phenomenon in the *Bulletin of the Atomic Scientists*, the residual heat is 'the fire that you can't put out, the generation of heat from fission fragments now and weeks from now and months from now, heat that must be removed' (Socolow 2011). This problem did not find mention in the statement some months later by the Secretary of the DAE, who disingenuously asserted, 'It was not a nuclear accident … immediately after the earthquake, the nuclear reactor shut down and nuclear chain reaction stopped' (PTI 2011b). In other words, the DAE equated nuclear accidents with ones where a chain reaction persists, thereby ignoring large classes of severe accidents that could occur even after the reactor is shut down.

Another set of arguments were aimed at making the accidents seem inconsequential for human health and the environment. On 10 November 2011, the DAE Secretary claimed that the 'spread of radiation was not as high as it was projected,' and went on to assert that the total casualty due to the nuclear accident was zero (PTI 2011b). Some months later, on 26 February 2012, he claimed, inaccurately, that among 'the three known such incidents – Three Mile Island, Chernobyl and Fukushima – the casualty due to nuclear plant malfunction is 50–60 in Chernobyl and zero in the other two incidents' (TNN 2012).

It is worth clarifying here that the term incident has a technically precise meaning and is only used for any nuclear event that ranks at level 3 or below on the International Nuclear Event Scale (INES). Levels 4–7 are, according to the scale, accidents with local consequences, wider consequences, a serious accident, and a major accident, respectively (IAEA 2012).[8] Of the three accidents, Three Mile Island was ranked at level 5 and Chernobyl and Fukushima were ranked at level 7. Thus, it would be negligent to describe those as incidents – unless it were the case that the goal was to avoid drawing attention to these events.[9]

Nor are the casualty figures accurate. It is established that one of the primary impacts of exposure to radiation, the incidence of cancer, occurs many years after the exposure.[10] Therefore, while no one is likely to have died of cancer so far, the Fukushima accidents will likely lead to something of the order of a thousand cancers globally over the next few decades (Beyea, Lyman, and Von Hippel 2013; Hoeve and Jacobson 2012; von Hippel 2011).[11] Further, hundreds of square kilometers will remain unusable for agriculture for decades because of the contamination of Cesium-137, which has a radioactive half-life of 30 years. In the case of Chernobyl, there has been a longstanding debate over the number of casualties. However, even the International Atomic Energy Agency (IAEA) and the World Health Organization concluded in 2005 that in the areas that are most contaminated alone, there are projected to be approximately 4000 fatal cancers from the accident (Chernobyl Forum 2006, 106). Modelling by independent epidemiologists leads to the expectation that by 2065, there would be about 16,000 cases of thyroid cancer and 25,000 cases of other cancers just in Europe (Cardis and Krewski 2006).

Perhaps, the only accident of these three on which there is widespread consensus is the Three Mile Island accident, and most experts believe that it resulted in no fatalities; but even in that case, there is some evidence of cancer incidence as a result of the radioactivity released during the accident (Wing 2003; Wing et al. 1997).

A further assertion by the DAE following the Fukushima accident was that even if such an accident were to occur, the DAE and its constituent organizations could manage the situation effectively. On 30 September 2011, for example, the Secretary of the DAE claimed: 'We are prepared to handle an event like Fukushima' (HT Correspondent 2011). Having seen the difficulty in dealing with Fukushima in a country that has had a history of dealing with major natural catastrophes and with more developed physical and organizational capital, these claims are dubious.[12] Furthermore, just the previous year, the issue of dealing with a nuclear accident had been raised in the Indian Parliament, as it dealt with a bill on liability for nuclear accidents. At that time, the Secretary of the Ministry of Health and Family Welfare had testified to the Parliamentary Standing Committee on Science & Technology that the Ministry was 'nowhere to meet an eventuality that may arise out of nuclear and radiological emergencies' (Parliamentary Standing Committee on Science & Technology 2010). Yet again, we see the attempt by the DAE and NPCIL to bend reality while making the Fukushima accidents seem inconsequential.

But, the main thrust of the statements on Fukushima by DAE and NPCIL officials has been to assert that the accident is irrelevant, because no nuclear accident will ever occur in India.[13] On more than one occasion, the DAE's Secretary has made assertions to the effect that the probability of a nuclear accident in India is zero, that Indian nuclear plants are one hundred percent safe. On 20 March 2011, an interview of the Secretary of the DAE by the science journalist Pallava Bagla for the popular television channel NDTV involved this dramatic exchange (Bagla 2011):

> NDTV: Are Indian reactors safe?
>
> Srikumar Banerjee: One hundred per cent.
>
> NDTV: Can you keep your hand on your heart and say Indian public need not fear Indian reactors?
>
> Srikumar Banerjee: Yes I am keeping my hand on my heart to say that Indian public need not fear Indian reactors and if there is any chance of any accident then we will be the first one to say, close it.

These related set of openly professed beliefs do not appear to be merely a case of maintaining a public position. In fact, the former chairman of the Nuclear Power Corporation has stated that it is 'important' that 'the people (operating the nuclear plant) should be confident about safety' (Subramanian 2000). This appears to be a directive that the confident view of safety be internalized and in line with routine exhortations by DAE officials to employees to be confident of the safety of their operations. An example of this is when, in the aftermath of the Tokaimura criticality accident in Japan, a former AERB chairman had warned that nuclear accidents could not be ruled out in India (Tribune 1999). Delivering a speech that year at the Bhabha Atomic Research Centre, the DAE head's response was that:

> such a statement, made without any scientific basis, was a symptom of the technological diffidence in some persons who considered that as a nation, India was not capable of dealing with high technology … I do not think so. And there is no doubt that all of you, who have a spectacular record of achievements, do not think so. (Chidambaram 1999)

Because there is little openness in the activities of the nuclear establishment, and workers in various nuclear facilities mostly live in close-knit communities, it seems likely that elite views on nuclear risk have prevailed over much of the organization.

Safety recommendations and implementation

The more formal response to Fukushima by NPCIL was the appointment of six task forces 'to review consequences of occurrences' of situations similar to Fukushima and assess the safety of Indian nuclear power plants 'assuming non availability of motive power and design water supply routes' (NPC 2011b). These task forces were to examine (1) the Boiling Water Reactors at Tarapur; (2) the first two PHWRs at Rawatbhata (RAPS 1&2); (3) the third and fourth PHWRs at Kalpakkam (MAPS 1&2); (4) the subsequently standardized PHWRs; (5) the Pressurized Water Reactors (Russian VVERs) at Koodankulam; and (6) the 700 MWe PHWRs under construction at Kakrapar and Rawatbhata. These task forces submitted their interim report in May 2011, which is available on the NPCIL website (NPC 2011b). However, there does not seem to be a final report in the public domain.

Likewise, the AERB constituted a committee 'to review the safety of Indian NPPs [nuclear power plants] against external events of natural origin.' In its work, this committee focused on what it called 'Beyond Design Basis Events (BDBE) of natural origin' and prolonged station blackout, i.e. the nuclear power plant being cut off from all external sources of electricity for a long period. The AERB's committee 'considered the assessment made by the NPCIL task force(s) in its report' (AERB 2012), and appears to have used it in developing its recommendations and final report.

It is beyond the scope of this paper to cover the content of these reports comprehensively. Though they are somewhat different in their emphasis, the predominant message from both reports is that the safety of nuclear power plants in India is assured. The NPCIL task forces' interim report, for example, states that their review of safety and earlier reevaluations 'indicate that *adequate provisions exist* at Indian nuclear power plants to handle station blackout situation and *maintaining continuous cooling* of reactor core for decay heat removal' [our emphases] and that the:

> design of Indian nuclear power plants take into account a wide range of postulated initiating events from low probability high consequences events to high probability low consequences events to ensure that radiological consequences of postulated events are well below acceptable levels for public safety. (NPC 2011b)[14]

Likewise, the AERB Committee claims that the 'magnitude of postulated design basis natural events and the related requirements for siting and design of NPPs, as specified in AERB safety regulations, are found to be *appropriate and sufficiently conservative*' (AERB 2011, 5) [our emphasis]. It even goes on to suggesting that in India, unlike 'some other NPP operating countries', which are left unnamed, 'all key operating personnel as also station management personnel in Indian NPPs are graduate engineers who are formally authorized to carry out their respective duties only after successful completion of intense training' and are therefore 'better placed to handle off-normal situations in the plant compared to their counterparts in several other countries' (AERB 2011, 4).

However, the sub-committees that AERB had set up to look at specific power plants did uncover several concerns, especially in the case of the Tarapur Boiling Water Reactors that are of a design very similar to the Fukushima Daiichi I reactor. Therefore, AERB went on to laying down a set of recommendations for action at various nuclear power plants.

Implementing recommendations in Koodankulam

One test of how serious NPCIL and AERB were about undertaking safety measures to avoid a Fukushima-like situation, and enhance safety in general, was to come when the KKNPP had to be loaded with fuel and the reactor made critical. Accordingly, we briefly list the recommendations pertaining to just this facility (AERB 2011, 109–110).

Based in part on the experience at Fukushima, where the reactor cores melted down because of a lack of cooling, the AERB committee's focus was on ensuring that there is adequate water for cooling in the event of an accident. Hence, it recommended that there should be backup provisions from alternate sources for various reactor components that need water, such as steam generators, the reactor coolant system, and spent fuel pools. It also recommended that an emergency water storage facility be constructed so as to withstand earthquakes and that this facility should be capable of storing enough water for decay heat removal from the core for a period of at least one week. Another key concern was that absence of electricity might hamper various essential functions and hence the committee recommended the acquisition of 'mobile self powered pumping equipment for emergency use,' 'additional backup power supply sources for performing essential safety functions,' and setting up a facility 'for monitoring safety parameters using portable power packs.' AERB assured that 'NPCIL's response would be reviewed … before initial fuel loading.'

NPCIL's task force report also suggests that it would implement any recommendations. The task force report maintained that in response to earlier accidents, the 'lessons learnt requiring corrective measures *were implemented* not only for affected sites but also at other stations and projects under construction *as an established practice of enhancing safety levels* for all plant' [our emphasis].

However, as the time approached for loading the first unit of KKNPP with fuel, NPCIL had not yet implemented many of the AERB review committee's recommendations. In contrast to what a diligent regulator would have done, AERB allowed NPCIL to go ahead with fueling the reactor. In contrast to the spirit of the report produced by its own review committee, AERB now emphasized that its review had concluded that KKNPP had 'adequate safety measures against external events' and that in order to 'further enhance safety, as an abundant measure, some additional safety enhancements proposed by NPCIL were reviewed in depth and accepted for implementation in a phased manner' (Gopalakrishnan 2012).

In the AERB's revised view, the 'additional arrangements … are in the nature of a backup to backup' and that 'it was never the intention of AERB that the recommendations … have to be mandatorily put in place/implemented before permitting initial fuel loading and continuous commissioning of the plant thereafter' (AERB 2012). While the AERB committee began its report by stating that one of its two main foci was on BDBE, now AERB was content with the fact that the VVER-1000 'is one of the advanced reactor in the world, having built in design features for Beyond Design Based Accidents management.'

The context for this decision to push for commissioning the reactor before all the recommended safety measures had been implemented was the intense protests by local citizens (Kaur 2012; Ramdas and Ramdas 2011). Catalyzed by the Fukushima accidents, prolonged protests were conducted by people living near all the different proposed nuclear sites across India, in addition to some of the currently operating nuclear power plants. Such opposition is not new; there has been significant opposition to every new nuclear reactor and uranium-processing facility that has been planned since the 1980s (Ramana 2012).

The Koodankulam protests, however, were more sustained and widespread than any other protests that happened before in India. The government's response was twofold; it started a violent police crackdown on the protesters (Kaur 2012), and it tried its best to accelerate commissioning of the reactors.

In parallel, a grassroots activist group initiated a public interest petition that was to eventually reach the Supreme Court of India, which held several hearings (IANS 2012; PTI 2012a; Vaidyanathan 2012). In a somewhat surprising statement, the Court made it clear that it could stop the commissioning of the KKNPP if it found that the mandatory safety requirements for it have not been put in place (PTI 2012b).

It was in its affidavit to the Supreme Court during the public interest case that the AERB declared that the plant was safe even though its recommendations had not been implemented. Though it is all but impossible to provide evidence, it appears possible that AERB must have been under pressure to accelerate commissioning of KKNPP and not insist on the construction of a seismically qualified emergency water storage facility and acquisition of various additional devices to deal with a loss of power as a precondition for regulatory clearance. The former Chairperson of the AERB, and now a vocal critic of the state of nuclear safety in the country, has termed its actions 'a total volte face … and contrary to the spirit and recommendations of the AERB Post-Fukushima Safety Evaluation Committee' (Gopalakrishnan 2012).

One cause of the Fukushima accident was the Japanese regulatory system's inability to force nuclear utilities, including TEPCO, to incorporate its safety recommendations (Funabashi and Kitazawa 2012).[15] The volte face of AERB, whatever the reasons behind it, represents a weakness in the nuclear regulatory system in India. The reasoning that AERB offered for not insisting that NPCIL undertakes all of the safety measures recommended by the AERB's committee has parallels with the type of overconfidence displayed by the Japanese Nuclear Safety Commission, which 'did not include provisions for an extended loss of power in its accident-management policy' because they felt that it was 'reasonable to expect that transmission lines will be restored or emergency power systems repaired quickly' (Funabashi and Kitazawa 2012, 12). In parallel with this reasoning, AERB's affidavit in the Supreme Court insisted that the:

> existing inventory of water available … in storage tanks hydro accumulators, steam generators and spent fuel pool can cater to the safety requirements related to reactor

> cooling and spent fuel stored for a limited duration even assuming non-availability of external water' source. This limited duration is however adequate to arrange alternate source of water from outside the site.

Knowledge of seismicity

There was similarly a disconnect between recommendations and implementation in the case of seismic safety, a reasonable concern because of the triggering of the Fukushima accident by an earthquake whose magnitude was much larger than Japanese decision-makers had taken into account in their planning and design (Nöggerath, Geller, and Gusiakov 2011). The AERB committee:

> recommended that treatment of uncertainties in data and certain computational procedures should be improved to obtain an even higher degree of conservatism in the assessment of the magnitude of design basis external events of natural origin. The revised guidelines so generated may be considered for inclusion in AERB regulations. (AERB 2011, 7)

It specifically highlighted 'limitations of current methodologies because of lack of sufficient and relevant earthquake data and other uncertainties concerning site tectonics.'

One particular site where seismicity was a significant issue was Jaitapur in western India, which has been selected to host up to six 1600 MW reactors. In the last half century, major earthquakes have occurred at Latur and Koyna that are at distances of 100–400 km from Jaitapur. The main limitation with current methodologies for predicting the likelihood of earthquakes in Jaitapur was precisely what the AERB committee had highlighted – a lack of sufficient data. Independent seismologists have pointed out that the:

> historical seismic record near Jaitapur extends reliably back for only 200 years, with scant additional data prior to 1800. Due to the long interregnum between earthquakes in continental India (millennia), historical seismic data from a few hundred years cannot be taken as a guide to future seismic hazard. (Bilham and Gaur 2011, 1280)[16]

Therefore, they concluded that an earthquake of magnitude 6.5 or more, 'although unlikely … could occur within the lifetime of the nuclear power plant' in close vicinity of Jaitapur (Bilham and Gaur 2011, 1275). In a subsequent paper, they emphasized that:

> Jaitapur lies in a region where plate tectonic stresses are locally close to critical failure, and where minor perturbations in stress can trigger earthquakes. Geologically, the Jaitapur region meets many of the criteria known to be conducive to intra-plate seismicity. Tectonically, the Jaitapur region is precisely in the same state of seismic quiescence and historical ignorance as the regions of Latur or Koyna were, prior to the damaging earthquakes [in 1993 and 1967] for which they are now famous. (Gaur and Bilham 2012, 1276)

The response to the first article from NPCIL came in the form of a press release entitled 'Seismicity Considerations for Jaitapur NPP' on its website, where it argued that it had 'approached seismicity in a comprehensive way' by 'specifying and building reactors on conservative basis with ample design margins above what is considered possible risk' (Nagaich 2012, 2). In another brochure on its website, NPCIL was confident enough about the robustness of reactor designs as to declare, 'Nuclear power plants are the safest places to be during an earthquake' (NPC 2011a).

This overconfidence is manifest in other respects. On 20 March 2011, the Chairman of NPCIL declared, 'We have got total knowledge and design of the seismic activities. Worst seismic events and tsunami have been taken into consideration in our designs' (TNN 2011). A previous case of overconfidence was the assertion by the DAE's Reactor Safety Analysis Group in 1986: 'For coastal sites, flooding may be due to tropical cyclones, tsunamis, seiches and wind waves. In India, tsunamis and seiches do not occur. Hence cyclones alone have been singled out for detailed study' (RSAG 1986, 18). The DAE appears to have stopped making this assertion after the December 2004 tsunami, which also hit South India.

Such overconfidence, especially about factors as uncertain as earthquakes and the response of reactors to them, does not indicate a good safety culture. Geological knowledge is constantly evolving and geologists do not always know of the existence of various seismic faults and 'unknown faults that cause damaging earthquakes are fairly common'; for example, the magnitude 6.3 earthquake that struck Christchurch, New Zealand, in 2011 occurred on an unknown and unexposed fault (Macfarlane 2011). Although Japan's seismicity and fault structures have been very intensively studied, the country's new Nuclear Regulatory Authority has recently discovered previously unknown active earthquake faults under nuclear power plant sites, for example at the Tsuruga reactor (Inajima and Okada 2013).

Fire at Narora atomic power station

We now discuss a historical example where NPCIL failed to effectively learn from an accident elsewhere – the fire in Brown's Ferry plant in the United States on 22 March 1975. The fire lasted nearly seven hours, disabled most of the emergency core cooling systems of the two co-located reactor units, and the reactors came close to melting down. The fire led to significant changes mandated at all US nuclear plants, as prescribed in the US Code of Federal Regulations Part 50 (Ramsey and Modarres 1998, 106). Some of these regulations entailed changes to physical and electrical systems, including built-in redundancies designed to prevent fires. Other countries adopted similar measures.

The Narora station has two 220 MW (megawatt) heavy water reactors that started operating in 1991 and 1992. By this time, the recommended practice in the nuclear design community was that electricity supply should be encased in separate and fire-resistant ducts. This was not done at Narora, which underwent a major fire in March 1993.

The accident, involving the first of the two reactors, started when two blades of the turbine broke off due to fatigue, sliced through 16 other blades, and caused it to vibrate significantly.[17] These vibrations led cooling pipes carrying hydrogen gas to break, and the released hydrogen caught fire. This was accompanied by leakage of oil from the turbine system. Fire spread to the oil and the entire turbine building. Cables supplying power to the secondary cooling systems also caught fire, leaving these cooling systems inoperable. All the four different cabling systems in the plant lost supply as part of the general blackout in the plant. Smoke entering the control room made it uninhabitable. The accident was categorized as level 3 on the INES scale.

Operators had to manually actuate the reactor's shutdown systems (Koley et al. 2006). They also climbed onto the top of the reactor building aided by battery-operated portable lighting and manually opened valves that released liquid boron into the core to slow down the reaction and avoid recriticality. Operators intervened in ensuring safety in yet another way: by starting up diesel-driven fire pumps and using water meant for fire suppression to circulate water in secondary pipes of reactor's steam generators, to ensure heat removal from the primary coolant system (NEI 1993).

The organization ignored a number of warning signs prior to the accident, and failed to learn from its own experience and that of others. The immediate cause of the fire was, of course, the turbines. In 1989, General Electric (GE) Company conveyed information to the turbine manufacturer in India about a flaw in the design that led to cracks in similar turbines around the world. It recommended design modifications and the Indian manufacturer as a result prepared detailed drawings for NPCIL, the plant's operator. GE and the turbine manufacturer also recommended to the NPCIL that the blade designs be replaced to avoid an accident. However, NPCIL did not act on this advice until after the accident (Gopalakrishnan 1999).

We do not know why the NPCIL failed to act. Was the leadership made aware of the recommendations but chose not to act because of other priorities such as urgency to generate electricity? Or were they unaware of these recommendations by GE and the turbine manufacturer? Both possibilities, involving either inaction in light of knowledge critical to safety or failure of information flow to decision-makers, contradict recommendations of safety researchers for improving safety in high hazard organizations. As James Reason writes: in an organization with good safety culture, 'those who manage and operate the system have current knowledge about the human, technical, organizational and environmental factors that determine the safety of the system as a whole' (Reason 1997, 195).

Coming to the underlying reason for why the fire led to an accident rated at level 3 by causing a station blackout, it was not for a lack of knowledge. The DAE was well aware of the hazards of fires and the need for safe practices to avoid fires. The probabilistic safety assessment for Narora performed in 1989 by DAE analysts, including Anil Kakodkar who was to become the head of the DAE in 2000, stated that common cause failures involving fires of excessive heating would be reduced if 'physical diversity and fire barriers are provided' (Babar et al. 1989, 109). Following the Narora accident, the DAE acknowledged in a paper that after 'the Browns Ferry experience in the United States, fire risk evaluation and hazard analysis has been considered as an essential component of safety evaluation and assurance' (Vinod et al. 2008, 2359). There is no information in the public domain about why the DAE, despite this widespread understanding about fire hazards, adopted the design it did for the Narora reactors, but it is clear that the DAE failed to learn the lesson from prior accidents and warnings.

Nor was Brown's Ferry the only reason to be concerned about fires. There had been at least three fires at the Rajasthan reactors before the Narora fire (Ramana and Kumar 2010).[18] Turbine blade failures had occurred repeatedly at Indian reactors prior to the Narora accident. Oil leak incidents were also common in the turbine generator systems of many reactors. Moreover, the factors that combined to produce the Narora accident in 1993 were not only endemic in India's nuclear plants, but also combined elsewhere in a manner that prefigured the Narora accident. In 1989, a reactor in Spain experienced turbine vibrations, which led oil to leak and hydrogen coolant to escape. The hydrogen burned violently and the oil caught fire. Fire spread through cabling and disabled operation of the emergency cooling and heat exchange pumps in the reactor (Ramsey and Modarres 1998, 325).

The Narora accident indicates a widespread failure within the DAE to generalize from these failures in India and abroad, and act on safety-critical information. Failure to effectively learn, and have this learning reflected in its actions, would make it more difficult for it to prevent future accidents. Asked by an interviewer about recurring turbine blade failures at nuclear reactors, the DAE Secretary side-stepped the issue by suggesting 'this kind of failure at Narora has happened for the first time … two blades failing' and then offering the non sequitur, 'you must remember that as far as nuclear reactor is concerned, there was no problem at Narora. The reactor worked perfectly according to design' (Chidambaram 1993). While the DAE Secretary may have been trying to mystify the interviewer by noting the difference between the turbine and the reactor, there is little doubt that the DAE had ignored early warnings and set the stage for the Narora failure.

Organizational learning from accidents and overconfidence

The origins and antecedents of the Narora accident, and its response following the accident, raise doubts about the DAE's attitudes towards safety and its effectiveness at ensuring that learning occurs within the organization. Its interpretations of failures do not seem to contribute to learning, either. As mentioned, after the Narora accident, the DAE has tried to interpret the events in a positive light, including making the claim that the accident illustrated the inherent safety of PHWRs (Wagh and Singh 2001). This claim is untenable because it took repeated intervention by the operators to keep the accident contained. First, operators manually actuated the primary shutdown system 39 s into the accident. Then, they initiated cooling down of the reactor by opening valves that discharged steam from the secondary system (Bajaj and Gore 2006). Operators also added boron to keep the reactor subcritical.[19] They also fed the secondary loop with water from the fire protection system (Koley et al. 2006).

Because of the need for human intervention, which is eventually needed to make continued heat removal by the secondary coolant possible in the event of a station blackout (Hajela, Datta, and Bajaj 1992; Nagdaune, Vijayan, and Venkat Raj 1992), the claim of inherent safety is not valid. More generally, DAE's interpretation of the accident sequence takes the operator actions for granted. These interventions are assumed to be a normal part of the task sequence, but these do not fall under the definition of inherent safety.[20] The DAE's interpretation, therefore, neglects to consider what could have occurred in Narora if operators had not intervened rapidly. This is another example of its failure to generalize lessons from the accident about what might go wrong in nuclear reactors.

In contrast, where operator actions are the proximate cause, as in the accident at the Kalpakkam Atomic Reprocessing Plant (KARP) where workers were subjected to radioactive exposures after a valve failure in 2003 (Anand 2003; Ramana and Kumar 2010), they are highlighted by the DAE in its efforts to deflect responsibility. This accident at KARP also suggests how the absence of openness and responsibility within the DAE, and failure to create an environment 'in which all individuals regardless of rank feel responsible for every detail of plant operation they can observe, and in which they feel free to point out their observations without fear of adverse consequences to themselves' (Rochlin and Von Meier 1994), can inhibit learning about safety-relevant information (Ramana and Kumar 2010).

In its interpretation of the accident at Narora, the DAE's attention is from the outset focused away from the conditioning set of failures that led to the accident in the first place. As mentioned earlier, these failures were common in the DAE's reactors and, in the case of Narora, combined into an accident that was rated level 3 on the INES scale. The lessons of previous fire accidents, and the specific warnings about the turbine blade design, had also not been acted upon.

Similar warnings had been ignored in the case of the Kakrapar Atomic Power Station (KAPS) where, in June 1994, heavy rains in South Gujarat led to heavy flooding in the turbine building and also made the control room inaccessible. As a result of floodwater entry, equipment in the turbine building, including water pumps used to cool the reactor core, became submerged. Floodwater also entered the solid waste-management facility, and carried away canisters of radioactive waste into the open (Gadekar 1994a, 1994b, 1994c). It is not known how many were swept away or released their contents into the water. The main cause of the Kakrapar flooding was the low elevation of the site. This had been recognized previously and the site had been recognized to be at risk of flooding and corrective measures were reportedly taken (Srinivasan 2002, 139). These were not effective when the flooding occurred in 1994. Furthermore, despite similar flooding previously in the reactors in Rajasthan, the lessons had not been learned and according to the former head of the AERB, 'sealing arrangements were not provided to prevent water ingress through cable trenches and valve pits. Similar flooding had occurred twice at RAPS in 1976 and 1982, owing to the very same construction errors as at KAPS' (Gopalakrishnan 1999). Neither had measures been taken to protect the waste-management facility.

This is in contrast to what deliberately occurs within organizations with good safety culture.

> Since catastrophic failures are rare events, collectively mindful organizations work hard to extract the most value from what little data they have … They work on the assumption that what might seem to be an isolated failure is likely to come from the confluence of many 'upstream' causal chains. (Reason 2000, 10)

As illustrated through their reactions to Fukushima and Narora, the DAE's organizational elites also exhibit a tendency to trivialize accidents and interpret even major accidents as inconsequential failures. The interpretation of minor accidents is worse. After a leak of heavy water at the Madras-2 reactor in 1999, the Station Director claimed that it 'did not involve an unusual situation' and that the amount of heavy water that escaped 'was only an insignificant quantity'; the Secretary of AERB stated, 'the safety of the reactor was ensured' (Subramanian 1999). This is unlike how HROs operate. Weick et al. observe that while 'most organizations tend to localize failure, effective HROs tend to generalize it' (Weick, Sutcliffe, and Obstfeld 1999).

The Indian nuclear establishment illustrates one of the many paradoxes about safety: 'if an organization is convinced that it has achieved a safe culture, it almost certainly has not' (Reason 2000, 4). As the examples discussed earlier show, nuclear officials frequently make claims about how safe its facilities and operating practices are.

Complacency and discounting of risks have been observed to be one of the root causes of many accidents (Leveson 2011). Scott Sagan identifies overconfidence within the US nuclear weapon complex as a 'serious problem' (Sagan 1993, 10). The Rogovin Commission appointed by the US Nuclear Regulatory Commission to investigate the accident at Three Mile Island found that:

> an attitude of complacency pervaded both the industry and the NRC [Nuclear Regulatory Commission], an attitude that the engineered design safeguards built into today's plants were more than adequate, that an accident like that at Three Mile Island would not occur – in the peculiar jargon of the industry, that such an accident was not a 'credible event' (NRC Special Inquiry Group 1980, 90).

Overconfidence is likely to have a causal influence on safety, and therefore, might be an important source of problems in the DAE's facilities.

A good example of how the two factors identified above played a part in the adoption of more risky designs is the case of the Prototype Fast Breeder Reactor that is under construction. The DAE has chosen a reactor core design with a positive void coefficient that could lead to uncontrolled and fast power excursions rather than alternative designs with negative coefficients, and included a containment that is not designed to withstand a full-scale accident (Kumar and Ramana 2008). The reasons for this adoption include both overconfidence and the desire to cut costs. Both these factors come together in a revealing comment by a former DAE official, who 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 to 1 in 10^8 per demand, why invest more funds for safety features. (Paranjpe 1992, 513)

Conclusion

Nuclear power plants, as sociologist Charles Perrow described nearly three decades ago, are 'interactively complex' and 'tightly coupled' systems prone to 'normal accidents' (Perrow 1984), even in the best of circumstances when operated reliably. Given the catastrophic consequences of nuclear accidents and the complexity of nuclear power plants, organizations in charge of them need to have good safety culture if they are to operate these facilities reliably. However, as shown in this paper through description of the nuclear establishment's response to the accidents at Fukushima and Narora, the establishment does not satisfy the demanding organizational conditions for safe operations of a complex, high-hazard technology. It exhibits various characteristics that contradict safety theorists' observations about HROs.

Contrary to providing 'the cognitive infrastructure that enables simultaneous adaptive learning and reliable performance' or exhibiting a 'reluctance to simplify interpretations' for a organization's health or possessing 'the willingness and the competence to draw the right conclusions from its safety information system', the DAE appears to show a systematic disposition to ignore inconvenient facts, simplify interpretations in its favor, and draw inappropriate conclusions from evidence. In other words, decision-makers within the DAE appear to have deeply flawed models of the world and this can only work against ensuring safe and reliable operation of its facilities.

In the AERB's and NPCIL's mental models of nuclear reactors, there is zero probability of accidents. Within that framework, the safety measures that are being undertaken are in fact unnecessary, because the occasions when they would be used can never occur. Near-misses, rather than being wake-up calls, are argued as proving inherent safety of reactor designs. It is this sort of thinking that, following Fukushima, the Japanese term the 'safety myth' (Nöggerath, Geller, and Gusiakov 2011; Onishi 2011). The consequences of such myths could be catastrophic.

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Notes

1. It could have been pointed out, for example, that Japanese reactors had survived numerous earthquakes before March 2011.
2. In many, if not all, of the countries that have continued supporting nuclear power, government or nuclear officials were quick to dismiss the possibility of a Fukushima-scale nuclear accident in their own countries (Ramana 2013).
3. For example, the Konkan Bachao Samiti, a local environmental organization, had raised concerns about the safety of the EPR design proposed for Jaitapur in a meeting with the Nuclear Power Corporation on 4 August 2010, citing problems identified by regulatory authorities in the United Kingdom and lack of regulatory approval in the United States (Pednekar 2010). Likewise, the Peoples Movement Against Nuclear Energy identified numerous safety concerns with Koodankulam, including volcanic and tsunami hazards, and inadequate fresh water availability onsite (PMANE 2011).
4. An exception to the AERB's regulatory purview is facilities that have potential nuclear weapons applications, including fuel cycle facilities such as reprocessing plants.
5. The constitution of the AEC states that 'the Secretary to the Government of India in the Department of Atomic Energy shall be the ex-officio Chairman of the Commission'.
6. In 2012, the national Comptroller and Auditor General assessed the AERB in detail and found legal and practical hurdles in the way of independent functioning of the organization (CAG 2012).
7. Though AERB and NPCIL looked at various nuclear power plants and associated spent fuel pools, there was no examination of reprocessing plants that contain large quantities of radioactive materials, both as spent fuel in pools (while awaiting reprocessing) and as high and intermediate level waste produced during reprocessing.
8. One way to quantify these levels has been on the basis of the amount of radioactivity of Iodine-131 (I-131) released to the atmosphere; Level 4 corresponds to approximately 50 to 500 TBq of I-131 release, Level 5 to approximately 500 to 5000 TBq, Level 6 to approximately 5000 to 50,000 TBq, and Level 7 to above approximately 50,000 TBq (IAEA 2012, 17–18).
9. The use of the term plant malfunction to refer to what happened at Chernobyl is also troubling.
10. The example of thyroid cancers resulting from the 1986 Chernobyl accident might be useful to recall in this context. In 1991, the International Atomic Energy Agency concluded that 'there is no clear pathologically documented evidence of an increase in thyroid cancer of the types known to be radiation related' (International Chernobyl Project and International Atomic Energy Agency 1991, 388). Since the latency period of thyroid cancers from Chernobyl is around 5.8 years (Nikiforov and Gnepp 1994), attempting to assess thyroid cancer incidence before this period has elapsed would, naturally, lead to the inference that there is no correlation between thyroid cancer incidence and the accident. As time passed, the increased incidence of thyroid cancers became more apparent. In 2000, the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) recorded that there were an 'unusually high numbers of

thyroid cancers observed in the contaminated areas during the past 14 years', and went on to observe that 'the number of thyroid cancers (about 1,800) in individuals exposed in childhood, in particular in the severely contaminated areas of the three affected countries, is *considerably greater* than expected based on previous knowledge' (UNSCEAR 2000, 504–514).

11. At Fukushima, about 90% of the radiation is said to have blown over the ocean, which may explain part of the differences between estimates for Chernobyl and Fukushima (personal email from Jan Beyea, 10 February 2013). For an estimate of what might happen if the radioactive materials were to have fallen over populated areas, see (Gering et al. 2013).
12. The DAE's emergency plans are similarly dubious and are conspicuously inadequate to the circumstances expected during an accident (Ramana 2012, 214–216). The sociologist Lee Clarke has argued that such implausible plans 'are rhetorical instruments that have political utility … for organizations and experts' rather than having any actual use in the event of a real emergency (Clarke 1999, 13).
13. This is not a new claim. In 1999, following the Tokaimura criticality accident in Japan, the AEC Chairman categorically asserted that there 'is no possibility of any nuclear accident in the near or distant future in India' (ToI 1999).
14. The AERB and NPCIL do not publicly specify what initiating events they postulate, but it is entirely possible that like the safety authorities in Japan, who chose not to postulate the kind of earthquake and Tsunami that occurred on March 11, 2011, real events might fall out of the bounds of postulated initiating events.
15. The Independent Investigation Commission on the Fukushima Daiichi Nuclear Accident found that 'regulatory authorities … had encouraged the company [TEPCO] to incorporate new findings regarding tsunami risks into its safety plans, but such measures were not made mandatory' (Funabashi and Kitazawa 2012).
16. The long interregnum that the authors talk about refers to the recurrence of an earthquake on the same fault, and this may not occur for many thousands of years.
17. To the extent possible, we derive this description from documents put out by the DAE and its sister organizations. If these are not available, or as a supplement, we use news and media reports. We assume that these are accurate unless there is reason to believe otherwise.
18. The source of information for these event details has been the annual reports entitled 'Operating Experience with Nuclear Power Stations in Member States' that the IAEA puts out. These reports are based entirely on information that the DAE provides the IAEA. See (Ramana and Kumar 2010) and (Ramana 2012) for specific references.
19. The former chairman of the Atomic Energy Regulatory Board, the organization responsible for the safety of DAE installations, has said that a local core-melt and explosive fuel-coolant interaction could not have been ruled out if the crew had not acted as they did (Chanda 1999).
20. According to the IAEA, an inherent safety characteristic is a 'fundamental property of a design concept that results from the basic choices in the materials used or in other aspects of the design which assures that a particular potential hazard can not become a safety concern *in any way*' [our emphasis] (IAEA 1991, 10). The phrase 'in any way' implies that operator inaction or even error should not lead to an accident.

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Nuclear Safety in India: Theoretical Perspectives and Empirical Evidence

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Abstract

This paper examines lessons from the operating experience in India's nuclear facilities about factors influencing the risk of potential accidents. Different perspectives on safety in hazardous facilities have identified organizational factors coincident with reliable and accident-free operations; these include functional redundancy and compensation for failures, the importance of organizational leaders in setting and maintaining safety standards, healthy relationships between management and workers, and sophisticated learning from failures. Using publicly available information about incidents and failures, we find that these conditions are frequently violated.

Introduction

India plans a large expansion of power from nuclear energy over the next few decades. Many reactors and other facilities associated with the nuclear fuel cycle, and operated by the country's Department of Atomic Energy (DAE) and its subsidiary organizations, have had accidents of varying severity (Ramana 2012).[1]

A major accident in a densely populated country like India can be catastrophic. Therefore, a study of the safety performance and culture in India's nuclear facilities is of inherent interest.

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In addition, if nuclear energy is to substantially contribute to reducing greenhouse gas emissions, it would have to expand significantly over the next few decades. Such expansion would especially have to occur in industrializing or developing countries with fast-growing electricity requirements and relatively low levels, or complete absence, of nuclear generation capacity. India offers a case study for understanding the challenges facing expansion of nuclear power in such countries. How India's nuclear establishment manages safety is therefore of interest not just to people living in India but to the larger international community.

A number of studies on nuclear safety have noted the importance of characteristics of the organizations managing and operating these facilities. The goal of this paper is to shed some light on the organizational culture and behavior within India's nuclear establishment. Our inquiry relies partly on the previous work of scholars of organizations, who have observed common behaviours in those organizations managing to operate hazardous technologies in a safe and reliable manner. On the basis of the examination of the safety record of India's nuclear facilities, we derive lessons about the prospects for safe operations therein. We also seek to understand how the choices made by the DAE and its everyday practices affect the risk of accidents at its facilities.

We begin by describing theories of accidents and safety that are relevant to our study, and examine the causal role that various factors can play. Then, we analyse two safety related incidents in Indian nuclear facilities. From the public record, the first event appears to be unique. However, the second incident that we analyse is one among a number of such events, and we examine some of the underlying reasons for why efforts to stop their recurrence have been unsuccessful. This examination points to lacunae in how nuclear facilities are managed. As further illustration of some underlying lacunae, we also list some other recurring failures at India's nuclear facilities. All of these problems suggest that the DAE's actions are inconsistent with actions recommended by safety theorists for lowering the risk of accidents. Finally we briefly discuss latent influences on organizational behaviour within the DAE, such as economic and political pressures as well as attitudes towards risk.

Theoretical Perspectives on Safety

The origins of accidents and factors contributing to safe operation have been discussed previously in the literature and we briefly summarize some of the different approaches to the subject. Broadly speaking, these approaches can be divided into those focused on aspects of the technology and those focused on aspects of the operating organization and management. One may further subdivide approaches focused on technology into those that are optimistic about avoiding accidents through the use of appropriate design, especially what is termed "defense in depth" (Glasstone and Sesonske 1981; Knief 1992); and those that are pessimistic about avoiding accidents, notably the school of thought that goes back to Charles Perrow's analysis of what happened at the Three Mile Island nuclear plant in 1979 as a "normal accident" whose origins lay in the structural features of the system (Perrow 1984).[2] We do not delve into these approaches that are focused on technology because nuclear power in India is not significantly different from other countries in this aspect. In contrast, our case studies are revealing about the organizational behaviour in DAE's facilities, where a country-specific examination merits interest owing to its unique institutional situation.

There is a vast literature on the origins of accidents that seeks to identify the organizational factors contributing to safe operation. In the context of nuclear hazards, much attention has been paid to the concept of safety culture, especially in the aftermath of the 1986 Chernobyl accident (Pidgeon 1991). A prominent example of approaching safety through the study of organizational practices has been the work of the High Reliability Organization (HRO) School, led by a group of scholars at the University of California, Berkeley. HRO scholars tried to explain what allowed some organizations to operate hazardous technologies with what they felt was "an extraordinary level of safety and productive capacity" (La Porte 1996). Their task was to identify those organizational factors that allowed the management of risky technologies with a relatively high degree of safety. The HRO group maintains, though, that they have only uncovered “conditions that were necessary for relatively safe and productive management of technologies” but do not wish to imply that “these conditions were sufficient” (La Porte and Rochlin 1994).

The common features of organizations with a record of relatively safe operation that HRO theorists identify involve: the importance of political elites and organizational leaders placing a high priority on safety in design and operations; setting and maintaining safety standards and practices; sophisticated learning from failures; and ensuring a healthy relationship between management and workers (Roberts 1989, 1990; LaPorte and Consolini 1991; Sagan 1993; La Porte 1996; Bigley and Roberts 2001). Other safety theorists identify similar factors. James Reason calls for an organizational culture where managers are knowledgeable and pay attention to safety in the organization as a whole, showing the ability to learn appropriate lessons from the safety information system and act upon those, and where the relationship between managers and workers encourages reporting of errors (Reason 1997, 195-196).

Redundancy in system operations, by making allowance for failures, is also widely emphasized. Because of the reliance on safety devices to ensure accident-free operations, all these perspectives, implicitly or explicitly, require organizations to ensure that that these devices should be reliable and built with high quality control. One approach that does not fit well into our dichotomy of technology-focused and organization-focused approaches is the Systems Safety (SS) approach (Leveson 2002, 2004; Marais, Dulac, and Leveson 2004; Leveson et al. 2009; Leveson 2011). Here, safety is an emergent property that can be evaluated only at the system level and not at the component level; and accidents do not necessarily result only from individual failures. In their view, accidents result from inadequate enforcement of constraints on behaviour (where the constraints can arise from the physical system, engineering design, management, or from regulatory practice) at each level of the socio-technical system (Leveson et al. 2009). However, SS theory too has implications for organizational behaviour in particular, it emphasizes tolerance for dissenting views and avoiding blame (Leveson 2011, 415-443), drawing in part on the notion of a "just culture" (Dekker 2007; Reason 1997).[3]

We do not select between these different perspectives on safety. Rather, we focus on what they share in their recommendations for increasing safety. We now examine two specific events at the DAE's facilities and evaluate how the organization performed with respect to these characteristics.

Safety Events in Indian Nuclear Installations

The presence or absence of severe accidents, or more generally the frequencies of accidents, do not by themselves point to underlying characteristics of a system: for example, whether it is safe or whether the managing organization acts to promote high reliability. Instead, following the observations of HRO scholars, we ask if the available evidence suggests a high priority to safety at all levels of the organization, whether the management is open to inputs from workers, and whether there are efforts to improve safety at all levels and to learn from mistakes. Some of the evidence on these questions emerges around accidents triggered by prosaic failures, which are likely to have been easily prevented if the practices had been different. There have been many such failures at Indian reactors and other nuclear facilities (Ramana and Kumar 2010; Ramana 2012). We now examine two in detail.[4] These are by no means the most severe accidents or the ones with greatest potential consequences,[5] but they clearly illustrate problems such as the failure to learn from experience of repeated failures, biased reporting and interpretation of accidents by management, lack of openness and transparency, low priority to worker safety, and inadequate attention to safety in general.

1. Kalpakkam Reprocessing Plant Accident

The DAE has three reprocessing plants to deal with irradiated spent fuel produced by nuclear reactors.[4] The reprocessing of spent fuel produces chemical and radioactive wastes, which are usually classified into low (LLW), intermediate (ILW) and high level waste (HLW) depending on the radioactivity level or concentration. In January 2003, a valve failure at the Kalpakkam Atomic Reprocessing Plant (KARP) led to HLW entering a stainless steel tank (Tank-3) intended to hold LLW. Six employees, who were instructed to collect samples from Tank-3, ended up collecting this highly radioactive material (Venkatesh 2003).

At the time of the valve failure, about five years after the plant started operations in 1998, no monitors had been installed to check for radiation levels in that area. Neither were there any mechanisms to detect the valve failure. Therefore workers had no way of knowing that the samples they went in to collect were emitting high levels of radiation. The accident was recognized only after a sample collected was taken to a different room and radiation measured. In the meantime the six workers had received extremely high radiation doses (280-420 mSv) (Anand 2003).

What is of greatest relevance to an evaluation of the safety culture of the DAE is the response of the management. KARP is administratively under the Bhabha Atomic Research Centre (BARC). Despite a safety committee's recommendation that the plant be shut down, the management of BARC decided to continue operating the plant (Anand 2003). Then, the employees union, the BARC Facilities Employees Association (BFEA), wrote a letter to the director setting forth ten safety related demands, including the appointment of a full time safety officer. The letter also recounted two previous incidents where workers were exposed to high levels of radiation in the past two years, and how higher officials had always cited urgency of operations as a reason for the Health Physics Department not following safety procedures. Once again there was no response from the management. Finally, some months later, the union resorted to a strike.

The management's response was to transfer some of the key workers involved in the agitation and give notice to others; this had the desired effect, and two days later all the striking workers joined back. The BARC Director's interpretation was smug: "If the place was not safe, they would not have joined back" (Mohapatra 2003). Ultimately, the union leaked information about the radiation exposure to the press. Once the news had become public, the management grudgingly admitted that this was "worst ever radiation exposure incident" in its history (Das 2003).

But the management blamed the whole accident entirely on the employees. According to the Director of BARC, the accident was due to a "little bit of error in judgment, miscalculation and over-enthusiasm" on the part of employees (Radhakrishnan 2003). He went on to assert that "failure of equipment went unnoticed" in the facility. Finally, he went on to directly accuse the workers by suggesting that their "mistake was that they didn't mount area gamma monitors before entering the area" (Anonymous 2003). But there were no gamma monitors in that area. Indeed, installing such monitors had been one of the ten demands made in the BFEA letter to the management. Asked about this, the head of BARC's waste management division could only offer the excuse: "We were in the process of installing these when the unfortunate incident occurred" (Anand 2003), thus belying their own accusation.

The second accusation leveled by the management was that some of the workers were not wearing their thermoluminescent dosimeter (TLD) badges (Anand 2003). But this has nothing to do with the accident; TLD badges would not have warned the workers about radiation levels until after the fact.[5] They would only help assess each worker's radiation exposure after the event. Furthermore the fact that TLD badges were frequently not used suggests a low priority to radiation safety. For its part, the BFEA claimed that because of the unrelenting pace of work at KARP and "unsafe practices being forced on the workers", accidents have become a regular feature (PTI 2003). In other words, practices that promote safety had to be ignored in order to meet work pressures.[6] The accident, again, illustrates how the DAE violates most of the recommendations offered by the different perspectives on safety.

First, there was no redundancy that would protect the workers in case of valve failure. SS theorists argue that in addition to learning from an accident, there should be “an increasing emphasis on preventing the first one”, i.e., the focus should be on preventing precursor failures and other accident triggers (Leveson 2011, , 5).

Second, though the plant had been operating since 1998, there had neither been any monitors in the region to detect radiation levels, nor any procedures to alert workers to radiation levels. In organizational terms, this lack of redundant mechanisms to deal with valve failure and of monitors to detect such failure suggests the relatively low importance given to safety by the leadership.

Third, rather than trying to work with them to prevent the occurrence of events of this kind, the BARC management blamed the workers and took disciplinary action against employees who demanded information about the accident (Sri Raman 2003). More generally, KARP operations were marred by discontent and opacity, and the management repeatedly disregarded worker’s attempts to have safety features installed.

All of these might be contrasted with the findings of safety theorists. For example, the HRO school’s studies reveal that high performing nuclear power plants possess an atmosphere of openness and responsibility, “in which all individuals regardless of rank feel responsible for every detail of plant operation they can observe, and in which they feel free to point out their observations without fear of adverse consequences to themselves” (Rochlin and von Meier 1994). Such organizations “reward the discovery and reporting of error, without at the same time peremptorily assigning blame for its commissions. This obtains even for the reporting of one’s own error…” (La Porte 1996, , 64). Likewise, the Systems Safety approach posits that “blame is the enemy of safety” (Leveson 2011, , 56-57, 531). Finally, many safety theorists emphasize the importance of trust between workers and managers (Cox, Jones, and Collinson 2006), and this characteristic again is lacking at KARP. In a similar vein, James Reason has called for "an atmosphere of trust in which people are encouraged, even rewarded, for providing essential safety-related information"(Reason 1997, , 195).

2. Heavy Water Leaks

In March 1999, some personnel at the second unit of the Madras Atomic Power Station were using a device called BARCCIS (Bhabha Atomic Research Centre Channel Inspection System), which is used to inspect coolant tubes in reactors. Suddenly, a plug that sealed one of the coolant channels—through which heavy water was to flow and remove the heat produced during reactor operations—slipped away and a large quantity of radioactive heavy water leaked out.[7] Even though the reactor was shut down for maintenance, a plant emergency was declared, which could be seen as an indication of the seriousness of the event.

The station director's statement to the press, on the other hand, creates the impression that it was a scheduled release of heavy water: "We undertook the operation of re-seating the plug in the form of replacement which involves *planned escape* of heavy water from the channel inside the fuel machine vault" (Subramanian 1999) [our emphasis]. He went on to characterize the leaked heavy water as being of "an insignificant quantity."

A number of public statements by others associated with the nuclear programme, however, indicated that the amount was not insignificant. The secretary of the Atomic Energy Regulatory Board stated that it was less than 4 tons (Subramanian 1999, , 28). Soon afterwards, the Press Trust of India reported, quoting officials from the Nuclear Power Corporation, that it was about 6 tons (Xinhua 1999). A former chairperson of the AERB went further and speculated that about 14 tons of heavy water might have leaked and supported his speculation by asking, "Why was a plant emergency declared (during this period, the reactor was shut down)? If the leak was only like that from a tap, why declare a plant emergency?" (Subramanian 1999, , 28). Even the lowest of these estimates cannot be considered insignificant.

Leaks of heavy water at Indian nuclear power stations have been a regular occurrence, starting with the Rajasthan Atomic Power Station (RAPS)— the first heavy water reactor constructed in India (Ghosh 1996). But, despite a lot of effort—quite understandable because heavy water is expensive and hard to produce—the DAE has not managed to contain them. Just in 1997, such leaks occurred at the Kakrapar-I, Madras Atomic Power Station-II, and Narora-II reactors (IAEA 1998, , 301-320). In 2004, leaks at RAPS resulted in large release of tritium to the atmosphere (AERB 2009, , 37). The previous year, high levels of tritium were recorded in the liquid discharges from the Narora and Kakrapar Atomic Power Stations (AERB 2008, , 38).

There appear to be multiple causes for such heavy water leaks. On 2 July 2007, high tritium levels were detected at the RAPS-II reactor, which turned out to be because of a "pin-hole" opening in the primary heat transfer system. In turn, the opening was a result of a "substantial reduction in wall thickness due to flow induced erosion/corrosion" (AERB 2008, , 19-20). During January to March 2009, there were three heavy water leaks in different nuclear reactors, all due to "fretting damage" [a special wear process that occurs at contact areas] (GoI 2010, , 33). A heavy water leak in the Madras Atomic Power Station in 1988-89 was due to the failure of the moderator inlet manifolds, a device meant to withstand the impact of the moderator heavy water entering the calandria at high velocity (Sundararajan, Parthasarathy, and Sinha 2008, , 95).

The amounts of heavy water that leak can be significant. For example, on 15 April 2000, there was a leak involving seven tons of heavy water at the Narora-II reactor (AERB 2001, , 13). Three years later, on 25 April 2003, there was another heavy water leak at the same reactor, this time involving six tons (AERB 2004, , 18).[8] In contrast, leaks at Canada's heavy water reactors involve much less quantities, typically tens or at most hundreds of liters.[9]

Following the 1999 heavy water leak, the Atomic Energy Regulatory Board undertook a review of the BARCCIS system and suggested a number of changes in design, operating procedures, and training (AERB 2004, , 18). The occurrence of numerous heavy water leaks since, including a leak at the Narora reactor that was similar in character to the MAPS leak, despite these changes suggests weaknesses of regulation, failure to learn from earlier accidents, or continued operator errors.

3. Problems with equipment maintenance, design and practice

Heavy water leaks are not the only recurring problem in the DAE's facilities. Another frequent issue is with inadequate and inoperative safety equipment. These are required to maintain control of the reactor under unanticipated circumstances, so if they are not working there is an increased probability that an initial event could cascade. A related problem is of safety devices being left in an inoperative state or maintenance of equipment being neglected. There are examples in the case of the Narora reactor, which experienced an accident in 1993 when turbine blade failure led to a fire in the turbine building and a complete loss of power in the station; and operators had to intervene in multiple ways to shutdown the reactor, avoid recriticality and facilitate decay heat removal; details of this accident has been discussed elsewhere (Ramana and Kumar 2013; Ramana 2012; Ramana and Kumar 2010). During this accident, the smoke sensors in the power control room at Narora did not detect the fire as soon as it started (Srinivas 1993); the fire was detected only when the flames were noticed by plant personnel.[10] Similarly, three hours and fifty minutes into the accident, the two operating diesel driven fire water pumps failed due to causes that have not been identified (Nowlen, Kazarians, and Wyant 2001). A third pump was out of service for maintenance.

Two contributing factors to the Narora accident in 1993 had also been prevalent in DAE's facilities: excessive vibrations in the turbine bearings and oil leaks. In 1981, Rajasthan-2 was shut down twice because oil leakage in the turbine building led to high levels of sparking in the generator exciter (IAEA 1982, , 235). After it was restarted, the reactor had to be shutdown yet again when a large oil leak from the turbine governing system was observed. Only when the reactor was restarted a third time, in early 1982, were the high vibrations of the turbine bearings and the failure of the turbine blades noticed (IAEA 1983, , 250). This then led to a prolonged shutdown of more than 5 months. Even after this problem had apparently been fixed, the reactor had to be shutdown once again because of high turbine bearing temperatures (IAEA 1983, , 230). Again in 1983, high vibrations were noticed in turbine generator bearings and it was revealed that two blades in the second stage of the high pressure rotor had sheared off at the root (IAEA 1984, , 292). In 1985, the first unit of the Madras Atomic Power Station (Madras-1) was shutdown repeatedly because of high bearing vibrations in the turbine generator (IAEA 1986, , 240). Rajasthan-1 had to be shutdown due to high bearing vibrations in 1985, 1989, and 1990 (IAEA 1986, , 242; 1990, , 302; 1991, , 298).

Oil leaks were also common. In 1988, Madras-2 was shutdown due to an oil leak from the generator transformer (IAEA 1990, , 288). In 1989, a heavy spark was observed from slip rings on the exciter end of the turbine in Madras-1; there were also two other fires in the same reactor near the primary heat transport system (IAEA 1990, , 298). Oil leaked from a turbine bearing in Madras-2 in 1989 (IAEA 1990, , 300). In 1992, there was an oil leak in the turbine stop valve in Madras-2 (IAEA 1993, , 288). In addition in 1992, in the Narora-1 reactor there were two separate oil leak incidents in the turbine generator system (IAEA 1993, , 289). There has been at least one hydrogen gas leak prior to the Narora fire accident: in 1991, in the generator stator cooling water system of Madras-2 (IAEA 1992, , 390).

All these failures should have caused serious concern because the factors that combined to produce the Narora accident in 1993 had combined elsewhere earlier to disable all the safety systems. In 1989, a reactor in Spain experienced turbine vibrations, which caused oil to leak and hydrogen to escape. The hydrogen burned violently and the oil caught fire. The fire spread through the cabling and disabled operation of the emergency cooling and heat exchange pumps (Ramsey and Modarres 1998, , 325). The reactor was permanently shutdown.

Another set of examples of repeated failures in DAE facilities involves failures of heat transport pumps. In 1980, Rajasthan-1 experienced unanticipated shutdowns four times during power system fluctuations; at least thrice this happened after the disabling of primary heat transport pumps, and heat generated during operation could not be removed from the core (IAEA 1981, , IN-3, 3). At least once, some pumps were already inoperative when power fluctuations caused additional pumps to fail. That year in Tarapur-2 generation was restricted for nearly two weeks because only one recirculating pump was in service (IAEA 1981, , IN-2, 3). Subsequently that year, the unit had to be shutdown for 12 days to attend to the failure of the sole recirculating pump.[11] Such problems recurred through the 1980s and 1990s. In 2004, Madras-2 was shutdown for 8 days because the two main primary coolant pumps were unavailable (IAEA 2005, , 324). After being restarted, the reactor had to be shutdown again because motor bearings of one of the pumps had to be replaced.

These examples indicate that organizational elites pay insufficient attention to small failures, and also to maintenance and inspection. Lack of attention to such factors has been identified as one of the underlying causes of the 1988 Piper Alpha accident (Paté-Cornell 1993). In that sense, these failures also indicate inadequate attention to safety by the DAE's leaders to safety in design and operations. SS theorists argue that there should be “an increasing emphasis on preventing the first one”, i.e., the focus should be on preventing precursor failures and other accident triggers (Leveson 2011, , 5). The public record does not suggest any such increased emphasis.

The other problem that the continuing series of small leaks and other failures offers evidence of is the inability to engage in sophisticated learning from failures. James Reason points out that an organization with good safety culture “must possess a learning culture—the willingness and the competence to draw the right conclusions from its safety information system" (Reason 1997, , 196). As discussed earlier, because of the recurrence of similar failures despite avoidance efforts, there is reason to question the competence of the DAE. A further problem might be a tendency to simplify interpretations, a tendency that safety theorists warn against (Vogus and Welbourne 2003). For example, After the 1999 leak, the Director of the Madras Atomic Power Station claimed that it "did not involve an unusual situation" and that the amount of heavy water that escaped "was only an insignificant quantity"; the Secretary of AERB stated, "the safety of the reactor was ensured" (Subramanian 1999). This is unlike how HROs operate. Weick *et al* observe that while "most organizations tend to localize failure, effective HROs tend to generalize it" (Weick, Sutcliffe, and Obstfeld 1999).

Conclusions and Discussion

By studying organizations that operate hazardous technologies, safety theorists have identified various factors that are common to organizations that manage to operate these facilities in a manner that is relatively free of errors. Our case studies suggest that the Indian Department of Atomic Energy does not possess all of these characteristics, and goes against some of the recommendations offered by a number of safety theorists. Specifically, we find evidence of political elites and organizational leaders not placing a high priority on safety in design and operations. For example, at the Kalpakkam Reprocessing Plant, there was clearly inadequate redundancy to protect against valve failure, and no radiation monitors to detect that highly radioactive waste had entered an area that was not designed to deal with the material. The DAE also does not appear to be learning the appropriate lessons from failures and this is demonstrated both through repeated failures and by its benign interpretations of events. After the heavy water leak at the Madras Atomic Power Station, the director tried to dismiss the significance of the leak in multiple ways. The KARP management drew wrong lessons about the safety of the facility from the fact that striking employees rejoined work. Finally, there is evidence that the relationship between management and workers is strained, with employees being blamed for failures that they could not possibly be responsible for.

It is clear that there is much to be gained by the DAE developing these organizational characteristics that are prescribed by safety theorists. The question that emerges is whether there are factors that decisively work against the acquisition of such characteristics.

Unlike many utilities running nuclear plants that have been studied in the safety literature, the DAE is a state owned ("public sector") organization. Thus, in contrast with private companies, profitability is not an overarching goal. However, the DAE has frequently stated that it aims to produce economical nuclear power. There are several instances where economic motivations have been cited as a motivation to cut back on activities promoting safety.[14] Furthermore during various phases in its five-decade long experience with operating reactors and other nuclear facilities the DAE has been under pressure, in part because of its inability to meet production targets it set itself, to accelerate construction, reduce maintenance time for reactors, and cut costs. In this, there may be some parallels with organizations such as the National Aeronautics and Space Administration (NASA) in the United States (Leveson et al. 2009). Thus, pressures on the DAE to prioritize efficient and economical delivery of its products, i.e., nuclear electricity and or related services, over improving safety might have weighed against the adoption of practices followed at HROs. These measures, while offering the benefit of safe operations, are expensive, especially in labor and management resources, and these might be seen as unproductive and not worth the cost if facilities are perceived by management to have been operating smoothly for many years (Pool 1997, , 277).

A second important factor that might work against the DAE adopting practices more conducive to lowering risk is the confidence that the DAE seems to have that the facilities that it builds and operates are completely safe. For example, in the aftermath of Fukushima the head of the DAE asserted that nuclear reactors [in India] are "one hundred percent" safe, and the Chairman of the Nuclear Power

Corporation went as far as denying what happened in Fukushima: “There is no nuclear accident or incident in Japan’s Fukushima plants. It is a well planned emergency preparedness programme which the nuclear operators of the Tokyo Electric Power Company are carrying out to contain the residual heat after the plants had an automatic shutdown following a major earthquake” (PTI 2011b). Some months later, the head of the DAE asserted that the probability of a nuclear accident at the DAE’s reactors is “one in infinity”, seeming to imply zero (PTI 2011a).

Complacency and discounting of risks has been observed to be one of the root causes of many accidents (Leveson 2011). Scott Sagan identifies overconfidence within the U.S. nuclear weapon complex as a “serious problem” (Sagan 1993). One of the lessons learnt through the analysis of a 2001 accident in the Netherlands was the importance of avoiding over confidence and “to avoid relying on a past successful history” (Mengolini and Debarberis 2012). Organizations where “past good performance is taken as a reason for future confidence (complacency) about risk control” has been identified as a weakness (Hale and Heijer 2006, , 136). James Reason points out that one of the many paradoxes about safety is that “if an organization is convinced that it has achieved a safe culture, it almost certainly has not”; HROs, on the other hand, “seem excessively bleak” (Reason 2000). There is plentiful evidence that the DAE is anything but bleak when it considers the safety of its facilities. Overconfidence is likely to have a causal effect on safety, and therefore might be an important source of problems in the DAE’s facilities.

In summary, we have offered evidence here of instances of accidents and failures of safety systems at the DAE’s facilities, as well as organizational characteristics that violate the recommendations of safety theorists. This combination suggests that the DAE does not meet the demanding organizational requirements for safe operations of a complex, high hazard technology. Our analysis of two safety related events shows some of these problems, including the repeated occurrences of accidents triggered by prosaic failures in its facilities, poor organizational learning from previous failures and system elites not being sufficiently interested in safety and not listening to employees. These factors together have been identified as playing a causal role in improving safety, and their absence makes accidents in DAE facilities more likely.

The organizational weaknesses of the DAE are a reminder of how hard it is to establish a strong safety culture. India's nuclear power programme dates back to 1948, when its Atomic Energy Commission was first established; its first power reactor started operating in 1969. At the institutional level, it seemed to be paying attention to safety regulation by establishing bodies to oversee the various facilities in the country ever since the constitution of an internal Safety Review Committee in 1972 (Gopalakrishnan 2002, , 384-385). Numerous documents verbalize the importance of safety culture and the DAE has benefited from reviews by international bodies like the World Association of Nuclear Operators (Koley et al. 2006; GoI 2007). If, despite these efforts, there are ongoing and serious concerns about the safety of nuclear facilities in India, the problems would magnify if nuclear power were to expand manifold. If nuclear power is seen as an important part of the solution to climate change, this should be borne in mind, especially in the context of countries with limited experience of nuclear power. It also prods us to reiterate a popular adage in the nuclear industry: a nuclear accident anywhere is a nuclear accident everywhere.

Notes:

1. Examples of such subsidiary or affiliated organizations are the Nuclear Power Corporation of India Limited (NPCIL), the Bhabha Atomic Research Centre (BARC), and the Indira Gandhi Centre for Atomic Research (IGCAR). Safety regulation is the responsibility of the Atomic Energy Regulatory Board (AERB), except for those facilities that have potential nuclear weapons applications, including fuel cycle facilities such as reprocessing plants. Since its inception, the AERB has reported to the AEC, which is headed by the operational head of the DAE. Following the Fukushima accidents, there have been widespread expressions of concern about the safety of Indian nuclear facilities, including the AERB's

lack of independence. As a result, the Indian government has proposed changing this arrangement. In this paper, we use DAE as an umbrella term to refer to all these subsidiary organizations. We recognize that the events described in this paper are classified as incidents according to the International Atomic Energy Agency's International Nuclear Event Scale, but deliberately use the more commonly used term accidents in order to emphasize that all these events have safety significance, especially when studying organizational culture.

2. Since then, Perrow's work has spurred an enormous range of analyses of a variety of systems (Sagan 2004).

3. A just culture provides "an atmosphere of trust in which people are encouraged, even rewarded, for providing essential safety-related information—but in which they are also clear about where the line must be drawn between acceptable and unacceptable behaviour" (Reason 1997, , 195).

4. To the extent possible, we derive these descriptions from documents put out by the DAE and its sister organizations. If these are not available, or as a supplement, we use news and media reports. We assume that these are being accurate unless there is some strong reason to not believe that. Another source of information has been the detailed annual reports entitled "Operating Experience with Nuclear Power Stations in Member States" that the International Atomic Energy Agency (IAEA) puts out. These reports are based entirely on information that the DAE provides the IAEA.

5. That distinction probably befits the 1993 fire at the Narora atomic power station.

6. This does not include pilot-scale reprocessing plants and hot cells.

7. These badges measure cumulative exposure over a period of time, and are meant to be submitted to the health physics department for assessment.

8. It may be mentioned that, from the limited public information available, Indian reprocessing plants appear to have generally operated at low efficiencies (IPFM 2010).

9. The heavy water loaded in a reactor becomes radioactive because some of the deuterium (heavy hydrogen) nuclei absorb a neutron to become tritium (a hydrogen atom with two neutrons). Further, this tritium will be

in the form of tritiated water, which is easily absorbed by the body as it is chemically identical to water.

10. On 28 June 2007, Narora II experienced a heavy water leak (AERB, 2008, 38). In June and July of 2012, there were two significant heavy water leaks at the Rajasthan Atomic Power Station, one involving radiation exposure to 38 workers (Sebastian, 2012; Sundaram, 2012).

11. See for example, http://ca.news.yahoo.com/300-litres-heavy-water-spilled-point-lepreau-124018967.html

12. This problem continues to recur. In 2005, for example, the Atomic Energy Regulatory Board (AERB) found instances of failures in fire detectors at Kakrapar and power supply for emergency cooling at Madras (PTI 2005).

13. Another recurring problem at this reactor are leaks from a primary feed water pump recirculation line.

14. For example, the Tarapur I & II reactors suffered regularly from vibrations, but the DAE chose not to make design and other changes to eliminate these vibrations "for economic reasons" (Nanjundeswaran and Sharma 1986).

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Negligence, capture, and dependence: safety regulation of the design of India's Prototype Fast Breeder Reactor

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Negligence, capture, and dependence: safety regulation of the design of India's Prototype Fast Breeder Reactor

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This article contributes a case study of regulation of the design of India's Prototype Fast Breeder Reactor (PFBR). This reactor is the first of its kind in India, and perceived by the nuclear establishment as critical to its future ambitions. Because fast breeder reactors can experience explosive accidents called core disruptive accidents whose maximum severity is difficult to contain, it is difficult to assure the safety of the reactor's design. Despite the regulatory agency's apparent misgivings about the adequacy of the PFBR's design, it eventually came to approve construction of the reactor. We argue that the approval process should be considered a case of regulatory failure, and examine three potential factors that contributed to this failure: institutional negligence, regulatory capture, and dependence on developers and proponents for esoteric knowledge. This case holds lessons for nuclear safety regulation and more generally in situations where specialized, highly technical, knowledge essential for ensuring safety is narrowly held.

Keywords: nuclear safety; safety regulation; breeder reactors; regulatory capture; India

Introduction

The multiple accidents at Fukushima have led to widespread attention to failures of regulatory systems in overseeing safety in the nuclear industry. Fukushima has been widely described as a human-caused disaster, resulting from the close relationship between Japanese regulators and utilities operating the reactors. All three main reports on the accident – by the government investigation committee, by the independent investigation commission set up by Japan's Diet, and by the independent investigation panel established by the Rebuild Japan Initiative Foundation – list multiple deficiencies in Japan's nuclear regulatory establishment (Fukushima Nuclear Accident Independent Investigation Commission 2012; Funabashi and Kitazawa 2012; Investigation Committee 2012).

This article examines how the potential for severe accidents in India's Prototype Fast Breeder Reactor (PFBR) has been handled by the country's regulatory agency. The PFBR is the first of many demonstration fast breeder reactors that India's Department of Atomic Energy (DAE) plans to build. It is designed to produce 1250 megawatts of thermal power, which in turn would generate 500 megawatts electric power.

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The DAE includes two organizations to develop and construct fast breeder reactors, the Indira Gandhi Centre for Atomic Research (IGCAR) and construction company BHAVINI. The Atomic Energy Regulatory Board (AERB) is responsible for nuclear safety regulation in the country.

Construction of the PFBR began in 2004 and the plant is to be commissioned sometime in 2015 in Kalpakkam, Tamil Nadu (Hindu 2014). Its designers and other DAE personnel have repeatedly asserted that everything has been done to ensure that the reactor is safe. For example, two leading members of IGCAR state:

> Safety has been given highest attention in the design of PFBR. The design complies with robust, nationally as well as internationally acceptable safety criteria. The safety has been well demonstrated through analytical and numerical analyses as well as through extensive experimental investigations under environments such as sodium and high temperatures as prevailing in the reactor … The design and safety aspects have been reviewed thoroughly at all the stages starting from design to component erection stage by well qualified experts in the country, under AERB. (Raj and Kumar 2011)

Contrary to these assertions, as briefly described below and as elaborated elsewhere, evidence for significant weaknesses in the PFBR design appears compelling (Kumar and Ramana 2008, 2009a, 2009b, 2011). These weaknesses result partly from design choices that contradict internationally accepted safety recommendations. The DAE's own safety analyses of the PFBR indicate that, under certain conditions, severe accidents could occur that can overwhelm the reactor's safety systems including its containment structure (Singh and Harish 2002; Kumar and Ramana 2008; Kumar and Ramana 2009a). How could DAE have been permitted to build a reactor that is inadequately protected against severe accidents? We argue below that weaknesses in the system of nuclear safety regulation played a role in creating this outcome. Examining the case of the PFBR in detail offers a window into these weaknesses.

This article examines deficiencies in the regulatory process that were involved in the approval of the PFBR design by AERB. We summarize the main weaknesses in the PFBR design and explain why we consider the construction of the PFBR a failure of the regulatory process. We then consider three factors that likely contributed to the failure of the AERB to regulate the outcome more rigorously: negligence on the part of the AERB, capture of this regulatory body by the regulated institutions, and dependence on proponents of the technology for the esoteric knowledge required to analyze accidents and their consequences. The article concludes with a few preliminary lessons about regulating hazardous technologies where relevant knowledge is narrowly held. Some of these lessons are more general, not being limited to the case of the PFBR or to nuclear safety regulation in India.

Core disassembly accidents and regulatory failure

A critical challenge to the safety of fast breeder reactor designs is what is called a 'core disassembly accident' (CDA). The problem of CDAs originates in the fact that in fast breeder reactors, the fuel is not in its most reactive configuration when the reactor is operating normally (Maschek et al. 2012).[1] Therefore, if fuel geometry were to change in an accident, for example as a result of melting, the reaction rate can increase. If that occurs rapidly enough, it would cause an explosion in the core,[2] and if the explosion releases sufficient energy to breach the protective structures surrounding the reactor, radioactive material can be released into the environment.

Safety assessments of fast breeder reactors have generally included calculating the energy release in CDAs, so that it can be determined if the design can contain the effects of such accidents. Such calculations were pioneered by Nobel Prize winning physicist H. A. Bethe along with J. H. Tait (Bethe and Tait 1956). Subsequently, CDA studies have been conducted for nearly all FBRs constructed or proposed in the United States and Western Europe (Wilson 1977). Despite decades of experience acquired during fast reactor accident studies, there are many uncertainties in calculating CDA energy releases and it remains difficult 'to accurately model the transition phase and to ensure the prevention of core disruption and recriticality' as a result of which the possibility of a CDA remains 'a continued concern' (Lee and McCormick 2011, 392).

In the case of the PFBR, too, the designers have calculated that the explosive energy released by what they conceive as the worst-case core disruptive accident would release an explosive energy of 100 megajoules (MJ, or a million joules) (Singh and Harish 2002).[3] However, this 100 MJ value is small compared to other fast reactor designs that have been developed elsewhere, and this fact alone should have been sufficient for the regulatory body to doubt the assertions put forward by the reactor's designers, and require the PFBR to be designed to withstand an explosion with much larger energies.

The reason that the PFBR should have been designed to withstand much more than 100 MJ of energy is not difficult to understand. The amount of radioactive material in a reactor core increases with the power that can be produced by the reactor. Since the amount of radioactive materials in the core is a key factor in determining the amount of energy released during a CDA, the latter will increase with the power rating of the reactor. The PFBR is designed to produce more power than most fast breeder reactor designs developed in other countries, and can therefore be expected to release more energy in the event of a severe accident.

One measure of the conservativeness of a reactor design with respect to containing a CDA, relative to its size, is the ratio of two quantities: the maximum explosive energy level that can be contained by the reactor with high confidence, and the amount of heat (thermal power) the reactor is designed to produce. For the PFBR, the thermal power is 1200 MW and the maximum CDA energy is assumed to be 100 MJ, and so this ratio is about 0.08. In contrast, the SNR-300 reactor that was constructed (but never operated due to safety concerns) in Germany had a ratio of about 0.5: it was designed to contain 370 MJ of explosive energy, although it had a much smaller core that could produce 760 MW of thermal power (Waltar and Reynolds 1981). The FFTF reactor that attained criticality in the United States in 1980 had a thermal power of 400 MW and its maximum CDA energy was estimated to be around 350 MJ, for a ratio of 0.9. These are just two of many other such examples (Kumar and Ramana 2008).

These comparisons suggest that the designers of the PFBR have likely underestimated the amount of energy that could be released by a CDA. As described in Appendix 1, there are significant uncertainties behind the official estimate of the maximum energy that can be released by a hypothetical CDA. Furthermore, previous work has shown that not only is the PFBR design inadequate for containing the possible range of effects of a CDA, but also that DAE's claims about PFBR safety contradict its safety studies (Kumar and Ramana 2008, 2009a, 2009b, 2011).

That the regulator (AERB) permitted construction despite these weaknesses indicates, prima facie, regulatory failure. This section examines this contention based on secondary sources available to the authors, and considers three potential factors affecting the regulatory process: negligence, regulatory capture, and dependence on technology developers for esoteric knowledge. These are not water-tight categories and some of evidence we present to substantiate one factor could also be applied to another factor.

Regulatory negligence

There is evidence of negligence by the regulator in assessing both the probability and consequences of a CDA.

Assessing the probability of a CDA

Claims about safety are claims about the future; and any such claims require specifying what future situations can the design features of the technical artifact, such as the nuclear reactor, protect against. The nuclear engineering community usually terms these situations as the 'design basis'. According to the International Atomic Energy Agency (IAEA), the design basis is the 'range of conditions and events taken explicitly into account in the design of a facility, according to established criteria, such that the facility can withstand them without exceeding authorized limits by the planned operation of safety systems' (IAEA 2007, 51). The design basis includes what are termed 'design basis accidents' (or DBA), which comprises 'accident conditions against which a facility is designed according to established design criteria, and for which the damage to the fuel and the release of radioactive material are kept within authorized limits' (IAEA 2007, 145). More severe accident conditions not included in the design basis but which can lead to damage to the fuel or to radioactive release to the environment are termed beyond design basis accidents (BDBA). The terms design basis event (DBE) and beyond DBE are sometimes used in making this distinction.

There is judgment involved in deciding what gets included in the design basis, and these frequently turn out to have been wrong. The potential for fallibility in characterizing the DBE was well demonstrated by the Fukushima accident. The design basis for the Fukushima Daiichi reactors included only earthquakes of at most a magnitude of 7.9 and a tsunami of 6.7 m. In comparison, the earthquake that occurred on 11 March 2011 released more than 1000 times the energy of the largest earthquake considered in the design basis and the actual tsunami was almost 9 m high (Macfarlane 2011). While not always as consequential, various assumptions and judgments involved in characterizing the design basis can often turn out to have been mistaken.

A key regulatory decision about the CDA was whether it shall be treated as a DBA or a beyond DBA because this classification determined what criteria were to be applied by the regulator in evaluating the PFBR design. As argued elsewhere, the nuclear establishment in India is prone to over-confidence (Ramana and Kumar 2014). The designers of the PFBR considered it very unlikely that the reactor could undergo a CDA, and categorized it as beyond design basis.[4]

However, the AERB, at least initially, took a different view and stipulated that CDAs should be considered as DBAs (Chetal 2010). In time, AERB switched to treating CDAs as a BDBA.[5]

IGCAR officials did not just argue that the CDA should be treated as beyond design basis but even implicitly disparaged the expertise of AERB staff to undertake a review of the PFBR. In June 2010, the director of IGCAR described the members of the AERB that were evaluating the safety of the PFBR as 'competent persons who have designed and reviewed thermal reactors' and then went on to complain that it had been 'not easy to make them realize the different response for PFBR as sometimes the system response and behavior are quite different from PHWR' (Chetal 2010).[6]

In principle, this difference of opinion could have been settled through a quantitative analysis. Both IGCAR and AERB seem to generally agree that DBEs are those with a probability of occurrence that is greater than once in a million for each year of a reactor's operation, whereas those with probabilities below this threshold are termed Beyond Design Basis Events (BDBE) (AERB 2000; Raghupathy et al. 2004). Thus, if the probability of a CDA were found to be less than one in a million reactor years, then it should be considered beyond design basis.

The estimate of the probability of occurrence of an accident sequence is often computed using what is known as the probabilistic risk assessment or probabilistic safety analysis (PRA/PSA) methodology. Under this methodology, an accident is assumed to result from one of many combinations of a series of failures. Based on assumed, estimated, or otherwise calculated probabilities for each individual failure, reactor designers, vendors, and regulators reach conclusions about the probabilities of different kinds of accidents (per reactor year) at various reactors.[7] Had the designers of the PFBR and the regulatory body independently conducted such risk assessments, then they could have agreed on whether these assessments can support CDAs being considered beyond design basis.

The reason to recommend multiple independent risk assessments is that the probability estimate depends on how the calculation is done in its details and there is no unique answer even when accident models are broadly similar.[8] To take an example from the 1990s, two different power companies and analysis teams carried out PSAs for two nearly identical Swedish reactors, Forsmark 3 and Oskarshamn 3 (Simola 2002). Starting with similar overall goals, both PSAs chose similar initiating events, but analyzed these quite differently, treated common cause initiators in different ways, used different human error events and their probabilities, and so on, and the net effect was that these two PSAs reached significantly different results.

Similarly, the US Nuclear Regulatory Commission (NRC) landmark PRA for five nuclear reactors in the early 1990s included two different estimates of the risk of a major accident triggered by an earthquake.[9] In the case of the Surry power plant with two Westinghouse Pressurized Water Reactors, the mean core damage frequency (CDF) using these two methods varied by a factor of 4.8 (NRC 1990, 3–4).[10]

Therefore, even under best-case conditions where implementers of PRAs have no motives other than estimating the risk, two reliable and competent groups could come up with different probability estimates. In arguably more realistic situations, institutional interests and biases might lead to even more drastic differences.[11] Thus, there is every reason to assume that, had the AERB carried out its own calculation of the probability of a CDA, it would have come to a different answer from the designers of the PFBR, one that would likely have been less optimistic than any figure that the developers of the reactor design might arrive at.

Publicly, one PRA is available; but this was carried out by IGCAR and was published only in 2010 (Ramakrishnan et al. 2010), long after construction of the PFBR had commenced and at a time when a significant revision in design would be difficult and expensive.[12] A revised and more detailed version was published in 2012;

these found that the 'desired target value for overall CDF of PFBR' is 0.9×10^{-6} per reactor year (Ramakrishnan et al. 2012). Even though the CDF includes contributions from various accident pathways that would not be categorized as a CDA, it is worth noting that the result obtained by the designers of the PFBR is very close to the one in a million reactor year threshold that both parties to the dispute agreed on for categorizing the CDA as a DBE.

Furthermore, this PSA has a serious shortcoming of scope: it is 'limited to internal events at full power' (Ramakrishnan et al. 2010, 239). In other words, it only calculates the probability of damage to the reactor core when that is initiated by some failure or other event within the reactor, such as the rupture of a pipe. It does not consider operations at low power, and more importantly, events originating external to the reactor, such as earthquakes, Tsunamis, or volcanoes.[13]

The hazards to nuclear reactors stemming from such events were made clear in Fukushima. The quantitative significance of including external events in PSA calculations of the CDF is high. For the Clinch River Breeder Reactor Project in the USA, the CDF estimate due to purely internal events was 3.7×10^{-6} per reactor year; the contribution from external events was 3.2×10^{-5} per reactor year, that is, contributions from external events made core damage nearly ten times as probable (Ramakrishnan et al. 2012, 669). In other words, inclusion of external events could well change the categorization of an accident from beyond design basis to within the design basis.

Given the closeness of the calculated core damage probability to the one in a million cutoff for deciding on what is a DBA and furthermore the exclusion of external events, the precautionary course of action would have been to include CDAs within the design basis. It is also likely that if the calculation had considered any uncertainty in estimating the probability of various initiating events, there is a significant chance that the CDF probability would be above this cutoff value.

There is one more serious problem that AERB ought to have taken note of. As noted by the panel appointed by the US NRC to examine the 1975 Reactor Safety Study,[14] it is 'conceptually impossible to be complete in a mathematical sense in the construction of event-trees and fault-trees' and this 'inherent limitation means that any calculation using this methodology is always subject to revision and to doubt as to its completeness' (Lewis et al. 1978, ix). Thus, the real likelihood of severe accidents in complex systems such as nuclear reactors can quite possibly be underestimated by PSA techniques, and there is considerable residual ambiguity. This uncertainty, again, should have led a cautious regulator to insist that the CDA be included as a DBA and the reactor be designed and constructed according to the criteria that govern such accidents. Specifically, the PFBR design would have had to contain the possible consequences of such accidents.

There is likely a good reason why AERB does not seem to have taken into account any of these considerations that may have prompted inclusion of CDAs in the design basis as a measure of prudence: the AERB had approved the PFBR's design long before the publication of the 2010 PSA. According to its standard guide for process through which the AERB issues consent for construction of a reactor, it had to do so in three stages: 'clearance for excavation, clearance for first pour of concrete and clearance for erection of major equipment' (AERB 2006, 5). In the case of the PFBR, the AERB issued excavation clearance on 13 July 2002 (AERB 2003, 4); the authorization for 'first pour of concrete' was issued on 15 December 2004 (Sundararajan, Parthasarathy, and Sinha 2008, 64). In May 2006, AERB granted clearance for construction of the reactor vault up to 26.715 m elevation (AERB 2007, 10). The clearance for installation of the safety vessel was granted on 4 February 2008 (AERB 2008, 9).

To summarize, because of the many sources of uncertainty, the cautious decision would be to err on the side of safety. The IAEA recommends that accidents 'be analyzed in a conservative manner' and the use of 'conservative assumptions, models and input parameters in the analysis' (IAEA 2012, 22). This was clearly not done in the case of the PFBR.

Assessing the consequences of a CDA

The AERB was also negligent when it came to overseeing calculations of the consequences of a CDA. The conventional approach to safety in nuclear reactors is to incorporate multiple protective systems so that they would all have to fail before a radioactive release occurs. This concept is called defense in depth (Glasstone and Sesonske 1981; Knief 1992). In line with this approach, the PFBR 'is designed with various engineered safety features … [and] core catcher and containment are provided as defense in depth for BDBE' (Chetal et al. 2006, 859). According to the Preliminary Safety Analysis Report for the PFBR, its

> containment is designed to provide a leak-tight boundary that contains the release of radioactive core fission products and fuel, and withstands the pressure resulting from burning of sodium in air through potential leak paths in case of [a] core disruptive accident … (IGCAR 2004).

But as shown elsewhere, the PFBR's containment is really not capable of withstanding a severe CDA (Kumar and Ramana 2008).

Constructing a strong reactor containment building is essential to limit the consequences of severe accidents. But this would improve safety only if the RCB is designed for a sufficiently severe accident; otherwise, it is only a half-measure and cannot actually contribute to safety under the range of accident conditions possible. One measure of the capacity of the containment to withstand an accident is what is called the design pressure; when the pressure in the containment building due to the accident is significantly above this nominal value, the integrity of the containment cannot be guaranteed.[15]

Around the time the AERB was reviewing the PFBR design, IGCAR, consistent with its confident outlook, seems to have 'considered' a value of 20 kPa 'for the design pressure for RCB' (Bhoje 2003, 10). During the 1990s, the earlier version of the PFBR design envisioned a containment that could withstand an overpressure of 25 kPa (Vaidyanathan et al. 1995, 165). By contrast, the AERB insisted that the containment be stronger, albeit marginally. In 2003, after it

> completed the review of the design on Reactor Containment Building with reference to the sodium release and pressure release during a probable core disruptive accident, the PDSC [of the AERB] recommended that the design pressure of the Reactor Containment Building should not be less than 30 KPa. (AERB 2003, 12)

There is not enough information in the public record to understand the process through which this difference was resolved. However, the end result was that the reactor has a design pressure of 25 kiloPascals (kPa) (Chetal et al. 2006; Sundararajan, Parthasarathy, and Sinha 2008, 65). In other words, to the extent the

AERB did make recommendations to improve safety, its recommendations were not implemented.

As described in the appendix, for reasons involving the uncertainties in estimates of the energy released by a core disruptive accident, as well as calculations carried out on the pressure on the containment in the case of a more energetic accident (Kumar and Ramana 2008), the PFBR design is not adequate to withstand the range of pressures that could develop as a result of a CDA. Further, going by a standard metric to measure how strong the containment of a reactor is,[16] the PFBR turns out to be much weaker than other fast reactors. Again, the AERB permitting the construction of a reactor with a relatively weak containment building does not bespeak of rigorous or adequate regulation.

In a careful regulatory process, these problems and uncertainties in IGCAR's safety assessments would have been thoroughly understood independently by the AERB, and transparently so that anyone doubtful of the safety of the PFBR could have evidence of adequate oversight. This is especially important in the case of the PFBR because of the aforementioned uncertainties about the probability of a CDA as well as its consequences, which could be disastrous. But AERB staff have not published any papers with their own analyses.

There have been cases in other domains where safety regulators have not made independent investigations of their own. The US Safety, Reliability, and Quality Assurance unit of the National Aeronautics and Space Administration (NASA), for example, did identify problems with the O-rings of the Challenger spacecraft but left the problem unresolved. In part, their lack of firm action may be because they 'did not originate engineering data and analysis. They *reviewed* the determinations of the working engineers' (Vaughan 1996, 269). The results of that failure of safety regulation, among various other failures, became spectacularly evident on 28 January 1986, when the Challenger Space Shuttle broke apart 73 s into its flight, leading to the deaths of its seven crew members.

Regulatory capture

Theoretically, the most widely deployed argument for why regulators fail to perform satisfactorily involves the idea of regulatory capture, where the regulatory agency is taken over by people representing the interests of the regulated entity. This idea originated in studies of efforts by governments to stop corporations from forming monopolies (Stigler 1971; Peltzman 1976). Regulatory capture is a charge that has been leveled against nuclear regulators in many countries. Japan in the aftermath of Fukushima is a prominent example. The investigations of the independent investigation commission set up by Japan's Diet made it clear that 'the necessary independence and transparency in the relationship between the operators and the regulatory authorities of the nuclear industry of Japan were lost, a situation best described as "regulatory capture" – a situation that is inconsistent with a safety culture' (Fukushima Nuclear Accident Independent Investigation Commission 2012, 15 (Chapter 5)).

Likewise, the regulation of nuclear power in the United States by the NRC has been called a 'textbook example' of capture (Von Hippel 2011). Perhaps the best example of the effects of such capture in the United States is what happened at the Davis–Besse nuclear plant in the state of Ohio. At that reactor, leaking boric acid almost ate through a reactor pressure vessel head before it was discovered in 2002; although there had been indications earlier, routine inspections had failed to detect the ongoing process of corrosion. The plant was to be shut down by December 2001 for a full inspection, but the operating organization 'provided additional information to the USNRC and obtained approval to postpone a full inspection' by some months (Ghosh and Apostolakis 2005, 208). The NRC's inspector general explained this approval 'in large part by the desire to lessen the financial impact on (the licensee) that would result in an early shutdown' (NRC Inspector General 2002, 23). Just prior to the discovery of the hole, the Davis–Besse plant received the highest ratings possible in the NRC's Reactor Oversight Process (Keystone 2007, 65). Writing about NRC's regulatory performance, the inspector general concluded that the NRC 'appears to have informally established an unreasonably high burden of requiring absolute proof of a safety problem, vs. lack of a reasonable assurance of maintaining public health and safety' (NRC Inspector General 2002, 23).

One mechanism through which regulatory capture happens is through movement of personnel from the entity being regulated to the regulatory agency, inducing regulators to make decisions in the interests of the regulated entity when regulators have been socialized in the latter environment (Bó 2006, 214).[17] The independent investigation panel established by the Rebuild Japan Initiative Foundation pointed explicitly to the 'sweetheart relationships and revolving door that connected the regulatory bodies and electric companies, academics, and other stakeholders in the nuclear community' (Funabashi and Kitazawa 2012).

This is on display in the case of the AERB. A few years after the AERB cleared the PFBR for construction, one of the authors of the IGCAR's studies of the energetics of a CDA at the PFBR, and someone who worked for over three decades at IGCAR, became the Secretary of the AERB (Singh 2010). Personnel from the DAE and associated organizations are often inducted into the AERB's safety evaluation committees (Gopalakrishnan 1999). Two of the Chairpersons of the Project Design Safety Committee (a part of the AERB) for the PFBR, G. R. Srinivasan and S. S. Bajaj, and the chairman of the Site Evaluation Committee, S. Krishnan, all came from the Nuclear Power Corporation (Sundararajan, Parthasarathy, and Sinha 2008, 64).

The other mechanism has been through selection of personnel regarded as being less confrontational with the DAE and its plans and interests. In contrast to this is the experience of Dr A. Gopalakrishnan, one Chairperson of AERB who tried to make the agency independent of the DAE. During his tenure as chair of the AERB, the organization conducted a detailed assessment of safety deficiencies in the DAE's nuclear installations and the need for safety upgrades (Gaur 1996, xv). The AERB prepared a comprehensive document on this subject and submitted it to the Atomic Energy Commission (AEC) and the highest levels of government, much to the discomfort of the DAE. Gopalakrishnan also proposed ways to strengthen the nuclear regulatory system and make it more effective. The DAE and AEC, not surprisingly, did not extend his term after his first three-year term as Chairman of AERB ended on 16 June 1996. In doing so, they also conveyed a message about what sort of regulators were welcome.

Such a relationship of subordination and cohabitation contradicts understanding of what amounts to good practice, in addition to India's international treaty obligations. Globally, ever since the Chernobyl accident there has been consensus that it is essential 'to effectively separate nuclear power development from nuclear safety oversight functions' (IAEA 2006, 10). Article 8 of the international Convention on Nuclear Safety, which India has signed and ratified, calls upon signatories to 'take the appropriate steps to ensure an effective separation between the functions of the regulatory body and those of any other body or organization concerned with the promotion or utilization of nuclear energy' (CNS 1994). The absence of such separation has been identified as one of the factors leading to the Fukushima accidents by the Independent Investigation Commission.[18]

The same lesson has been learnt in non-nuclear domains too. For example, in June 1993, a small commuter aircraft flying out of Sydney, Australia, crashed, killing seven people in all. A subsequent investigation conducted by the Bureau of Air Safety Investigation placed a major part of the responsibility on the regulator, the Australian Civil Aviation Authority (ACAA), arguing that 'activities of the ACAA's Safety and Standards Division appeared to be biased towards promoting the viability of the operator (Monarch Air-a small commuter airline in commercial difficulties) rather than serving the safety needs of the travelling public' (Reason 1997, 165).

There is also the question of flawed institutional culture, widely discussed in the literature. Assessments of specific facilities or technological designs are often flawed because of systemic institutional biases.[19] While there is value in viewing the AERB's actions through the lens of regulatory capture, that approach is also inadequate because it does not reflect the extent to which the regulatory agency itself sees its interest aligned with that of the regulated, that is, the breeder program.[20] In the context of electricity regulation in India, it has been observed that 'regulatory behavior is in part explained by a form of self-censorship or self-regulation, arising from the schizophrenic relationship between regulators and the government' (Dubash and Rao 2008, 330).

This characterization does carry over to the AERB and there appears to be considerable self-censorship in nuclear regulation. Thus, for example, the AERB's recommendation only required that the containment be built to withstand 30 kPa. Had the board calculated the overpressure that would result from the release of energies on the order of 1000 MJ, which are clearly possible even from IGCAR's own calculations (Singh and Harish 2002), then it would have found that the containment would have to withstand over 50 kPa (Kumar and Ramana 2008). Further, worse scenarios can be easily envisioned: if the primary vessel ruptures and the sodium enters the containment building, the overpressure on the containment can exceed 150 kPa; similarly, high overpressures have been calculated in some accident scenarios for the American Clinch River Breeder Reactor (Gluekler and Huang 1979); higher overpressures would result if higher values of the thermal to mechanical energy conversion efficiencies were to be used in analysis than the low figure that the IGCAR used in its studies (Kumar and Ramana 2008).

Dependence on technology developers for esoteric knowledge

Finally, could the AERB, even if it had been independent, well-intentioned, and possessing the will to carry out a critical safety evaluation, have actually achieved careful and comprehensive scrutiny of the PFBR design? While such a counterfactual question cannot be answered definitively, there are reasons why this task would have been significantly difficult. FBRs are complex systems whose analysis requires quite specific expertise and extremely detailed calculations. Moreover, these calculations are beset by uncertainty because they must grapple with inadequate knowledge of how accidents will propagate. The underlying models used by designers are only weakly constrained by experimental knowledge, which in the case of the PFBR covered only a very small part of what designers would have actually need to know to obtain an upper bound on the explosive energy released in a CDA.

Furthermore, these calculations tend to use complex software codes, many of which are developed by FBR designers. IGCAR researchers used an internal code and a software program developed by at the US Argonne National Laboratory (Jackson and Nicholson 1972; Singh and Harish 2002). There are limits to how far independent experts can penetrate these codes and understand the implications of the assumptions made in them. Given the role of latent knowledge acquired through developing and implementing these codes, it is likely that AERB relied on IGCAR to analyze their reactor design and inform them about how it would behave under conditions that could challenge its safety.

Many of the assumptions used in FBR safety assessment are not only empirically untested, but perhaps untestable: to put it baldly, it is not possible do an experiment wherein a reactor is made to explode. This means that the conclusions derived using these codes are based significantly on judgment of the designers at IGCAR or other fast reactor designers, and these assumptions are not always made explicit. Thus, in general, it would be very difficult for a regulatory body such as AERB to confirm the adequacy of IGCAR's designs unless it has undergone a similar process involving detailed model development and its use. This difficulty is not limited to regulating fast breeder reactors or to the case of nuclear power in India.

Given the inherent challenges facing the domestic nuclear safety regulation system, one question that emerges is whether there are appropriate bodies within the international nuclear safety community that could have offered a corrective and been able to evaluate the DAE's claims. There are several such possibilities that come to mind, including the IAEA, OECD's multinational design evaluation program, and professional societies involved in standards development.[21] The main difficulty in this case is that there are few operating fast breeder reactors in the world and the vast majority of the experience possessed by the IAEA, as well as most other international nuclear organizations, pertains to thermal reactors that are typically moderated and cooled by water.

Therefore, even though the designers of the PFBR tried to base their general approach to safety on an IAEA Safety Series document (NS-R-1 'Safety of Nuclear Power Plants: Design'), they had to admit that the guidance contained in this publication was applicable only to water cooled reactors. Indeed, the IAEA document says explicitly,

> It is expected that this publication will be used primarily for land based stationary nuclear power plants with water cooled reactors designed for electricity generation or for other heat production applications (such as district heating or desalination). It is recognized that in the case of other reactor types, including innovative developments in future systems, some of the requirements may not be applicable, or may need some judgment in their interpretation. (IAEA 2000, 1)

How about those individuals with experience in fast reactor analysis and design? A small part of the peer community at the IAEA and other international conferences are breeder reactor designers and operators from other countries. This international fast reactor community is unlikely to serve as an error correcting mechanism where the AERB has failed. The community is embattled by a number of factors: lack of funds, opposition to reprocessing, shifts in nuclear energy policies over the years, concerns about safety, criticism of particular fast reactors, and repeated challenges in large and visible fast reactor projects in a number of countries.

Success of the PFBR and subsequent fast reactors in India is therefore as much in the interest of the international community of fast reactor engineers as of the DAE. This international community failed to challenge or critically examine India's fast breeder establishment on its claims about safety. DAE personnel have presented talks on the design and safety of the PFBR at various international conferences, many of which have been printed by the IAEA or other such bodies. The discussions at these conferences do not seem to have extended to any critical examinations of the PFBR's design and its ability to withstand severe accidents. Instead, breeder scientists from other countries have praised DAE for persisting with the technology [see e.g. (UPI 2006)].

The absence of critical examinations also reflects the confidence that the breeder reactor community has developed over the years in the safety of these reactors. Over the decades, fast reactor designers have chosen to highlight some physical characteristics that these reactors possess, often using terms like 'intrinsic safety' (Bagdasarov et al. 2001), or something equivalent to describe them. Their hope is that these 'inherent safety' concepts 'can be used to increase the level of safety to the point where it is highly unlikely, or perhaps even not credible, for such severe accident consequences to occur' (Wigeland and Cahalan 2011, 516). The deployment of such arguments cannot be separated from broader strategies to build support for fast breeder reactors,[22] but as a result, it would be difficult to imagine any member of this community helping the AERB make a case for, say, a stronger containment.

A key problem for breeder reactors is economic viability and they are widely seen, even by other nuclear power enthusiasts, as more expensive than the more common light water reactors. The general expectation is that this cost difference will persist into the future; for example, an expert elicitation study involving 30 European and 30 US nuclear technology experts found that under business-as-usual conditions, they expected the next generation of designs (Gen. IV), including sodium cooled fast reactors, to be remain more expensive than current generation of light water reactors even till 2030 (Anadón et al. 2012).

Proponents of breeder reactors sometimes argue that these higher costs are due to the unnecessary imposition of safety systems. Thus, for example, speaking at a specialists' meeting on passive and active safety features of liquid metal fast breeder reactors organized by the IAEA, an IGCAR 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 [1 in 100 million, or much lower than typical safety requirements] per demand, why invest more funds for safety features. (Paranjpe 1992)

In other words, fast reactors are already so safe, at least by the designers own evaluations, that they do not need any more safety features.[23]

Given this disposition toward overconfidence as well as the economically beleaguered state of breeder programs, it is not to be expected that many in this community would admit the PFBR's design to be inadequate. Thus, personnel in the AERB are unlikely to have found any independent sources of expertise to critically evaluate the PFBR or to support their misgivings.

Discussion

An important part of the context to nuclear safety regulation in India is the administrative subordination of the AERB to India's Atomic Energy Commission. As the Comptroller and Auditor General pointed out,

> AERB was constituted under Section 27 of the AE [Atomic Energy] Act, 1962, which allows the Central Government to delegate any power conferred or any duty imposed on it by this Act to any officer or authority subordinate to the Central or State Government. Section 27 of the Act currently does not provide for constitution of any authority or Board and merely provides for delegation of powers to a subordinate authority. Therefore, the legal status of AERB can be seen to be more of a subordinate authority with powers delegated to it by the Central Government than of a statutory body with independent powers. (CAG 2012)

The report goes on to point out:

> Article 8 of the Convention on Nuclear Safety of the IAEA, ratified by the Government of India on 31 March 2005, stipulates that each contracting party should take appropriate steps to ensure an effective separation between the functions of the regulatory body and those of any other body or organization concerned with the promotion or utilization of nuclear energy. A regulatory body must be able to exercise its key regulatory functions (standard-setting, authorization, inspection and enforcement) without pressure or constraint.

It then observes, based on its audit, that 'The fact that the Chairman, AEC and the Secretary, DAE are one and the same negates the very essence of institutional separation of regulatory and non regulatory functions'. The Chairman of the AERB reports to the Chairman of the AEC.

The CAG in its audit concludes that 'AERB has no effective independence as per the criteria laid down by IAEA', that it does not have the 'authority for framing or revising the Rules related to nuclear or radiation safety', and that it has 'no powers with regard to imposition of penalties' (CAG 2012). The lack of independence of the regulator creates difficulties that are not unique to the PFBR.

The natural question that emerges is what can be done. Our analysis suggests some ways forward, but also some far more serious problems. The easy-to-state solutions are that there should be administrative reforms to increase the independence and efforts to enhance the institutional capacity of the AERB. Alert public interest organizations and suitable government watchdog agencies, such as the Comptroller and Auditor General, could help in averting regulatory capture and lowering of standards. These would improve regulation of designs of future fast breeder reactors, and also other aspects of the civilian nuclear program.

But this recommendation has to be balanced with the recognition that the observed weaknesses in regulation of the PFBR, and indeed all decision-making in India related to nuclear power, are also related to the lack of accountability.[24] There is evidence from around the world that shows that shortfalls in accountability can undermine regulatory performance (May 2007). In countries where the state strongly supports nuclear energy, however, it is difficult to impose accountability on the regulatory body. After all, if the government wants many reactors to be constructed rapidly, then a lengthy regulatory process would be seen as deleterious to those aims. Unfortunately, as the cases of Japan's Nuclear and Industrial Safety Agency and the United States NRC demonstrate, accountability might begin to develop only after accidents such as Fukushima and Three Mile Island. An accident as impetus is certainly not desirable, but it is not clear if anything else can counteract the political power of the DAE and IGCAR when they seek to keep the AERB or any successor regulatory body in a subservient position.

Another step that would help is much more transparency. This is related to one limitation in our work: it is based entirely on the publicly available record, which is very limited. While we do know there have been disagreements between the AERB and IGCAR, we do not have access to the minutes of the meetings between these agencies and thus cannot shed light on, for example, what arguments were used by members of these institutions.

Despite our dependence on secondary sources, there are important conclusions that can be drawn. The available documentary record, for example, shows that there was some conflict between the AERB's safety committee and the designers of the PFBR, and the viewpoint of the designers won over the regulatory viewpoint as the design was being developed. This can be interpreted in two ways. First, the AERB did not have adequate expertise to judge safety in the PFBR and it was told so during its meetings with IGCAR. It realized its technical inadequacy and decided to settle for IGCAR's point of view. Alternately, the AERB did have the technical capacity and concluded that CDAs were serious enough to build a stronger containment but was overruled by IGCAR. While we cannot infer which is the more accurate interpretation based on the available evidence, neither is conducive to ensuring that the reactors that will be built in the country are designed to be adequately safe.

The third element of our analysis – dependence on esoteric knowledge – raises another set of challenges to safety regulation that do not appear to have straightforward solutions. What can a regulatory agency do to ensure that it has independent sources of information and is not reliant on the technology proponent's models and interpretations of what the safety issues are? Can it create conditions for error avoiding and error correcting behavior under conditions unfavorable to such behavior? At the very minimum, dealing with these problems would require the cultivation of independent expertise both within the regulatory agency and outside. Such expertise would have to be simultaneously interested in the critical issues of safety arising from particular reactor designs and disinterested in the outcomes of the regulatory process for the proponents of these designs. Given the nature of the knowledge needed, this would require the institution of and sustained funding for specialized training programs, but that would of course be insufficient in itself. This poses inherent difficulties for regulation, not just pertaining to safety regulation of fast breeder reactor designs, but of technically complex and hazardous systems in general.

Conclusion

Modern societies, with their increasing reliance on hazardous technologies often based on arcane and exclusively held technical knowledge, have turned toward regulatory bodies, such as the AERB, to make sense of the risks they are confronted with. The extent of this reliance varies between countries, but it is hard to imagine any country that deploys hazardous technologies wherein policy makers do not appeal to some group of experts to reassure themselves and society at large about the safety of these technologies. The sociologist of technology John Downer observes, 'These regulators have become a twenty-first century clergy, standing between the public and the esoteric knowledge with which they must contend, and both the public and policy makers are prone to accept their conclusions at face value with minimal reflection or circumspection' but goes on to warn that 'case studies of technological practice repeatedly suggest that such obeisance is misguided' (Downer 2010, 84).

Our case study of the regulation of the PFBR confirms this warning. It strongly suggests the AERB neither performed its own safety assessment nor did it act as if it had critically examined the safety studies of the designers of the PFBR, but instead settled for a design that was based on the DAE's flawed interpretations of its studies. We have argued that different factors appear to have played a role in this regulatory failure. AERB displayed negligence by allowing CDAs to be considered as beyond the DBA, despite the absence of a full-fledged probabilistic risk analysis to justify this choice, and does not seem to have independently carried out its own calculations of the probability of a CDA. AERB also failed to enforce its limited recommendation of increasing the design pressure for the containment. The weakness of this recommendation by itself points to self-censorship within the regulatory body, because it had access to DAE's studies, which themselves imply that much higher design pressures might be necessary to contain severe accidents although that was not the way the DAE interpreted them. Such censorship could result from multiple factors: AERB is administratively subservient to the DAE, it includes several high-ranking former staff from the DAE, and it is likely that it sees its interests as closely aligned with those of the latter organization. The last difficulty above probably cannot be resolved through administrative reform alone. These difficulties are not unique to the PFBR licensing process, and affect safety regulation in general.

Even if these problems were solved, a fundamental difficulty with nuclear safety regulation would remain: regulators must inevitably rely on breeder reactor developers for the esoteric knowledge needed to analyze accidents. This has consequences for regulating complex hazardous systems wherever they occur, not just nuclear reactors in India. As far as India's fast breeder reactor program is concerned, these challenges seem to offer little grounds for optimism that future reactor designs will be regulated more rigorously.

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Notes

1. CDAs can occur in any reactor that works on the basis of fast neutrons, that is, neutrons that do not get slowed down by design before continuing the chain reaction, and not just in those fast neutron reactors called fast breeder reactors. In all these reactors, there are two other complicating factors. First, because of the higher levels of fissile material, the core can become critical even after undergoing an accident. Second, these events could take place over much shorter time scales. While the core melting seen at Fukushima took place on the scale of hours, core disruption in fast reactors can take place within seconds or shorter.
2. The term explosion is used to mean a rapid release of energy; the magnitude of the energy released in such explosions is orders of magnitude smaller than a nuclear weapon explosion.
3. For perspective, 1 kg of TNT upon exploding releases approximately 4.2 MJ of energy.

4. For example, a group of senior designers stated in their analysis of the consequences of a CDA, 'Core Disruptive Accident (CDA) resulting in core melt down is a very low probability event in an FBR and hence it is considered as a Beyond Design Basis Event' (Chellapandi et al. 2003, 1).
5. According to the official history of the AERB, 'Proceedings in the PDSC (Project Design Safety Committee of the AERB) were often marked by intense debates arising from differences in the design approach followed in PFBR and PHWR. For instance, it took several sessions to accept the proposition that hypothetical core disruptive accident (CDA) in PFBR is a beyond design basis accident (BDBA)' (Sundararajan, Parthasarathy, and Sinha 2008, 65).
6. The IGCAR director was speaking at an IAEA workshop on Operational and Safety Aspects of Sodium Cooled Fast Reactors. PHWR stands for pressurized heavy water reactors, the kind of reactor that constitutes the overwhelming majority of India's nuclear reactor fleet. In other words, the IGCAR official was saying that AERB officials were really suited only to regulate PHWRs and not the PFBR.
7. There are major problems with the probabilistic risk assessment approach used to calculate these figures (Ramana 2011). Many of these problems have been identified and acknowledged for decades, and there is growing realization that there is no approach that would overcome all of these problems. At the same time, experts do find ways of 'disowning' failures of probabilistic safety assessments using a variety of narratives (Downer 2014).
8. A corollary of this observation is that what is considered design basis and what is considered beyond design basis is to some extent a matter of judgment.
9. As the NRC explained, 'Two sets of hazard curves are used (and reported separately) in the seismic analysis. One set was prepared by Lawrence Livermore National Laboratory under contract to NRC. Analysis performed using these hazard curves … suggest that relatively rare but large earthquakes contribute significantly to the risk from seismic events. A second set of hazard curves was also prepared for sites east of the Rocky Mountains for the Electric Power Research Institute … Although both projects made extensive use of expert judgment and formal methods for obtaining these judgments …, there were some important differences in methods. Nonetheless, the NRC believes that at present both methods are fundamentally sound' (NRC 1990, 1–4).
10. David Lochbaum of the US based Union of Concerned Scientists lists numerous instances where PRAs for technically identical reactors come up with vastly different results on key measures like the CDF (Lochbaum 2000, 13–17).
11. This was alluded to by the former director of the US Environmental Protection Agency, William Ruckleshaus, who said 'We should remember that risk assessment data can be like the captured spy: if you torture it long enough, it will tell you anything you want to know' (Percival and Alevizatos 1997, 339).
12. AERB regulations do not require an early submission of the PSA.
13. This does not mean that external events have not been considered while designing the PFBR. Some of them have (Raghupathy et al. 2004), but the probabilistic safety assessment does not include the cases that are triggered by external events.
14. The 1975 Reactor Safety Study, which was led by Professor Norman Rasmussen of the Massachusetts Institute of Technology, was among the earliest prominent applications of PRA techniques to nuclear safety.
15. There are other determinants of the adequacy of the containment to withstand accidents, for example, its volume.
16. The metric involves the ratio of two quantities. The numerator is the product of two measures of the ability of the containment to withstand an accident, its design pressure and its (large) volume. The denominator is the power level of the reactor. It is expected that the energy that would potentially be released during an accident would increase with the power rating of the reactor.
17. Other mechanisms such as 'cultural capture' and 'cognitive capture' have also been suggested, especially in the case of the financial industry and its regulation (Wilmarth 2013, 1417).
18. As the Fukushima Nuclear Accident Independent Investigation Commission's Official Report to Japan's Diet put it, 'The TEPCO Fukushima Nuclear Power Plant accident

was the result of collusion between the government, the regulators and TEPCO, and the lack of governance by said parties. They effectively betrayed the nation's right to be safe from nuclear accidents' (Fukushima Nuclear Accident Independent Investigation Commission 2012, 16).

19. Charles Perrow illustrates this with the case of the Occupational Safety and Health Administration (OSHA) inspection of the Union Carbide chemical plant in Institute, West Virginia (Perrow 2011). In December 1984, following the accident at the Union Carbide plant in Bhopal, India, at a Congressional Hearing, OSHA officials maintained that the plant in West Virginia was safe (Shabecoff 1984). It was only after an accident at the plant on 11 August 1985 that OSHA found plant supervisors failing 'to follow standard company procedures and to properly monitor chemicals in storage' which resulted in the 'overheating of a reactor tank and the discharge of toxic gases' (Jackson 1985).
20. This was the case also in the early years of the nuclear program in the United States because there were no separate business interests to 'capture' governmental actors (Duffy 1997, 20).
21. The World Association of Nuclear Operators (WANO) largely focuses on safety in operations at nuclear reactors after they have been commissioned.
22. The rhetorical vision of generating risk-free energy because of reactors that are said to be 'inherently safe' is also widely prevalent in the case of small modular reactors (Sovacool and Ramana 2015).
23. The overconfidence demonstrated by the breeder community is mirrored by many nuclear designers' beliefs about the safety of nuclear power plants. As a nuclear engineering consultant pointed out in the aftermath of the 1986 Chernobyl accident: 'At some point in their careers, most nuclear engineers and scientists have had to resolve in their own minds the awful dilemma posed between the potentially catastrophic effects of a serious accident at a nuclear power plant and the remoteness of the possibility of its occurrence. Most have squared the circle by concluding, consciously or otherwise, that the remoteness of the chance of an accident was such that serious accidents likely *would* never happen and even *could* never happen. Even if one did happen, it would not be at your plant' (Davies 1986, 59).
24. On the problems related to accountability in the Indian nuclear program, see, for example Sharma (1983), the essays in (Gaur 1996), Ramana and Suchitra (2009), and Ramana (2012).

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Appendix 1.

The calculation of the 100 MJ value for the maximum possible energy release involves two crucial assumptions: that only about half the core can melt and collapse in a CDA, and that only one percent of the thermal energy produced during the accident is converted to mechanical work, that is, explosive energy (Singh and Harish 2002). Not only are both these assumptions doubtful, but the effects on the containment are also very sensitive to alternate assumptions.

It is not possible to conduct empirical tests of the effects on a reactor in a severe accident when safety systems also fail. Therefore, designers and engineers rely on mathematical models to study such accidents. Modeling severe nuclear accidents is a challenging task, as there are several interacting physical processes involved during any such accident. Many of these processes are difficult to represent precisely in models, especially once fuel has melted (as it would in a CDA) and the geometry of the core is changing. All this makes the results of severe accident modeling quite uncertain (Bell 1981).

To calculate what happens within the reactor core and pressure vessel immediately after an accident and how the fuel and coolant behave, IGCAR uses an internally developed mathematical model. Using this model, the DAE estimates that following a loss of coolant accident, at most about half the core could melt (Singh and Harish 2002). It would be conservative to base containment design on such a result if such an estimate was accurate and without uncertainty, or alternately if the outcome were demonstrably insensitive to changing assumptions in the model. Neither of these conditions holds for the PFBR. Not only is energy released in a CDA very sensitive to the assumption of how much of the core collapses in the accident, but there also are omissions in the analysis so that the accident could be considerably worse than asserted by the DAE. Broadly, these omissions involve ignoring physical process or changes to the fuel as they are progressively irradiated in the core, and these changes could make the accident progress further – rendering inadequate the DAE's conclusions about how much of the core can melt and thus participate in a CDA.

The second crucial assumption made by PFBR designers is the fraction of the heat produced during a CDA that would be converted to explosive energy, and this quantity is also subject to significant uncertainties. Small-scale experiments attempting to reproduce this aspect of an accident situation have found that as much as 4% of the thermal energy might be converted into mechanical energy (Berthoud 2000), much higher than the 1% conversion efficiency assumed in safety studies of the PFBR. In reactor safety calculations in France for the PHENIX fast breeder reactor and in the United States for the Fast Flux Test Facility (FFTF), another fast reactor, conversion efficiencies of 5–10% have been used in analyses (Wilson 1977). Using such values would lead to much higher explosive energies in a CDA than the maximum value of 100 MJ stated by the DAE.