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# Kudankulam Nuclear Power Plant: How Safe is Safe Enough?

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With India's first 1,000 MWe light water reactors at Kudankulam in Tamil Nadu attaining safe criticality on July 13, 2013, the stage is set for electricity generation. While it may take another 30 to 40 days for Kudankulam Nuclear Power Plant (KKNPP) to fully synchronise with the power grid and generate electricity; the criticality of the reactor is an important milestone achieved by the Nuclear Power Corporation of India Limited (NPCIL). This milestone, however, has come in the wake of significant public opposition generated after the Fukushima crisis in Japan. Local people at Kudankulam, who laid an eight-month siege to the plant, expressed serious concerns over the safety of the reactors in the wake of the Fukushima nuclear disaster.

To allay the fear of local peoples over safety of the plant, the Atomic Energy Commission had set up an expert committee which concluded in its report that KKNPP meets with all current safety requirements and is thus safe for operation. The committee which held several rounds of discussions with the local people's representatives, however failed to break the impasse over the KKNPP. Similarly, in responding to one of the petitions seeking a moratorium on Kudankulam project, the Supreme Court unambiguously stated that, "apprehension, however legitimate it may be, cannot override the justification of the project". Nevertheless, doubts are continued to be raised about the safety of VVER reactors at Kudankulam plant. The persistence of safety related concerns thus raises pertinent questions about 'how safe is safe enough' for the Kudankulam reactors? And more importantly, what do nuclear risk assessment methods tell us about the possibility of future accidents at KKNPP?

## Reactors and Safety

How safe is safe enough for a nuclear reactor? From the very outset, the engineers and technicians have relied on long-established engineering methods of designing structures and machines to prevent accidents. The potential danger of fuel meltdown and radioactive releases from reactor accidents were identified well before the first commercial nuclear reactor became operational. To deal with various types of risks and failures in nuclear reactors, the reactor designers have evolved two important approaches: deterministic and probabilistic. The deterministic approach involves *pre-determination of accidents* through effect chains, understanding their consequences and developing safety systems and enables engineers to anticipate the physical causes of system failure and prevent such failures by introducing redundancy, duplication, or strengthening of crucial system elements. The probabilistic approach, on the other hand, involves *estimating the likelihood of an accident*, using mathematical calculations of the probability that systems and subsystems in a reactor would fail. More fully developed in the 1970s, this approach was called as 'probabilistic risk assessment' (PRA).

The deterministic and probability approach widely complements each other in designing and operation of nuclear power reactors. A reactor is first designed using the deterministic logic, and probabilistic calculations are then made to confirm the robustness of the design, operational weaknesses (if any) and to address these weaknesses if necessary. In addition to these safety approaches, the nuclear designers and operators have adopted various qualitative and quantitative safety criteria in designing and operating a nuclear reactor. Although, the qualitative criterions adopted by regulatory agencies declaratory in nature, they are mainly aimed at 'preventing unreasonable risk to the public and the environment and ensuring that the use of nuclear energy must be safe.' The quantitative criteria on the other hand assigns specific 'numeric safety values' using probabilistic approach that the designers and operators must meet while designing and operating a nuclear reactor.

The probabilistic and qualitative safety criterions have been progressively adopted by regulatory bodies and utilities all over the world. The application of these safety criteria ensures that the likelihood of accidents with serious fuel meltdown is low, and the potential radiological fallouts from such accidents are limited. The probabilistic safety criteria mainly involve assessing compliance for two important metrics such as: Core Damage Frequency (CDF) and Large Early Releases Frequency (LERF). The CDF refers to the ability of the NPPs to prevent accidents. It is a sum of frequencies of all event sequences that could cause serious damage to the reactor core. The LERF is an estimate of the frequency of those accidents that would lead to large radioactive releases into the atmosphere. An accident involving large scale radioactive release requires implementing the off-site emergency arrangements. Typically, the release is of the order of (a) more than $1x10^{14}$ Becquerel of cesium-137 into the atmosphere due to core meltdown or (b) as a fraction of the inventory of the core. The radioactive releases from Fukushima crisis were almost 10 percent of the total reactor core inventory, whereas the Chernobyl accident released 20-40 per cent of Caesium-37 and 50-60 per cent of I-137 of the total reactor core inventory. The definition of the criterion for CDF and LERF differs considerably with the reactors technologies but it is mostly expressed as a single value.

In the US, the Nuclear Regulatory Commission which introduced the PRA method in 1975, requires that calculated core damage frequency (CDF) for NPPs is of the order of $1x10^{-4}$/RY which means that probability for the event of reactor core damage in one reactor year is one in 10,000. Subsequently, the US nuclear industry through a ‘Requirement Document’ issued by the Electric Power Research Institute in 1990, adopted a core damage frequency limit of 1x10-5/RY for future Light Water Reactors. The International Atomic Energy Agency (IAEA) too established a group in 1985, specifically tasked with considering matters of nuclear safety called as ‘International Nuclear Safety Advisory Group’ (INSAG). In its October 1999 Report (12) on safety principles, INSAG set a severe core damage frequency criteria of $1x10^{-4}$/RY for existing reactors. It further suggested that the application of appropriate safety principles and objectives to future plants “could lead to achievement of an improved goal of not more than $1x10^{-5}$ severe core damage events per plant operating year.”

While the INSAG report did not assign any quantitative value to large early radioactive frequency; it suggested that “an objective for future reactors is the practical elimination of accident sequences that could lead to large early radioactive releases.” Although, the total elimination of events that could unleash large radioactive fallout into the atmosphere may not be possible, numerically it calls for achieving LERF as low as achievable in design and operation of NPPs. In many countries the LERF is adopted below or equal to $1x10^{-6}$/RY.

In accordance with the prevailing international best practices in reactor safety and the INSAG criteria, the AERB in India has adopted the quantitative values for CDF and LERF for new reactors designs as $1x10^{-5}$/RY and $1x10^{-6}$/RY, respectively. The aforementioned safety criterions approved by the AERB underscores its appraisal of what constitutes a ‘minimal risk’ for new nucler plants in the country. The AERB and NPCIL have developed dedicated expert groups for conducting PRA studies and the probabilistic assessment for design basis events have been carried out for all operating NPPs.

**KKNPP: Meeting the Safety Criteria**

The twin VVER-1000 reactors that have been built at Kudankulam belong to family of Russian designed Pressurised Water Reactors (PWRs). The first VVER-1000 reactor was commissioned at the Novovoronezh, Russia in 1981. There are 11 VVER-1000 reactors currently operational in Russia, and seven more in countries such as Bulgaria, Czech Republic, China and Iran. The US-NRC in collaboration with the Russian Federal Nuclear and Radiation Safety Authority, Gosatomnadzor, carried out probabilistic risk assessment for VVER-1000 reactors of the earlier design (V-338) operational at the Kalinin plant during 1995 to 2004. This study estimated the probability of reactor core damage accident for Kalinin Unit 1 from internal initiating events (IEs) as $2.39x10^{-4}$/RY.

The VVERs at Kudankulam are the modified versions of another Russian export design (V-392) with several enhanced safety features, which brings it on par with the IAEA's Generation III category of reactors. The newly incorporated safety features in KKNPP includes four safety trains instead of three, passive heat removal system, higher redundancy for safety system, double containment, additional shut down systems like quick boron and emergency boron injection systems, core catcher in the unlikely event of fuel melt-down, passive hydrogen re-combiners inside the containment etc.

These new safety features taken into account in the design drastically improves the safety of VVER reactors at Kudankulam. The PRA study conducted by the NPCIL has calculated the CDF value for KKNPP Unit I as $1x10^{-7}$/RY which means the probability of serious core damage accident at Kudankulam is one in one million reactor years. Statistically, this designed CDF value increases the safety of these reactors at least by a factor of 100 compared to earlier reactor designs. Similarly, the lower quantitative value of LERF for KKNPP as $1x10^{-9}$/RY is almost close to zero as can be meaningfully considered. Thus, one can surmise that, KKNPP is one of the safest, if not the safest reactors operating in the world today.

*Views expressed are of the author and do not necessarily reflect the views of the IDSA or of the Government of India.*