REPORT

OF

AERB COMMITTEE TO REVIEW

SAFETY OF INDIAN NUCLEAR POWER PLANTS

AGAINST EXTERNAL EVENTS OF NATURAL ORIGIN

August 2011

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Annexures

August 31, 2011

REPORT OF AERB COMMITTEE TO REVIEW SAFETY OF INDIAN NUCLEAR POWER PLANTS AGAINST EXTERNAL EVENTS OF NATURAL ORIGIN

1. Background

The Great East Japan Earthquake of magnitude 9 that occurred on March 11, 2011 generated a series of large tsunami waves that struck the Fukushima Dai-ichi and Fukushima Dai-ni nuclear power plants (NPPs) on the east coast of Japan. As per reports from the Japanese authorities, magnitudes of both the earthquake and the tsunami were beyond the design basis for these NPPs. The flooding of the site due to tsunami waves incapacitated the power supplies which led to a nuclear accident encompassing units 1, 2 and 3 of Fukushima Dai-ichi NPP. Cumulative impact of the accident was unprecedented in terms of damage to the reactor cores in units 1, 2, and 3 (core of unit-4 had been unloaded at that time into the spent fuel pool) and impairment of cooling to the irradiated fuel in the spent fuel storage pools of all the four units. Four hydrogen explosions also occurred that caused damage in the reactor buildings of units 1, 3 and 4 and the wet well of unit-2. The accident resulted in significant release of radioactivity into the public domain, including to the sea, and emergency measures like evacuation of people in the vicinity of the NPP and some restrictions on consumption of food items etc. had to be implemented. It took a large effort over several days before units-1, 2 and 3 could be stabilized and cooling of fuel in the storage pools could be restored. The accident was rated at level-7, the highest level of the International Nuclear and Radiological Event Scale. There was no significant adverse impact on the safety of units 5 and 6 of the Dai-ichi NPP and the 4 units of Dai-ni NPP as well as on the Onagawa NPP (3 units) and the Tokai NPP (1 unit) located nearby.

Chairman, AERB constituted a committee on March19, 2011 to review the safety of Indian NPPs against external events of natural origin, in the light of the Fukushima accident (annexure-1).

2. Committee's work approach

Work plan

After review of the various reports on the Fukushima accident it was realized that from the information available presently it is not possible to draw any firm conclusions on the causes of the accident in terms of design shortcomings and/or inadequacies in procedures or in their implementation (a brief description of the Fukushima accident is given in annexure-2). However it was apparent that the accident was mainly caused by

- Severe flooding caused by the beyond design basis tsunami, and
- Consequent prolonged station black out (SBO) i.e. loss of off-site as well as on-site AC power supplies at the NPP.

Accordingly, the committee drew up its work plan focusing on these two major areas viz. Beyond Design Basis Events (BDBE) of natural origin and prolonged SBO. The current regulations of AERB on the safety of NPPs against natural events (annexure-3) were kept in view while drawing up the work plan. The committee's work plan is given in annexure- 4.

Work methodology

The committee constituted expert working groups to

a) Develop the guidelines for deciding on the magnitude and related issues concerning beyond design basis external events of natural origin,

b) Review all relevant aspects related to external events of natural origin and prolonged SBO for the boiling water reactors of TAPS-1&2

c) Examine all relevant aspects related to external events of natural origin and SBO for the Pressurized Heavy Water Reactor (PHWR) based NPPs. In this context Working Groups were constituted to conduct reviews in the following areas:

   1. Station blackout & ultimate heat sink
   2. Electrical systems
   3. Control & instrumentation
   4. Spent fuel storage facilities.
   5. Heavy water upgrading plants

d) Examine the safety issues related to radioactive waste disposal facilities at the NPPs under extreme natural events, and

e) Examine the severe accident management guidelines for NPPs.

Summarized interim reports of the working groups are given in annexure-5. Reports of the working groups were discussed in detail in eight meetings of the committee to arrive at the review findings (annexure-6) and the committee’s conclusions & recommendations. Information contained in the report of the IAEA Fact Finding Mission that visited the Fukushima NPP during May 2011 was also kept in view while arriving at the conclusions and recommendations.

3. Observations

3.1 Use of operating experience feedback for enhancing safety of NPPs has all along been accorded high importance in India. Lessons learned from Three Mile Island and Chernobyl accidents and from several other operational incidents in NPPs in India and abroad, as also from new knowledge gained through research have been appropriately used for design and procedural improvements to enhance safety of our NPPs. The present exercise of reviewing the safety of our NPPs in the light of the Fukushima accident is to be viewed in this context.

3.2 As per AERB regulations, 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. In addition, operating personnel have to periodically renew their licenses after clearing simulator training, written examination and interview by an expert committee. In some other NPP operating countries there is no requirement of engineering degree as educational qualification for reactor operators. Thus the NPP operators in India are better placed to handle off-normal situations in the plant compared to their counterparts in several other countries.

3.3 All NPPs in India undergo Periodic Safety Reviews (PSR) following the procedure prescribed in AERB regulations. The PSRs comprise of a detailed design and operational safety review that is conducted every 10 years and a brief but comprehensive review which is done every 5 years. For older NPPs special safety reviews have been carried out when they were approaching the end of their originally proclaimed design life, which is based on the plant’s estimated economic life or, in some cases, on technical considerations as understood at the time of their design. These reviews are done based on current safety standards.

3.4 A large number of safety upgrades have been implemented over the years in our NPPs, especially in the old units, based on the outcome of the various safety reviews mentioned above. Some upgrades have been made on the basis of seismic reevaluation of the structures, systems and components (SSCs) of the NPPs and the outcome of studies on potential flooding due to postulated failure of dams upstream of the water reservoirs at the NPP sites. The committee noted that these safety upgrades have substantially enhanced the safety of our NPPs including their capability to withstand natural events.

3.5 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. However in the light of Fukushima experience it is considered prudent to further enhance this conservatism and also postulate the magnitude of beyond design basis natural events.

3.6 The submarine faults capable of generating tsunamis are located at very large distances of more than 800km from the Indian coast. Thus, unlike in the Fukushima case, the possibility of simultaneous occurrence of an earthquake and a tsunami at our NPPs, is almost non-existent.

3.7 A major lesson learned from the Fukushima experience is that capability to cool irradiated fuel in the reactor core and in the spent fuel storage facilities must remain available in the event of prolonged SBO as also in the face of beyond design basis natural events.

3.8 PHWRs form the back bone of the current Indian nuclear power programme. In this design, cooling of the reactor core, with the plant in hot shut down state, is achieved by natural convection flow of reactor coolant through steam generators. With the design provision for charging water to the secondary side of the steam generators using diesel engine driven pumps, this mode of core cooling can be maintained even under extended SBO. The efficacy of this design feature got amply demonstrated during the 17 hours long SBO caused by the turbine hall fire incident at Narora unit-1 in 1993 when reactor core cooling could be successfully maintained.

3.9 In the case of TAPS-1&2, which is a boiling water reactor (BWR) based NPP, core cooling under SBO can be maintained up to about 8 hours by natural convection circulation of reactor coolant through the emergency condenser. Heat from the coolant is removed by boiling of water present on the secondary side of the emergency condenser. The inventory of water on the secondary side of the emergency condenser is sufficient for cooling the core for about 8 hours only and thereafter it must be made up.

3.10 The heat load from irradiated fuel stored to design capacity in the spent fuel storage pools is much less and the inventory of water in the pools much larger at our NPPs in comparison to the corresponding heat load and water inventory in the spent fuel storage pools at Fukushima NPP. Consequently, for the Indian NPPs, submergence of the fuel in the pool water is assured for a time period of at least one week under SBO, even with the most conservative assumptions on the quantum of decay heat from the stored fuel and without any credit for operator action.

4. Conclusions and recommendations

External events

4.1 Though AERB regulations with respect to design basis for external events are sufficiently conservative, it is 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.

4.2 Regulatory requirements on derivation of seismic design basis ground motion (DBGM) should address the limitations of current methodologies because of lack of sufficient and relevant earthquake data and other uncertainties concerning site tectonics. The AERB guide on seismic studies and DBGM may be revised accordingly.

4.3 It is observed that seismic signal based automatic reactor trip is presently provided in NAPS and KAPS only. In other operating units, seismic alarms are provided and in the event of an earthquake the reactor has to be tripped manually. Seismic signal based automatic reactor trip should be implemented in all reactor units. Also, seismic switches and sensors that are located outside the reactor buildings should be protected against any flooding at the site.

4.4 The Fukushima accident has shown that occasionally the magnitude of natural events can be higher than what is considered in design. It is therefore prudent to make additional design provisions such that at least the basic safety functions for the NPPs are not impaired even under beyond design basis natural events (or extreme events). Towards this aim it is recommended that the parameters for each postulated extreme natural event be defined conservatively using the best available analytical methods. While design basis external events should govern the design of SSCs, functionality of the most safety relevant SSCs should still be maintained under extreme events.

4.5 In spite of the conservative estimates of the design basis external events of natural origin, there is a residual risk of exceeding these estimates. While absolute quantification of beyond design basis events is not feasible, their probable magnitudes should still be defined for safety margin assessment. The expert group on external events, after detailed deliberations on the physical bounds of the underlying parameters as currently understood, and the inherent uncertainties, has recommended interim measures for evaluating extreme natural events other than earthquake and tsunami. These recommendations are provided in annexure-7. For assessment of extreme tsunami event it is recommended that all physically possible combinations and variations of tsunamigenic source parameters and accurate near shore data should be considered.

The group is working on developing methodology to define extreme earthquake event at a level higher than SSE. This work should be completed at the earliest possible.

4.6 Assessing the magnitude of extreme natural events is a highly challenging task due to the inherent uncertainties involved, especially in respect of tsunami wave heights. In this context the committee noted that a detailed exercise is in progress at AERB, using a computer code for analysis that is validated using the data from the 2004 Indian Ocean tsunami, for estimating the maximum tsunami wave heights that can possibly be generated from the sub-sea faults around the Indian coasts. This work should be completed expeditiously.

The work done so far indicates that the maximum postulated flood level at Kalpakkam coast is likely to get revised upwards and consequently the corresponding design improvements for MAPS will have to be considered. The flood level assessment is based on a tsunami generated from a sub-sea earthquake caused by the Andaman-Nicobar-Sumatra fault and takes into account, in a most conservative manner, the fault parameters and the directivity of tsunami propagation towards the Kalpakkam coast. The Prototype Fast Breeder Reactor at this site is likely to remain unaffected due to this revision as its grade level is sufficiently high. For all other coastal NPP locations there will be no change in the maximum postulated flood level.

4.7 Design provisions should be made to ensure safety even for the conservatively estimated magnitude of extreme events without any unreasonable demand on operator actions. For example, provision of air cooled diesel generators (DGs) capable of remaining operational even under extreme events, and, portable power packs that could be easily hooked up at pre-identified points, to supply back up power for performing essential safety functions and obtaining information on important safety parameters, could be considered as a further measure of defense in depth.

4.8 A beyond design basis external event may disable the facilities available at the NPP site for monitoring and control of important reactor parameters. It may also result in physical isolation of the site such that it may not be possible to receive outside help for a considerable period of time. Creation of an emergency facility at each NPP site which will remain functional under such conditions is therefore recommended. The facility should have adequate radiation shielding and should be seismically qualified. It should also have provisions for communication with relevant agencies and for obtaining information from all units at the site to help decide on further course of actions, as also for food, resting etc. for essential personnel for a period of about one week.

PHWRs

4.9 As already stated, the PHWR based NPPs, which account for 18 of the 20 operational NPP units in India presently, have a distinct advantage in respect of core cooling capability under SBO. This is because core cooling in these reactors can be sustained under SBO condition by natural convection flow of the reactor coolant through steam generators (SGs). Heat from the SGs is removed by boiling of water on their secondary side and the steam so produced is discharged to the atmosphere. Design provisions exist for charging water to the SGs by diesel engine driven pumps, without any need for electric power, and the water inventory available for this purpose, without any replenishment, is sufficient for more than one week's requirement.

4.10 To ensure natural convection flow of the reactor coolant through SGs, any significant voiding in the reactor coolant system should be prevented, which may appear in the system if losses due to leakages from the system are not made up. It is therefore recommended that a reliable back-up provision should be made for PHT make-up during extended SBO.

4.11 Presently the safety analysis has been done for SBO duration of 24 hours. This should be extended to the period beyond 24 hours. Temperature rise of moderator and vault water during SBO beyond 24 hours should also be assessed and means to limit these temperatures should be provided.

4.12 Assured operability of the fire water system during extended SBO is extremely important. Towards this, instituting periodic maintenance and surveillance programme on fire water system piping is recommended. The fire water pumps should be qualified for sustained operation by endurance testing. The starter batteries and their chargers for these pumps should be relocated at a higher elevation to protect them against flooding. Logistics for manual pumping of fuel to diesel operated fire water pumps in case of non-availability of the normal pumps need to be confirmed. Similarly the vent lines of the underground diesel storage tanks should also be suitably raised to prevent any water ingress into the tanks during flooding of the NPP sites.

TAPS -1&2

4.13 In the case of the boiling water reactors of TAPS-1&2, core cooling under SBO can be maintained for about 8 hours by natural convection cooling of reactor coolant by the water present on secondary side of the emergency condenser. To ensure reactor cooling in this mode beyond 8 hours, back up provisions should be made for replenishing loss of inventory by injection of water to the reactor coolant system as well as to the secondary side of the emergency condenser.

There is also a need to enhance compressed air back up to the relief valves in the auto blow down system (ABDS), to ensure their operability to depressurize the reactor during extended SBO. Presently operation of these valves during SBO is possible for a limited number of operations, till their local compressed air accumulators drain out.

4.14 Some of the safety systems including class III power supply system in TAPS-1&2 are located below the revised reference flood level for the site and therefore external flooding at TAPS has the potential to cause SBO. The equipments for emergency core cooling and filtered containment venting are also located below the revised reference flood level.

A detailed study is hence necessary to identify the design improvements required to ensure availability of the above systems during external flooding and the requisite corrective actions should be implemented at the earliest.

Interim arrangements such as alternate means to inject water into the secondary side of the emergency condenser and to the reactor coolant system should be considered if the permanent solutions are likely to take considerable time for implementation (NPCIL has recently informed that these interim arrangements have already been made).

4.15 There are also certain issues with regard to spent fuel pool integrity and pool make up capability for TAPS-1&2 consequent to a beyond design basis seismic event, such as integrity of fuel pool gates, loss of pool water due to sloshing and operability of service/demineralised water pumps for pool water make up. These pumps are also located below the revised reference flood level. All these issues should be examined in detail and appropriate modifications carried out to enhance safety margins and availability of essential equipment under such severe external events.

KKNPP

4.16 The KKNPP units-1&2 are now in advanced stages of construction with initial commissioning activities for unit-1 already initiated. The safety review of KKNPP, in the light of Fukushima accident, is being done by the AERB's Advisory committee on project safety review of light water reactors (ACPSR-LWR). However for the sake of completeness, a short note on the subject is attached (annexure-8).

It can be seen that KKNPP design already has several advanced safety features including those for ensuring safety against external events of natural origin and for management of design basis as well as beyond design basis accidents. However as a matter of abundant caution a review is being done in the wake of the Fukushima accident to identify further improvements if any that need to be made.

Spent fuel storage facilities

4.17 For the spent fuel storage pools at our NPPs it is seen that the stored fuel remains submerged in water for a period ranging from 9 to 16 days in the older plants viz. TAPS-1&2, RAPS-1&2 and MAPS-1&2, and for over one month in other plants, without any cooling or addition of water. Nevertheless an external water hookup provision for charging water in the pools for all the operating plants should be implemented. This make up capability should remain unaffected by the external events and SBO. Provision for monitoring the level and temperature of pool water and radiation fields inside the spent fuel storage buildings under SBO should also be made.

4.18 Detailed site specific safety assessment of spent fuel storage bays should be carried out with respect to structural integrity and leak-tightness of pools, loss of pool water from sloshing and, stability of fuel racks and mechanical handling equipment in case of extreme earthquake event.

Severe accidents

4.19 In spite of all the safety features provided, the extremely remote possibility of an accident leading to partial or total melting of fuel in the reactor core due to unforeseen reasons should still be deterministically taken into consideration. Provisions for management of such an accident, termed as severe accident, need to be made such that the operators are able to control its progression and mitigate its consequences in terms of preventing, or at least minimizing, any significant adverse impact in the public domain.

4.20 In the area of severe accident management significant progress has been made in our country in the recent past in terms of analysis and R&D work. Broad guidelines for management of severe accidents in PHWRs including the management of hydrogen generated from the reaction between overheated fuel & its cladding and the reactor coolant have also been worked out. It is seen that in case of PHWRs severe accident management in terms of arresting the progression of the accident is comparatively simpler. This is on account of the presence of the large quantity of moderator heavy water at low pressure and temperature inside the reactor vessel and the large inventory of water in the vault that surrounds the vessel, both capable of acting as a heat sink to absorb decay heat from the fuel. Therefore the strategy for severe accident management essentially comprises of maintaining sufficient inventory and adequate cooling of moderator and vault water.

The analysis and R&D referred above for severe accident management should be expeditiously translated into design provisions together with related procedures for the operating as well as under construction PHWRs.

4.21 In the case of TAPS-1&2, preparatory work for inerting the primary containment, for management of any hydrogen escaping from the reactor pressure vessel in case of a severe accident, has been taken up. Similarly, work on development of severe accident management guidelines has also been initiated. These tasks should be completed on priority.

Other recommendations

4.22 In case of damage to the off-site power supply lines and to the station switchyard during an external event, it is important that these be repaired and brought back into service at the earliest. Necessary preparedness for this purpose including stocking requisite spares and logistics of obtaining services of expert agencies should be looked into.

4.23 Functioning/operability of all safety related control and instrumentation (C&I) including their backup instrument air accumulators, and integrity of their supports/panels, should be checked for beyond design basis earthquake level, after the relevant parameters for such an event are available.

4.24 The practice of storing spent radioactive ion exchange resins in underground tanks should be discontinued as in case of earthquake or severe flooding this can cause spread of radioactive contamination. The resins presently stored in such tanks at TAPS and MAPS should be appropriately treated and disposed off.

4.25 Functional integrity of radioactive liquid effluent storage tanks and surrounding dykes at NPPs should be assessed under beyond design basis external events and corrective measures implemented as necessary

4.26 Capabilities need to be developed to treat large quantities of liquid waste that may get generated in case of an accident. Large capacity transportable radioactive effluent treatment modules, which can be speedily deployed at any NPP site, could be one possibility.

4.27 Site specific assessment of existing structures and equipment, specifically the tall structural steel towers and distillation columns of the heavy water upgrading plants, should be carried out for postulated level of external events, especially earthquakes, and any impact of their failure on nearby plant facilities should be checked.

4.28 In addition to the recommendations listed above, various other suggestions have also been made in the detailed reports of the working groups. Actions on those suggestions should also be taken as appropriate. In this regard, the committee noted that even while its deliberations were in progress, NPCIL has proactively initiated work towards implementation of the recommendations of the committee and those from its own review and has drawn up an action plan for this work. It is also seen that pending implementation of permanent design improvements which require procurement of materials, components etc. and working out detailed engineering, some interim arrangements for meeting the intent of the recommendations have already been made.

(Dr. I.D. Gupta) (Dr. R. Krishnan) (Prof. C.V.R. Murty)

(On behalf of Dr. B.N. Goswami)

(S.K. Chande) (S.A. Bhardwaj) (Dr. A.K. Ghosh)

(R. Chowdhury) (A.G. Chhatre) (L.R. Bishnoi)

(S. K. Sharma)

Annexure-1

परमाणु
ऊर्जा
नियामक
परिषद

सत्यमेव जयते
भारत सरकार
GOVERNMENT OF INDIA

Atomic
Energy
Regulatory
Board

जय हिन्द

25 Years of
Safety Regulation
1983 - 2008

अध्यक्ष
CHAIRMAN

No.CH/AERB/25/2011/ ९७७   March 19, 2011

Sub: Committee to review the safety of Indian Nuclear Power Plants in the light of earthquake and tsunami in Japan

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The severe earthquake of March 11, 2011 in Japan followed by high intensity tsunami has resulted in damage in a number of reactors at Fukushima Nuclear Power Station. A committee consisting of the following is hereby constituted to review the safety of Indian Nuclear Power Plants in the light of the above incident.

1.	Shri S.K. Sharma, Former Chairman, AERB	- Chairman
2.	Shri S.K. Chande, Vice Chairman, AERB	- Member
3.	Shri S.A. Bhardwaj, Director(Tech.), NPCIL	- Member
4.	Dr. A.K. Ghosh, Director, HS&E Group, BARC	- Member
5.	Prof. C.V.R. Murthy, IIT, Madras	- Member
6.	Dr. I.D. Gupta, Director, Central Water & Power Research Station, Pune	- Member
7.	Dr. B.N. Goswami, Director, Indian Instt. of Tropical Meteorology, Pune	- Member
8.	Shri R. Chowdhury, Former Director, RG, BARC	- Member
9.	Shri A.G. Chhatre, Associate Director (SA&S), NPCIL	- Member
10.	Shri L.R. Bishnoi, Head, S&SED, AERB	- Member-Secretary

The Committee may review

(1) Capability of Indian Nuclear Power Plants to withstand earthquakes and other external events such as tsunamis, cyclones, floods, etc.

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Organisation

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(2) Adequacy of provisions available to ensure safety in case of such events, both within and beyond design basis.

The Committee may co-opt additional members and/or constitute task forces as required.

Since the events at Fukushima are still unfolding, and full analysis and understanding will become available in due course, the Committee may consider issuing an interim report as early as possible followed by a final report subsequently.

SSBajaj

(S.S. Bajaj)

Chairman & Members of the Committee

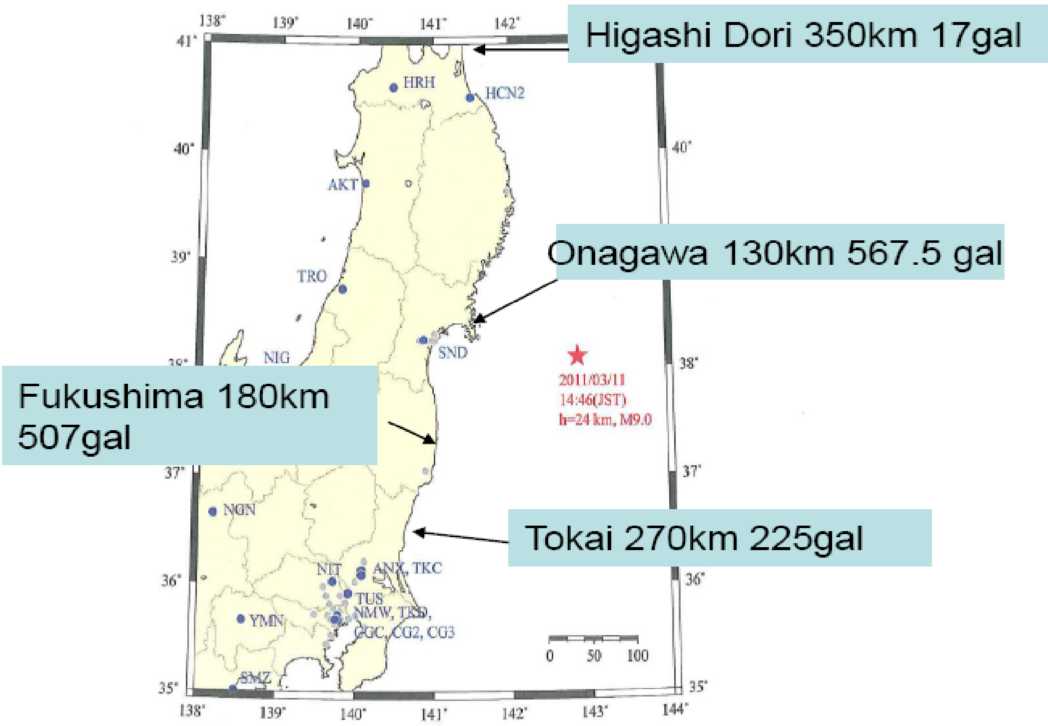
Copy to: Chairman, AEC
CMD, NPCIL
Director, BARC
Vice Chairman, AERB
Secretary, AERB

Annexure-2

BRIEF DESCRIPTION OF THE FUKUSHIMA ACCIDENT

Background

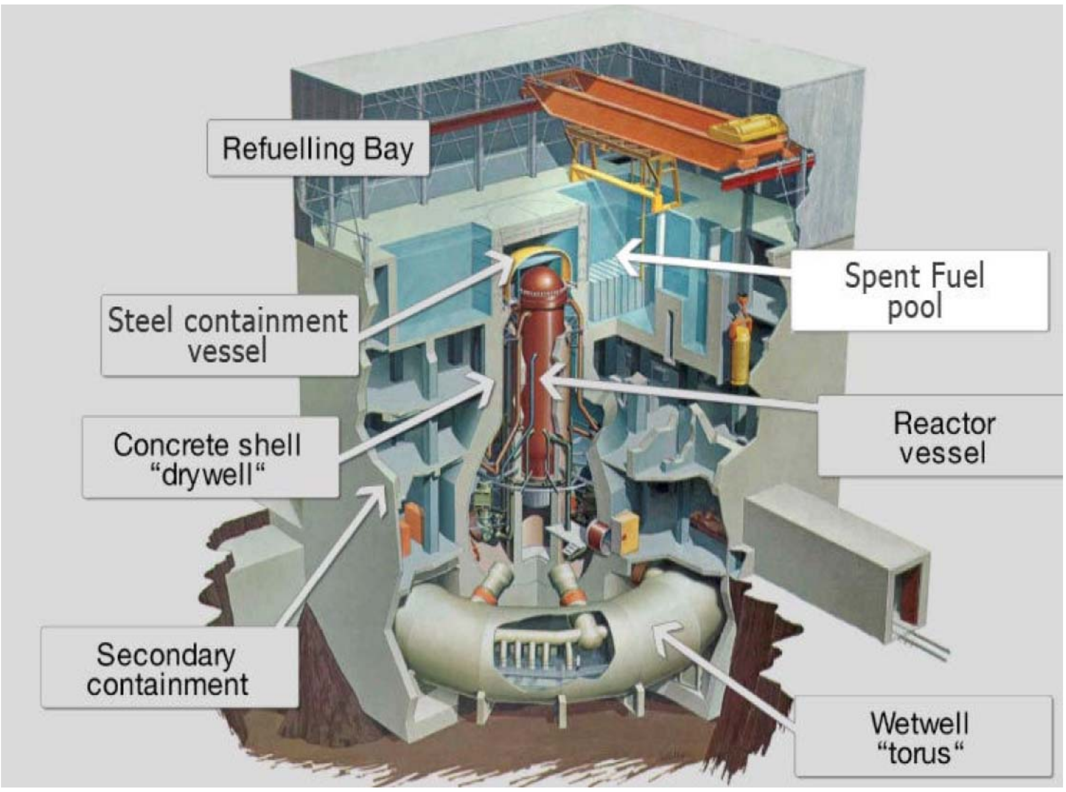
The pacific coast of east Japan was struck by an off shore earthquake of magnitude (M_w) 9.0, the largest in Japan's recorded history, at 14:46 (JST) on March11, 2011. The earthquake generated large tsunami waves that struck the Tohoku district in a series of waves, the highest being 38.9m at Aneyoshi, Miyako. The earthquake and the tsunami caused widespread destruction of life and property. As well as other enterprises, four nuclear power plant (NPP) sites on the northeast coast of Japan were affected to varying degrees by the earthquake and the tsunami. These sites (shown below with distances from the earthquake epicenter and recorded maximum ground accelerations) were Higashi Dori (one unit), Onagawa (three units), Fukishima Dai-ichi (six units) & Dai-ni (four units) and Tokai II (one unit).



Of these fifteen units, eleven that were operating at the time of the earthquake were shutdown automatically on sensing the earthquake. Four units (one unit of Higashi Dori and units 4, 5 and 6 of Fukushima Dai-ichi) were already in shutdown state. The large tsunami waves that followed affected all the NPPs to varying degrees, with most serious consequence occurring at Fukushima Dai-ichi. Further description in this write up pertains to Fukushima Dai-ichi NPPs only.

The Fukushima Dai-ichi Nuclear Power Plant

The Fukushima Dai-ichi NPP site has six Boiling Water Reactor (BWR) units. Unit-1 is 460 MWe BWR-3 with mark-1 containment, units-2 to 5 are BWR-4 reactors of 784 MWe each with mark-1 containments, and unit-6 is BWR-5 reactor of 1100 MWe with mark-2 containment. The reactors were made operational progressively from 1971 to 1979. The main features of typical BWR with mark-1 containment are shown in the figure below.



The earthquake and the tsunami

The seismic source of the magnitude 9.0 earthquake was at latitude 38.1^{o}N, longitude 142.9^{o}E and focal depth was 23.7 km. The main shock was followed by many aftershocks, three among them being of magnitude more than 7.0. The acceleration response spectrum generated at the reference point of the Fukushima Dai-ichi NPPs due to this event exceeded the response spectrum corresponding to the design basis ground motion for the safe shutdown earthquake in some partial frequency bands.

The tsunami generated due to the earthquake hit the Tohoku District in a series of seven waves. The first and second large waves struck the Fukushima Dai-ichi NPP at 15:27 (JST) and 15:35 (JST), 41 minutes and 49 minutes respectively after the occurrence of the earthquake. The maximum height recorded at Fukushima Dai-ichi site was over 14 m. The maximum design basis tsunami height considered for the Fukushima Dai-ichi NPP at licensing stage was 3.1 m. In 2002 it was reassessed as 5.7 m following Japan society of civil engineers (JSCE) method for tsunami assessment of NPPs in Japan.

Fukushima Dai-ichi response to the external events

At the time of the accident, the Fukushima Dai-ichi units 1, 2 and 3 were operating, and units 4, 5 and 6 were shutdown for refueling and maintenance activities. At unit-4, reactor fuel had been offloaded to its spent fuel pool and the reactor vessel was open.

The three operating reactors (units-1, 2 and 3) of Fukushima Dai-ichi were automatically shutdown on sensing the earthquake. All external AC power supplies failed due to damage to the switchyard and collapse of the transmission towers due to the earthquake. The emergency diesel generators started at all six units to supply power to essential loads as designed.

However, the subsequent tsunami inundated the site causing extensive damage to the site facilities and disabled all emergency diesel generators. Only one air-cooled diesel generator at unit-6, located at a higher elevation and which was retrofitted as a backup power supply source, survived. This could be connected to supply AC power to essential loads of units 5 & 6, eventually achieving cold shutdown of these two units.

There was near total darkness at the site after nightfall and hardly any instrumentation and control system functioning to assist the operators. Total loss of all sources of electrical power in units 1 to 4 resulted in Station Black Out (SBO). All pumps including those required for cooling the reactors were inoperable.

The operators struggled for many hours to provide some cooling to the reactor cores of units 1, 2 and 3 without much success. Ultimately core cooling was completely lost in these units. The timing of loss of cooling varied depending on the specific design features of the units and the extent of success of operator actions to utilize available DC power sources of limited capacity. Damage to the nuclear fuel in the core commenced shortly after the loss of core cooling and thus the onset of the nuclear accident.

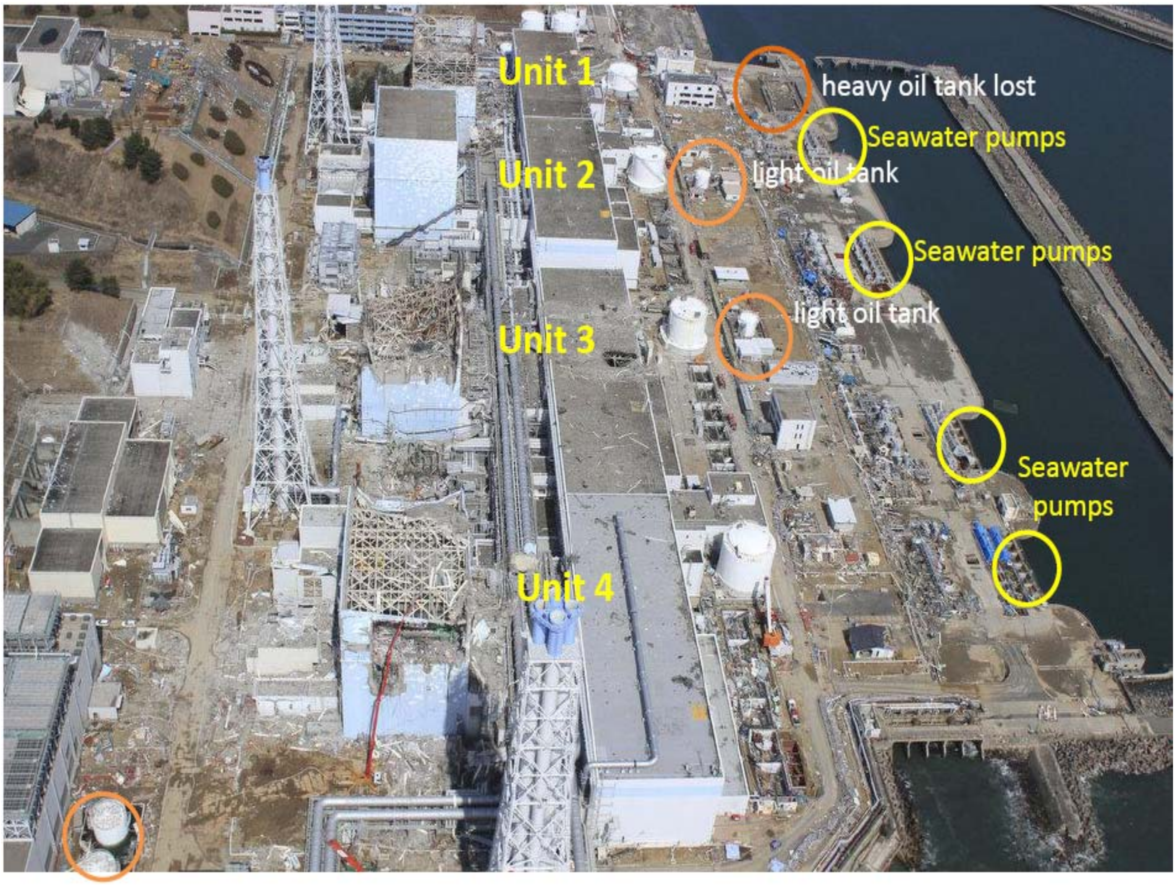
The Nuclear Accident at Fukushima Dai-ichi

The operators then resorted to alternate means of cooling the reactor cores by injecting freshwater or seawater into Reactor Pressure Vessels (RPV) of units 1 to 3 but with little success. The reactor cores finally got exposed leading to core melt. Parts of the melted cores settled at the bottom of the RPVs and are suspected to have caused partial damage to the RPVs. The exothermic reaction between steam in the RPV and zirconium (used in fuel cladding) produced large amounts of hydrogen. This hydrogen along with the radioactive material from the RPVs found its way into the primary containment vessel (PCV) through safety valves causing pressure rise in the PCV.

The gases in the PCV had to be vented several times to the atmosphere through ventilation stack to limit the pressure in the PCV to prevent its failure. However, some amount of gases containing hydrogen and radioactive fission products leaked into the surrounding reactor building. The leaked hydrogen accumulated in the upper part of the reactor building and caused explosions in units 1 and 3.

These explosions totally damaged the operation floor in units 1 and 3 and caused discharge of large amount of radioactive materials into the atmosphere. The hydrogen explosion in unit-2 is presumed to have occurred in the suppression chamber area. The hydrogen from unit-3 had possibly leaked into the reactor building of unit-4, through a common ventilation duct, which caused explosion in unit-4 also and damaged its operation floor. The physical condition of the plants after hydrogen explosions is depicted in the picture below.

The operators were also unable to monitor the water level and charge make up water to the spent fuel pools of units 1,2,3and 4. However, subsequently, water injection to the spent fuel pools was tried with the help of external agencies (viz. the Self-Defense Forces, the Fire and Disaster Management Agency and the National Police Agency) using helicopters and water cannon trucks but this met with little success. Ultimately water charging to the pools could be achieved by deploying concrete pump trucks, injecting freshwater from nearby reservoirs after the initial sea water injection.



Radioactive material leaked into the environment during explosions, during venting of primary containment vessels and also through the water leaked from the reactor pressure vessels during the continued efforts of cooling the reactor cores. Based on the quantity of radioactive material released to the environment, this accident has been rated at level-7, the highest level on the International Nuclear and Radiological Event Scale.

In order to prevent undue exposures, Japanese authorities decided to evacuate the population in the 3 km area around the Fukushima Dai-ichi plant on 11 March. This was extended to 20 km on 12 March based on monitoring and assessments carried out subsequently. On March 15th, the residents living at a distance of 20 to 30 km were advised to stay indoors for sheltering.

In spite of the large releases, there was not a single person seriously affected by radiation (deterministic effect). In fact monitoring of about 2,00,000 members of the public indicated that nobody received any significant dose and no radiological health effects are expected.

Restoration work at Fukushima Dai-ichi after the accident

A road map prepared by TEPCO (the owner of the plant) addresses the three basic safety functions of shutdown, cooling of reactor core and containment of radioactivity. Boron was added in the water that was injected in the reactor vessels in the early phase of the accident to ensure long term sub-criticality. The road map aimed to achieve stable cooling in the first 3 months and a complete cold shutdown condition within 6 months. The stable cooling, characterized by stabilizing the Reactor Pressure Vessel (RPV) bottom head temperature, was achieved by mid-July 2011. Since then, water is being continuously injected to reactor core at a rate 3-5 m^3/hr for sustained cooling. The spilled water is collected, treated and injected back into RPV. For achieving the cold shutdown conditions installation of closed loop system containing pump, heat exchanger and its associated secondary cooling system is under consideration.

For stable cooling of the spent fuel pools, a closed loop cooling system with its associated pump, heat exchanger and secondary cooling system has been installed and made operational for units-1, 2, 3 and 4.

A system of continuous injection of nitrogen gas into the primary containment (PCV) was implemented in units-1, 2 and 3 to obviate possibility of another hydrogen explosion. TEPCO is in the process of installing temporary structural reactor covers on all the affected reactor buildings.

Many other short and medium term restoration measures involving monitoring of reactor parameters, treatment of radioactive water, removal of debris inside the plant, control of contamination spread in ground water and sea water, and radiation dose monitoring outside the plant have been implemented. Long term restoration measures like achieving long term cold shutdown conditions in all the affected units, installation of permanent reactor building covers, protection against stress corrosion cracking of the structural material, installation of supporting structure below the spent fuel pool in unit-4, remediation of contaminated soil, continuous environment monitoring and commencement of work for removing the fuel from the core are being implemented.

Annexure-3

AERB REGULATIONS ON NPP SAFETY AGAINST EXTERNAL EVENTS

AERB was created in 1983 to formally regulate safety in nuclear and radiation facilities in the country. Over the years AERB has evolved a robust procedure for safety review and issue of consents at various stages of setting up of these facilities in line with the best international practices and IAEA guidelines. The main consenting stages for nuclear power plants are siting, construction, commissioning, operation and decommissioning. At each stage a comprehensive review in a multitier structure of safety committees is carried out before issue of consent.

AERB's safety review process

The basic safety philosophy of defense-in-depth is followed in the design of nuclear power plants (NPPs). Important features of this philosophy are overlapping provisions in design and operation to prevent plant failures, detection and intervention of any failures should they occur, provision of multiple barriers against radioactivity releases, and mitigation of the radiological consequence through emergency procedures in case of any releases. All operating plants undergo a periodic safety review (PSR) to address changes in safety standards, ageing effects, and operating experience feedback including those pertaining to external events.

Design of 220 MWe Indian PHWR based NPPs was standardized from NAPS onwards. The currently operating PHWR plants based on this standardized design are NAPS-1&2, KAPS-1&2, KGS-1 to 4 and RAPS-3 to 6. TAPS-3&4, though based on the basic features of standardized Indian PHWR, has been designed to produce 540 MWe. The standard design, which has several improved design safety features, has undergone rigorous safety review. For example, these plants are provided with two failsafe independent and diverse shutdown systems to achieve guaranteed reactor shutdown state with very high reliability, besides reactivity control through reactor regulating system for normal operation. The concrete vault housing the calandria (reactor vessel) is filled with water. This provides a large heat sink against progression of any accident caused by loss of coolant. Availability of large volume of low pressure moderator in the calandria is also an inherent advantage for core cooling in case of an accident.

All the standardized PHWR based NPPs are provided with double containment with inner primary containment of prestressed concrete acting as a primary barrier against release of any radioactivity to the atmosphere in the event of an accident. The primary containment is designed conservatively to withstand a pressure much higher than that estimated during the postulated design basis accident. Thus it can prevent releases even in case of certain level of beyond design basis accidents.

For the plants constructed before NAPS (viz. TAPS-1&2, MAPS-1&2, RAPS-1&2), significant safety improvements have been carried out through back-fits and safety upgrades based on periodic safety reviews and special reviews conducted when these NPPs were approaching the end of their originally proclaimed design life. Major improvements are related to seismic safety, emergency core cooling and ageing management.

India has also witnessed a few significant events at its NPPs, namely a large fire at NAPS in 1993, flooding at KAPS-1&2 in 1994, delamination of the inner containment dome during prestressing operation in Kaiga-1 in 1994 and tsunami at MAPS-1&2 in 2004. Lessons learnt from these events as also from relevant events at NPPs abroad have been incorporated by appropriate improvements in design and operating procedures.

Safety regulations

The major safety regulations of AERB are issued in the form of safety standards comprising of safety codes and safety guides. Developing and updating of safety regulations is a continuing process at AERB. For NPPs, the safety regulations structure corresponds to various stages of consenting, namely siting, design & construction, commissioning, operation and decommissioning. Besides this, standards on the regulatory process have been issued to lay down procedures for issue of consents, design and operational safety review, regulatory inspections, and also for development of safety regulations. Assurance of quality in each of the activities pertaining to NPPs during their life cycle is institutionalized through another set of regulatory standards.

The regulatory standards in India are developed with safety concepts, requirements and methodologies derived from IAEA safety standards and other international nuclear safety regulations, which collectively represent enormous experience in design, construction and operation of NPPs.

AERB follows a three tier system of regulatory standards. Safety codes establish the objectives and set the requirements to be fulfilled to provide adequate assurance for safety. Safety guides elaborate on the requirements specified in the codes and describe the approaches for their implementation. Safety manuals deal with specific topics and include detailed scientific and technical information on the subject. These standards are prepared by experts in the relevant fields and are extensively reviewed by advisory committees of AERB before their publication. The standards are periodically reviewed and revised as necessary, in the light of experience and feedback from users as well as new developments in the field.

The list of regulatory standards developed by AERB is exhaustive covering all facets of nuclear power plant safety and is available on AERB website. A selective list of standards pertaining to NPP safety against external events is given below.

AERB Safety standards on NPP safety against External Events

S.No.	Identification Number	Title
1.	AERB/SC/S	Code of practice on safety in Nuclear Power Plant Siting
2.	AERB/SG/S-1	Meteorological Dispersion
3.	AERB/SG/S-2	Hydrogeological dispersion – methodology and modelling
4.	AERB/SG/S-3	Extreme values of meteorological parameters
5.	AERB/SG/S-4	Hydrogeological considerations of NPP siting
6.	AERB/SG/S-5	Principles of control of exposures in public domain
7.	AERB/SG/S-6A AERB/SG/S-6B	Flooding and flood analysis a) Inland flooding b)Coastal sites
8.	AERB/SG/S-7	Human induced events and establishment of design basis
9.	AERB/SG/S-8	Influence of site parameters on emergency preparedness
10.	AERB/SG/S-9	Population distribution and its analysis
11.	AERB/SG/S-10	Quality Assurance in Siting
12.	AERB/SG/S-11	Seismic Studies and Design Basis Ground Motion for Nuclear Power Plant Sites

AERB safety standards on NPP Design and Operation relevant to Safety against External Events

S.No.	Identification Number	Title
1.	AERB/NPP-PHWR/SC/D (Rev. 1)	Design of Pressurised Heavy Water Reactor Based Nuclear Power Plants.
2.	AERB/SC/O	Code of Practice on Safety in Nuclear Power Plant Operation.
3.	AERB/SC/QA	Code of Practice on Quality Assurance for Safety in Nuclear Power Plants.
4.	AERB/NRF/SC/RW	Management of Radioactive Waste.
5.	AERB/NF/SS/FPS(Rev. 1)	Fire Protection Systems for Nuclear Facilities.
6.	AERB/SC/G	Regulation of Nuclear and Radiation Facilities.
7.	AERB/SG/D-5	Design Basis Events for Pressurised Heavy Water Reactors.
8.	AERB/SG/D-7	Core Reactivity Control in Pressurised Heavy Water Reactors.
9.	AERB/SG/D-11	Emergency Electric Power Supply Systems for Pressurised Heavy Water Reactors.
10.	AERB/SG/D-15	Ultimate Heat Sink and Associated Systems in Pressurised Heavy Water Reactors.
11.	AERB/SG/D-18	Loss of Coolant Accident Analysis for Pressurised Heavy Water Reactors.
12.	AERB/NPP-PHWR/SG/D-20	Safety Related Instrumentation and Control for Pressurised Heavy Water Based Nuclear Power Plants.
13.	AERB/NPP-PHWR/SG/D-21	Containment System Design for Pressurised Heavy Water Reactors.
14.	AERB/NPP-PHWR/SG/D-23	Seismic Qualification of Structures, Systems and Components of Pressurised Heavy Water Reactors.
15.	AERB/NPP-PHWR/SM/D-2	Hydrogen Release and Mitigation Measures under Accident Conditions in Pressurised Heavy Water Reactors.
16.	AERB/SG/O-1	Training and Qualification of Operating Personnel of Nuclear Power Plants.
17.	AERB/SG/O-6	Preparedness of the Operating Organization for emergencies at

		Nuclear Power Plants.
18.	AERB/SG/O-9	Management of Nuclear Power Plants for Safe Operation.
19.	AERB/SG/O-10	Core Management and Fuel Handling for Nuclear Power Plants.
20.	AERB/SG/O-13	Operational Safety Experience Feedback on Nuclear Power Plants.
21.	AERB/SG/EP-1	Preparation of Site Emergency Preparedness Plans for Nuclear Installations.
22.	AERB/SG/EP-2	Preparation of off-site Emergency Preparedness Plans for Nuclear Installations.

Safety against external events of natural origin

The AERB Safety Code on design of PHWR based NPPs stipulates that structures, systems and components necessary to assure the capability for shutdown, residual heat removal and confinement of radioactive material shall be designed to remain functional throughout the plant life against postulated natural events. Design basis for the structures, systems and components shall include:

(i) Consideration of the highest specified intensity of the postulated natural events or other external events; and

(ii) Consideration of the radiological consequences of such events.

Details of design basis external events that need to be considered are described in the AERB code on safety in NPP siting. The siting code stipulates that site characteristics, which may affect the safety of the nuclear power plant, shall be investigated and assessed. Proposed sites shall be examined with respect to the frequency and the severity of external events and phenomena that could affect the safety of the plant. For an external event (or combination of events) the choice of values of the parameters upon which the plant design is based should be such as to ensure that structures, systems and components important to safety in relation to that event (or combination of events) will maintain their integrity and will not suffer loss of function during or after the design basis event. If, after thorough evaluation, no engineering solution can be found to provide adequate protection against design basis external events, the site shall be deemed unsuitable for locating a nuclear power plant of the type and size proposed. Design bases shall be derived for each identified external event by adopting appropriate methodologies presented in relevant safety guides.

Historical records of the occurrences and severity of the important natural phenomena shall be collected for the region. The data shall be carefully analyzed for reliability, accuracy and completeness. If data for a particular type of natural phenomenon are incomplete for a particular region, then data from other regions which are sufficiently similar to the region of interest may be used in the formulation of the design basis event.

Based on data of historical seismicity, the code IS 1893 published by Bureau of Indian standards has divided India into four seismic zones (zone II to zone V) with higher zone numbers denoting increasing levels of seismic hazard. As per AERB requirements, no NPP shall be located at a site that falls in seismic zone–V, which has a potential to generate earthquakes beyond Magnitude-7. In addition, it is also verified that no earthquake generating faults are located within 5km radius of the site.

For evaluation of design basis ground motion for NPP, site specific studies are carried out within a region of 300 km. These include collection of information on historical seismicity, earthquake generating faults/areas, possibility of reservoir induced seismicity etc. Field investigations are carried out by expert agencies like Geological Survey of India. The level of detail of investigations increases as the location of investigation nears the site.

Design basis ground motion in all operating NPPs in India, from NAPS onwards have been arrived at following the above stated methodology, which is considered to be reasonably conservative. The older generation nuclear power plants, viz TAPS 1&2, RAPS 1&2, and MAPS 1&2 have been re-evaluated in recent years with respect to the site specific ground motion as applicable to new NPPs. Based on the findings, the structures, systems and components have been modified/ strengthened as necessary. These include provision of new emergency diesel generator buildings, modifications of battery banks, strengthening of masonry walls etc.

Flooding potential at an NPP site and the related hazards to be considered in design depend on whether the NPP site is inland or coastal. For a coastal site, the design basis flood level is estimated considering maximum tsunami wave height or the combined effect of a cyclone and rainfall. It is also assumed that cyclone or tsunami is coincident with the maximum tide level. The estimated flood level is summation of the maximum tide level, the maximum of storm surge or tsunami level and wave run-up. Various parameters of cyclone (radius, speed, pressure drop, etc) are chosen such that the flood levels estimated would have a mean recurrence interval of 1000 years or above.

AERB guidelines for flooding due to tsunami were based on historical data of tsunami wave heights prior to the 2004 Indian Ocean tsunami event. A value of 3m for tsunami wave height was specified for the west coast north of Karwar and 2.5m for east coast and rest of the west coast south of Karwar.

Though flood levels at NPP sites on the east coast due to 2004 tsunami were lower than these estimated design basis flood levels, AERB recognized the need for a more rigorous treatment of tsunami hazard for coastal NPP sites. For example, for the recently cleared NPP site for KKNPP units-3 to 6, assessment of worst case scenario from various tsunamigenic sources, as applicable to the site, was done as per AERB requirement.

For an inland site, the hazards to be evaluated include probable maximum flood in the water body near the site along with maximum rainfall, and, flood caused from failure of any upstream dam. The value of probable maximum flood is chosen such that the mean recurrence interval of the flood is 1000 years or above. Conservatively formulated AERB guidelines exist for detailed evaluation of the flood hazard at inland sites.

Annexure- 4

COMMITTEE’S WORK PLAN

The following plan was developed for the work to be done by the various expert working groups and review by the committee.

1. Review the postulated design basis external events of natural origin, as given in AERB safety regulations, to determine whether their magnitudes require any revision.
2 Suggest the magnitudes that could possibly be used as guidelines for beyond design basis situations, where possible.
3. Examine the structures, systems and components (SSCs) of existing NPPs to withstand the intensity of external events in terms of maintaining their integrity, as determined in 1 and 2 above.
4. Check adequacy of provisions for the following in operating NPPs in case of external events of maximum postulated intensity.
   (a) Capability to shutdown and maintain the reactor in shutdown state.
   (b) Capability to maintain reactor containment integrity.
   (c) Capability to adequately cool the reactor core on a sustained basis.
      - ✓ Ensuring availability of class 1,2 and 3 power supplies
      - ✓ Availability of relevant equipment and components like DGs, pumps, piping and valves
      - ✓ Availability of control equipments
      - ✓ Availability of water in required quantities
      - ✓ Availability of ultimate heat sink
   (d) Capability to adequately cool the reactor core under station black out condition.
5. Safety of irradiated fuel in spent fuel storage pools.

6. Safety of other facilities at the NPP site that have a potential for release of radioactivity or spread of radioactive contamination like, near surface radioactive waste storage facilities and heavy water upgrading plants.

In addition to the areas identified in the work plan given above, the committee decided to bear in mind the following areas also based on the various discussions:

i) Severe accident management guidelines (SAMG) should be developed, duly accounting for postulated accidents initiated by beyond design basis external events.
ii) Combination of related natural events like earthquake and tsunami/flood should be taken into account.
iii) Secondary effects of the external events like flooding of operating areas or fire breakout, wind/flood induced missiles, disruption of transport and communication need to be taken into account.
iv) A major earthquake is generally followed by a number of aftershocks. The design of SSCs including engineered safety features as well as SAMG need to take the aftershocks also into account.
v) PSAs for external events should be undertaken to identify cliff edge effects and measures for dealing with them.
vi) Capability of the plant to withstand beyond design basis external events (BDBEs) should be examined in the safety analysis.
vii) The concept of ‘Dry site’ i.e. protection against flooding by adopting high safe grade level rather than by flood protection measures like bunds/dykes etc., should be preferred for new NPPs.
viii) Safety assessment should include common cause failure in multiple units at the same site.

ix) Tsunami warning should be available to NPPs at coastal sites through active tsunami warning systems.

x) Source models and methodologies for estimation of seismic and tsunami hazards need to be revisited in the light of Fukushima events.

xi) Safety against external events should form part of PSRs.

xii) There should be provision of a shielded and seismically qualified on-site emergency facility at each NPP site, well equipped for obtaining information on plant parameters and for communication with relevant agencies to decide on the course of action in the event of an accident. These facilities should also have provision of food and rest for the occupants for about one week.

xiii) Design provisions should be made in all NPP units for preventing hydrogen explosions in case of a core damage accident.

Annexure-5

SUMMARIZED INTERIM REPORTS OF WORKING GROUPS

1. Expert group to review external events in relation to the safety of nuclear power plants

General

The magnitude 9.0 Tohoku Earthquake (Japan) and the associated tsunami brought into limelight the need for developing methodologies to derive parameters for NPPs corresponding to beyond design basis extreme events. The AERBSC-EE constituted an Expert Group for review of external events with the following terms of reference.

1. Review the list of postulated design basis external events of natural origin for completeness and examine their magnitudes, as given in current AERB safety documents, to determine whether these require any revision
2. Suggest the magnitudes of Beyond Design Basis External Events that should be used as guidelines, where possible
3. Propose guidelines for deciding on safe grade levels for new NPP sites
4. Examine adequacy of the current methodologies for evaluating integrity of the existing structures, systems and components (SSCs) of NPPs to withstand external events, both within and beyond design basis, and suggest revision/modification, as appropriate

The Expert Group examined the current methodologies for estimation of design basis parameters and design approaches followed for NPPs. Parameters to define beyond design basis extreme events are proposed, where possible at this stage. In cases like probable maximum flood in inland sites, the estimation depends on several site specific parameters. In such cases a general guideline is proposed for arriving at the site specific estimates.

Earthquake

Current approach

Assessment of seismicity and related hazards constitute a major part of the siting criteria for NPPs. The Indian Seismic design Code [IS:1893, 2002] groups the country into four seismic zones. Zone V is associated with the areas of maximum seismicity and zone II with the minimum. As per AERB siting code, AERB/SC/S, an NPP is not allowed to be located in zone V. Potential for ground rupture is also assessed during site evaluation. Presence of a fault in the vicinity of the site increases the chance of ground rupture during an earthquake. Hence, if there is an evidence of an active or a capable fault within 5km of site, the site is deemed unacceptable. Similarly, sites prone to liquefaction (an earthquake related phenomena) are also rejected during site evaluation.

Adequate precaution is taken at detailed design stage to ensure that SSCs of an NPP are capable of withstanding the effects of vibratory ground motion arising from strong earthquakes derived from study of site characteristics. The design basis ground motion (DBGM) for this purpose is evaluated for each site.

The DBGM is derived for two levels of earthquakes, S1 or Operating Basis Earthquake (OBE) level and S2 or Safe Shutdown Earthquake (SSE) level. SSE represents the maximum potential vibratory ground motion that can be expected for the region (with mean recurrence interval (MRI) of the order of 10,000 years). In the event of this level of earthquake, the considerations are to shutdown the reactor, keep it under shutdown condition, including removal of decay heat and containing postulated radioactive release within containment structure. Hence the safety systems, needed to meet these requirements, are designed for SSE level of earthquake. Another vibratory ground motion, OBE, (with MRI ~100 years) is also specified. All plant systems, including those for normal operation, are expected to continue to function when subjected to OBE. If the plant experiences ground motion above this, it shall be shutdown and inspected. These motions are expressed by appropriate parameters such as site specific response spectra for various damping factors, time history and its duration.

For estimating the DBGM parameters of a site, the earthquake sources (*e.g.*, faults) around the site need to be identified and maximum earthquake potential of each source need to be estimated. This is achieved by conducting a detailed investigation of geological, tectonic and seismological environment of the site. The data on recorded historical and pre-historical seismicity are also collected.

The investigations are conducted in four scales, regional (300 km minimum), intermediate range (50 km radius), local (5 km radius), and site area (within plant boundary). Each set of study leads to progressively more detailed investigation resulting in large volume of data and information as it gets closer to the site. The investigations are carried out through specialist agencies in the field like Geological survey of India (GSI). The areas are investigated through satellite imageries, aerial photographs, detailed maps to determine tectonic structures that could be considered as the sources for earthquakes. The historic earthquake data available in earthquake catalogues and any other sources are also collected.

For estimation of ground motion corresponding to S2 level, the maximum potential of each fault is estimated. This also takes into account the maximized value of historical/recorded seismicity attributable to the fault, by increasing the recorded magnitude by at least one intensity equivalent for calculation of ground motion. Subsequently, this maximized earthquake is brought to the point on the fault which is closest to the site and for this magnitude and distance combination, earthquake acceleration is determined. The exercise is repeated for all other faults surrounding the site and the maximum of acceleration arrived at from all these faults, is adopted as design basis S2 level acceleration.

Detailed guidelines for derivation of S1 and S2 level ground motions including PGA, and response spectra are given in AERB guide AERB/SG/S-11.

Expert group deliberations

SSE level event

The Expert group (EG) discussed the existing method of deriving safe shutdown earthquake and no significant shortcomings could be noted in the method. However, the EG recognized limitations of the method because of lack of sufficient and relevant earthquake data and other uncertainties regarding site tectonics. These limitations and possible means of alleviating these were deliberated as follows:

In peninsular India, scarcity of tectonic and historical earthquake data is a major challenge for defining the earthquake potential. Earthquake events on many faults have not resulted in manifestations on earth surface. Some major faults are embedded well below the upper strata that are several kilometers thick; only sub-features from such fault propagate towards the surface. There are difficulties in quantifying the seismic hazard because of the limitations of data and uncertainties in defining the seismic sources. Hence the following aspects need to be kept in mind while assigning the SSE level event.

1. All possible sources of data should be consulted in preparing the tectonic map of the region. DELPHI method (or weighted expert opinions) may be employed to arrive at the final recommendation
2. All lineaments shall be examined to identify faults. All mapped faults should be considered as *capable*, unless proven otherwise following well established procedures. A procedure needs to be arrived at for ascertaining whether fault is *active* or *capable* or not.
3. The attenuation relations shall correspond to the tectonic and geologic characteristics of the region.
4. The ratio of vertical acceleration to the horizontal (presently being considered as 2/3) can be un-conservative, especially in near field events. This ratio is frequency dependant, and an estimate of this ratio can be derived from the existing attenuation relationships.

The Expert Group recommends that the above aspects should be considered while revising the AERB guide on seismicity so that the concerns are appropriately addressed.

Beyond Design Basis Earthquake Event

The Fukushima earthquake event has highlighted that the safety of NPPs need also to be assessed for beyond design basis events of natural origin like earthquake and tsunami. The Expert Group considered the following approach to have a reasonable level of such beyond design basis extreme event for safety assessment.

As with SSE, estimating a beyond design basis earthquake event for an NPP site in India is a challenging task because of the shortcomings in the data, as enumerated below:

(1) Lack of paleoseismic data on slips at faults on which earthquakes are known to have occurred in the past and reliable slip rates associated with each fault;
(2) Lack of statistically sufficient records of historic earthquake, instrumented data from M>5 events in the recent decades and attenuation relations for each region in the country, especially in peninsular India;
(3) Lack of correlation of epicenters with faults/lineaments; and
(4) Some faults that do not have surface features may be missed

Therefore a postulated beyond design basis earthquake event may be used for assessing the safety margin of existing NPPs and of the new NPPs. This could be arrived at based on a comparative study of ground motion parameters derived from:

a) A postulated level of expected maximum acceleration/intensity of shaking at site, guided by the regional seismicity and local soil/rock site conditions, irrespective of earthquake source location, and
b) Maximization of earthquake hazard as evaluated following the procedure for SSE level earthquake.

Further work and discussion on estimation of postulated beyond design basis earthquake event following both methodologies is in progress.

Flood hazard

<u>Current approach</u>

External flooding is one of the important natural events that have potential to induce common-mode failure in an NPP. Generally NPPs are safe guarded against external flood hazard by having finished ground level of the plant above the design basis flood level (DBFL). In some cases, protection structures are built around the plant site to safeguard against DBFL.

The regulatory requirements for safety of an NPP in India in relation to external flooding hazard are outlined in AERB Safety Code [AERB/SC/S, 1990]. The sites are generally categorized into two types, coastal site and inland site. The Code provides requirements for both types of sites. Similarly, Safety Guides are published by AERB to elaborate requirements of the Code including methodology for assessment of hazard due to external floods. The guides allow use of probabilistic or deterministic approaches for arriving at the design basis flood for NPP sites. While following probabilistic approach, the design basis flood is calculated for a 1000 year mean recurrence interval of occurrence of the causative phenomena (viz. precipitation, storm, etc). While following deterministic approach, the biggest historical storm in the region is transported to the site area and is oriented in such a way that it maximizes the flood in the river or storm surge in the sea. Based on the estimated storm or flood, the design basis flood level *at site* is estimated with the use of detailed numerical models. Expert government agencies like Central Water Power Research Station, Central Water Commission, carry out this assessment.

For inland site, the flooding could occur due to flood in the adjoining river/lake, upstream dam break or intense precipitation in the surrounding region. The failure of the upstream dam/weir may result from seismic or hydro-geological causes or from faulty operations of these structures or channel obstruction due to landslides, log or debris jams, etc. Guidelines for evaluation of probable maximum precipitation and flooding due to failure of water control structures are covered in AERB/SG/S-6A. If the site is on the bank of inland water body such as reservoir or lake, the effect of seiches (long period vibration of water body induced by earthquake) is to be considered in determination of DBFL. In addition, simultaneous occurrence of intense precipitation in the local site could cause local flooding. The adequacy of site surface drainage to cater to this demand is also verified.

For a coastal site, the flooding hazards include those caused by cyclonic storms, tsunamis and local intense precipitation. Guidelines for evaluation of flooding due to cyclonic disturbance in coastal sites are covered in AERB/SG/S-6B, which allows use of probabilistic or deterministic estimation of design basis flood level due to storm surge. In the deterministic method a set of maximised hypothetical storms are considered, selected and moved to the location critical for formation of a surge at the site. It is then used as input for an appropriate numerical model for surge calculation. It is assumed that the maximum historical tide level is coincident with the storm surge and values of the maximum tide, storm surge and wave run-up are added to arrive at the worst estimate of flood level above reference level (generally mean sea level) during cyclonic storms.

The design basis levels for tsunami are specified in AERB/SG/S-11 (1990). The guidelines provided were based on the historical data. As per this guide, Indian coast was divided into two regions, locations above Karwar on west coast and locations below Karwar including east coast. The specified tsunami heights were 3m and 2.5m respectively for these regions. The levels due to the maximum tide and wave run-up were to be added to this height to arrive at the design basis flood level due to tsunami.

Though flood levels due to 2004 tsunami were lower than the estimated design basis flood levels for the NPP sites, AERB recognized the need for a more rigorous treatment of tsunami hazard for coastal NPP locations. In order to ascertain the maximum hazard that could be posed by tsunami to a site, expert inputs from many national agencies, like Geological Survey of India, National Geophysical Research Institute, Center Water Power Research station, Indian Institute of Technology Kanpur, and Indian Institute of Technology Madras were taken. Based on the discussions, the postulated tsunamigenic sources and corresponding scenario events (some even bigger than the 2004 event) were estimated.

Expert Group Deliberations

The expert group reviewed the existing AERB guidelines for flood hazard assessment. It was noted that no major pitfalls exist in procedures followed for estimation of design basis flood levels except in case of tsunami hazard assessment. It was noted that in AERB guide, the recommendations were based on some data prior to 1990. The EG noted that AERB has initiated actions towards more rational estimation of tsunami hazard based on numerical evaluation of tsunami wave heights arrived at from maximum potential tsunamigenic sources around Indian coast. As regards the existing methodologies of estimating DBFL, the EG noted that some statistical models were used in past for rainfall prediction irrespective of their fitness to the site specific data. Uncertainties in the estimates were also not addressed systematically. In view of this, the following recommendations were made by the EG with regard to evaluation of flood hazard.

Evaluation of design basis flood

1. The analysis of site specific rainfall data indicates wide variations in the trends shown by each data. Hence, there is a need for qualification of probabilistic models with respect to the data before using the same. In addition to statistical models currently covered in AERB guide (Gumbel and Frechet), log Pierson type-3 distribution may also be considered for statistical analysis of data.
2. The assessment of flooding shall be with respect to a mean + 1 sigma estimate of rainfall, corresponding to 1000 year return period using appropriate probabilistic models.
3. Irrespective of the return periods used in estimation, some natural phenomena/parameters have physical upper bounds. Existence of such deterministic upper bounds, if any, shall also be taken into account in the evaluation.
4. While assessing inland sites, scenario involving combination of flood due to dam break and earthquake should be considered.

Assessment of Beyond Design Basis Flood

A site specific analysis needs to be conducted in all plants with respect to the beyond design basis extreme event to assess the extent of flood hazard with the parameters defined below. The phenomena/numbers should be revised and reviewed by appropriate expert groups in project mode so that the gaps in current assessment can be addressed. The interim recommendations, which can be considered as possible guidelines for reasonable quantification of beyond design basis level of extreme flooding events for NPP assessment at this stage, are as follows:

Inland Flood

- In case of flooding caused due to dam break, the postulation (size, extent and duration) and assessment involves uncertainties and therefore a conservative upper bound analysis is suggested for beyond design basis extreme event of dam break along with a rainfall/flood of 100 year return period
- Analysis of past rainfall data of Mumbai city and three NPP sites, Tarapur, Kakrapara and Rajasthan, shows that a 15% increase in design basis rainfall (i.e. mean + one standard deviation corresponding to 1000 years return period) leads to a rainfall event of over 10000 years return period (i.e. one order higher event). Therefore, the volume/flow considered for extreme flood assessment during design basis conditions in Inland sites (i.e. value corresponding to mean + one standard deviation estimate for 1000 years return period) may be increased by 15% to arrive at a first order estimate of flood levels for inland sites as well as for carrying out the capacity assessment of site drainage corresponding to an extreme event.

Coastal flooding

- The EG looked into the available data for past storms. Based on these data, it recommended that a pressure drop of *100 milli bar*, associated wind speed of *300 kmph* for east coast and 240 kmph for west coast and radius of *50 km* may be taken as an upper bound value for the postulated beyond design basis cyclonic storm. The translational speed of storm may be considered as 40kmph. The total height of the wave shall be summation of (a) tidal variation, (b) storm surge height, (c) wave set up and (d) wind induced wave run-up.

Tsunami hazard

- Major contribution of tsunami hazard to NPP sites arise from Burma-Andaman-Sunda region and Makran coast of Pakistan with some local tsunamis reported near Chagos ridge/Nascent zone in Indian Ocean.
- The faults associated with Carlsberg ridge on Owen Fracture Zone are transform faults and earthquakes in the region are generated due to strike slip mechanism. Though large number of earthquakes occurred in this fracture zone, no tsunamis had been reported and no large earthquakes had occurred. Hence, tsunamigenic potential of these sources is considered very insignificant.
- The tsunamigenic zone along east coast of India (viz Burma-Andaman-Sumatra region) is more than 1300km away from nearest NPP site (Madras/Kalpakkam). Similarly, the tsunamigenic zone along west coast (viz. Makran region near Pakistan) is about 800 km away from Tarapur. Hence, unlike in Japan, NPPs along Indian coast would be subjected to either a local earthquake or a tsunami caused by a far away earthquake.
- The submarine faults/lineaments located near to the coast are small in nature and does not have the capability to produce large vertical displacements that are tsunamigenic in nature. In view of this, there is hardly any possibility of near source tectonic tsunamis hitting NPP sites.
- A detailed site specific analysis using a validated numerical model shall be carried out to arrive at accurate estimates of tsunami run-ups under all possible combinations and variations of input parameters. The evaluation shall include accurate near-shore data.

As an interim estimate, median values of tsunami heights calculated from tsunami simulation study for east coast of India based on maximum possible magnitude of tsunamigenic earthquakes sources and maximum possible variations of parameters defining earthquake source mechanisms are considered. Median values of simulated tsunami heights (with respect to mean sea level) at Kalpakkam site range from 6-8m where as at Kudankulam site it is about 4m. In Tarapur, the predicted median tsunami wave heights generated from Makran source are less than 2m. However, it may be noted that possible contribution from a co-seismic land slide may have to be taken into account for assessments carried out for Makran source. Therefore, in the interim, the tsunami height value of 3m (as suggested in AERB guide SG/S-11) may be considered appropriate for margin assessment of NPPs along the West coast of India. The tidal variations, as applicable for each site shall be considered while evaluating the total height due to tsunami.

- For a multi-facility site, plant specific modifications like protection walls may cause modification of impact of phenomenon on the neighboring areas. A global analysis that ascertains impact on all facilities shall be conducted before implementing any protection measures.
- In some locations, shore line bathymetry may be such that it causes very high amplifications in wave amplitudes. This shall be appropriately considered and if necessary, the site should be engineered against tsunami hazard.

Other Meteorological Hazards

For purpose of design of NPPs, wind and temperature constitute major meteorological parameters that can have an impact on design. Majority of structures in an NPP have lateral dimensions comparable to their heights and lateral forces due to earthquake generally governs the design. However, in slender structures like stack, wind could govern.

The Fukushima earthquake event has highlighted that the safety of NPPs need also to be assessed for beyond design basis events of natural origin like earthquake and tsunami. In this respect, the possible upper bound parameters that could be associated with beyond design basis extreme meteorological phenomena were also deliberated.

Wind

Current approach for design basis wind

Two levels of wind effect are considered during design of NPPs, namely (a) Severe Wind, and (b) Extreme Wind. The *Severe Wind* is postulated to be that event having a return period of *1,000 years, a*nd the *Extreme Wind* of *10,000 years*. NPP structures are designed for *severe wind* velocities. The extreme wind velocities are used to assess whether wind induced missiles could be generated at an NPP site and if so then their effects on items important to safety are also evaluated. Wind velocities corresponding to both categories are calculated using probabilistic approaches using site specific data and/or code of practice for wind loads published by Bureau of Indian standards (BIS-875, part-3, 1987)

The EG considers the procedures followed for estimation of design basis wind speed adequate. It was also opined that for the current need of defining beyond design basis extreme events, it may be considered that the parameters defined for each phenomena already includes/absorbs the effect of climate change.

Assessment of beyond design basis wind

The EG looked into the past data on extreme winds in the country and recommended, based on judgment for a first order estimate, that a beyond design basis extreme wind speed may be postulated corresponding to 1000 year return period increased by 50% and rounded off to nearest 10m/s speed.

The site specific values may be arrived at using the relevant procedures brought out in IS 875.

Temperature

Current approach for design basis temperature

Extreme temperature in atmosphere could create thermal stresses in structures. In addition, this could also impact the performance of systems that use air as ultimate heat sink. At present, no specific requirements exist with regard to the design basis values to be adopted for atmospheric temperature during design of structures, systems and components. However, the mean + 1 sigma estimate corresponding to 1000 year return period is being used in the design of structures in recent projects.

Assessment of beyond design basis temperature value

For the purpose of structural evaluation of SSC as well as functional evaluation of safety systems related to ultimate heat sink, with respect to beyond design basis extreme temperature, (mean + 2 sigma) for higher values and (mean - 2 sigma) for lower values corresponding to 1000 year mean return period may be considered as guidelines for beyond design basis value of temperature.

2. Working group to review safety of TAPS-1&2 against external events

The Working Group (WG) noted that Fukushima experienced natural events, magnitudes of which were higher than those considered in the design of the plant. These natural events led to sustained station blackout. WG opined that similar natural events can be postulated with magnitudes higher than that considered in the design of the station/ units, to see the capability of the TAPS -1& 2 units to withstand them. WG observed that AERB SC-EE has constituted a working group to arrive at maximum credible magnitudes of natural events for different NPPs. Till the time these values are available it decided considering the following values:

(i) Seismic Event: Design basis value of 0.2g for the safety related components and systems.

(ii) Tsunami Height: Upto 108'elevation of the plant considering a tsunami height of 3m for the western coast due to an underwater seismic event at Makran.

The WG reviewed the capability of the plant SSCs to perform the basic safety functions of Shutdown, Continued core cooling and Containment under the seismic and tsunami events. It also observed that station blackout would form a subset of tsunami event in the present plant configuration and studied the effect of it as well.

The following engineered safety features were identified for the handling of a serious event/ accident for achieving the basic safety functions considered above:

a) Shutdown:
   (i) Safety Control Rods:
   (ii) Liquid Poison System (Shared - Active System):

b) Core Cooling:
   (i) Emergency condenser system (Passive System)
   (ii) Low pressure core spray and post incident system (CS&PI) (Shared - Active System)
   (iii) Auto Blow Down System (ABDS)
   (iv) Shutdown Cooling System (RHRS)
   (v) Emergency Feed Water System
   (vi) RBCW and SSW cooling Water systems

c) Containment:

(i) Primary and secondary containment structures

(ii) Containment isolation valves

(iii) Emergency Ventilation System

d) Emergency Power Supply System

(i) Emergency Diesel Generators (Class III)

(ii) Station control power supply system (Class II)

(iii) Station Batteries (Class I)

Based on the above the following were analyzed:

<u>Seismic Event:</u>

WG noted that the critical SSCs have been designed to withstand SSE of 0.2g. Hence the SSCs would tolerate a seismic event upto a design basis value of 0.2g. Seismic re-evaluation of the plant has been carried out during the safety upgradation activity carried out in 2002-2006 wherein the SSCs have been evaluated for design basis seismic event. These studies were conducted as per the methodology prescribed by IAEA safety report series 28.

WG noted that some of the critical components like secondary steam generators, their supports, steam generator secondary lines, ECCS, Liquid Poison System etc. have not been included in the seismic re-evaluation studies conducted as it was beyond the scope of the study as per the IAEA safety standard used. WG opined that these systems and other support systems as have been identified by the WG also need to be studied for seismic withstand capability while examining the beyond the design basis natural events in the wake of Fukushima accident.

The WG also felt that till maximum credible magnitudes of natural events are available, the capability margins available to withstand beyond design basis seismic loads may be arrived at by extrapolating the analysis results available with NPCIL from the earlier seismic re-evaluation studies. Accordingly WG has requested NPCIL to provide the values of the SSCs identified.

With the above background the WG studied the capability of the plant systems to withstand a design basis seismic event and fulfilling the basic safety functions. The reactor will be shutdown by the control rods with the energy stored in the control rod drive accumulators. Cooling of the reactor will be possible using emergency condenser, ECCS, CRD feed pumps etc. Containment isolation function will be achieved by the fail safe containment isolation valves.

As the SSCs have been designed to withstand the design basis seismic event, concurrent LOCA was not considered. It was seen that the basic safety functions will be achieved and capability to feed the reactor also will be available. Filtered containment venting function will not be available as the filters are not designed seismically. Table I below gives the fulfillment of the basic functions and the cope up time available.

<u>Tsunami Event:</u>

The WG studied the capability of the plant SSCs to withstand a tsunami of height 3m based on the value given in AERB guide for siting (AERB/SG/11).

Considering a tsunami height of 3.0 m the highest water surge level comes to 6.84m above mean seal level, which translates into a plant elevation of 107.63 ft. for TAPS 1&2. Including a conservative margin, it was decided to consider a total water surge level for tsunami wave upto an elevation of 108’. Based on this, a Tsunami Vulnerability Chart (TVC) was prepared which lists out critical equipment along with their elevation and the impact on the safety functions due to their failure.

Due to the elevation of the equipment, many of the systems including Class III emergency power supply systems will be affected by tsunami. The event will transgress into a Station Black Out (SBO). The basic safety functions will be achieved. There will be no feed capability to make up reactor water level. Core cooling through emergency condenser will be possible for a limited period. ECCS system will not be available. Filtered containment venting function will not be available. Table I gives the details of fulfilling of the basic safety functions and the cope up time available for handling this event.

<u>Station Black Out:</u>

On Station Black Out conditions, the reactor will be shutdown by the control rods with the energy stored in the control rod drive accumulators. Cooling of the reactor for a limited period will be possible using emergency condenser. Containment isolation function will be achieved by the fail safe containment isolation valves.

During extended SBO, as indicated above, the basic safety functions will be achieved. The coping time for the event depends on the emergency condenser function (8 hrs), reactor water inventory (6hrs) and availability of Auto Blow Down System (ABDS) to depressurize the reactor for limited number of operation of RVs and the station batteries service time. Feed capability will not be available. Filtered venting of Containment will not be available due to unavailability of power supply to the fans. Table I below gives the fulfillment of the basic functions and the cope up time available.

<u>Concurrent Failures:</u>

a) LOCA:

As the safety related systems and components are designed for a value of 0.2g, Double Ended Guillotine Break (DEGB) is not likely to occur during a design basis seismic event. Emergency feed capability of 410 lpm (110 gpm) exists for make up at high pressure (100 kg/cm2) through CRD feed pumps which are designed for withstanding design basis seismic events.

b) Fire:

Fires can be possible along with a seismic event. The nature and extent of the fire will affect the recovery actions and the resources available for handling the external events. The fire protection lines at present are not seismically qualified. Fixed fire protection system can be used if they survive the external event. Portable fire extinguishers can also be used. Fire order for handling the fires is available. The site is serviced by a centralized fire station.

<u>Spent fuel storage:</u>

The spent fuel pool and the components are seismically designed. The water inventory in the fuel pools will give 9 days of margin before uncovering of the bundles takes place with the full core unloaded in the spent fuel pools. However this margin will be considerably reduced if the fuel pool gates towards the reactor cavities lose their integrity and/ or sloshing takes place during a seismic event. Worst case calculations for a gross failure of the fuel pool gates and available water inventory showed a margin of nearly 1.3 days for the fuel pool storage racks to get uncovered. The fuel pool water make up capability will be affected by the non-availability of service water or demineralised water pumps as they are not seismically qualified and also will be affected by tsunami.

<u>Severe Accident Management:</u>

There are no severe accident management guidelines available for TAPS 1&2. These should be made available based on severe accident analysis.

Conclusion:

The above studies indicate that for all the events considered, shutdown of the reactors will be achieved by control rods. Limited passive cooling capability exists for six to eight hours using emergency condensers. Containment isolation capability exists through fail safe isolation valves.

Handling of the events further will depend on the availability of the equipment/ resources and the approachability of various equipments/ locations during the progression of the events.

The short comings of different SSCs in achieving their intended function under different postulated scenarios were brought out in the interim report. Action should be initiated to address these issues.

Table-I <u>Safety Function Capability and Cope up Time for External Events</u>

Safety Function	**SIESMIC (0.2g PGA)**	**TSUNAMI (3.0 m) (108' ele)**	**Extended SBO**
Shutdown (control of reactivity)	**Yes.** (i) SCRAM through Control Rods. (ii) Liquid Poison System Available	**Yes** (i) SCRAM through Control Rods. (ii) Liquid Poison System Not Available ATWS mitigation not possible (Emergency Power Supply	**Yes** (i) SCRAM through Control Rods. (ii) Liquid Poison System Not Available ATWS mitigation not possible (Emergency

		system (EPS) not available)	Power Supply system (EPS) not available)
Core Cooling	**Yes** (i) Emergency Condenser (8 hrs) (ii) CRD feed facility (iii) ECCS & ABDS available	**Yes** (i) Emergency Condenser (8 hrs) (ii)Reactor Water Inventory (6hrs) (iii)Station Battery (8hrs) <u>Following NOT Available:</u> Reactor feed facility, ECCS, ABDS & RHR - No Emergency Power Supply	**Yes** (i) Emergency Condenser (8 hrs) (ii) Reactor Water Inventory (6hrs) (iii)Station Battery (8hrs) <u>Following NOT Available:</u> Reactor feed facility & ECCS, ABDS & RHR - No Emergency Power Supply

Containment	**Yes** Containment Isolation –Fail Safe Filtered Venting not possible - Filters Not designed for SSE	**Yes** Containment Isolation –Fail Safe Filtered Venting not possible. - Flooding of areas & No Emergency Power Supply	**Yes** Containment Isolation –Fail Safe Filtered Venting not possible. - No Emergency Power Supply

3. Working group to review safety of PHWR based NPPs under prolonged station black out and loss of ultimate heat sink

The Working Group (WG) decided to review the following set of events for all PHWR based NPPs in India.

i. Extended station black out (SBO)
ii. SBO+flood level above design value
iii. SBO+loss of Ultimate Heat Sink

The Working Group interim report is based on the reviews carried out so far on the operating PHWR NPPs capability to withstand station blackout for at least 7 days. NPPs capability to withstand multiple failures as listed above will reviewed in the forthcoming meetings of WG.

All Indian PHWR stations are provided with backup diesel generator sets to provide class-III power supply to essential station loads in case of off-site power failure (class-IV). Stations are also provided with class-II (UPS) and Class-I (battery banks), which cater to very essential loads. These sources provide power for sufficient period for taking initial actions in case of station blackout (i.e. loss of class off-site and on-site power). With the redundancy and reliability of DG sets, likelihood of SBO is very rare.

All PHWR NPPs, except RAPS-1&2 have three diesel operated fire pumps. Provisions exist for injecting fire water to steam generators and end shields during SBO conditions. In case of RAPS-1&2 two additional diesel generator of 625KVA were retrofitted to take care of floods caused by Gandhi Sagar dam break. These are located at higher elevation and will power essential loads during SBO.

All NPPs have emergency operating procedures for dealing with SBO. Routine surveillance and testing of diesel fire pumps and flood DGs is carried out for ensuring their availability. The carbon steel fire water ring header used for injecting water to SGs, end shields is not covered under any surveillance programme. Failures in the form of pin holes have been observed in the past in these lines.

Though SBO is considered as a beyond design basis event, all standardized PHWR final safety analysis reports have considered SBO and has analysed it considering crash cool down at the 6^{th} & 30^{th} minute of station blackout to enable injecting fire water into steam generators secondary side for further decay heat removal . The analysis indicates that core cooling can be maintained through thermo-syphoning in both the conditions by injecting fire water to the steam generators and end shields. In the FSAR analysis is given for first 24 hours of SBO. It is also seen that temperature of moderator water and vault water reaches to about 90 & 80 deg C respectively in case of SBO of 24 hours. The likely temperatures in case of extended SBO beyond 24 hours and their acceptability need to be assessed and addressed accordingly.

For assuring continuity of thermo-siphoning, PHT system needs to be made up if losses due to shrinkage and leakages are high during extended SBO. At present no dedicated back up provision exists in PHWR stations for PHT system make-up under prolonged SBO conditions.

Sufficient inventory of water is available at all sites for catering to water supply requirements of SBO for at least seven days. The diesel fire pumps can operate for about 7 to 8 hours after which need for replenishment of diesel will arise. The makeup diesel for the fire water pumps would have to be done manually, which is possible. The capability of the diesel fire pumps for continued operation needs to be proved by operating for an extended period.

For the interim report, the working group concluded that SBO for a period of 7 days may be manageable for all PHWRs without external support, with respect to fuel cooling, subject to:

i. Induction of quality assurance programme for fire water injection systems and incorporation of suitable surveillance & ISI activities.
ii. Dedicated provision to inject water from a different source to be identified and kept ready for maintaining system full on PHT side to effect continuous thermo siphoning.
iii. Demonstration and maintenance of required logistics to transfer diesel to fire water pump from main diesel tanks.
iv. Inclusion of appropriate steps in the EOP to take care of non availability of compressed air and power supplies to I & C, both for actuation of certain essential valves and indication of process parameters.

v. Alternate means may have to be evolved to limit the moderator and vault water temperature to be within acceptable limits under prolonged SBO.

Working group proposes to continue the review of the multiple events of SBO + beyond design basis flood level and SBO + loss of ultimate heat sink.

4. Working group to review safety of electrical systems of PHWR based NPPs against external events

Review scope

This Working Group was constituted to examine electrical systems of the Indian PHWR based NPPs to ensure availability of Class I, II, and III power supplies to all essential equipment/components in the case of external events of natural origin, including combination of related events, of maximum postulated intensity, required for a) Safe shutdown of the reactor and maintaining it in guaranteed shutdown state b) Adequate cooling of the reactor core on a sustained basis and for c) Ensuring containment function.

Towards this WG obtained a firsthand knowledge of the current status of the location of all Class I, II and III Electrical system equipment of the stations vis-a-vis its requirement under postulated external events such as DBFL, seismic withstand level etc, its Present condition and the flood proof condition of cable entry from trenches and tunnels to the building where this equipment are located in view of various external events postulated form the respective stations through answers to the queries send by the WG to them in tabular form and presentations by the stations.

Observations

It was tried to assess the margin available at various stations to prevent flooding of site based on the Design basis flood level and Elevation of DGs, Class I, II & III equipment of all the operating stations based on the presentations made by the stations and it is found that margin exists in all the stations. Similarly DGs, Class I, II & III equipment in the station at this site, as presented by the site are designed as per Design basis Earth quake level of each site except in RAPS 1 and 2 where the Design basis Earth quake level was revised. Subsequently the plant was re evaluated for the new design value and presently this station qualifies for the new earthquake level.

The effect of earth quake on switch yards and the time within which the switch yard and hence the class IV power supply can be normalized post earth quake was also reviewed in the light of the presentations made on the infrastructure available and various arrangements existing at stations for it. Stations needs to carry out a detailed study to work out the components in switchyard vulnerable to earth quake ,spares required to be stored at site and expertise required for certain jobs to bring back switchyard and the grid on line subsequent to an earth quake within a short period.

On WG request station conducted walk through to confirm the healthiness Class I, II and III electrical power supply equipment. In the walk through it was observed that all the equipment are properly supported and the equipment presently at location meets the design specifications. WG had requested stations to confirm that if fire barriers at cable penetrations perform as seal against flooding then it is required to be confirmed that it is a flood cum fire barrier and not fire barrier alone.

From the presentations made WG has generally observed that Class I, II and III power supply equipment of all the stations will be available under presently specified Design Basis Flood Level and Design basis earth quake.

Further work

The committee will review the class I, II and III power supply availability for beyond design basis accident situation once the parameters to be considered are identified.

WG has identified a few no. of specific issues pertaining to various stations which needs to be considered while considering BDBA and it will be reviewed in the forth coming meetings of the WG.

5. Working group to review safety of control & instrumentation of PHWR based NPPs against external events

The Working Group on C&I was formed to examine the control & instrumentation systems in Indian PHWRs for their functioning in the case of external events, including combination of related events of maximum postulated intensity. The review process adopted by the Working Group considers the site specific relevance to the external events taking into account of plants same vintage. Plant specific data on C&I systems was obtained through a detailed questionnaire, prepared covering five major areas,

- Reactor trip and actuation circuits and associated C&I components along with monitoring for long term guaranteed shutdown status,
- Instrumentation for ensuring sustained core cooling and its status monitoring,
- Instrumentation for ensuring containment functions and monitoring of containment performance,

- Displaying important process parameters required during prolonged SBO, and
- Instruments used for monitoring radiological status within and outside the plant.

The Working Group on C&I insisted the station/NPCIL to provide data on C&I systems / components and associated panels about their, location, mounting and elevation, power supply sources, environmental qualification and seismic qualification and also fire rating / qualification of cables. Initial review of plant data from all the PHWRs under operation is completed and it revealed the following strengths and weaknesses,

(i) Seismic qualifications for typical instruments at newer NPPs (KGS-1 onwards) are in the range of 3 to 3.5 g. For older units tests/analysis was conducted for some important instruments as well as associated panels at ECIL. Analysis for older units shows seismic qualification for instruments in the order of 1 g. Seismic qualification data for Alarm units and Field Panel are typically in the range of 2.5 g to 2.88g. In general the safe shutdown earthquakes for Indian NPPs are in the order of 0.1g to 0.3g.

(ii) Environmental qualification includes operability of instruments at specified temperature, humidity and radiation level if inside RB. Temperature and Humidity rating for instruments outside RB are generally 45 to 55 Deg C at 95 RH.

(iii) Sensors necessary for reactor tripping function & instrumentation linked to moderator and its fire water injection are located inside RB. Hence, their locations with respect to elevations are not important from flooding consideration.

(iv) I&C related to fire water injection in SGs and ECCs are located outside RB at 100m EL, which is grade level. The grade levels are above design basis flood in all the NPPs.

(v) Station battery back-up for control power supplies are adequate for more than 1 hr without any load cut off, except for RAPS & MAPS, where they can supply for 30 minutes. But with the shedding of un-important loads as per station operating procedure, the availability can be extended. However, there is a need to establish vital parameter monitoring for prolonged SBOs upto 7 days.

(vi) Some components of the various control power supplies (250V AC/48V DC/24V DC) are located at 100m El, except for RAPS and MAPS where they are generally at higher elevation.

(vii) Although availability of COIS/CRCS/PIS is not so important considering prolonged SBO, their functions with respect to monitoring of post-accident monitoring of some parameters need to be re-looked.

(viii) The pumps and related C&I used or can be used for sustained core cooling like ABFP, ECCS are located at 91m elevation, which is below flood level, but flooding is possible at these areas only after the water level raises above 100m El. Any water leakages through cable or pipe penetrations need to be checked.

(ix) Manual cranking of diesel engine driven fire pumps are not always possible. The availability of starting batteries and chargers become important during beyond design basis flood. Presently they are located at 100m El (grade level).

(x) Some seismic switch/sensors are outside RB and at lower elevations than grade level. Water tight enclosures for them need to be ensured.

(xi) Seismic trip is provided only at NAPS. In multi-unit sites like KGS and RAPS only one unit is having seismic instrumentations, where as other units are sharing the alarm signals. However, NPCIL decided to provide seismic trip at all the NPPs

(xii) Provisions for post accident sampling of atmosphere inside containment and important process fluid need to be strengthened further.

(xiii) Although emergency operating procedures (EOPs) available at all the NPPs, indications available to confirm guaranteed shutdown state of the reactor under extended SBO (about 7 days) should be re-looked.

(xiv) Backup instrument air accumulators are provided for important individual valves / instruments. Availability of those accumulators under a beyond design basis seismic event need to be ensured.

(xv) In RAPS/MAPS, Power and Control battery banks shared. Aspects related to availability of critical control functions under SBO should be reassessed.

(xvi) In older plants the supplementary controls rooms (SCRs) were back-fitted as an abundant safety measure. The control and monitoring facilities provided at these SCR need to be re-look from point of view of sensor separation, adequacy of Power Supply, etc. WG felt that the strengthening of SCR will be easier task than increasing availability of MCR.

The preliminary review conducted for each NPPs by the Working Group were to check the adequacy of the existing provisions. Further detailed review is under progress. This WG is dependent on the outcome of other WGs and the guidance of AERBSC-EE on the following:

- Adequacy of postulated events/combinations to be considered
- SBO duration to be considered
- Maximum postulated intensity of earthquake & flood
- Systems required for Severe Accident Management scenario.

6. Working group to review safety of spent fuel storage facilities at NPPs against external events

Scope

The following are the scope of review for WG

a) Capability to adequately cool the irradiated fuel stored in these facilities on a sustained basis and in this regard, adequacy and availability of the following as appropriate.

   i) Power supply of appropriate class.
   ii) Ventilation system to prevent contamination spread.
   iii) Equipment and components like DGs, pumps, heat exchangers, piping and valves
   iv) Control equipments and instrumentation
   v) Provisions for cooling of fuel and make up pool water level in case of prolonged station blackout

b) Structural integrity of pools and liners and their leak tightness.

c) Water loss due to sloshing and capability for making up the water level.

d) Stability of fuel storage racks/trays and mechanical handling facilities.

e) Prevention of falling of heavy objects in the pool or on the dry storage casks.

Observation

WG held a total of five meetings and discussed about the terms of reference, scope, review approach and methodology to be adopted for systematic and early completion of review work. WG reviewed preliminary analysis and calculation results based on the thermal load.

From the analysis results it is observed that,

- The time available before the water level reaches minimum shielding level for arranging the makeup water supply under the postulated events ranges from minimum 3.9 days (MAPS) to maximum 26.3 days (NAPS). The studies were done on the basis of the preliminary data available for the existing plants.

- In case of RAPS- 1&2 and MAPS- 1&2 the emergency preparedness for makeup water should match the requirement of 3.9 and 5.75 days respectively.
- The time available before the top layer of fuel bundles is exposed is 16.81 days for RAPS- 1&2, 11.4 days in case of MAPS- 1&2 and for other reactors it is more than one month.
- WG also observed that the bulk boiling of water starts in as low as 1.9 days (MAPS) to 7 days (NAPS) which may affect the structural integrity of the pool.
- WG felt that an external water hookup provision for maintaining the required level in SFSB for all the operating plants should be implemented irrespective of any other design provision. Make up flow rate should be commensurate with the water loss due to evaporation.
- WG observed that present monitoring of SFSB water level, temperature and radiation field may not be adequate during BDBA condition like Fukushima. Accordingly, suitable passive monitoring/ indication facility should be provided which will be capable of monitoring the above parameters.

Preliminary safety assessment on various SFSBs indicates that for decay heat removal there is no immediate concern in Indian nuclear power plants (PHWRs) except for RAPS- 1&2 and MAPS- 1&2 as indicated by the analysis.

Further Work

Detailed site specific safety assessment of SFSB with respect to following will be considered based on the available design information:

Earth quake, tsunami and their combination, Rain water flooding, Dam break/river diversion leading to flooding, Fire due to hydrogen, Earth quake induced sloshing, water loss; spread of contamination and make up provision, Adequacy of power supply for SFSB cooling and its availability during external events, Effectiveness of ventilation system to prevent spread of contamination and control of fire due to hydrogen explosion or any other reason, Stability of fuel storage racks and mechanical handling equipment, Falling of objects /casks in SFSB.

Man-made external events and certain natural external events which are not relevant in Indian context are not proposed to be considered.

7. Working group to review safety of heavy water upgrading plants at NPP sites against external events

The working group reviewed the safety of Heavy Water Upgrading Plant and Heavy Water Clean-up Facility at Nuclear Power Plant (NPP) sites against the release of radioactivity or spread of radioactive contamination due to external events of natural origin including combination of related events of maximum postulated intensity. The observations of the committee are as follows:

i) The various systems involved in Heavy Water Upgrading are :
   a) Downgraded Heavy water storage & transfer facility
   b) Heavy water evaporation & clean-up facility
   c) Heavy water Upgrading Plant.

During the nuclear reactor operation, heavy water which has leaked from the reactor moderator system and the Primary Heat Transport (PHT) system is degraded with impurities in the surroundings. The isotopic purity of the leaked heavy water comes down. The collected downgraded Heavy water is stored in tanks of various capacities in the Downgraded heavy water storage & transfer system. All impurities such as oil, ionic impurities, dirt etc, are removed in Evaporation & clean-up plant by filtration, alkaline-evaporation, condensation and ion-exchange unit. In the Heavy Water Upgrading Plant isotopic purity of downgraded heavy water is improved to reactor grade by distillation under sub atmospheric pressure condition.

ii) Building Layout:

a) Downgraded storage tanks of various capacity (250 m^3 /180 m^3 etc.) are provided for storage of downgraded heavy water in an RCC dyke. Capacity of tanks and dyke varies from site to site.

b) Evaporation & Clean-up plant is single storey building. The size & location of the plant varies from site to site. Generally, clean-up plant, is located adjoining Upgrading Plant in most of the sites, however in few sites, the plant is located in Service building. Equipments located in clean-up plant are small capacity storage tanks, evaporators, filters etc.

c) Upgrading Plant building is a single storey RCC structure. Two Distillation columns are housed in a structural steel tower of overall size 7 m x 7 m x 58 m height. Number of distillation columns also varies from site to site. Major equipment in the plant is Distillation column assembly. Each distillation column assembly consisting of column sump, column sections (14 nos. each), vapour hood – reflux condenser assembly and re-boiler. Column sump is having 1.4 m diameter and 5.3 m height. Each column section is of 1.05 m diameter and 3.25 m height. The total height of distillation column assembly of 14 sections including reflux condenser and top vent condenser is 55 meters. The height of structural steel tower housing the distillation column assembly is 58 meters. Other equipment in the plant are Storage tanks of various capacity (10 m^3 /6 m^3/ 3 m^3 etc.), pumps, Motor Control Center (MCC), UGP control panel etc.

iii) Heavy Water Upgrading Plant (UGP) is an auxiliary system to nuclear power plant and its availability or operation does not have any effect on power plant operational safety. In case of total power failure after any external event, the Upgrading Plant will go to safe shutdown mode.

iv) Location of UGP & Clean-up facility with respect to other site structures varies from site to site. Also downgraded storage capacity and UGP processing capacity is different at different sites.

v) During sitting, design and construction of facility, the parameters like site characteristics and to some extent of the external natural events are taken into account.

vi) These facilities are designed and constructed as per the safety requirements. An earth quake of considerable magnitude and total flooding of these facilities will not cause undue radiological impact to the public as downgraded heavy water is contained in tanks and drums. Also the Tritium content is expected to be very less due to low Isotopic Purity (I.P.) of downgraded heavy water. Storage tanks at NPPs are designed and constructed to meet the requirements of operational basis earth quake (OBE). The radioactive release from the plant is limited in view of lower I.P. of downgraded heavy water stored in the system.

vii) In case of external events like earthquake, tsunami, flood, cyclone beyond design level, there can be immediate safety related issues of the plant or people around it but it will not initiate series of catastrophe.

viii) Further study and analysis of existing building, structure and equipment specifically tall structural steel tower and distillation column need to be carried out to assess margins available in the postulated level of earthquake. This issue is site specific and individual plants are to be addressed independently rather than in a combined effort based on location of Upgrading plant and nearby safety structures.

8. Working group to review safety of near surface radioactive waste disposal facilities and liquid waste storage tanks at NPP sites against external events

After Fukushima nuclear accident in Japan, AERB has constituted a Committee to review the safety status of Indian NPPs in case of external events. AERB-EE Committee has constituted a Working Group to examine the safety of Near Surface Disposal Facilities (NSDF) and radioactive liquid waste storage tanks at all NPP sites with respect to the release of radioactivity or spread of radioactive contamination due to external events of natural origin including combination of related events of maximum postulated intensity.

The Working Group met on four occasions and reviewed the existing design and construction features of NSDFs and radioactive liquid effluent storage tanks at various NPPs in the country. A preliminary report of Working Group was submitted to AERBSC-EE and was reviewed. A summary of study carried out so far is presented here.

Near Surface Disposal Facilities (NSDF)

In India, NSDFs are co-located within the exclusion boundary of NPPs. These are designed, constructed and operated based on a combination of engineered (structure, waste matrix, backfill materials) and passive safety (geospheric environment) features. During siting, design and construction of NSDF, the parameters like site characteristics and some extent of the external natural events are taken into account. Therefore, the radiological consequence due to failure of NSDF depends on various parameters like site characteristics, structure of the disposal module, characteristics of back fill materials, waste matrix, layout etc.

Safety assessments of NSDF are available at all NPP sites. These assessments are based on the ingress of water into the disposal system. The safety assessment scenario involves the leaching of radionuclides from the waste matrix and subsequent migration of radionuclides from the disposal system to the groundwater. A net work of monitoring bore-holes is provided at each NSDF to assess the performance of disposal system by sampling and analysis of ground water. The assessment of radiological impact due to such scenarios is reported to be within the acceptable limits. Preliminary study indicates that even in the case of Fukushima type extreme natural events, the possibilities of any undue radiological impact are minimal.

Spent Resin Storage Tanks

At TAPS (1&2) and MAPS (1&2), spent resins generated from the reactors are stored in underground tanks. The ingress of water into the storage tanks or failure of storage tanks has the potential of spreading contamination in the surrounding environment. At present the radionuclide monitoring data in bore wells around the spent resin storage facility shows safety of storage as activity levels are below the detection limit (BDL). This establishes the integrity of storage tanks. The practice of storing spent resin in water at underground tank needs further investigation. However, early immobilization of these resins and discontinuation of practice of storage of spent resin in underground tank is recommended.

Radioactive Liquid Effluent Storage and Treatment

Radioactive liquid effluents generated from PHWRs are collected in the liquid effluent segregation and storage system (LESS) or Liquid Effluent Management Plant (LEMP) and subsequently transferred to the treatment / disposal facility. The liquid waste storage tank area i.e. civil structure of LESS and dykes at NPPs are designed and constructed as per the seismic qualification of OBE. The LESS/LEMP tanks are designed for storage capacity equivalent to two days effluent generation from the plant. The total storage capacities of dyke tanks located in a multi-unit site is approximately equivalent to 21 days of waste volume generation.

In the event of failure of dyke tanks in a design basis accident, the entire volume of water is expected to be retained within the dyke. The management and disposal of the stored water inside the dyke wall is carried out on campaign basis. However, assessment of the consequences beyond design basis accident due to external events e.g. failure of tanks as well as dyke needs further studies.

Liquid waste management system at NPPs are designed and constructed to meet the operational requirements and management of low active liquid waste. The pipes and systems used at liquid effluent management system may not meet the requirements of natural events of maximum postulated intensity beyond OBE. Since the nuclear accident is expected to generate large volume of waste having significant concentration of radionuclide, the existing waste treatment / conditioning system at various NPPs would have to be suitably augmented for management of such waste by large capacity mobile/transportable radioactive effluent treatment system.

Conclusion

NSDFs are co-located with NPP. These are underground structures storing waste in solid form. Therefore, potential of spread of contamination in the event of extreme events is minimal. However, storage of spent resin in underground storage tanks needs to be discontinued along with immobilization of stored resin.

In the event of a nuclear accident due to external event, large volume of contaminated water is expected to be generated. Every site has the capacity to hold limited volume of radioactive effluents provided the integrity of the storage tanks are not affected by the event. During detailed study, an estimate of volume and activity of the radioactive effluent likely to be generated during nuclear accident due to EE needs to be made based on postulations of accident scenario. Depending upon site condition and nature of event, the parameters may vary. Subsequently, scheme for treatment of such effluent is worked out based on the experience available in the country on treatment of radioactive effluents. Such systems are recommended to be modular in design which can be transported to the affected site and pressed in service on short notice. Specific ion-exchangers for radioisotopes like ^{137}Cs have been developed in BARC and are in use for treatment of radioactive effluents. Further site specific safety review / studies are required for NSDFs and liquid waste storage tanks under extreme natural events.

9. Working group to review severe accident management provisions and guidelines in PHWR based NPPs

AERB committee to review safety of Indian NPPs against external events constituted a working group (WG) to review severe accident management guidelines (SAMG) and provisions for PHWR based NPPS. The WG reviewed the possible severe accident scenario resulting from extended station blackout and other initiating events, and this note summarizes the salient points of the review.

In general the WG found that the approach of SAMG development by NPCIL for Indian PHWRs is in conformance with SAM framework as defined in IAEA safety standards and with SAMGs being developed internationally.

For Indian PHWRs, the SAMG under development are based on IAEA safety standard NS-G2.15 and cover the following objectives:

- Preventing significant core damage,
- Terminating / Mitigating the progress of core damage once it has started (by taking intervention actions),
- Maintaining the integrity of the containment as long as possible,
- Minimizing releases of radioactive material, and
- Achieving a long term stable state.

The design features of PHWRs present inherent and significantly better safety features in terms of SAM in comparison to light water reactors. In standard PHWRs, the fuel is located in horizontal coolant channels inside a cylindrical calandria vessel containing heavy water moderator. The calandria is surrounded by a large vault filled with light water. Analysis indicates that any event progression leading to severe core damage accident is relatively slower and much longer time is available for handling the event than in light water reactors of comparable power levels. This is on account of the large quantities of light and heavy water surrounding the fuel which act as heat sink to remove decay heat from fuel. The PHWR design permits various levels of operator intervention such that the progression of an accident can be arrested before occurrence of severe core damage. Severe accident progression can also be controlled and a stable state can be achieved by continuous addition of water to the primary heat transport system, calandria and calandria vault to replenish the water inventory lost by boiling. The design features to facilitate appropriate operator actions under accident conditions and to obtain information on relevant process parameters, in terms of required hardware and instrumentation, are being developed for the operating PHWR units, whereas for 700 MWe PHWRs, which are under construction, such provisions are part of their design. There are certain design differences between the PHWRs from NAPS onwards and the older PHWRs viz. RAPS-1&2 and MAPS-1&2, as the older plants do not have a water filled vault surrounding the calandria. However WG has observed that the SAMGs outlined for standard PHWRs, like addition of water to PHT system and to the calandria will still be effective for these reactors also.

It is seen that with the availability of moderator and vault water as heat sink under accident conditions and operator actions to restore the normal heat sink through the steam generators, the molten core can be retained inside the calandria. Hence there is no possibility of any failure of calandria vault and reactor building raft due to molten fuel-concrete interaction.

All Indian PHWRs use natural uranium as fuel and heavy water as both moderator and coolant. The reactor automatically trips at the beginning of a postulated accident and the design includes systems to ensure adequate sub-criticality thereafter on a continuing basis. Further SAMGs include mitigating actions by injecting light water in various systems and criticality of natural uranium fuel bundles in light water is not possible.

SAM programme envisages actions from control room/field with advice from technical support centre located on site. This centre should have facilities to obtain information on plant data as required for effective management of the situation. It is also seen that off-site emergency plans are well established for Indian NPPs and the off-site actions would not require any modifications due to implementation of SAMGs.

Annexure-6

REVIEW FINDINGS

A. Magnitude of external events and related issues

General

The magnitude 9.0 Tohoku Earthquake (Japan) and the associated tsunami brought into limelight the need for developing methodologies to derive parameters for NPPs corresponding to beyond design basis events. In this respect the current methodologies for estimation of design basis parameters and design approaches followed for NPPs were examined. Parameters to define beyond design basis events are proposed, where possible at this stage. In other cases guidelines are proposed for arriving at the site specific estimates.

Earthquake

<u>Current approach</u>

Assessment of seismicity and related hazards constitute a major part of the siting criteria for NPPs. An NPP is not located in seismic zone V, which is the zone of high seismicity in India as per the seismic design code published by the Bureau of Indian Standards (IS:1893-2002). Sites prone to ground failure phenomena during seismic events are also rejected. As an additional measure against ground failure, if there is an evidence of a seismic capable fault within 5 km of a site, the site is deemed unacceptable.

Adequate precaution is taken at detailed design stage to ensure that safety related SSCs of a NPP are capable of withstanding the effects of vibratory ground motion arising from the strongest earthquake derived from site specific studies following regulatory norms.

For estimation of ground motion corresponding to safe shutdown earthquake (SSE) level, the probable maximum earthquake potential of each seismogenic source (fault) is estimated. This also takes into account the maximized value of historical/recorded seismicity attributable to the fault. The source point of this maximized earthquake on the fault is brought nearest to the site and for this magnitude and distance combination, earthquake acceleration is determined. The exercise is repeated for all faults surrounding the site and the maximum of accelerations thus derived is adopted as design basis SSE level acceleration. The spectral shape is derived conservatively considering an ensemble of past earthquake records on geoseismically similar regions and local site soil/rock conditions. Detailed guidelines for derivation of design basis ground motion are given in AERB guide AERB/SG/S-11.

Recommendations

Safe shutdown earthquake

The current method of estimating ground motion corresponding to SSE is found to be quite appropriate. However, certain limitations because of lack of sufficient and relevant earthquake data and other uncertainties regarding site tectonics were recognized and these should be considered while revising the AERB guide on seismicity.

Beyond design basis earthquake event

Estimation of earthquake vibratory ground motion strongly depends on site data and the approach for addressing uncertainties. Therefore, defining a beyond design basis earthquake event for the purpose of siting and design of NPPs is a challenging task for Indian sites because of data inadequacy.

Keeping these limitations in mind, it is recommended that a beyond design basis earthquake event may be defined based on a comparative study of ground motion parameters derived from:

c) A postulated level of expected maximum acceleration/intensity of shaking at site, guided by the regional seismicity and local soil/rock site conditions, irrespective of earthquake source location, and
d) Maximization of earthquake hazard as evaluated following the procedure for SSE level earthquake.

Further work on estimation of magnitude of postulated beyond design basis earthquake event following the above recommended methodologies is in progress.

Flood hazard

<u>Current approach</u>

The current regulation allows use of probabilistic or deterministic approach for arriving at the design basis flood level (DBFL) for NPP sites. While following probabilistic approach, the design basis flood is calculated for a 1000 year mean recurrence interval of occurrence of the causative phenomena (viz. precipitation, storm, etc). While following deterministic approach, the biggest historical storm in the region is transported to the site area and is oriented in such a way that it maximizes the flood in the river or storm surge in the sea. Based on the estimated storm surge or flood, the DBFL at a site is estimated using detailed numerical models. The DBFL estimation is generally assigned to expert governmental agencies like the Central Water and Power Research Station and the Central Water Commission.

For inland sites, flooding could occur on account of overflowing of an adjoining river/lake, upstream dam break or intense precipitation in the surrounding region. Guidelines for evaluation of probable maximum precipitation and flooding due to failure of water control structures are covered in AERB/SG/S-6A. If the site is on the bank of an inland water body such as a reservoir or lake, the effect of seiches is also to be considered. In addition, intense precipitation in the local site can also cause flooding and hence the adequacy of site surface drainage has to be verified.

For a coastal site, the flooding hazards include those caused by cyclonic storms, tsunamis and local intense precipitation. Guidelines for evaluation of flooding due to cyclonic storms in coastal sites are covered in AERB/SG/S-6B. Values of the maximum tide, storm surge and wave run-up are added to arrive at the most conservative estimate of flood level above a defined reference level, generally mean sea level.

Flood levels due to tsunami are specified in AERB/SG/S-11 (1990), which are based on the historical data. As per this guide, Indian coast is divided into two regions (i) locations north of Karwar on west coast and, (ii) locations south of Karwar on west coast and the entire east coast. The specified tsunami heights are 3m and 2.5m respectively for these regions. These tsunami heights are superimposed on the maximum astronomical tide and are then added to the wave run-up to arrive at the DBFL. With the experience of the 2004 Indian Ocean tsunami, a more rigorous treatment of tsunami potential for coastal NPP locations was considered necessary. Accordingly AERB initiated work towards more rigorous estimation of tsunami hazard based on maximum potential tsunamigenic sources around the Indian coast.

Recommendations

Design basis flood

The existing procedures followed for estimation of DBFL are found to be appropriate except that the tsunami heights should be revised based on the results of the detailed work that is in progress.

It is noted that some statistical models were used in past for rainfall prediction irrespective of their fitness to the site specific data. Uncertainties in the estimates were also not addressed systematically. These aspects were deliberated upon and suitable recommendations have been made to correct these deficiencies.

It is also recommended that while assessing inland sites, a scenario involving combination of earthquake and flood due to dam break should be considered.

Beyond design basis flood

Interim recommendations, which can be considered as possible guidelines for reasonable quantification of beyond design basis flood level for safety assessment of NPPs, are as follows:

- In case of flooding caused due to dam break, a conservative upper bound analysis (in terms of postulated size, extent and duration of break) is suggested for beyond design basis event of dam break along with a rainfall/flood of 100 year return period
- The volume/flow considered for design basis flood conditions in Inland sites (i.e. value corresponding to mean + 1 sigma estimate for 1000 years return period) may be increased by 15% to arrive at a first order estimate of flood levels for inland sites as well as for carrying out the capacity assessment of site drainage corresponding to beyond design basis flood event. This is based on the analysis of past rainfall data at NPP sites. The analysis shows that a 15% increase of design basis rainfall (i.e. mean + one standard deviation corresponding to 1000 years return period) would

lead to at least one order higher rainfall event (i.e. return period increases from 1000 years to over 10000years).

- Considering available data for past storms, it is recommended that a pressure drop of 100 millibar, associated wind speed of 300 kmph for east coast and 240 kmph for west coast and a radius of 50 km may be taken as an upper bound value for the postulated beyond design basis cyclonic storm. The translational speed of storm may be considered as 40 kmph. The total height of the wave shall be summation of (a) maximum tide height, (b) storm surge height, (c) wave set up and (d) wind induced wave run-up.
- Major contribution of tsunami hazard to Indian NPP sites arises from Burma-Andaman-Sunda region which is about 1300 km from the nearest NPP site at Kalpakkam and, from Makran coast of Pakistan which is about 800 km from the nearest Tarapur site. Based on current understanding no possibility of any near source tectonic tsunami hitting our NPP sites can be visualized since the near shore faults are not large enough to cause tsunamigenic sea water displacement. Hence, unlike in Japan, NPPs along Indian coast would be subjected to either a local earthquake or a tsunami caused by a far away earthquake. However, as stated earlier, the existing postulates of tsunami heights may be revised based on the results of the detailed and rigorous analysis that is being done presently.
- A detailed site specific analysis using a validated numerical model should be carried out to arrive at accurate estimates of tsunami run-ups under all possible combinations and variations of source parameters. The evaluation shall include accurate near-shore data.
- For a multi-facility site, plant specific modifications like protection walls may cause modification of impact of phenomenon on the neighboring

areas. A global analysis that ascertains impact on all facilities shall be conducted before implementing any protection measures.

- In some locations, shore line bathymetry may be such that it causes amplifications in wave amplitudes. This shall be appropriately considered and if necessary, the site should be engineered against such amplified tsunami hazard.

Other Meteorological Hazards

Possible upper bound parameters that could be associated with beyond design basis meteorological phenomena were deliberated. For the purpose of design of NPPs, wind and temperature constitute major meteorological parameters that can have an impact on design.

<u>Current approach</u>

NPP structures are designed for severe wind corresponding to a return period of 1000 years. The extreme wind corresponding to a return period of 10000 years is used to assess whether wind induced missiles could be generated at an NPP site and if so their effects on items important to safety are also evaluated. Wind velocities are calculated following probabilistic approaches with site specific data and/or following code of practice for wind loads published by Bureau of Indian standards (IS:875 (part-3)-1987).

At present, no specific requirements exist with regard to the design basis values to be adopted for atmospheric temperature for design of structures, systems and components. However, the mean + 1 standard deviation estimate corresponding to 1000 year return period is being used in the design of structures in recent projects.

Recommendations

Design basis wind and temperature

The current procedures for estimation of design basis wind speed are considered adequate. Requirements for design basis temperature should be specified in regulatory documents.

Beyond design basis wind and temperature

Considering past data on extreme winds in the country and engineering judgment, it is recommended that, as a first order estimate, a beyond design basis wind speed may be postulated corresponding to 1000 year return period wind increased by 50% and rounded off to nearest 10m/s speed. The site specific values may be arrived at using the relevant procedures brought out in IS 875. This beyond design basis wind may be used for structural safety assessment.

For the purpose of structural evaluation of SSC as well as functional evaluation of safety systems related to ultimate heat sink, with respect to beyond design basis temperature, (mean + 2 standard deviation) for higher values and (mean - 2 standard deviation) for lower values corresponding to 1000 year mean return period may be considered as guidelines for beyond design basis value of temperature.

B. TAPS-1&2

Seismic Event

The critical SSCs of TAPS-1&2 have been designed to withstand SSE of 0.2g and this was confirmed during the seismic re-evaluation of the plant done during 2002-2006. However, some of the critical components like secondary steam generators and their supports, steam generator secondary lines, ECCS and Liquid Poison System have not been included in the seismic re-evaluation studies conducted as it was beyond the scope of the study as per the IAEA safety standard used for the purpose. These systems and components need to be studied for their seismic withstand capability.

During a seismic event the reactor will be shutdown by the control rods with the energy stored in the control rod drive accumulators. Cooling of the reactor core will be possible using emergency condenser, ECCS, CRD feed pumps etc. and containment isolation function will be achieved by the fail- safe containment isolation valves.

Thus it is seen that in case of a seismic event the basic safety functions will be achieved and capability to feed water to the reactor will also be available. However, filtered containment venting, if needed, may become unavailable as this system is not seismically designed.

Tsunami event and SBO

Considering a tsunami height of 3.0 m for the TAPS site as per AERB/SG/S11, the highest water surge level comes to 6.84m above mean seal level, which translates into a plant elevation of 107.63 ft, say 108 ft, for TAPS 1&2. For a tsunami surge height corresponding to 108 ft El., some of the safety systems including Class III emergency power supply system will be adversely affected. The event may therefore transgress into SBO.

Under SBO condition, while the reactor will be shutdown by the control rods with the energy stored in the control rod drive accumulators, there will be no feed capability to make up reactor water level. Core cooling through emergency condenser will be possible for a period of only 6-8 hrs. Availability of Auto Blowdown System (ABDS) to depressurize the reactor will be for limited number of operation of RVs till the station batteries drain out. ECCS system and filtered containment venting function will also be not available. It is therefore necessary that appropriate actions are taken to rectify these deficiencies at the earliest.

The committee was informed that NPCIL has already initiated action to address these problems. These actions also include inerting of the primary containment to preclude any hydrogen explosion in the event of a loss of coolant accident.

Concurrent failures

LOCA

As the safety related systems and components are designed for a value of 0.2g, the primary coolant piping is not expected to fail during a design basis seismic event. Emergency feed capability of 410 lpm (110 gpm) exists for make up at high pressure (100 kg/cm2) through CRD feed pumps which are designed to withstand design basis seismic events.

Fire

Fires are possible due to secondary effects of a seismic event. The nature and extent of fire will affect the recovery actions and the resources available for handling the external events. The fire water piping is not seismically qualified. This deficiency should be corrected.

Spent fuel storage

The spent fuel pool and the associated components are seismically designed. In the event of loss of cooling of water in the pool, with the water inventory available in the fuel pools, it will take 9 days before uncovery of the stored fuel bundles takes place with the full core unloaded in the spent fuel pool. However this time will be considerably reduced if the fuel pool gate towards the reactor cavity loses integrity and/ or there is loss of water due to sloshing during a seismic event. In the worst case of gross failure of the fuel pool gates this time gets reduced to 1.3 days. The fuel pool water make up capability may also be adversely affected by the non-availability of service water and demineralised water pumps as they are not seismically qualified, and also by flooding from a tsunami event. These deficiencies are to be corrected.

Severe accident management

While preparatory work on inerting of the primary containment has been taken up, there are no specific severe accident management guidelines available presently for TAPS 1&2. These should be made available based on a detailed severe accident analysis.

C. PHWRs

C.1 Station black out (SBO)

Provisions for SBO

All Indian PHWRs have emergency diesel generators to provide class-III power supply to essential station loads in case of off-site power failure (class-IV). Stations are also provided with class-II (UPS) and Class-I (battery banks) power which can cater to essential safety functions for a limited period, determined by the capacity of the battery banks, in case of SBO (i.e. loss of class-IV and class-III power). The likelihood of a SBO is remote with the redundancy and high reliability of DG sets and this is the reason that SBO is considered a beyond design basis event. All PHWRs, except RAPS-1&2, are also provided with 3x100% capacity diesel engine operated fire water pumps with provision for injecting fire water to steam generators, reactor core and end shields during SBO.

In case of RAPS-1&2, two additional diesel generators of 625 KVA capacity each (called flood DGs) were retrofitted and located at a high elevation to supply back-up power to essential loads during the postulated event of severe flooding caused by Gandhi Sagar dam break that may incapacitate the station DGs. This flood DG also caters to the power supply needs of fire water pumps. In addition water from the dousing tanks can be injected to SGs for ensuring thermo-siphon cooling of the core under SBO.

All NPPs have emergency operating procedures for dealing with SBO. Surveillance checks and testing on diesel fire pumps and flood DGs is carried out regularly. However the carbon steel fire water ring header is not covered by any surveillance programme and failures in the form of pin holes have been occasionally observed in the past. This deficiency needs to be corrected.

SBO upto 24 hours duration

Though SBO is considered as a beyond design basis event, SBO up to 24 hour duration has been analyzed considering crash cool down of PHT at the 6^{th} & 30^{th} minute after the onset of SBO. The analysis results indicate that core cooling can be maintained through thermo-siphoning for both the conditions by injecting fire water to the steam generators. It is also seen that temperatures of moderator water and vault water reach 90 & 80 deg C respectively at 24 hours after onset of SBO, without any cooling.

SBO up to 7 days duration

The temperature rise of moderator water and vault water in case of SBO beyond 24 hours needs to be assessed and alternate means are to be provided to limit these temperatures to within specified limits.

For assurance of thermo-siphoning, PHT system must be kept solid by inventory make-up to cater to system shrinkage and losses on account of leakages. Back-up provision for PHT make-up, that will remain operational during extended SBO, needs to be provided.

Sufficient inventory of water is available at all sites for catering to water supply requirements during SBO for at least seven days.

The diesel fire pumps can operate for 7 to 8 hours with diesel available in their day tanks. After this period, diesel can be transfered to the day tanks using hand operated pumps. This mode of transfer of diesel should be firmly established and demonstrated. Also, capability of the diesel fire pumps for sustained operation during extended SBO should be established by endurance testing and, periodic maintenance and surveillance checks should be instituted for assurance of this capability.

Availability of compressed air and power supply to I & C for actuation of essential valves and for indication of relevant process parameters also has to be ensured.

Further work

In PHWRs, the atmosphere is the ultimate heat sink (UHS) for decay heat removal till the PHT temperature is brought down to 150 deg. C by thermo-siphon cooling through SGs. Therefore, in this mode of decay heat removal, the availability of UHS is assured. However for long term maintenance of core cooling through shut down cooling heat exchangers after PHT temperature has come down, sustained availability of UHS needs to be ensured. It is proposed that a detailed study of all aspects for the situation of extended SBO coupled with loss of UHS be taken up in future. Similarly, a detailed review for assured core cooling for the case of simultaneous occurrence of SBO and beyond design basis flooding should also be done.

C.2 Electrical systems

External flooding

Class I, II, & III equipment and DG sets in all operating PHWR stations are located at elevations higher than the design basis flood levels (DBFL) considered in their original design. Subsequently the DBFLs for MAPS 1/2 and RAPS 1/2 were revised upward considering latest data and new postulates made for these sites. Accordingly, one extra DG set, called black-out DG, was installed at a suitable higher elevation in MAPS 1&2 to cater to both units. However *there is no standby provided for the black-out DG in MAPS*. In the case of RAPS 1&2, two additional DGs were provided and located at an elevation higher than the estimated level of flood caused by postulated break of Gandhi Sagar dam. However, pumps for transfer of diesel oil from main diesel storage tanks to day tanks, for the new DG sets and for the old DGs, are common. It has also been observed that the *vent lines of the main diesel storage tanks may get submerged under DBFL. This deficiency is to be corrected.*

For KAPS site, flood level expected in case of Ukai dam break is presently under study and safety of class I, II & III electrical equipment may need reassessment based on the outcome of this study.
All other stations meet the latest DBFL requirements.

Earthquake

Class I, II, & III equipment and DG sets in all stations were qualified for original design basis earthquake level for the corresponding sites. Design basis earthquake levels at MAPS-1&2 and RAPS-1&2 sites were recently updated. Both these plants have been re-evaluated for the current design basis earthquake levels and necessary back-fits were done. Other stations meet current earth quake requirements.

Switchyard and power grid

The effects of earthquake and/or flood on switchyards and off-site power supply lines and the time period in which these can be normalized after any such event need to be assessed. *Stations need to carry out a detailed study to assess the components in switchyards that are vulnerable to damage, spares required to be stored at site and expertise required to be outsourced for specialized jobs to bring back switchyard and the grid on line within a short period, subsequent to an earthquake/flooding event.*

Functional Readiness of Electrical Systems

In the plant walk-through conducted by stations, it was observed that all the equipment are properly supported and their present locations meet the design specifications.

Further work

Availability of class I, II and III power supplies and related issues in case of BDBEs of natural origin will be reviewed after the parameter values of PGA, flood levels etc. for such situations are available.

C.3 Control and instrumentation

Plant specific data on C&I were reviewed for five major safety areas

- Reactor trip and actuation circuits and associated C&I components along with monitoring for long term guaranteed shutdown status,
- Instrumentation for ensuring sustained core cooling and its status monitoring,
- Instrumentation for ensuring containment functions and monitoring of containment performance,
- Important process parameter displays required during prolonged SBO, and
- Instruments used for monitoring radiological status

Earthquake

Seismic trip is provided only at NAPS. In other sites only one unit is having seismic instrumentation and other units share the alarm signals. NPCIL informed that it has been recently decided to provide seismic trip at all the NPPs.

Seismic qualification for safety related instruments at newer NPPs (KGS-1 onwards) is for a PGA in the range of 3 to 3.5g at their anchor points. Tests/analyses conducted for safety important instruments including associated panels for older units indicate their seismic qualification for PGA of 1g. Seismic qualification for alarm units and field panel are in the range of 2.5 g to 2.88 g. Adequacy of these seismic withstand levels need to be confirmed.

Backup instrument air accumulators are provided for important valves/instruments. Availability of these accumulators under a beyond design basis seismic event needs to be ensured.

External flooding

Sensors for reactor tripping function and instrumentation linked to moderator and fire water injection are located inside RB and hence their location with respect to elevations is not important from flooding consideration.

Starting batteries and chargers of diesel engine driven fire water pumps are located at 100 m EL (grade level) and hence may get submerged during beyond design basis flood. As manual cranking of the engines may not always be possible, relocation of the batteries and chargers to a higher level should be done.

Some components of control power supplies (250 V AC/48V DC/24V DC) are located at 100 m EL, except for RAPS-1&2 and MAPS where they are at higher elevation.

The C&I required for ABFPs and ECCS are located at 91 m elevation but inside the buildings. Flooding is possible in these areas only if the water level outside the buildings rises above 100 m EL. However, possibility of any water leakage through cable or pipe penetrations needs to be checked.

Seismic switch/sensors are located outside RB and at lower elevations than grade level. Water tight enclosures for them need to be ensured.

Station blackout

Station battery back-up for control power supplies are adequate for about 1 hr without any load cut off, except for RAPS-1&2 & MAPS where they will suffice for only 30 minutes. With shedding of un-important loads as per station operating procedures, the availability of batteries can be slightly extended. However, it is necessary that battery bank capacities are augmented such that power supply for vital I&C items for performance of all safety functions and for monitoring of related parameters is assured during an

SBO of up to 7 days duration. Details in this regard have to be worked out to decide on the extent of augmentation.

In RAPS-1&2 and MAPS, power and control battery banks are shared. Aspects related to availability of critical control functions under SBO should be reassessed for these plants.

Miscellaneous

Temperature and humidity ratings for instruments outside Reactor Building (RB) are in the range of 45 to 55 Deg C at 95% RH. Further review is required with respect to environment qualification to ensure operability of instruments at specified environmental conditions.

Although functioning of COIS/CRCS/PIS is not important during SBO, their availability with respect to recording of information for any post-accident analysis is to be examined.

Consideration need to be given for making provisions for post accident sampling of the atmosphere inside containment and of important process fluids.

In older plants the supplementary controls rooms (SCRs) were back-fitted. The control and monitoring facilities provided at these SCRs need to be re-looked from the point of view of sensor separation, adequacy of power supply, etc.

C.4 Spent fuel storage facilities

Current status of cooling of spent fuel pools

An assessment of water loss due to boiling-off from the spent fuel storage pools at the PHWRs was done to check the time available before the pool water level goes down to a) below the minimum level specified from shielding consideration, and b) below the top of the stored spent fuel bundles, considering no cooling of the pool water and no water make-up to the pool. For this assessment, it was conservatively assumed that the pool is loaded up to its design capacity with spent fuel discharged at design average burn-up from the reactors operating at rated power including one full core charge that was recently transferred into the pool over a period of 50 days. The results of the assessment are shown in the table below.

Sr. No.	Parameter	RAPS-1&2	MAPS -1&2	NAPS - 1&2	KAPS - 1&2	KGS-1-4 & RAPS-3-6	TAPS - 3&4
1.	Time taken by bay water to reach 60^0 C (Days)	0.93	0.63	2.35	2.11	1.57	1.71
2.	Time taken by bay water to reach 100^0 C (Days)	2.8	1.89	7.04	6.34	4.72	5.14
3.	Time taken to reach minimum shielding level (Days)	5.75	3.89	26.31	23.7	17.67	23.26
4.	Total time taken for exposure of top fuel bundles	16.81	11.37	52.59	47.37	35.25	39.99

	to air (Days)						
5.	Boil-off rate / minimum makeup required (Te/Hr)	1.72	2.53	2.86	2.73	2.76	4.34

The following can be observed from the table:

- The time available before the water level reaches minimum level prescribed for shielding ranges from minimum 3.9 days (MAPS) to maximum 26.3 days (NAPS).
- In case of MAPS-1&2 and RAPS-1&2 the emergency preparedness for making up water should match the requirements of 3.9 and 5.75 days respectively with sufficient safety margin.
- The time available before the top layer of fuel bundles is exposed is 16.81 days for RAPS-1&2, 11.4 days in case of MAPS-1&2 and for other reactors it is more than one month.
- The bulk boiling of water can start within as low as 1.9 days (MAPS) to 7 days (NAPS). It should be checked whether such boiling can affect the structural integrity of the pool.
- An external water hookup provision for maintaining the required level in SFSB for all the operating plants should be implemented irrespective of any other design provision. This make up capability should remain unaffected by the impact of the external events including SBO.
- The present instrumentation for monitoring level and temperature of water in the pools and the radiation fields inside the SFSB may not be adequate during BDBA conditions like those experienced at Fukushima.

Accordingly, suitable passive monitoring/indication facilities should be provided for these parameters.

Further work

Detailed site specific safety assessment of SFSBs should be done further with respect to the following.

- Structural and leak tightness integrity of pools to withstand external events
- Earthquake induced water sloshing losses
- Need and adequacy of power supply for SFSB water cooling and its availability during external events
- Stability of fuel storage racks and mechanical handling equipment like overhead cranes, including possibility of their falling and creating secondary damages

C.5 Heavy water upgrading plants

Current safety status

The Heavy Water Upgrading Plant (UGP) is an auxiliary system to nuclear power plant and its availability or operation does not have any effect on power plant operational safety. In case of total power failure after any external event, the Upgrading Plant will go to safe shutdown mode. Upgrading Plant building is a single storey RCC structure. Two Distillation columns are housed in a structural steel tower of overall size 7 m x 7 m x 58 m height. The total height of distillation column assembly is 55 meters. These facilities are characterized with low/limited radioactivity and hence classified as the lowest safety class and seismic category corresponding to operating basis earthquake (OBE). An earthquake of OBE level and flooding will not cause any significant radiological impact at the site and no impact in the public domain. However, the steel tower and distillation columns would require assessment of structural integrity for beyond design basis natural events.

Further work

In case of external events like earthquake, tsunami, flood, cyclone of beyond design basis level, a limited plant emergency around this facility may occur. Further study and analysis of existing building structure and equipment, specifically tall structural steel tower and distillation columns, need to be carried out to assess margins available to cater to postulated level of earthquake and impact of any failures on nearby plant facilities. This issue is site-specific.

D. Radioactive waste management facilities at NPP sites

Near Surface Disposal Facilities (NSDF)

In India, NSDFs for disposal of solid radioactive waste generated from NPP operation are co-located with the NPP within the exclusion boundary. These are designed, constructed and operated based on a combination of engineered (structure, waste matrix, backfill materials) and passive safety (geospheric environment) features. During siting, design and construction of NSDFs, site characteristics and external natural events, to the extent applicable, are taken into account.

Safety assessments of NSDFs are available for all NPP sites. These assessments are based essentially on the postulated ingress of water into the radioactive waste disposal system and leaching of radionuclides from the waste matrix and their subsequent migration into the groundwater. A network of monitoring bore-holes is provided at each NSDF to assess the performance of disposal system by sampling and analysis of ground water. The radiological impact during normal conditions, as seen from these analyses, is found to be well within the specified limits. Preliminary study indicates that even in the case of a Fukushima type of extreme natural event, the possibility of any significant radiological impact, especially in the public domain, is minimal.

Spent Resin Storage Tanks

At TAPS-1&2 (and earlier at MAPS-1&2), spent ion exchange resins generated are stored in underground tanks. The ingress of water into the storage tanks or failure of storage tanks has the potential of spreading contamination in the surrounding area. The radionuclide monitoring data in bore wells around the spent resin storage facilities shows safe storage as activity levels are below the detection limit (BDL). However, in case of any severe event of natural origin, the integrity of these tanks may be lost that has the potential of causing widespread radiological hazard though limited to within the NPP premises. Removal of the resins from these tanks and immobilization of the resins should be done at the earliest and the practice of storage of spent resin in underground tanks should be discontinued.

Radioactive Liquid Effluent Storage and Treatment

Radioactive liquid effluents generated from PHWRs are collected in the liquid effluent segregation and storage system (LESS) or Liquid Effluent Management Plant (LEMP) and subsequently transferred to the treatment / disposal facility. The liquid waste storage tank area i.e. civil structure of LESS and the surrounding dykes are designed and constructed as per the seismic qualification of OBE. The LESS/LEMP tanks are designed for storage capacity equivalent to two days effluent generation from the plant.

The holding capacity of dyke tanks located in a multi-unit site is equivalent to about 21 days of waste volume generation.

In the event of failure of tanks in a design basis accident, the entire volume of water will be retained within the dyke. The management and disposal of the water inside the dyke wall will be carried out on a campaign basis. Assessment of the postulated consequences of beyond design basis external events e.g. simultaneous failure of tanks and dykes requires further study. However, even in such a case, there is no possibility of any significant impact in the public domain due to the large distance of over 1 km between the dykes and the exclusion boundary.

Further Work

An estimate of volume and activity of the radioactive effluents likely to be generated during any postulated nuclear accident including any accident due to extreme natural events needs to be made. Subsequently, a scheme for treatment of such effluents is to be worked out. Such systems are recommended to be modular in design which can be transported to the affected site and pressed into service on short notice.

Annexure-7

RECOMMENDATIONS FOR EXTREME NATURAL EVENTS OTHER THAN EARTHQUAKE AND TSUNAMI

(i) Extreme flood due to cyclones at coastal sites: Considering available data on past cyclonic storms, a pressure drop of 100 *millibar*, associated wind speed of 300 *kmph* for east coast and 240 *kmph* for west coast, translational speed of 40 *kmph* and a radius of 50 *km* may be taken as upper bound values for the postulated extreme cyclonic storm.

(ii) In case of flooding caused due to postulated dam break, a conservative upper bound analysis (in terms of postulated size, extent and duration of break) is recommended along with a rainfall/flood of 100 year return period.

(iii) Considering past rainfall data at NPP sites, an extreme flood event to be used for assessment of safety margin against flooding at NPP sites, which would correspond to at least one order higher return period, may be defined as follows:

- Site surface drainage: 15% additional over the calculated design basis rainfall (mean plus one standard deviation corresponding to 1000 years return period).
- Inland sites extreme flood: 15% additional volume/flow of water over the calculated probable maximum flood of 1000 year return period.

(iv) Regulatory requirements for design basis atmospheric temperature should be specified as (mean + one standard deviation) corresponding to

1000 year mean return period for high value and (mean - one standard deviation) corresponding to 1000 year mean return period for low value. Beyond design basis temperature values for margin assessment may be specified as (mean + two standard deviations) corresponding to 1000 year mean return period for high value and (mean - two standard deviations) corresponding to 1000 year mean return period for low value.

(v) An extreme wind event for safety margin assessment should correspond to over 10000 year return period. In lieu of this, the design basis wind speed increased by 50% and rounded off to nearest 10 m/s speed may be considered as an extreme wind event for margin assessment.

Annexure-8

SAFETY ASSESSMENT OF KUDANKULAM NUCLEAR POWER PLANT UNITS-1&2 (KK NPP1&2) IN THE WAKE OF FUKUSHIMA ACCIDENT

Two Units of VVER Pressurized Water Reactors (Model V-412) each of 1000 MW rating are being built at the Kudankulam Site in Tamil Nadu. Initial commissioning activities for Unit # 1 have started with AERB issuing clearance for "Hot-Run" on June 30, 2011. Construction of Unit # 2 is in an advanced stage of completion.

The design of KK NPP incorporates a number of engineered safety features (ESFs) for catering to design basis accidents (DBAs) and beyond design basis accidents (BDBAs), and several other design safety features.

ESFs for catering to DBA

- Emergency Core Cooling System (ECCS)
- Secondary circuit protection against over-pressurisation
- Emergency Gas Removal System
- Fission Products Removal and Control Systems
- Emergency Safety Boron Injection System
- Quick Boron Injection System (QBIS)

ESFs for catering to BDBA

- Passive Heat Removal System (PHRS)
- Additional system for core passive flooding
- Annulus passive filtering systems (APFS)
- System for retaining and cooling of molten core

Other salient design safety features

- 4 x 100% active safety system trains and 4 x 33% passive safety system trains
- Large water inventory in I & II stage ECCS hydro-accumulators
- Automatic Reactor Scram on seismic signal
- Battery banks with 24 hrs capacity
- Sea water pumps located at more than 2.2m above design basis flood level (DBFL).
- Safety related buildings and structures located at least 3.0m above DBFL.
- A shore protection rubble wall

Post-Fukushima safety Assessment

A Task Force (TF) constituted by NPCIL carried out safety assessment of KKNPP-1&2 in the light of Fukushima accident and its findings were reviewed by the AERB's Advisory committee on Project safety review of light water reactors (ACPSR-LWR) and the AERB committee on safety review of Indian NPPs in the light of Fukushima accident. Salient points emerging from the assessment and its reviews are given below.

- Back up provisions from alternate sources should be made for
  - Charging water to secondary side of SGs
  - Make-up of borated water to spent fuel pools
  - Injection of borated water in the reactor coolant system
- Seismic qualification of emergency water storage facility and augmentation of its storage capacity for core decay heat removal for a period of at least one week
- Mobile self powered pumping equipment for emergency use

- Facility for monitoring safety parameters using portable power packs
- Finalization of emergency operating procedures for BDBA conditions
- Primary Containment to be assessed for ultimate load bearing capacity.
- Doors and barrels of airlocks to be qualified for proof test pressure
- Ensuring that highly active water used for cooling the core catcher vessel under BDBA is contained inside the primary containment
- Reconfirmation of design adequacy of hydrogen management system
- Environmental qualification of core catcher temperature monitoring system
- Adequacy of design provision for remote water addition to core catcher
- Adequacy of instrumentation for monitoring plant status during BDBA
- Details of margin available on location of various safety related SSCs above DBFL should be reviewed again.
- Need for design provision for containment venting, that has been deleted, should be re-examined.
- The backup sources for water injection to SG secondary side should be seismically qualified.
- Provisions for addition of water to core catcher require a detailed study, to ensure that there is no possibility of any steam explosion.
- Provision of additional backup power supply sources for performing essential safety functions, like air cooled DGs located at a high elevation, should be considered.

The recommendations are being examined and NPCIL's response would be reviewed in ACPSR-LWR before initial fuel loading in unit-1.

Annexure - 9

LIST OF ABBREVIATIONS

ABDS	Auto Blow Down System
ABFP	Auxiliary Boiler Feed Pump
AC	Alternating Current
ACPSR	Advisory Committee for Project Safety Review
AERB	Atomic Energy Regulatory Board
AERBSC-EE	AERB Committee to Review Safety of Indian NPP against External Events in Light of Fukushima
BARC	Bhabha Atomic Research Centre
BDBA	Beyond Design Basis Accident
BDBE	Beyond Design Basis Event
BDL	Below Detection Limit
BWR	Boiling Water Reactor
C&I	Control & Instrumentation
COIS	Computerised Operator Information System
CRCS	Control Room Computer System
CRD	Control Rod Drive
CS&PI System	Core Spray & Post Incident System
DBA	Design Basis Accident
DBE	Design Basis Event
DBFL	Design Basis Flood Level
DBGM	Design Basis Ground Motion
DC	Direct Current
DEGB	Double Ended Guillotine Break
DG	Diesel Generator
ECCS	Emergency Core Cooling System
EG	Expert Group
EL	Elevation

EOP	Emergency Operating Procedure
GSI	Geological Survey of India
IAEA	International Atomic Energy Agency
INES	International Nuclear and Radiological Event Scale
IP	Isotopic Purity
ISI	In-Service Inspection
JSCE	Japan Society of Civil Engineers
KAPS	KakraparAtomic Power Station
KGS	Kaiga Generating Station
KKNPP	Kudankulam Nuclear Power Plant
KVA	Kilo Volt Ampere
LEMP	Liquid Effluent Management Plant
LESS	Liquid Effluent Segregation and Storage System
LOCA	Loss of Coolant Accident
LWR	Light Water Reactor
MAPS	Madras Atomic Power Station
MRI	Mean Recurrence Interval
MWe	Mega Watt electrical
NAPS	Narora Atomic Power Station
NPCIL	Nuclear Power Corporation of India Limited
NPP	Nuclear Power Plant
NPSD	Nuclear Projects Safety Division
NSDF	Near Surface Disposal Facility
OBE	Operating Basis Earthquake
PCV	Primary Containment Vessel
PGA	Peak Ground Acceleration
PHT system	Primary Heat Transport System
PHWR	Pressurized Heavy Water Reactor
PIS	Plant Information System
PSA	Probabilistic Safety Analysis

PSR	Periodic Safety Review
PWR	Pressurized Water Reactor
R&D	Research & Development
RAPS	Rajasthan Atomic Power Station
RB	Reactor Building
RCC	Reinforced Cement Concrete
RH	Relative Humidity
RHRS	Residual Heat Removal System
RPV	Reactor Pressure Vessel
RV	Relief Valve
SAMG	Severe Accident Management Guidelines
SBO	Station Black Out
SCR	Supplementary Control Room
SFSB	Spent Fuel Storage Building
SG	Steam Generator
SSC	Structures, Systems and Components
SSE	Safe Shutdown Earthquake
TAPS	Tarapur Atomic Power Station
UGP	Upgrading Plant
UHS	Ultimate Heat Sink
UPS	Un-interrupted Power Supply
WG	Working Group