CONVENTION ON NUCLEAR SAFETY

Special National Report by the Government of the

Islamic Republic of Pakistan

for

Second Extraordinary Meeting

August, 2012

Prepared on behalf of

The Government of
Islamic Republic of Pakistan

by

Pakistan Nuclear Regulatory Authority

and

Pakistan Atomic Energy Commission

ABSTRACT

On behalf of the Government of Pakistan, Pakistan Nuclear Regulatory Authority submits this Special National Report for peer review at the Second Extraordinary Meeting of the Convention on Nuclear Safety at the International Atomic Energy Agency in August 2012. The report presents the steps taken by the Government of Pakistan to ensure safety of its nuclear installations through implementation of lessons learnt so far in response to the accident at Fukushima Daiichi Nuclear Power Plants.

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ABBREVIATIONS

AAC	Alternate AC
AC	Alternating Current
ABCC	Auto Boiler Crash Cooling
AECL	Atomic Energy of Canada Limited
AOPs	Abnormal Operating Procedures
ATOG	Abnormal Transient Operation Guidelines
BDBA	Beyond Design Basis Accidents
BYW	Bay Water System
C-1	Chashma Nuclear Power Plant Unit 1
C-2	Chashma Nuclear Power Plant Unit 2
C-3	Chashma Nuclear Power Plant Unit 3
C-4	Chashma Nuclear Power Plant Unit 4
CANDU	Canada Deuterium Uranium
CAs	Change Approvals
CERO	CHASNUPP Emergency Response Organization
CFR	Code for Federal Regulations
CHASCENT	CHASNUPP Center for Nuclear Training
CHASNUPP	Chashma Nuclear Power Plant
CNPGS	Chashma Nuclear Power Generating Station
COG	CANDU Owners Group
CPDS	Critical Parameters Display System
CSS	Committee on Safety Standards
D_2O	Heavy Water
DBA	Design Basis Accident
DC	Direct Current
DDMA	District Disaster Management Authority
DG	Diesel Generator
DNPO	Directorate of Nuclear Power Operations
DNS	Directorate of Nuclear Safety
DQA	Directorate of Quality Assurance
DSW	Dousing Spray Water System
DX	Diesel Generator building
ECC	Emergency Control Center
ECCS	Emergency Core Cooling System
EDGs	Emergency Diesel Generators
EFW	Emergency Feed Water system
EHWT	Emergency Heavy and Light Water Transfer system
EMG	Emergency Management Group
ENSREG	European Nuclear Safety Regulators Group
EOPs	Emergency Operating Procedures
EP	Emergency Planning
EPA	Environmental Protection Agency
EPIPs	Emergency Plan Implementing Procedures
EPP	Emergency Preparedness Plan
EPR	Emergency Preparedness and Response
EPZ	Emergency Planning Zones

ERC	Emergency Relief Cell
ERT	Emergency Response Team
ESF	Engineered Safety Features
ESG	Emergency Support Group
EST	Emergency Sump Transfer System
FATA DMA	Federally Administered Tribal Areas Disaster Management Authority
FIJW	Medium Pressure Forced Emergency Injection Water System
FRAP	Fukushima Response Action Plan
FX	Fuel Building
g	Gravitational Acceleration
GoP	Government of Pakistan
HDTRs	High Density Tray Racks
HH-HSI	High High Head Safety Injection
HPME	High Pressure Melt Ejection
HPSI	High Pressure Safety Injection
IAEA	International Atomic Energy Agency
ICRC	International Committee of the Red Cross
ICT DMA	Islamabad Capital Territory Disaster Management Authority
IEC	Incident and Emergency Center
IJW	Emergency Injection Water System
IPR	Internal Peer Review
IPSART	International PSA Review Team
IRS	International Reporting System for Operating Experience
IRRS	Integrated Regulatory Review Services
IRRT	Integrated Regulatory Review Team
ISARMAP	Integrated Safety Review Master Plan
ISMES	Istituto-Sperimentale-Modelli-E-Strutture
K-1	Karachi Nuclear Power Plant (Unit-1)
KI	Potassium Iodide
KINPOE	Karachi Institute of Nuclear Power Engineering
KOFREP	KANUPP Off-site Emergency Response Plan
KONREP	KANUPP On-site Emergency Response Plan
KRERC	KANUPP Radiological Emergency Response Committee
LBLOCA	Large Break LOCA
LOCA	Loss of Coolant Accident
LPSI	Low Pressure Safety Injection
MCCI	Molten Core Concrete Interaction
MCR	Main Control Room
MPa	Mega pascal
Mph	Miles per hour
MSL	Mean Sea Level
MW_e	Mega watt electrical
MW_{th}	Mega watt thermal
NACs	National Assistance Capabilities
NADMA	Northern Area Disaster Management Authority
NaOH	Sodium Hydroxide
NCA	National Competent Authority
NDMA	National Disaster Management Authority

NCMC	National Crisis Management Cell
NDMC	National Disaster Management Commission
NDMO	National Disaster Management Ordinance
NDRP	National Disaster Response Plan
NEMS	Nuclear Emergency Management System
NERSP	National Environmental Radioactivity Surveillance Program
NEWS	Nuclear Event Web based System
NGOs	Non Governmental Organizations
NPP	Nuclear Power Plant
NPS	Nuclear Power Station
NRECC	National Radiation Emergency Coordination Center
NREP	National Radiological Emergency Plan
NUSSC	Nuclear Safety Standards Committee
NWP	National Warning Point
NX	Nuclear Auxiliary Building
OBE	Operating Basis Earthquake
OPPs	Operating Policies and Principles
OSART	Operational Safety Review Team
PAEC	Pakistan Atomic Energy Commission
PAMS	Post Accident Monitoring System
PAR	Passive Autocatalytic Recombiners
PAZ	Precautionary Action Zone
PDMA	Provincial Disaster Management Authority
PGA	Peak Ground Acceleration
PHT	Primary Heat Transport System
PHWR	Pressurized Heavy Water Reactor
PIEAS	Pakistan Institute of Engineering and Applied Sciences
PIEs	Postulated Initiating Events
PMD	Pakistan Meteorological Department
PMF	Probable Maximum Flood
PNRA	Pakistan Nuclear Regulatory Authority
PROSPER	Peer Review of Operational Safety Performance Experience
PRZ	Pressurizer
PSA	Probabilistic Safety Assessment
PSDP	Public Sector Development Program
PWR	Pressurized Water Reactor
QNPC	Qinshan Nuclear Power Company
RA SSC	Radiation Safety Standards Committee
RANET	Response and Assistance Network
RLE	Review Level Earthquake
RLO	Re-Licensing Outage
RPM	Radiation Portal Monitor
RWST	Refueling Water Storage Tank
RX	Reactor Building
SAC	Safety Analysis Center
SAF	Auxiliary Feedwater System
SAMGs	Severe Accident Management Guidelines

SAT	Self Assessment Tools
SB-LOCA	Small Break Loss of Coolant Accident
SBO	Station Blackout
SCF	Reactor Cavity Flooding System
SCI	Containment Isolation System
SCS	Containment Spray System
SCV	Chemical and Volume Control System
SDMA	State Disaster Management Authority
SDV	Screening Distance Value
SFP	Spent Fuel Pool System
SG	Steam Generator
SHR	Containment Hydrogen Recombining System
SHM	Containment Hydrogen Mixing System
SI	Safety Injection
SNERDI	Shanghai Nuclear Engineering Research and Design Institute
SOK	Safe Operation of KANUPP
SOPs	Standard Operating Procedures
SPD	Strategic Plans Division
SRC	Reactor Coolant system
SRH	Residual Heat Removal System
SSCs	Structures, Systems and Components
SSE	Safe Shutdown Earthquake
TEWS	Tsunami Early Warning System
TRANSSC	Transport Safety Standards Committee
TSC	Technical Support Committee
UHS	Ultimate Heat Sink
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation
UPS	Un-interruptible Power Supply
USNRC	United States Nuclear Regulatory Commission
VCW	Reactor Vault Cooling System
WASSC	Waste Safety Standards Committee
WANO	World Association of Nuclear Operators
WES	Essential Water System
WHO	World Health Organization

PART I

INTRODUCTION

1. Introduction

1.1 General

This Special National Report has been prepared to comply with the obligations set down in item 11 of the Summary Report from the 5th Review Meeting of the Convention on Nuclear Safety.

The report comprises four parts. Part I includes the introduction, general information about the national policy regarding use of nuclear power technology and summary of steps taken by the regulator and licensee in response to the accident at Fukushima Daiichi Nuclear Power Plants (NPPs). Part II provides general information about the design of operating nuclear power plants in Pakistan. This has been done to facilitate comparison with the Fukushima Nuclear Power Plants Design. Part III of the report presents the details of specific actions taken by Pakistan in various topical areas as identified in the IAEA guidelines. Part IV of the report provides lists of actions planned/taken by the regulator and licensee of nuclear power plants in response to the experience feedback of accident at Fukushima NPS.

1.2 Pakistan's Policy on use of Nuclear Power Technology

Growing energy needs necessitate exploration and use of all possible sources of energy including Hydel, Fossil fuel, Nuclear and Renewables. Pakistan's policy regarding use of nuclear power technology, therefore, remains the same as before, but with more stringent controls on safety and its implementation at Nuclear Power Plants. This was re-iterated by the Pakistan Delegation during the IAEA Ministerial conference in June 2011 as:

"Safe and sustainable nuclear energy is essential to advance our development agenda as also considered by many member states. My country attaches the highest priority to nuclear safety. We believe that it is imperative for the sustainability and expansion of nuclear power. There can be no compromise on this. There can be no room for complacency."

This policy was reaffirmed by Pakistan during the 55th Session of IAEA General Conference and the endorsement of the IAEA Nuclear Safety Action Plan.

It is the need of the hour to learn lessons from the Fukushima accident and the way it was managed by Japan. Pakistan has a strong will to improve the safety of its nuclear power plants to enhance confidence in their design and management systems and to thereby enhance confidence of the public in the use of nuclear energy. This requires detailed re-assessments and analyses. Sharing of relevant information by Japan with the international community is crucial in this respect. As additional information is received, assessments/improvements will continue to be performed accordingly by Pakistan.

1.3 Summary of Actions Taken Post Fukushima

Pakistan has been closely observing the events at Fukushima. It is understood that this accident occurred as a result of natural events far larger in magnitude than those considered in the design of nuclear power plants. Accordingly, Pakistan initiated a re-assessment of natural hazards to determine whether the current design bases remain valid. In this respect, Pakistan is revisiting and re-assessing the design & operational safety, accident management procedures, training and emergency preparedness plans.

To ensure public protection, Pakistan restricted import of edible goods from Japan without measurement of radiation levels by PNRA and provision of radiation free certificate issued by relevant authority of Japan.

Pakistan also actively participated in the development of the IAEA Nuclear Safety Action Plan and endorsed the finalized Action Plan during the IAEA Board of Governors Meeting and subsequently during the IAEA General Conference in September, 2011.

1.3.1 Actions taken by PNRA

The actions taken by Pakistan Nuclear Regulatory Authority (PNRA) in response to Fukushima accident included the following:

i. National Response and Emergency Coordination Center (NRECC) of PNRA was activated immediately from “Normal/Ready mode” to “Full Activation mode”. NRECC remained operational 24/7 for five months after the accident.

ii. Data shared through official website of Japanese authorities and Incident and Emergency Center (IEC) of the IAEA was analyzed and trended.

iii. Environmental monitoring was started at various locations within the country to monitor build up of any radioactivity.

iv. After enough information of the accident was available, PNRA asked Nuclear Power Plants in Pakistan to revisit /re-assess the design and safety features and accident management guidelines as well as emergency preparedness measures. PNRA also required the nuclear power plants to submit reports of the re-assessment identifying the actions needed for necessary safety improvements. The re-assessments required by PNRA comprised the following:

    a. Re-assessment of natural hazards
    b. Availability of infrastructure (necessary for plant safety such as AC power supply sources, heat sink)
    c. Consideration of the station black-out condition (loss of all AC power) for longer duration
    d. Re-evaluation of the design features provided at nuclear power plants for controlling and removing hydrogen such as hydrogen recombining system, hydrogen mixing system
    e. Re-evaluation of the Emergency Operating Procedures (EOPs) and Severe Accident Management Guidelines (SAMGs)
    f. Re-evaluation of the off-site Emergency Preparedness Plans (EPPs) including Emergency Plan Implementing Procedures (EPIPs)
    g. Re-evaluation of the training program in the light of the Fukushima accident with particular emphasis on the limitations of simulator since such scenarios may not be possible to simulate.

v. PNRA chalked out a plan to review its regulations in the light of Fukushima lessons learnt so far and identify areas for improvements. Several Modifications have been proposed in the existing regulations as a result.

vi. PNRA has again requested IAEA to conduct an IRRS mission in March 2013 (Preparatory mission by IAEA planned in 2012) in line with the IAEA Nuclear Safety Action Plan.

1.3.2 Actions taken by PAEC

Pakistan Atomic Energy Commission (PAEC) is cognizant of implementing safety at its Nuclear Installations. In the wake of the Fukushima accident, PAEC recognizes that in addition to the need of ensuring existing safety in design and operation, new measures for safety enhancement need to be identified.

PAEC carried out initial safety assessment of its NPPs based on available information on Fukushima accident and formulated a "Fukushima Response Action Plan" (FRAP) in May 2011, which identified certain actions in the following review areas:

i. External Natural Hazards
ii. Make-Shift AC Power
iii. DC Power Capacity
iv. Fire Protection and Control
v. Emergency Core Cooling
vi. Hydrogen Hazard
vii. Containment Integrity
viii. Spent Fuel Cooling
ix. EOPs, SAMGs (On-Site Actions)
x. Emergency Preparedness

A team of specialists from the Corporate office visited NPP sites to review the progress and scope of the FRAP, so that any gaps in meeting international concerns are also addressed. As a result, additional actions were proposed. Details are presented in specific topical areas in Part III of this report.

PART II

General Information about Nuclear Installations in Pakistan

2. General Information about Nuclear Installations in Pakistan

Pakistan has three nuclear power plants in operation: K-1, C-1 and C-2 with a total installed capacity of 725 MWe. Two more units, C-3 and C-4 are currently under construction. The operating license is held by Pakistan Atomic Energy Commission (PAEC).

2.1 General Information about K-1

K-1 is a heavy water moderated and cooled natural uranium fuelled, horizontal pressure tube reactor. The gross plant rating is 137 MWe and the corresponding net output is 125 MWe.

Plant Feature	Summary Description
Reactor	• Tubed calandria vessel of austenitic stainless steel • 208 coolant tubes (Zr-2.5%Nb alloy) • Coolant flow is in opposite directions in adjacent coolant tubes • The reactor vault filled with light water, serves both as a thermal shield and a cooling medium
Primary Heat Transport System	The Primary Heat Transport System (PHT) consists of the steam generators (boilers), pumps, piping, valves and the necessary auxiliaries. Each of the six boilers (three on each side) consists of an integral vertical steam drum and heat exchanger. Heavy water coolant is pumped through the coolant tubes to remove the heat generated in the fuel and to transport it to the boilers.
Safety Systems	The Emergency Core Cooling System (ECCS) is called Emergency Injection Water System in K-1, comprising two systems viz. Medium Pressure Forced Emergency Injection Water System (FIJW) and Low Pressure Emergency Injection Water System (IJW). The system is designed to detect the loss of primary system coolant, and supply water to cool the core. The Emergency Heavy and Light Water Transfer System (EHWT) is used for addition of heavy water to facilitate the operator to crash cool down and depressurize the Primary Heat Transport system in case of abnormal internal or external leakage. The reserve Heavy Water (D_2O) inventory will be helpful for the operator to gain time for leak isolation (if possible) before actuation of IJW system. Emergency Sump Transfer System (EST) is used to facilitate the transfer of spilled water from reactor building sumps located in south transmitter room and Process Water room to moderator area sump after Loss of Coolant Accident (LOCA) outside the boiler room.

	Containment Building Spray Cooling System has been provided to depressurize and reduce the temperature of the reactor building atmosphere following a major reactor accident. Depressurization is achieved by spraying chilled water (50°F / 10°C) in the reactor building through Dousing Spray Water System (DSW) which cools and condenses the air and vapor respectively, thus lowering the building pressure. Containment Isolation feature isolates the containment in the unlikely event of an accident. The containment atmosphere is isolated, on building pressure signal, from the environment by the use of isolation valves (dampers) and other barriers for all pipelines, which penetrate the containment.
Containment	Containment building has an internal diameter of 115 ft (35.05 m) and a height from the top of the base slab to the underside of the dome, of 122.5 ft (37.3 m). The building is designed for a leakage rate of 2.4%/day of containment air volume under design pressure conditions 28 psig (0.19 MPa).
Spent Fuel Storage	Irradiated fuel discharged from reactor is stored in the spent fuel storage located inside the service building. The Storage Bay is filled with water to provide shielding and to remove the heat of the spent fuel.
Electric Power System	The station generator delivers electrical power at 15 kV to the main output transformer, the station service running transformer and the 11 kV transformer of the switchyard extension building. The main output transformer supplies power at 132 kV to the utility system transmission lines through the station switchyard bus. When the normal source of power is not available, the two essential buses are fed by two diesel generators (1250 kW each). A third Diesel Generator of same capacity is available as SBO or AAC Diesel Generator. DC Battery capacity is 30 minutes for 230 V DC system and 10 hrs for 24V DC & 220V AC UPS as per design.

2.2 General Information about C-1 and C-2

C-1 and C-2 are two loop Pressurized Water Reactors (PWR). Gross electric output of C-1 and C-2 is 325 MWe each.

Plant Feature	Summary Description
Reactor	• 121 fuel assemblies • 37 Rod Cluster Control Assemblies • Multi-enrichment region type core • Cylindrical Reactor Pressure Vessel with hemispherical bottom internally cladded with Austenitic Stainless Steel.
Reactor Coolant System	The Reactor Coolant System (SRC) is arranged as two closed reactor coolant loops connected, each containing a reactor coolant pump and a steam generator. Pressurizer is connected to the hot leg of one of the reactor coolant loops.
Safety Systems	The Emergency Core cooling system includes a High Pressure Safety Injection (HPSI) subsystem, a Low Pressure Safety Injection (LPSI) subsystem which is a part of the Residual Heat Removal System (SRH), an Accumulator Injection subsystem and a Boron Injection system which is a part of the Chemical and Volume Control System (SCV). Containment Spray System (SCS) comes into operation after an accident that causes an increase in containment pressure and temperature (loss-of-coolant accident or secondary steam line break accident). In the first stage, it draws water from the Refueling Water Storage Tank (RWST). Sodium hydroxide is blended to this water to increase radioactive iodine absorption. At a later stage, the accumulating sump water in containment is re-circulated. Containment Isolation System (SCI) ensures containment leak tightness in case of an accident which could cause release of radioactive fission products from the reactor core. Containment Hydrogen Recombining System (SHR) in C-1 consists of two parallel, independent, full capacity trains. Each train includes an air purifier, a blower, a hydrogen recombiner, piping, valves and instrumentation & control. In C-2, this function is accomplished by Passive Hydrogen Recombiners, which are installed inside the containment. Containment Hydrogen Mixing System (SHM) is used at C-1 and C-2 to mix containment atmosphere during accidents in order to prevent

	localized concentration of hydrogen in the containment. The Auxiliary Feedwater System (SAF) at C-1/C-2 ensures sufficient water supply to the steam generators for removal of decay heat from the core in a variety of postulated conditions such as total loss of normal feedwater system; loss of reactor coolant (small breaks), steam or feedwater line break or loss of power source.
Containment	The C-1 and C-2 containment buildings are designed so that the leakage to the environment will not exceed 0.3% per day of the mass of gas contained in the containment, even in the unlikely event of a large break LOCA or main steam line break inside the containment. The containment provides sufficient free volume to contain the energy released in the event of a design basis accident. The C-1/C-2 containment system is designed such that for all break sizes, up to and including the double-ended severance of a reactor coolant pipe or secondary system pipe, the containment peak pressure remains below the design pressure, with adequate margins, and it can be reduced to half of the design value in 24 hrs.
Spent Fuel Storage	Space for Fuel storage is provided in the spent fuel storage pools of the fuel buildings of both C-1 and C-2 separately for 721 fuel assemblies each. The spent fuel storage racks are designed to allow adequate cooling of the spent fuel assemblies.
Electric Power System	The power generated by C-1 and C-2 is delivered to national grid through a common switchyard, connected to two independent grids by 220 kV double circuit transmission lines. Standby off-site power source is a 132 kV grid for both C-1 and C-2. C-1 and C-2 both have two independent emergency diesel generators each. The fuel oil tank capacity of each diesel generator is enough for a minimum of 14 days for C-1 and 7 days for C-2 under rated load condition. Separate Alternate AC Diesel Generators are provided to cope with Station Blackout (SBO) condition. The coping duration for C-1 and C-2 is 4 and 8 hrs respectively. DC Battery capacity is 2 hrs each for both C-1 and C-2.

PART III

Safety Assessment of Nuclear Power Plants Post Fukushima

3. Safety Assessment of Nuclear Power Plants Post Fukushima

3.1 External Events

3.1.1 General information on the K-1 site and site evaluation

K-1 site is on the Arabian Sea coast, 11 miles (17.7 km) west of Karachi. At the coastline, the land falls away sharply about 39 feet (11.9 m) to the beach and mean sea level.

3.1.1.1 Earthquake Hazard

The original design of K-1 was based on peak ground acceleration of 0.1g. The dynamic analysis of K-1 for five major structures associated with the reactor carried out at the time of design concluded that the structures could sustain accelerations considerably higher than 0.1g. Of the five major structures, the containment provides the largest safety margin against possible earthquake damage.

For seismic qualification of older NPPs, the global approach is re-assessment of seismic capacity of plant's structures and equipment through seismic walk down. The task of "Re-assessment of Seismic Hazard for KANUPP" was included in the "Integrated Safety Review Master Plan (ISARMAP)" for safety upgrade project "Safe Operation of KANUPP (SOK)". The ISARMAP and SOK were approved by IAEA in 1992.

As part of the seismic re-assessment task, geological investigation work for seismic input was undertaken by PAEC team in 1992 as per IAEA Guide 50-SG-S1 (Rev. 1991) to review the design 'g' value (Safe Shutdown Earthquake or SSE) and re-estimated this value to be 0.2g. For re-assessment of seismic capacity of the plant, an IAEA seismic walk down expert mission was invited in 1993, which carried out seismic safety review and walk down of K-1. The mission concluded that the plant can withstand seismic event of twice the original design value of 0.1g with minor seismic retrofits (easy fixes).

The mission recommended certain short term actions or easy fixes which were analyzed and designed for SSE value of 0.2g by local experts. The design work was reviewed and cleared by the IAEA mission in 2000 after the joint PAEC/IAEA national workshop on "Easy Fixes Program for NPPs" at K-1. Thereafter, the easy fixes were implemented resulting in enhancement of seismic resistance of the plant.

3.1.1.1.1 Re-assessment of K-1 Site –Post Fukushima and Actions Taken

The list of key structures, systems and components (SSCs) that are needed for achieving a safe shutdown state during an earthquake and those which are required to perform their intended function after the earthquake have been reviewed and found adequate. Surveillance walk-down of seismic supports was conducted to identify potential new hazards to critical equipment. As a result of this exercise, some additional safety/safety related equipment have been identified which require further strengthening.

A study was carried out to evaluate the integration of auto-shutdown scheme in the existing seismic activity monitoring instrumentation system on detection of SSE. According to the study, automatic shutdown of K-1 on SSE was not found feasible. An alarm will annunciate in Main Control Room (MCR) if the event exceeds 0.06g. Operator shall shut down the plant manually if the event exceeds 0.1g, in accordance with plant procedures (Operating Policies and Principles – OPPs or EOPs).

A fresh study to analyze the Seismic Hazard (without field work) along with Tsunami potential, as per new IAEA guidelines, for K-1 site has now been completed concluding results similar to those obtained from earlier studies conducted for earthquake and tsunami potential.

3.1.1.1.2 Schedule and milestones to complete the planned activities

Task	Target Date
Surveillance walk-down of seismic supports to identify any weaknesses or potential new hazards to critical equipment, and their resolution	30-06-12
Study to incorporate auto shutdown if SSE is detected	31-01-12
Detailed visual inspections of all the structures, especially the structures important to safety	29-02-12
Re-evaluation of seismic capacity of all structures for the revised seismic input of 0.2 g	31-12-12
Confirmation of seismic qualification of existing diesel fuel tanks & mounting platform	31-12-12
Re-assessment of earthquake hazard based on the new IAEA methodology (without field work)	31-03-12

3.1.1.1.3 Preliminary or final result of the activities, including proposals for further actions

Strengthening of supports for some safety/ safety related equipment is required. Auto-shutdown of K-1 on SSE was not found feasible.

Visual inspections of all the structures, important to safety have been performed. The Plant is considered safe against Safe Shutdown Earthquake (SSE) of 0.2g.

3.1.1.2 Tsunami / Flooding Hazard

A study was conducted in 2004, based on field evidences, for assessment of tsunami hazard for K-1 site, concluding that maximum height of the wave resulting from tsunamis would be up to 3.28 ft (1 m) under normal conditions and thus do not pose any hazard to K-1.

Since the K-1 site is about 39 feet (12 m) above Mean Sea Level (MSL), it can be considered safe from flooding due to winds/ storms/ tsunamis resulting from earthquakes. The maximum wave ride-up recorded for the worst storm 72 mph (116 km/h) at K-1 is 15 feet (4.57 m).

The heaviest rainfall recorded so far during last 50 years, at site, in a day is 9.97 inches (253 mm). The maximum rainfall caused flooding in lower parts of Karachi city but K-1 site remained unaffected. There are natural nullahs (water channels) near K-1 site which discharge most of the rain flood water into the Arabian Sea.

In the worst case, damage to the intake structure and severe choking of intake water system from debris can result in disability of the pump house. However, independent seismically qualified alternate heat sink, i.e., Emergency Feed Water (EFW) system was installed in 1991 to maintain the plant in safe shutdown state.

3.1.1.2.1 Re-assessment of K-1 site –Post Fukushima and actions taken

PAEC has completed re-assessment of tsunami hazard for K-1. In the worst case scenario maximum wave height at K-1 is assessed to be 9.31 ft (2.84 m), whereas, elevation of pump house floor is 9 ft (2.74 m), auxiliary area is 15 ft (4.57 m), electric & control cables is 33 ft (10.1 m) and distribution room & Emergency Feedwater and Medium Pressure Emergency Injection (FIJW) Diesel Generators are at 39 ft (11.89 m) above MSL.

K-1 has been enlisted with Tsunami Early Warning System (TEWS) of Pakistan Meteorological Department (PMD) for dissemination of Tsunami warnings. To enhance the survivability of plant safety systems in beyond design basis flooding, flood protection measures will be taken around the vulnerable SSCs after flooding re-assessments are completed. In case of flooding, there will be no effect on the plant due to simultaneous deterioration of offsite conditions, such as loss of offsite power, and inaccessibility of offsite personnel and equipment. Diesel Generators have air cooled radiators and will not be affected. Plant can be brought to safe state with the on-site resources.

Historically, there has been no instance of flooding due to storms at K-1 site. As an additional safety measure, water tightness of the entrance of Emergency Diesel Generator (EDG) building and distribution room has been planned. The possibility of flooding at K-1 site has been revisited in the light of Fukushima accident. Consideration of flooding is not necessary due to tsunami, but due to some extremely heavy downpour causing abnormal flooding of the area through choking of drains. Topography survey and hydrology study of the entire catchment area of the plant will be carried out to calculate rain water level at critical entrance /exit locations of the plant.

3.1.1.2.2 Schedule and milestones to complete the planned activities

Task	Target Date
Fresh Estimate of Tsunami hazard for K-1	31-03-12
Identification of the equipment vulnerable to unprecedented tsunami/flooding and feasibility to improve resilience	30-06-12
Feasibility of tsunami wall on sea side of the plant or around vulnerable equipment / systems	31-03-12
Fresh study of Hazards due to cyclones and flooding due to maximum probable precipitation	30-06-12

3.1.1.2.3 Preliminary or final result of these activities, including proposals for further actions

The Plant elevation of 39 ft (11.89 m) above MSL provides adequate natural protection against flooding from historically recorded tsunamis/ floods. Tsunami wall is therefore not needed. However, on the basis of topographic survey and hydrology study, water tight sluice gates may need to be installed at critical entrances of the plant for protection against possible flooding due to heavy rainfall or cyclones.

3.1.1.3 Storms/ Cyclones Hazard

In 1902, a cyclone storm crossed Sindh coast near Karachi at 24^{o}N, 67^{o}E. Its movement was in North-East direction. Another cyclonic storm of 36 mph (58 km/h) passed the same location on May 29, 2001. However, no effect was observed in Karachi. The outside open surface disturbances due to winds and storms can only result in minor oscillations in sea water level in the pump house fore-bay and do not alter the prevailing astronomical tide level (maximum tide level 6 ft {1.83m}) in the pump house fore-bay. Since, generation of storms and their predicted direction can be reported well in advance by local government departments or through electronic media, preventive measures can easily be taken.

3.1.1.3.1 Re-assessment of K-1 site –Post Fukushima and actions taken

Elevation of different locations / buildings above MSL at plant has been marked to identify protection of critical plant areas against flooding. Additionally water tight sluice gates will be installed at critical entrances of the plant for protection against flooding, if required.

Wind loading of all buildings is being checked and a table for effects of different categories of cyclones and safety significance of building will be prepared.

3.1.1.3.2 Schedule and milestones to complete the planned activities

Task	Target Date
Study of Cyclones including effects of different categories of Cyclones	29-02-12
Re-assessment of structures for wind loading	31-12-12

3.1.1.3.3 Preliminary or final result of these activities, including proposals for further actions

On the basis of topographic survey and hydrology study of K-1 site, appropriate measures will be taken at site to protect the critical components from flooding. Moreover, number of water drains will be increased.

3.1.1.4 External Fire Hazard

Since the K-1 site is not close to any forest or fire source therefore possibility of external fire is ruled out.

3.1.2 General information on the Chashma site and site evaluation

C-1/C-2 site is situated about 10 km from the Chashma Barrage on the left bank of the Indus River. The area around the site is classified as arid to desert and is characterized by sections of sand dunes, sparsely vegetated hills, sand-soil and bare rocky hills.

The site characteristics and its environs were investigated to establish bases for determining criteria for storm, flood, and earthquake protection, and to evaluate the validity of computational techniques for the control of routine and accidental releases of radioactive liquids and gases to the environment. Moreover, population distribution, nearby facilities, high wind, extreme temperatures, explosion, and aircraft crash were also considered. The impact of all relevant site related factors on plant and on individuals, society and the environment were found acceptable.

The average ground elevation at the site is about 200 m above MSL. The site area is bounded by a Canal on the northeast (originating from Chashma Barrage), by another Canal on the southeast and by 1.1 km wide Indus River on the northwest. The north-east canal is the ultimate heat sink for C1 and C-2. Water taken from the canal is returned to Indus River downstream of Chashma Barrage. As an alternate measure, cooling towers with independent water storage tanks are also provided to be used as ultimate heat sink.

3.1.2.1 Earthquake Hazard

The seismo-tectonic studies of the site are based on a detailed review of the geological and seismological reports prepared between 1975 and 1987 by different consultants. Additional geophysical, seismological, geological and tectonic studies recommended by IAEA Review Missions (1990, 1992), were carried out by PAEC and Istituto-Sperimentale-Modelli-E-Strutture (ISMES) of Italy which were reviewed and endorsed by IAEA in 1992. This was further re-confirmed by IAEA review mission during pre-construction stage of C-2 in 2005 and another follow up mission in 2006. Seismic data from 1992 to 2007 was plotted on the seismo-tectonic model to confirm the validity of the model. All seismic events recorded remained within the upper bound values of each seismic source.

During site evaluation, it was established that eighty four (84) earthquakes had epicenters within 80 km radius of the site during the 49-year period of record. These earthquakes are of rather small magnitude and some have been reported only recently by the Micro seismic Networks. The largest event reported in the immediate area is a magnitude 5.5 event at a focal depth of 3.5 km with an epicenter approximately 75 km northwest of the site (May 1, 1982). No instances of damage to village buildings or liquefaction have been reported in nearby areas as a result of these earthquakes. The earthquakes of this magnitude have not been reported as having any adverse effect on the site.

Taking into account the limited amount of historical seismic data and short operating time of local network in comparison with the longer recurrence time for large earthquakes, the magnitude of postulated event for each seismic source was assumed greater than the maximum recorded one. It was considered appropriate to assign the maximum potential generating capability to each seismic source by adding 1 to the magnitude (on Richter scale) of the strongest earthquake ever felt/recorded in a particular seismic source.

Site acceleration values are usually estimated using the empirical attenuation relations developed from actual records for different regions of the world. As such, relations developed for different regions of the world area were studied and most suitable ones, rich in database were used. For each relation, the mean values proposed by the authors, site effects, type and directivity of the faults have been taken into account. A standard deviation was added to the mean value of peak ground horizontal acceleration.

The Safe Shutdown Earthquake (SSE) or Seismic Level 2 (SL2) earthquake was determined by following the deterministic approach of IAEA Safety Guide No. 50-SG-S1 (1991) and all earthquake sources in 150 km radius were covered. The Operating Basis Earthquake (OBE) or Seismic Level 1 (SL1) earthquake was determined on the basis of seismicity during the last 100 years as OBE is determined for a return period of 100 years. The calculated accelerations vary from 0.036g to 0.25g. As such, 0.25g was adopted as the SSE for the site. As ground acceleration at site from the 100 years record did not exceed 0.10g, a value half of SSE, i.e. 0.125g was taken as OBE.

3.1.2.1.1 Re-assessment of Chashma site–Post Fukushima and actions taken

Surveillance walk-downs of seismically qualified buildings are underway at C-1 and C-2 to identify any weaknesses and potential new hazards from external natural hazards. Surveillance walk-downs of seismic supports are also currently in progress at C-1 and C-2 to identify any weaknesses and potential new hazards to critical equipment from external natural hazards. For C-1, walk down of about 125 seismic supports of 15 safety related systems have been completed. For C-2, 620 supports inside Reactor Building (RX) were checked (Room wise). Walk down checking of 2500 supports in C-2 Nuclear Auxiliary Building (NX) is underway.

3.1.2.1.2 Schedule and milestones to complete the planned activities

Task	Target Date
Seismic walk-down of SSCs to identify any non-conformances and potential new hazards to critical equipment from external natural hazards	30-06-12, 31-01-13(for Rx area)
Enhance seismic structures surveillance program	31-08-12
Feasibility study of auto shutdown if SSE is detected	31-08-12

3.1.2.1.3 Preliminary or final result of these activities, including proposals for further actions

All supports have been found to be satisfactory in the walk-downs of seismic supports completed so far at C-1 and C-2.

Feasibility of auto shutdown if SSE is detected is also being explored at C-1/C-2.

On the basis of existing seismo-tectonic parameters/data and updated relationships (1997), Review Level Earthquake (RLE) is determined as 0.32g for Chashma Site. Safety related structures of C-1/C-2 are already safe for PGA of 0.32g. The soil at Chashma site is not expected to liquefy below 0.35g.

3.1.2.2 Tsunami / Flooding Hazard

The Chashma site is situated on the left bank of a barrage on River Indus. The Chashma Barrage has created a reservoir with a live capacity of 0.68 MAF (0.838 billion cu. m). The normal pond level of the barrage is 642 ft (195.68 m) and the storage level is 649.0 ft (197.82 m). The barrage is designed to pass a flood of 942,200 cusecs (26,680 cumecs) at the flood level of 640.6 ft (195.25 m). The design upstream high flood level corresponds to the condition when all the gates of the barrage are open.

The following factors were considered while analyzing Chashma Site for floods:

- Floods Historical Records
- Probable Maximum Flood at different locations
- Probable Maximum Precipitation
- Precipitation Losses
- Dam Break Scenarios
- Water Level Considerations
- Historical Low Water

The phenomena of surge & seiche flooding, and tsunami are not relevant for the site and were therefore, not considered. The floods due to other natural causes as well as those due to failure of upstream planned dams have been analyzed.

Past floods in the Indus River have been caused by natural events such as precipitation, snow melt and bursting of ice or debris dams in the upper reaches of the Indus and its tributaries. The flood history of Indus has been studied by evaluating the records of floods for various sites.

In order to evaluate the impact of floods at the site, studies have been carried out to determine the worst possible flood that could occur at the site. Discharge measurements at Kalabagh, 64 km upstream of site, were started in 1922. The discharges at Chashma and Kalabagh are almost identical. In general, the flood peak neither increases nor decreases between Kalabagh and Chashma. In order to evaluate the impact of floods at the site, following floods have been evaluated:

- Maximum Flood historically recorded as 917,025 cfs (25,967 cumecs)
- Maximum Flood for a return period of 2,000 years as 1,284,000 cfs (36,359 cumecs)
- Maximum Probable Flood caused by probable maximum precipitation, snowmelt etc. 3,100,000 cfs (87,782 cumecs)
- Probable Maximum Flood (PMF) at proposed Kalabagh Dam break causes 6,500,000 cfs (184,000 cumecs) flood

The worst scenario for possible flooding of the plant site is the scenario of Kalabagh Dam break (proposed, yet to be constructed) under PMF conditions. In this scenario, the maximum discharge passing flood levels at the Chashma Site would be only about 1m above the average site level i.e., 201.03 m above MSL. It is the only scenario in which water reaches the site. Average flow velocity is only 4 ft/sec (1.2 m/s) which is not critical. Minimum level of finished grade of plant area is 201.3 m and for safety related structures is 203 meters above MSL to cater for wind induced wave action coincident with PMF, uncertainties, etc. No exterior access openings are provided below the grade floor elevation, in areas where flooding could damage Safety Class 1 structures or equipment.

The design of site drainage caters for the effect of local intense precipitation and adequate measures are in place to avoid localized flooding. For additional safety, the following measures are provided:

i. Water stops in all horizontal and vertical construction joints in all exterior walls up to the grade elevation.

ii. Water seals are provided for all penetrations in exterior walls up to the grade elevation.

iii. All exterior walls and the base mat are designed for the hydrostatic pressure considering submergence to the probable maximum flood elevation.

iv. The finished grade elevation adjacent to the plant is maintained at least 0.3 m below plant grade floor elevation.

3.1.2.2.1 Re-assessment of Chashma site –Post Fukushima and actions taken

Re-assessment of Emergency Control Centers (ECCs) robustness of both C-1 and C-2 against flooding and earthquake is presently underway. Studies have been initiated to determine the combined effect of earthquake and the dam break to determine how much it differs from Design Basis at C-1 and C-2.

3.1.2.2.2 Schedule and milestones to complete the planned activities

Tasks	Target Date
New study of flooding hazard potential for Chashma Site using updated historical information and considering Chashma Barrage break as a result of upstream dam break and other potential new scenarios	30-09-12
Re-assessment of Emergency Control Centers (ECCs) robustness of both C-1 and C-2 against flooding and earthquake	30-04-12

3.1.2.2.3 Preliminary or final result of these activities, including proposals for further actions

Design Basis Flood Level has been ascertained as 201.03 m above MSL while the Safety related SSCs are at 203 m above MSL. Strengthening of the surveillance program has been proposed at C-1 and C-2 to ensure that drains remain functional.

3.1.2.3 Other External Hazards

3.1.2.3.1 High winds

The Gumbel method of frequency analysis has been used for estimates of maximum wind speed up to 100 year return period. From the frequency analysis, the data at Chashma site gives a value of maximum wind speed of 42.26 m/s at 10 m elevation against a 100 year return period. Accordingly, a design wind velocity of 45 m/s at 10 m was used in the design of Seismic Category I structures. For Category II structures, a design wind velocity of 40 m/s, at 10 m was used, based upon a 50-year return period.

3.1.2.3.2 Temperature extremes

The measurement of temperature along with other meteorological parameters at Chashma site started in March 1976. To incorporate the long term records for estimation of temperatures, data available at other stations in the region is used. To do so, the available data at the site was correlated with the data of corresponding period at other stations. For correlation, the data of extreme temperatures at Chashma site were plotted against extreme temperature observed at each of the surrounding stations in the same periods. The line of best fit was drawn through plotted points by the method of least squares. For the Chashma site, the extreme maximum ambient temperature on record is 50.5°C, while the extreme minimum ambient temperature of -2.5°C. The extreme maximum dry and wet bulb temperatures are 49.3°C and 38.6°C and the extreme minimum dry and wet bulb temperatures are -3.0°C and -3.3°C, respectively.

3.1.2.3.3 Aircraft crash/Toxic Gas/Hazardous Clouds/External Fires/ Explosion

The Screening Distance Value (SDV) is used for evaluation of potential accidents for different external hazards. Detailed studies have been undertaken to locate chemical plants, refineries, storage facilities, mining and quarrying operation, military bases, missile sites, transportation routes (docks, anchorages and air ports), oil and gas pipelines, drilling operations and wells, underground gas storage facilities, military firing or bombing ranges, any nearby aircraft flights and holding or landing patterns, in the vicinity of the plant. No such hazard lies within SDV.

3.1.2.3.4 Re-assessment of Chashma site –Post Fukushima and actions taken

C-1 and C-2 have carried out the reassessment of vulnerability against hazards like storms, tornados, etc.

3.1.2.3.5 Schedule and milestones to complete the planned activities

Tasks	Target Date
Reassessment of vulnerability against hazards like storms, tornados, etc.	15-04-12

3.1.2.3.6 Preliminary or final result of these activities, including proposals for further actions

C-1/ C-2 are safe against external hazards such as storms, tornadoes, etc.

3.1.3 Activities carried out by PNRA

3.1.3.1 Actions taken for improvement

PNRA has made an initial review of the regulation PAK/910 “Regulation on the Safety of Nuclear Power Plant- Site Evaluation” in order to identify any changes that might need to be incorporated based on lessons learnt from Fukushima accident.

3.1.3.2 Schedule and milestones to complete the planned activities

Tasks	Target Date
Initial Review of Regulation PAK/910	31-12-11
Detailed Review of Regulation PAK/910	31-12-13

3.1.3.3 Preliminary or final result of these activities, including proposals for further actions

The initial review of PNRA Regulations has identified that PAK/910 “Regulation on the Safety of Nuclear Power Plant-Site Evaluation” needs to be revised and additional safety criteria may need to be incorporated in these regulations.

3.2 Design Issues

3.2.1 Assessment of Design Issues at K-1

3.2.1.1 Alternating Current (AC) electrical power

At K-1, essential 400V AC power is normally supplied from the station 4160V bus which is supplied in turn by the station running transformer or starting transformer. When the normal source of power is not available, the two essential buses are fed by two diesel generators (1250 kW each). A third Diesel Generator of same capacity is available as SBO or AAC Diesel Generator.

In addition to supplying power to 400V motors, the essential 400V supply is further stepped down to provide 120V AC, 24V DC, 220V AC UPS and 230V DC for instrumentation and control purposes.

3.2.1.1.1 Actions taken for improvement

In response to Fukushima accident, availability of diesel fuel at site has been reassessed and existing diesel fuel storage inventory is being increased. Arrangements have been made with offsite authorities to ensure supply of 2000 liters/day of diesel fuel. The following measures are being taken at the plant to enhance availability of essential 400V supply and / or 24V DC, 220V AC UPS and 230V DC keeping in view the Station Blackout (SBO) scenario for extended duration:

i. Integration scheme to connect a 300 kW, 400V mobile Diesel Generator set with the plant 400V essential supply (as a last resort). With the existing diesel fuel resources, this generator can provide power for more than 30 days.

ii. Energizing the 400V essential buses through existing diesel generators of Medium Pressure Emergency Injection (FIJW) System. The increased inventory of diesel fuel can be used for continued operation of FIJW diesel generators for up to 30 days.

iii. Installation of a new 80 kW, 400V Diesel Generator set (at a higher elevation) to provide electrical supply to essential valves of Emergency Core Cooling System (ECCS), plant emergency lighting and charging of 24V DC and 220V AC UPS system. In addition, a diesel driven pump will also be installed to inject water into Emergency FeedWater System (EFW), FIJW, Reactor Vault Cooling System (VCW), Bay Water System (BYW) and Dowsing Spray Water (DSW) System. (Please refer to section 3.2.1.2.1). These two engines can be operated for more than 50 days.

iv. Shifting of plant emergency lighting from the 230V DC system to 220V AC UPS system.

Feasibility studies related to the above have been completed and engineering design is in progress for implementation.

Apart from the above modifications, a study has also been conducted for increasing battery backup time of 230V DC system which is comparatively lower than 24V DC and 220V AC UPS system. 230V DC power remains available for 30-60 minutes. However, by conserving the DC power, this time can be extended to 7 hrs. For conservation, some of the DC loads have already been shifted to 24V DC & 220V AC UPS system which have a battery backup time of around 10 hrs. Certain actions are also identified which can be taken timely to conserve 230V DC power and thus increase the backup time.

3.2.1.1.2 Schedule and milestones to complete the planned activities

Task	Target Date
Development of Integration scheme for 300 kW, 400V mobile DG with Essential power supply	30-10-12
Provision for energizing the plant essential buses through FIJW-DG1/2	31-08-12
Installation of a new 80 kW, 400V DG set and its integration scheme	30-10-12
Shifting of emergency lighting from 230V DC to 220V AC (UPS)	30-10-12
Provision of alternate power supply to essential valves of ECCS	30-10-12
Feasibility study to arrange trolley mounted DGs from other organizations	29-02-12
Feasibility to increase diesel fuel storage capacity onsite	31-03-12
Identification of external sources of Diesel Fuel	31-03-12
Feasibility of use of natural gas in one of the Diesel generators for diversity	31-01-12
Procedure for conserving DC power to prolong its availability	30-06-11

3.2.1.1.3 Preliminary or final result of these activities, including proposals for further actions

Procedure for conserving 230V DC power has been modified to prolong availability from 30 minutes to 7 hrs.

To increase diesel fuel inventory, various on-site diesel fuel storage sources have been identified. This would increase minimum fuel inventory to run one EDG for about 9 days without any external support.

A feasibility study to explore the use of natural gas in one of the generators for diversity has been completed. Since the availability of natural gas after earthquake cannot be ensured, the proposal was not found feasible.

3.2.1.2 Emergency Core Cooling

Emergency Core Cooling System in K-1 consists of two subsystems: Medium Pressure Forced Emergency Injection Water (FIJW) System and Low Pressure Emergency Injection Water (IJW) System. IJW also serves to provide continuous long term core cooling.

Auto Boiler Crash Cooling System (ABCC) is provided for quick reduction in primary heat transport system pressure to facilitate water injection from FIJW/IJW. The Calandria is enclosed in a reactor vault containing large quantity of light water. In the highly improbable case of failure of all core cooling systems, this water can remove the heat from the core debris and retain it within the calandria for a long period of time.

3.2.1.2.1 Actions taken for improvement

After the Fukushima accident, preliminary studies have been carried out to identify additional possible points to inject fresh water /sea water to cool the core in extreme cases.

A diesel driven pump is being installed which takes water from tanks located at higher elevations to feed the Steam Generators, FIJW, VCW, BYW and DSW Systems. The modification also includes a provision to connect and supply water from the fire tender and seismic category I fire water ring. With appropriate lineup, gravity filling from the tanks to Steam Generators, FIJW, VCW and BYW will also be possible. Sea water may also be used as makeup into these systems.

Presently, one fire tender is available for injection into the systems. However, another fire tender is being procured.

3.2.1.2.2 Schedule and milestones to complete the planned activities

Task	Target Date
Feasibility study to determine the need of increasing pumping heads of EFW system	29-02-12
Provision of additional points for fresh water injection and use of fire fighting system (as a last resort) for emergency core cooling	31-10-12
Feasibility study of passive cooling such as natural circulation after shutdown and in case of SBO	31-03-12
Feasibility of steam-driven pumps to feed the boilers in extreme case	31-12-11

3.2.1.2.3 Preliminary or final result of these activities, including proposals for further actions

The study conducted to assess the need for increasing the existing pump discharge pressure of EFW during SBO condition concluded that such an increase is not required.

Existing Vault inventory at K-1 will take about 4 days to completely boil off without makeup after core melt down.

Provision for Installing Diesel Engine /Generator Driven Pump for Water Injection into EFW, VCW, FIJW, DSW and BYW systems in case of prolonged SBO is in-progress along with the Change Approvals (CAs) for each option.

Use of steam turbine driven pump for providing feedwater to boilers was not found feasible due to low pressure and low steam flow after plant shutdown especially in case of SBO.

3.2.1.3 Containment Integrity

K-1 containment building is designed for leakage rate of 2.4% per day at design pressure of 27 psig (0.186 MPa) whereas for design basis accident the maximum containment pressure rises to 22 psig (0.15 MPa). The entire internal surface of the containment building is coated with a plastic lining for leak tightness.

The containment isolation is based on pressure signal. The abnormal rise in containment pressure could be either due to Loss of Coolant Accident or Main Steam Line Break. Under normal operating condition, the boiler room pressure is slightly negative (-0.25”/-6.4 mm of H_2O column). If this pressure rises to 5” (127 mm) of H_2O column, the containment isolation takes place and the containment isolation dampers and motorized valves close to isolate the containment.

DSW system is available which initiates automatically at high radiation level and high pressure in reactor building to suppress containment pressure and limit radioactive releases.

3.2.1.3.1 Actions taken for improvement

The following studies have been conducted for severe accident conditions:

- Assessment of Hydrogen hazard
- Need for Hydrogen recombiners & igniters
- Possible methods for relieving containment pressure
- Determination of DSW adequacy for alleviating containment building pressure

3.2.1.3.2 Schedule and milestones to complete the planned activities

Task	Target Date
Assessment of hydrogen hazard	31-01-12
Feasibility and need for (passive) hydrogen recombiners and hydrogen igniters	31-01-12
Identification of measures that can be taken to ensure containment integrity in the worst case scenario	31-01-12
Review adequacy of DSW for severe accident	31-01-12
Feasibility of operating motorized relief valves (dampers) manually when power is not available	31-01-12
Feasibility of installing system for relieving containment pressure	31-10-12

3.2.1.3.3 Preliminary or final result of these activities, including proposals for further actions

Preliminary analyses reveal that in the short term, In-vessel hydrogen production in K-1 is too small to exceed the 4% lower upward flammability limit. The ex-vessel hydrogen due to Molten Core Concrete Interaction (MCCI) is quite small as compared to modern CANDUs due to very small core.

A feasibility study has shown the need for hydrogen re-combiners in K-1 containment. A limited, conservative analysis has been done to show the effectiveness of Passive Autocatalytic Recombiners (PARs). On the basis of the study, the number, capacity and location of PARs have been determined. Efforts are being made to acquire PARs of required capacity and various suppliers have been contacted.

Presently Hydrogen Monitoring System is not available at K-1. The specification and location for their installation has been finalized and Hydrogen monitoring equipment is being procured.

DSW system was found adequate for severe accident conditions. Operation of motorized dampers during SBO was not found feasible because of loss of instrument air and control power supply, therefore manual containment venting system will be installed.

3.2.1.4 Ultimate Heat Sink

The design provisions to prevent loss of Ultimate Heat Sink (UHS) i.e. sea water are: Steam blow-off valves to remove decay heat, spring loaded relief valves, redundant standby salt water pumps, two trains of Emergency Feedwater (EFW) system with dedicated DGs. In case of loss of primary UHS with SBO, the EFW is an alternate heat sink to maintain PHT system pressure, fuel and sheath integrity for 2 days. Some nearby tanks will be connected to this system to further increase the heat sink capacity to 6 days. However, this time can be increased indefinitely with water make-up to EFW tanks from other sources. If EFW system could not be taken into service, core damage is likely to occur after 3 hrs.

Operating experience as well as a theoretical study has revealed that even in case of loss of UHS (due to Process Water bulk draining), the change in Spent Fuel Pool (SFP) temperature is insignificant.

3.2.1.4.1 Actions taken for improvement

Re-assessment of the consequences of Loss of UHS has been carried out. Passive cooling was demonstrated (in cold shutdown conditions) to fill the boiler by gravity flow from Feed Water Storage Tank considering loss of UHS coincident with SBO.

3.2.1.4.2 Schedule and milestones to complete the planned activities

Task	Target Date
Demonstration of Passive cooling (in cold condition) such as natural circulation	31-03-12
Integrity Assessment of Intake Bay Structure	30-06-12

3.2.1.4.3 Preliminary or final result of these activities, including proposals for further actions

The filling of boilers in cold conditions under gravity driven flow has been demonstrated successfully during shutdown state. Diesel driven pumps will be used for providing feedwater to boilers to bring the plant to cold shutdown in case of loss of UHS coincident with SBO and failure of EFW system.

3.2.1.5 Spent Fuel Cooling

Spent fuel storage bay is designed to store spent fuel safely until it is removed for interim storage or final disposal. The Storage Bay is filled with water to provide shielding and to remove the heat of the spent fuel.

The spent fuel storage bay had design storage capacity of 23,760 spent fuel bundles. However, the capacity is being enhanced to 31,680 bundles by placing spent fuel bundles in High Density Tray Racks (HDTRs).

The spent fuel storage bay system includes a cooling circuit and a purification circuit. The cooling capacity of the system is adequate for the removal of heat generated in spent fuel bundles accumulated during 40 years of operation. The normal bay water temperature under such circumstances is 110°F (43.33°C) when process water is supplied at 90°F (32.22°C). Spent fuel storage bay low level alarm is available in MCR.

The bay water cooling capacity is so designed that if all the fuel bundles in the reactor core are dumped in the bay, the resultant bay water temperature would be limited to approximately 160°F (71.11°C).

3.2.1.5.1 Actions taken for improvement

Re-assessment of Spent fuel storage bay seismic design is being performed.

Safe dry out time (spent fuel uncovered condition) has been assessed in case of loss of cooling. The calculations of safe dry time show that operators would have more than 19 days to act before the whole bay water reaches its boiling temperature and uncovering of the top of spent fuel present in the bay occurs after 140 days. Measures against loss of spent fuel bay cooling have also been assessed. Source term estimation in case of dry out of spent fuel bay is being assessed. Criticality hazard of used and fresh booster enriched fuel has been evaluated.

In case of loss of cooling or drainage of spent fuel bay, possible points to inject water for makeup have been identified.

3.2.1.5.2 Schedule and milestones to complete the planned activities

Task	Target Date
Re-assessment of Spent fuel storage bay seismic design	31-03-12
Assessment of safe dry times of spent fuel	29-02-12
Provision of measures against loss of cooling or drainage of Spent fuel storage bay	30-10-12
Estimation of source term of spent fuel when water is lost or configuration is disturbed in Spent fuel storage bay	31-03-12
Study of criticality hazard of enriched (10%) fresh booster fuel assemblies stored at new fuel storage area in case of tsunami	31-01-12
Study of criticality hazard of Spent fuel bay in case of earthquake & tsunami due to presence of used enriched booster fuel	31-03-12

3.2.1.5.3 Preliminary or final result of these activities, including proposals for further actions

In case of loss of cooling or drainage of Spent Fuel Bay, possible points to inject water for makeup have been identified. Make up water sources have also been identified. Furthermore, it is proposed that a sprinkler system (or spray header system) be installed around the periphery of spent fuel bay to provide make up water, and to mitigate loss of spent fuel bay water events . The system would be designed with an assured supply of makeup water.

Assessment of safe dry time of spent fuel has been carried out. The calculations of safe dry time show that operators would have more than 19 days to act before the whole bay water reaches its boiling temperature and uncovering of the top of the spent fuel present in the bay occurs after 140 days (assuming that the Spent fuel bay is not drained, due to structural damage etc.). Since ample time is available to restore Spent Fuel Bay cooling, cladding oxidation and hydrogen production are highly unlikely.

K-1 also has an inventory of 32 fresh boosters (10 % enriched), stored in an aluminum box in the service building. Study of criticality hazard of these enriched fresh booster bundles in case of flooding has been conducted. The calculations show that there is no criticality hazard.

3.2.2 Assessment of Design Issues at C-1 and C-2

3.2.2.1 Alternating Current (AC) electrical power

The power generated by C-1 and C-2 is delivered by four transmission lines to the national grid through a common on-site switchyard connected to two 220 kV grids. Standby power source is an independent 132 kV grid for both C-1 and C-2.

The plant AC power system consists of main transformer, main generator, step-down transformer, auxiliary transformer, 6 kV AC auxiliary system, 380 V AC auxiliary system and 220 V AC power system. It is designed to transmit electrical power to the network and supply reliable power for auxiliaries to ensure the safe operation of the plant.

C-1 has two 3.2 MW EDGs, each with 14 days fuel storage capacity. C-2 has two 3.4 MW EDGs, each with 7 days fuel storage capacity. C-1 and C-2 are equipped with independent Alternate AC (AAC) power sources to cope with SBO condition. AAC Power sources can be manually connected with either of the two safety buses within 10 minutes in case of SBO condition for a safe reactor shutdown.

C-1 and C-2 Station batteries are designed for 2 hrs of maximum supply time after loss of AC power. The battery groups are housed in separate rooms which are ventilated well to ensure the hydrogen concentration to remain below 2% by volume within the battery areas, and are installed in seismically reinforced battery racks.

3.2.2.1.1 Actions taken for improvement

Feasibility of laying power cable (safe from external natural hazards) to connect a remote AC source to the plant electric power system (in case EDGs are disabled and there is radiation hazard near the plant) is being explored. Trolley mounted EDGs are currently not available. Feasibility of purchase of new trolley mounted EDG is under preparation.

Arrangements are in progress for sharing Diesel fuel between the units. Means to increase storage capacity for Diesel fuel for EDGs have been explored. Arrangements for the supply of Diesel fuel in case of natural disaster have been made.

Measures to improve protection of Diesel Generator fuel storage building against natural hazards are also being explored.

3.2.2.1.2 Schedule and milestones to complete the planned activities

Tasks	Target Date
Preparation of conceptual proposal for providing additional AC Power Source covering extreme natural hazards, all NPPs, interconnections of all installations, provision of trolley mounted small DGs, hookup of individual essential loads/buses, cables to remote connection points, resources available with the other organizations	30-06-12
Study to increase the storage capacity of Diesel fuel	31-12-11
Arrangements for supply of Diesel fuel in case of natural disaster	31-12-11
Feasibility for increase in DC Power capacity	31-12-12

3.2.2.1.3 Preliminary or final result of these activities, including proposals for further actions

In a pre-assessment, it has been established that Diesel Generator buildings (DX) at C-1 and C-2 are protected against flood and earthquake. However, some gaps related to water ingress were identified. Surveillance and maintenance of arrangements to prevent water ingress in DX building and cable trenches are being updated. Fire doors at all DX entrance are being made water tight especially at the bottom. Furthermore, enhancement in flood protection of Essential Service Water System (WES) for EDGs has also been proposed.

At present in C-1, 7 days fuel storage per train is available. By increasing diesel fuel stored up to 80% will enhance each train capacity to 14 days. Interconnection of fuel tanks will increase the availability of fuel at site to 28 days (at full load with one train). At C-2, 7 days fuel storage per train is available. Tanks are already filled up to 80%. Hence, further increase is not possible. Interconnection of fuel tanks will increase the availability of fuel at site to 14 days (at full load with one train).

Guidelines will be prepared to minimize load on EDGs in extreme conditions to prolong their availability. Furthermore, means to increase the storage capacity for lube oil in diesel building for more than 7 days are being assessed.

Feasibility for use of portable/remote instruments for temperature, humidity, radiation etc has been proposed for the case of potential failure of instrumentation.

3.2.2.2 Emergency Core Cooling

Emergency core cooling system includes a Refueling Water Storage Tank (RWST), High-High Head Injection Sub-System (HH-HSI), High pressure safety injection subsystem, low pressure safety injection subsystem which is a part of the Residual Heat Removal System (SRH), an accumulator injection subsystem and a boron injection subsystem which is a part of the Chemical and Volume Control System (SCV). In boron injection system, two centrifugal charging pumps and boric acid transfer pumps deliver borated water to SRC from the boric acid storage tanks.

In Design Basis Accidents (DBA), the safety injection system takes suction from the RWST and injects borated water into the SRC. After the inventory of borated water in the RWST is depleted, the borated water in the containment sump is re-circulated through the reactor core to provide long term cooling of the core. Changeover from RWST to recirculation sumps in containment is fully automatic.

The Auxiliary Feedwater System (SAF) is an Engineered Safety Feature (ESF) which comprises an emergency feedwater storage tank and two physically and electrically separated trains. Each train consists of one motor-driven pump and one diesel driven pump. Each diesel driven pump can be used to operate the SAF system for 68 hrs. The emergency feedwater storage tank is primary source with a separate demineralized water tank as backup. Seismically qualified fire water tanks are the long term water supply source for this system. The water supply is sufficient for about 10 days.

At C-2, a Reactor Cavity Flooding System (SCF) is also in place so that the water from RWST can be injected to flood the reactor cavity within 60 minutes in case of severe accident. This system can also provide containment spray water, if necessary.

3.2.2.2.1 Actions taken for improvement

Study to determine the feasibility of increasing pumping heads was conducted at C-1 and C-2. Possible additional points for fresh water injection into steam generator in extreme case have been identified. Use of Fire Fighting System for emergency cooling through steam generators has also been re-visited at C-1 and C-2.

Feasibility of back-fitting improved safety features of C-2 into C-1, or other ways to achieve the same objectives is also being conducted. Similarly, feasibility of incorporating Cavity Flooding system at C-1 similar to C-2 is being carried out.

3.2.2.2.2 Schedule and milestones to complete the planned activities

Task	Target Date
Exploration of possible additional points for water injection into steam generator using temporary pumping sources in extreme case	30-06-12
Feasibility study of interconnecting SAF system with Safety Injection system (SIS)	31-12-12
Revisit procedure for use of Fire Fighting system for emergency cooling through steam generators	30-06-12
Feasibility of installing Cavity flooding system at C-1	31-12-12

3.2.2.2.3 Preliminary or final result of these activities, including proposals for further actions

New injection points in existing feedwater injection lines have been identified for water injection into Steam Generator from temporary water sources. A feasibility study of interconnecting SAF system with Safety Injection system (SIS) has been proposed.

Procedure for use of fire water has been modified for improved injection into Steam Generator through SAF. Temporary arrangements have been re-validated.

3.2.2.3 Containment Integrity

Containment structure of C-1/C-2 is a pre-stressed concrete shell structure composed of a right cylinder with a shallow dome and is founded on a flat foundation base slab. The entire structure is lined on the inner side with a steel plate that acts as a leak tight membrane.

The containment encloses the reactor pressure vessel, steam generators, reactor coolant loops, and portions of the auxiliary and engineered safety features systems. It ensures that leakage of radioactive material to the environment does not exceed the acceptable dose limit as defined in PAK/910 even if a loss-of-coolant accident occurs. The specifications of containment are:

- Dry containment structure, 1 meter thick Steel-lined post-tensioned pre-stressed concrete walled circular building capped with concrete dome
- Diameter 36 m (inner)
- Height 57.5 m
- Thickness of Basemat 5.65 m
- Design pressure 0.26 MPa(g)
- Design temperature 127^{o}C
- Ultimate pressure capacity 0.86 MPa (without taking credit of liner)

- Free volume 50470 m^3

The containment building is designed to limit the leakage to 0.3% per day of the mass of air contained in the containment, at design pressure. It provides sufficient free volume to contain the energy released in the event of a LOCA. The internal structures and compartment arrangement provide equipment missile protection and biological shielding for maintenance personnel.

At C-1 and C-2, a Containment Spray system is in place to reduce containment pressure and radioactivity (using NaOH) to acceptable limits during Design Basis Accident inside the containment.

A Containment Isolation (SCI) system provides means for containment isolation in the event of Design Basis Accident at C-1 and C-2, and maintains the overall integrity of containment leak tightness.

The Containment Hydrogen Recombining System is used at C-1 and C-2 to remove the hydrogen generated inside the containment to avoid hydrogen explosion and combustion.

For C-1, two redundant trains of thermal hydrogen recombiners backed up with a post-LOCA hydrogen exhaust system are provided. For C-2, 18 passive autocatalytic recombiners (PARs) are installed inside the containment. Hydrogen Mixing System is used at both units to minimize accumulation of localized hydrogen in the containment.

The Hydrogen Concentration Monitoring System is used at C-1 and C-2 to continuously measure the concentration of hydrogen in the containment atmosphere following accidents for post accident management, including emergency planning. The containment hydrogen monitoring system is designed as class 1E and SSE.

3.2.2.3.1 Actions taken for improvement

Feasibility of filtered venting system for C-1/C-2 is currently underway.

Extent of hydrogen hazard has been re-analyzed at C-1 and C-2 along with review of earlier studies. Feasibility study of use of passive Hydrogen Re-combiners in C-1 similar to those in C-2 has been carried out.

3.2.2.3.2 Schedule and milestones to complete the planned Activities

Task	Target Date
Installation of PARs at C-1	31-07-14
Feasibility of installing SRC Pressurizer Throttle Valve at C-1	31-12-12
Feasibility of Filtered Venting System for C-1 and C-2	31-12-12

3.2.2.3.3 Preliminary or final result of these activities, including proposals for further actions

It has been established that even without containment spray, C-1 and C-2 containment buildings are robust enough to bear severe accidents for at least 72 hrs, conditional to the availability of PARS. Accordingly, installation of PARs has been decided at C-1.

3.2.2.4 Ultimate Heat Sink

The supply of cooling water (primary UHS) is from the C. J. Link canal which takes water from Chashma barrage. This supply can however, be disrupted in the case of an Operating Basis Earthquake (OBE). Hence, two seismically qualified inter-connectable closed circuit trains consisting of underground water tanks and cooling towers are provided as alternate UHS. No makeup water is required for 30 days but can be made available through installed tube wells.

For C-1, water intake for essential service water system (WES) is from canal water (once through mode) and utilizes cooling towers and water storage tank in recirculation mode. However at C-2, WES system is designed to operate in the recirculation mode only, to eliminate silt problem. During normal operation, the pump takes suction from the cooling tower basin and during SSE; the pump takes suction from water storage tanks.

3.2.2.4.1 Actions taken for improvement

Re-assessment of the consequences of Loss of Ultimate Heat sink has been carried out. Passive cooling is possible by filling Steam Generators through gravity injection from de-aerator storage tank for SBO.

3.2.2.4.2 Schedule and milestones to complete the planned activities

Task	Target Date
Re-assessment of the consequences of Loss of Ultimate Heat Sink	31-12-11
Improvement in design of essential service water pumping station entrance to prevent inundation in case of extreme flooding	30-09-12

3.2.2.4.3 Preliminary or final result of these activities, including proposals for further actions

As seismically qualified alternate heat sink is enough for 30 days without makeup requirements, no immediate concerns exist.

Improvement in design of essential service water pumping station entrance to prevent inundation during flood exceeding design basis is planned.

For complete loss of UHS (assuming diesel driven SAF available for 2 hrs), fuel damage may begin to occur after 8 hrs. For actions to cope with this case, refer to Section 3.2.2.2.2.

3.2.2.5 Spent Fuel Cooling

The Spent Fuel Pool Cooling and Clean-up System of C-1/C-2 serves the spent fuel pool in the Fuel Storage Building. Fuel storage building is Safety Class 3 and seismically qualified. It provides cooling, purification, make-up, filling and draining of spent fuel pools. Spent Fuel Pools capacity is sufficient for 721 fuel assemblies for each unit. The building is equipped with normal ventilation system along with accident exhaust ventilation system for pool area. The system consists of two 100% capacity trains, powered from independent essential buses.

The maximum abnormal heat load is determined based on heat generated by one full core (121 fuel assemblies) at 150 hrs after shutdown, 1/3rd of the core (40 fuel assemblies) at 480 hrs, and 560 assemblies of previous 14 year unloads. Two cooling trains can maintain the spent fuel pool water at or below 57.7 ^{o}C. With one train, the pool water temperature can be kept below 80 ^{o}C.

Borated water for normal makeup purposes is available from safety class 2 and seismic category I RWST through the purification pump. The de-mineralized water is provided to compensate the loss of water due to normal system leakage and pool water evaporation. The boron re-cycle makeup water serves as a backup source of water. System piping is arranged so that failure of any pipeline cannot drain the SFP below the safe water level.

3.2.2.5.1 Actions taken for improvement

Source term is being estimated for spent fuel in case water is lost or configuration is disturbed in SFP. Safe dry time (Fuel uncovered Condition) of SFP has been determined. Measures against SFP loss of cooling or drainage are being re-assessed.

3.2.2.5.2 Schedule and milestones to complete the planned activities

Tasks	Target Date
Estimation of source term for spent fuel if water is lost or configuration is disturbed, in SFP	30-06-12
Determination of safe dry time (Fuel un-covered) of SFP	31-12-11
Study of measures against SFP loss of cooling or drainage	30-06-12
Provision of SFP cooling with fire water	31-12-12

3.2.2.5.3 Preliminary or final result of these activities, including proposals for further actions

Provision of SFP cooling at C-1/C-2 with fire water (normal hydrants or fire tender) is planned. Study for provision of water spray into SFP is proposed. Although contingency procedures in context of refueling operations are available, Abnormal Operating Procedures (AOPs)/ Emergency Operating Procedures (EOPs) for loss of SFP cooling or inventory for other operating modes are planned. Based on international operating experience, following are in progress:

- Regular checking of SFP siphon breakers
- Provision of spares in close vicinity.
- Inclusion of SFP parameter monitoring in existing AOPs/ EOPs

Furthermore, a study to improve range and reliability of SFP instrumentation following BDBA has also been initiated.

3.2.3 Activities carried out by PNRA

3.2.3.1 Actions taken for improvement

PNRA has conducted an initial review of the following regulations on the safety of nuclear power plants to identify any changes that might need to be incorporated based on lessons learnt from the Fukushima accident:

i. PAK/911 "Regulation on the Safety of Nuclear Power Plant Design"
ii. PAK/912 "Regulations on the Safety of Nuclear Power Plants-Quality Assurance"
iii. PAK/913 "Regulations on the Safety of Nuclear Power Plants Operation"

Accident Analyses of C-1 and C-2 were revisited after Fukushima disaster based on insights from the Fukushima accident and international operating experience.

Following three cases were selected for analysis to simulate possible accident environment:

- SBO with Pressurizer (PRZ) safety valve stuck open
- SBO with Large Break LOCA (LB-LOCA) in cold leg
- SBO with Small Break LOCA (SB-LOCA) in cold leg

The objective was to assess the containment integrity for extended SBO condition. Based on the analysis, it was concluded that SBO with LB-LOCA in cold leg was the bounding case. The following were the main concluding points for SBO with LB-LOCA in cold leg:

a. It is unlikely that the global deflagration of hydrogen will occur because the average hydrogen concentration does not exceed 10% by volume in the containment.
b. The pressure of containment at the termination of calculations (07 Days) is about 0.58 MPa and containment temperature is 423 K.

Total concrete ablation thickness is 1.57 m, whereas the total thickness of the basemat is 5.65 m.

It was concluded that, if the mitigative measures are made available even after 7 days, the pressure can be maintained well below the ultimate pressure (~ 0.86 MPa) of the containment.

3.2.3.2 Schedule and milestones to complete the planned activities

Tasks	Target Date
Initial Review of Regulation PAK/911	31-12-11
Detailed Review of Regulation PAK/911	31-12-13
Initial Review of Regulation PAK/912	31-12-11
Initial Review of Regulation PAK/913	31-12-11
Detailed Review of Regulation PAK/913	31-12-13

3.2.3.3 Preliminary or final result of these activities, including proposals for further actions

The Initial review of PNRA Regulations identified that the PAK/911 “Regulation on the Safety of Nuclear Power Plant -Design” and PAK/913 “Regulations on the Safety of Nuclear Power Plants- Operation” need to be revised and additional safety criteria may need to be incorporated in these regulations.

The analysis of accident scenarios as mentioned in Section 3.1.3 revealed that the Large Break LOCA in connection with SBO is the most severe accident in term of containment pressure and the peak containment pressure for LBLOCA with SBO after 7 days reaches up to 0.58 MPa which is less than the containment ultimate pressure (~0.86 MPa). If the mitigative measures are made available even after seven days, the pressure can be maintained well below the ultimate pressure of the containment.

3.3 Severe Accident Management and Recovery (On-site)

3.3.1 Assessment of Severe Accident Management and On-Site Recovery Measures at K-1

Emergency Operating Procedures (EOPs) are essential tools to handle any nuclear emergency that arises during normal plant operations within the scope of design basis accidents. At K-1, Abnormal Transient Operation Guidelines (ATOGs) were used to handle nuclear emergencies or abnormal transients. K-1 prepared EOPs in 2008 with the assistance of IAEA, which were reviewed by IAEA Expert Mission in 2009. The suggestions /recommendations of IAEA experts were incorporated and revision-1 of EOPs was issued. In 2010, these EOPs were further revised to incorporate all the transient conditions mentioned in ATOG. The revised EOPs are being verified and validated by 'Table Top' and 'Walk Through' techniques.

External events will be covered in EOPs after implementation of K-1 Action plan in response to Fukushima accident. EOPs for external hazards such as flooding/tsunami, spent fuel cooling are to be developed. SAMGs (based on engineering judgment) are available to be consulted during severe accident conditions.

Emergency Management Group (EMG) and Emergency Support Group (ESG) are available to provide guidance, assistance and advice to Shift Supervisor in case of any emergency at the plant. EMG Group is headed by Director General K-1.

3.3.1.1 Actions taken for improvement

Fukushima Response Action Plan (FRAP) for K-1 was prepared and following actions were decided in relation to EOPs:

i. Verification, validation and training of operators on EOPs
ii. Updating EOP on earthquake safety & surveillance (Incorporation of mitigating actions for spent fuel storage bay)
iii. Development/improvement of EOPs for external flooding

Revised EOPs will be issued after validation.

Severe accidents analyses in the light of Fukushima accident (including SBO) were carried out to assess the following:

a. Need and feasibility of increasing pumping head (EFW system)
b. Extent of hydrogen hazard
c. Feasibility and need for (passive) Hydrogen Recombiners and Hydrogen Igniters
d. Flow requirements in different systems for long term core cooling
e. Adequacy of passive cooling such as natural circulation
f. Adequacy of DSW System for alleviating reactor building pressure
g. Feasibility of installing system for releasing containment pressure
h. Containment behavior

Additional supporting analyses for Severe Accident Management Guidelines (SAMGs) are being carried out indigenously. Technical Support Committee (TSC) and Emergency Response Team (ERT) have been established to assist the Emergency Management Group (EMG) to bring the plant in stable condition and minimize the radiological consequences in case of severe accident. Training / briefing has been given to EMG, ERT and TSC personnel for their intended response in case of severe accident conditions.

3.3.1.2 Schedule and milestones to complete the planned activities

Task	Target Date
Revisiting the existing SAMGs	30-06-11
Availability of all necessary equipment / gears for implementing EOPs, SAMGs	31-03-12
Review / ensure functionality of TSC in accident conditions on the basis of Fukushima experience	31-03-12
Development of a reserve force (ERT) of workers for coping with severe accident consequences	31-03-12
Supporting Analyses for SAMGs	31-12-12
Revision of SAMGs based on supporting analyses	31 -12-13

3.3.1.3 Preliminary or final result of these activities, including proposals for further actions

Currently, verification and validation of EOPs is being performed by 'Table Top' and 'Walk Through' techniques. SAMGs will be revised by December 2013 after completion of relevant supporting analyses.

3.3.2 Assessment of Severe Accident Management and On-Site Recovery Measures at C-1 & C-2

The following design features act as preventive measures against severe accidents

i. The general countermeasures are as follows (in C-1 and C-2 both):
    a. Large Steam Generator (SG) secondary water inventory
    b. Large pressurizer volume
    c. Large containment free volume
    d. Two relief valves in each main steamline
    e. Two diverse pumps in each train for Auxiliary Feedwater System (SAF)
    f. Additional startup-shutdown feedwater pump
    g. Safety graded Reactor Pressure Vessel (RPV) venting valve for release of non-condensable gases.

ii. Station Blackout (SBO) recovery: For C-1, diesel generators are provided as Alternate AC source (AAC) for each safety bus. For C-2, a diverse AAC diesel generator is provided on swing bus.

For C-2, specific additional measures are provided to mitigate the consequences of severe accidents. These include:

- Passive Autocatalytic Hydrogen Recombiners (PARs).

- Reactor Cavity Flooding System (SCF)
- A dedicated throttle valve on Pressurizer to preclude High Pressure Melt Ejection (HPME)
- An independent UPS for reactor coolant pump seal return water valves to address seal LOCA concerns during severe accidents

C-2 has generic SAMGs along with a number of computational aids supported by background documents. Initial training has been imparted to MCR & Technical Support Center personnel. A plan to update the SAMGs is underway to make it more plant specific.

CHASNUPP Emergency Response Organization (CERO) comprises two groups. One Group assembles at on-site Emergency Control Center (ECC) while the other assembles at Technical Support Center adjacent to plant MCR. The ECC, led by site Emergency Director, is the main command point for decision making and coordination of all activities during emergencies. This also includes radiological monitoring, off-site consequence assessment and initiation of corrective actions including maintenance jobs during emergencies. All external notifications and contacts are coordinated from this ECC. Technical Support Center assists MCR during some aspects of Design Basis Accident (DBA). The main function is its role for implementing SAMGs. It has no external contact function.

3.3.2.1 Actions taken for improvement

Minimum inventory of Boric Acid to cover potential emergencies has been increased. Based on initial assessment, requirement to install filtered venting system is proposed. For C-1, PARs and Pressurizer throttle valve are planned while feasibility study for cavity flooding system is underway.

3.3.2.2 Schedule and milestones to complete the planned activities

Task	Target Date
Increase in minimum inventory of Boric Acid to cover potential emergencies at C-1	31-12-11
Preparation of C-1 SAMGs	31-12-13
Preparation / Enhancement of onsite Emergency Plan Implementation Procedures (EPIPs) to address external natural hazards in light of Fukushima accident	31-08-12
Enhancement of capability of Technical Support personnel for severe accidents	30-06-12
Development of a reserve force of workers to cope with severe accident consequences at C-1/C-2	31-12-12
Preparation of proposal for common alternate ECC/ resource center for Chashma site	31-07-12
Availability of necessary equipment / gears for implementing SAMGs	31-12-13

3.3.2.3 Preliminary or final result of these activities, including proposals for further actions

Quantity of Boric Acid stored at site has been increased in the light of Fukushima experience {Beyond Design Basis Accident (BDBA) including SFP}. Study on the use of commercially available boric acid in case of unavailability of reactor grade in accident conditions has also been proposed.

Revision of EPIPs has been proposed to ensure Emergency Response Personnel availability at site in extreme situations.

Necessary competencies and qualification for the reserve force personnel especially with reference to Fukushima accident are being identified. Development of a cross training program to optimize capability of such personnel is proposed. Implementation of measures to maintain up to date information about the reserve force are also proposed.

It has been proposed that establishment of a common facility at suitable location as backup of ECC of all plants be explored. This common facility should have environmental monitoring information for the site and plant information for all plants. The building is to be hardened against extreme natural hazards and radiation from accidents. It can also be used for storage of all emergency equipment.

3.4 National Organizations

3.4.1 Pakistan Nuclear Regulatory Authority

Government of Pakistan (GoP) established Pakistan Nuclear Regulatory Authority (PNRA) as an independent regulatory body under an Ordinance in 2001. PNRA has been provided with necessary authority as well as human and financial resources to fulfill its assigned responsibilities of regulating nuclear safety and radiation protection in the country.

The existing workforce at PNRA stands at two hundred and fifty (250) professionals covering all core and support functions. PNRA continuously strives for excellence and is committed to improve its regulatory performance. It has provided opportunities for its staff to participate in international workshops/fellowships, on-the-job trainings and scientific visits in specialized fields for enhancing their technical competence.

3.4.1.1 Regulatory Effectiveness, Independence, Openness and Transparency

PNRA reports directly to the Prime Minister. PNRA has established two Technical Support Organizations, Center for Nuclear Safety at Islamabad and the Safety Analysis Center (SAC) at Karachi.

PNRA has been performing annual monitoring and evaluation of its regulatory activities since its inception. The monitoring of the regulatory performance of the PNRA is based on 12 strategic performance Indicators. (Please refer to Annexure–XIII of Pakistan's Fifth National Report for a list of Performance Indicators). PNRA also invites international experts for peer reviews. This process contributes to continuous improvement of regulatory effectiveness and efficiency, and drives PNRA towards improved performance in all of its activities.

PNRA regularly submits its annual report to the Government of Pakistan which is shared with the public. This has improved transparency and enabled the GoP and the public to keep abreast of regulatory activities.

Stakeholders and the general public are involved in the review process of national regulations. PNRA has established liaison with other national regulators to exchange information regarding licensing and regulatory oversight processes adopted by various regulatory authorities.

3.4.1.2 Actions taken for improvement

PNRA has reviewed its organizational capabilities and regulatory oversight processes in the light of Fukushima accident. PNRA revisited regulatory requirements for the safety of nuclear power plants to incorporate lessons learnt from Fukushima accident. As a result, a number of initial recommendations have been made which are under review process for revision of the regulations.

3.4.1.3 Schedule and milestones to complete the planned activities

Task	Target Date
Initial review of PNRA regulations related to safety of nuclear power plants	31-12-11
Revision of PNRA regulations in the light of feedback from Fukushima accident	31-12-13

3.4.1.4 Preliminary or final result of these activities, including proposals for further actions

Seven modifications have been proposed to be incorporated in National regulations after the initial review of regulations. Review of organizational capability and regulatory oversight processes did not identify any immediate need for changes.

3.4.2 Pakistan Atomic Energy Commission

Pakistan Atomic Energy Commission (PAEC) is the owner and licensee of a number of nuclear facilities which include nuclear power plants, research reactors, nuclear medical centers, agricultural centers, biotechnology centers and industrial units using radioactive sources. PAEC has the overall responsibility for the fulfillment of safety requirements of all its nuclear facilities. The design and operation of nuclear installations are in accordance with the national regulations and PAEC's Nuclear Safety Policy. The Government of Pakistan ensures that PAEC is provided with adequate resources to ensure safety of its nuclear installations throughout their life.

3.4.2.1 Commitment to improving Nuclear Safety, Openness and Transparency at NPPs

Commitment to nuclear safety is highlighted in PAEC's Nuclear Safety Policy which delineates specific safety rules and requirements for ensuring safety at Nuclear Installations. As a policy, PAEC always aims to exceed the regulatory requirements. As a result, most of them are met with comfortable safety margins. At the corporate level, the Directorate of Nuclear Safety (DNS) and Directorate of Quality Assurance (DQA) advise the corporate management on safety and quality issues. These offices conduct safety inspections and the QA audits of the nuclear facilities regularly. The recorded observations and recommendations serve as additional and independent source of information for the senior management. Independent reviews and assessments are also performed by international expert organizations. At the nuclear installation level, divisions with necessary authority and independence are in place, which are responsible for nuclear safety, licensing and quality assurance related activities. In addition, safety committees advise the management on safety and quality related issues as referred.

In accordance with the Nuclear Safety Policy, all the operating NPPs are conducting self-assessment at the plant level with reference to IAEA Nuclear Safety Action Guidelines. PAEC is committed to establishing and maintaining a capability at the corporate level to monitor and oversee all aspects of safety. Independent oversight is not a substitute for the responsibility of the management of nuclear facilities. Rather it serves as independent source of information for the senior management.

PAEC is also committed to establishing and promoting safety culture at its nuclear facilities, which encourages setting high standards, identifying and resolving problems and openness to questioning and criticism. A blame-free environment for the workers at its nuclear facilities is being maintained ensuring mutual respect and effective communication with various levels of management for continuous enhancement of safety in all the activities. International Peer Reviews and the operating experience feedback etc are used to verify the safety culture.

All activities at NPPs that may affect safety are performed by suitably qualified and experienced staff. The manpower requirements are met by establishing and implementing qualification and training programs. The same is verified through internal and external peer reviews conducted periodically.

PAEC is continually enhancing the capacity and quality of its key training institutes, such as Pakistan Institute of Engineering and Applied Sciences (PIEAS), Karachi Institute of Power Engineering (KINPOE), and Chasnupp Center for Nuclear Training (CHASCENT).

The activities at the nuclear power plants are appropriately reported and the reports are kept open to reviews at national and international level. All operating NPPs issue a number of reports on plant performance e.g. daily report, monthly technical reports covering all aspects of their operation and maintenance, annual reports analyzing their safety performance, quarterly performance indicator reports to WANO, reports about events and their analyses (besides those reportable under PNRA regulations), reports about significant outages, technical reports on specific topics, etc. K-1 also sends a monthly report to COG. All these reports are available to PNRA also, besides being analyzed and reviewed by the Directorate of Nuclear Power Operations (DNPO) at the corporate level.

3.4.2.2 Actions taken for improvement

Pakistan Atomic Energy Commission is carefully observing all the developments in Japan and countermeasures being taken in response to Fukushima accident since March, 2011. In perspective of the message delivered to the nuclear industry that nature is much more powerful and much more unpredictable than ever, PAEC had challenged its NPPs to come up with innovative ideas to provide greater safety assurance, a comprehensive plan was chalked out for re-visiting design of nuclear power plants to re-assess safety margins in line with IAEA Nuclear Safety Action guidelines. DNS, reporting to the Chairman, has been made responsible for periodical monitoring of the progress on measures being taken with regard to Fukushima Response Action Plans (FRAP) submitted by all three operating NPPs.

Although, Internal Peer Review (IPR) of the Operational Safety of NPPs at the corporate level was initiated in 2009, the scope was expanded after Fukushima accident to include Design Safety and Emergency Preparedness & Response (EPR). A team of the specialists from diverse disciplines conducted Internal Peer Reviews of all three operating NPPs and recommended various actions and measures to assure nuclear safety. IPR of K-1 was conducted in October 2011. The Review Team utilized 'IAEA Safety Action Plan' and 'ENSREG Stress Test specifications' as guidelines for the review of Fukushima Response Action Plan. IPR of K-1 was conducted by a team of 15 experienced professionals. The actions in Fukushima Response Action Plan of K-1 (FRAP-K1) were found to be progressing satisfactorily. Combined progress review of C-1 and C-2 on Fukushima Response Plan (FRAP-C12) was conducted in January, 2012 by a team of 7 specialists. There has been satisfactory progress on the actions identified.

3.4.2.3 Schedule and milestones to complete the planned activities

Task	Target Date
Review of progress of all NPPs on FRAP	31-01-12
Follow-up of the implementation of FRAP	Quarterly

3.4.2.4 Preliminary or final result of these activities, including proposals for further actions

The Review identified certain improvements need to be considered in already concluded studies for K-1, C-1 and C-2. It was decided that all insights will also be considered for future NPPs. These recommendations have been reported in the Topical Areas (3.1 to 3.6) of this report

3.5 Emergency Preparedness and Response and Post-Accident Management (Off-Site)

3.5.1 National Disaster Management Authority (NDMA)

West Pakistan National Calamities (Prevention and Relief) Act of 1958 provides for the maintenance and restoration of order in areas affected by calamities, and relief against such calamities. The Calamities Act 1958 was mainly focused on organizing emergency response. Emergency Relief Cell (ERC) created within the Cabinet Division in 1971, is responsible for disaster relief at national level. It provides assistance in cash and supplements the resources of the Provincial Governments in event of major disaster. National Crisis Management Cell (NCMC) was established in July 1999 under the Ministry of Interior. NCMC is responsible for coordinating plans for emergency relief services in case of emergency situations and its main function is to collect information regarding various emergencies in the country, along with coordination with Provincial Crisis Management Cell and other relevant agencies.

The need for strong institutional and policy arrangements was fulfilled by promulgation of the National Disaster Management Ordinance 2007 (NDMO) in the aftermath of the 2005 earthquake. Under NDMO[1], Government of Pakistan established a National Disaster Management Commission (NDMC) headed by the Prime Minister. It also established a National Disaster Management Authority (NDMA) to serve as the focal point and coordinating body to facilitate implementation of disaster management. All stake-holders including government departments / agencies and armed forces work through and form a part of NDMA in all stages of Disaster Risk Management.

Provincial Disaster Management Authorities (PDMAs) and District Disaster Management Authorities (DDMAs) have been established at the provincial and district levels of the country. The organization of Disaster Management in Pakistan is given in Figure 3.5-1. The National Disaster Management Authority has formulated a National Disaster Response Plan[2] (NDRP) after extensive cross-sectored consultations. NDRP seeks to upgrade the country's ability to cope with all conceivable disasters. To achieve this purpose, complete range of disaster management activities from preparedness to response has been addressed. The formulation outlines a framework for emergency response at different levels of the government structure; identifies roles and responsibilities of various stakeholders, and lays down coordination mechanism for activities involving the United Nations, Non Governmental Organizations (NGOs), civil society organizations, public & private sector and media to harness the full national potential for efficient disaster management.

Details on functions and responsibilities of NDMA/PDMAs/DDMAs and other stake holders are given in the NDRP.

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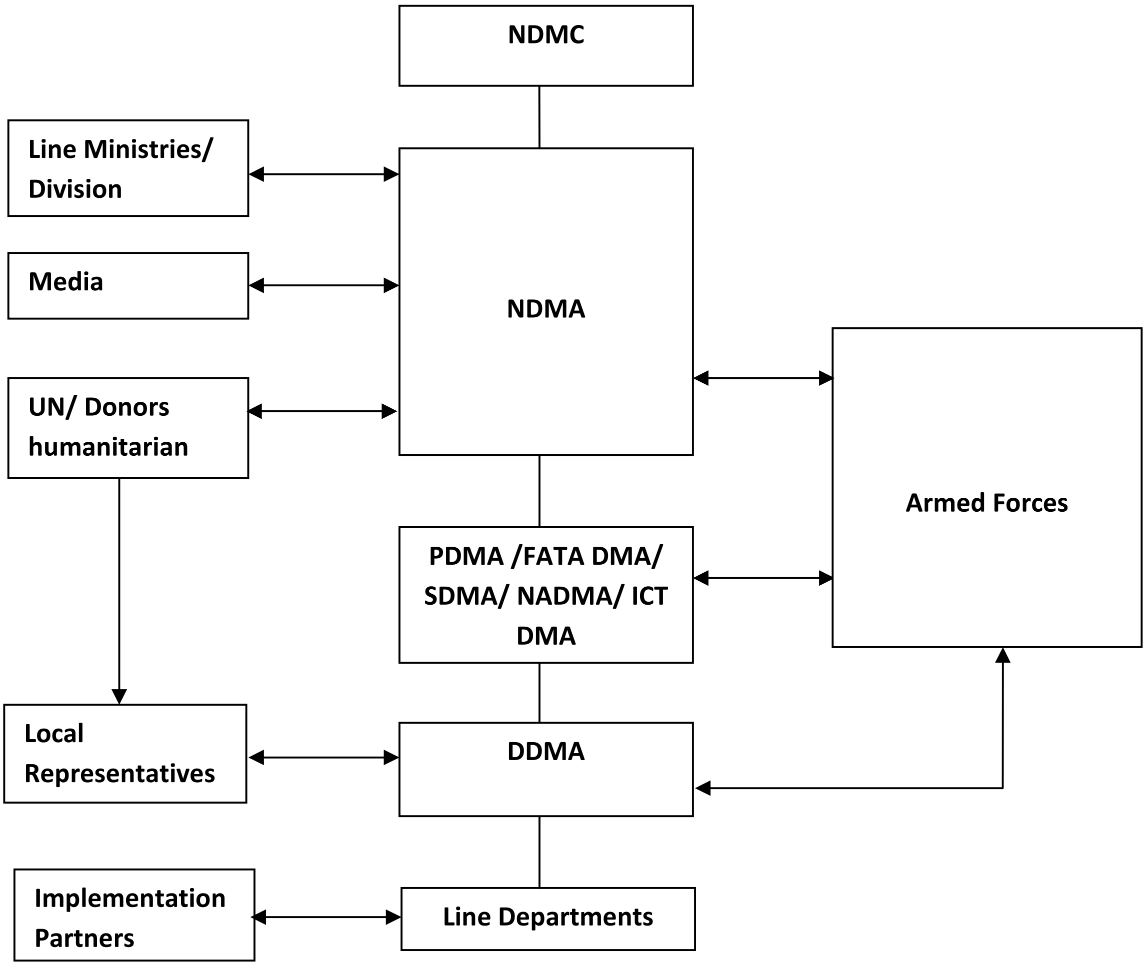
[1] http://www.ndma.gov.pk/publications/ordinance.pdf

[2] http://www.ndma.gov.pk/documents/NDRP/NDRP.pdf

Figure: 3.5-1: Organization of Disaster Management in Pakistan



3.5.2 Nuclear Emergency Management System (NEMS)

The emergency plans of the nuclear power plants are continuously evaluated and improved. Recently, with a view to ensure effective and efficient management of nuclear emergencies, the existing system has been provided an overarching arrangement under the concept of Nuclear Emergency Management System (NEMS). This concept has been built around the principles of centralized control, decentralized execution through tiered and graded approach and comprehensive involvement of operators and other organizations. Under this system a very comprehensive view is taken of the communication arrangements, flow of information, pre-arrangement of required assistance from different organizations / agencies and designation of authority to orchestrate such emergencies. The sole objective of this arrangement is provision of timely response to all nuclear and radiological emergencies. NEMS seamlessly dovetails with NDMA plans in case the nuclear emergencies begin to have off-site effects.

3.5.3 Off-Site Emergency Preparedness, Response and Post Accident Management at K-1

On-site and Off-site emergency plans of K-1 are in place and exercises are conducted periodically. On-site Emergency Control Center and Alternate Emergency Control Center (about 13 km from plant site) of K-1 are equipped with communication facilities, radiation monitoring system, post accident monitoring system, medical facilities, decontamination facilities, etc. Off-site Emergency Operating Center is located at PDMA office in Karachi city.

Off-site support is provided by KANUPP Radiological Emergency Response Committee (KRERC), chaired by DG (PDMA) Sindh for management of consequences of the accident and protection of public and environment.

3.5.3.1 Actions taken for improvement

Emergency Planning (EP) has been reviewed based on Fukushima experience. Improvements like development of recovery plan, policy to use Alternate Emergency Control Center as command point in severe accident, availability of DG & on-line Critical Parameter Display System (CPDS) in AECC have been made. Means for providing alternate communication links at plant and AECC are being explored. Reassessment of the Emergency Planning Zones (EPZ) is underway. In case of extensive damage to infrastructure around the plant, including communication facilities that may make technical support from outside more difficult, following steps will be taken for safety of plant personnel:

i. Storage of sufficient expendable items including food, fuel, water and other resources.
ii. Arrangements of alternate means of purveying different items of necessity, evacuation of persons / casualties and transportation of different types of experts.
iii. Training and motivation of workers to sustain themselves under prolonged conditions of adverse operating environment.

Newly formed TSC and ERT (Reserve Force), in consultation with EMG, will supplement plant personnel in case of impairment of work performance due to high local dose rates, radioactive contamination and destruction of some facilities.

3.5.3.2 Schedule and milestones to complete the planned activities

Task	Target Date
Reassessment of EPZs and corresponding emergency response actions	30 -06-12
Revision of Emergency Plans	31-03-12
Exploration of alternate communication links at plant and AECC	30-06-12
Revisit / enforce / support improvement of National Radiological Emergency Plan	30-06-12
Provision of Post Accident Monitoring equipment	31-12-13

3.5.3.3 Preliminary or final result of these activities, including proposals for further actions

Demographic data have been updated up to 16 km based on recent survey. Stock of Potassium Iodide (KI) tablets is being increased for population up to 16 km of the plant. Additional assistance and support required during radiological emergency from off-site authorities has been coordinated. Post Accident Monitoring equipment will be installed to monitor important parameters under accident conditions. On-site and Off-site Emergency Response Plan (KONREP & KOFREP) have been updated.

3.5.4 Off-Site Emergency Preparedness, Response and Post Accident Management at C-1 & C-2

On-site and off-site emergency plans of C-1 and C-2 are in place and exercises are conducted on regular basis. The coordination with local authorities and the response organizations are ensured through these plans. The emergency plans describe emergency facilities like emergency control centers (on-site and off-site), auxiliary emergency control Center, communication facilities, radiation monitoring system, post accident monitoring system, medical facilities, decontamination facilities, etc.

C-1 and C-2 have separate onsite emergency plans and a common offsite emergency plan. Both units have developed a joint procedure for interface during radiological emergency to establish communication link between ECCs and MCRs in case of emergency at C-1 and/or C-2. In case of emergency at any one unit, its MCR Shift Supervisor will notify the other unit to declare the same emergency class. Consequently, both the units will perform actions in accordance with their respective emergency plans and procedures.

The offsite emergency plan includes role and responsibilities of all the response organizations. District Government (Headquarter) is designated as offsite ECC. If the consequences are beyond its control, the offsite ECC may request support of provincial and federal government. These arrangements are exercised on regular basis according to the requirements of plans and procedures.

3.5.4.1 Actions taken for improvement

Improvement of EPPs has been initiated. Provision of alternate communication links in C-1 and C-2 MCRs is being explored. Reassessment of the Emergency Planning Zones (EPZ) is underway.

Provision of additional access routes to the site is being assessed. EPPs for C-1 and C-2 are being revised based on lessons learnt from Fukushima experience. Information regarding availability of portable post accident radiation monitoring equipment at various locations in Pakistan has been updated.

Personnel de-contamination facility in local hospital has been upgraded to handle increased number of contaminated personnel along with the relevant training of personnel involved. A reserve force of workers for coping with severe accident consequences has been established.

3.5.4.2 Schedule and milestones to complete the planned activities

Task	Target Date
Improvement of EPPs	30-04-12
Exploration of alternate communication links at MCR and ECC at C-1 /C-2	30-06-12
Reassessment of EPZ	30-12-11
Assessment of possible additional access routes to the site	30-06-14
Up-gradation of personnel de-contamination facility in local hospital	30-04-12

3.5.4.3 Preliminary or final result of these activities, including proposals for further actions

Means for providing alternate communication links at MCR and ECC at C-1/C-2 are being explored. A new bridge is proposed to be constructed for access to the plant site. Furthermore, a new escape route is proposed. Arrangement of a pontoon bridge has also been proposed in case of emergencies. Arrangements are in place to remove obstructions on access roads.

3.5.5 Role of PNRA in Off-Site Emergency Preparedness, Response and Post Accident Management

The role of PNRA is to ensure, co-ordinate and enforce preparation of emergency plans by licensee. PNRA has issued regulations to establish requirements for an adequate level of emergency preparedness and response to a nuclear or radiological emergency for minimizing its consequences.

National Radiation Emergency Coordination Center (NRECC) is established at PNRA Headquarter for coordination of response to nuclear accidents or radiological emergencies and remains functional round the clock. It is the focal point for regulatory response in case of an emergency (Abroad or Domestic) and also functions as the secretarial arm to Chairman PNRA, who is the National Competent Authority (NCA) for an emergency. NRECC is also the National Warning Point (NWP) for the Conventions on “Early Notification of a Nuclear Accident” and “Assistance in the Case of a Nuclear Accident or Radiological Emergency”.

During emergencies, the decision for implementation of protective measures is the responsibility of the licensee. The licensee keeps NRECC updated about the emergency situation and any protective measures taken.

PNRA is also coordinating with the response and law enforcing agencies to familiarize them with their role during a nuclear or radiological emergency.

A working group was formulated consisting of various stakeholders to develop the National Radiological Emergency Plan (NREP). The plan has been drafted and mainly includes the following:

- Legal requirements

- Responsibility of each stakeholder
- Coordination mechanism
- Emergency scenarios and their possible effects
- Contingency plans for each scenario
- Capacity building

3.5.5.1 Post Fukushima Actions by PNRA

The NRECC was activated on March 11, 2011 to study the evolving situation at Fukushima and its possible consequences. Being the national contact point, NRECC was receiving frequent updates from IAEA and accessing websites of Japanese national and other related organizations. The information received was continually reviewed and daily/weekly summary of updates was uploaded at PNRA website.

Pakistan has registered its National Assistance Capabilities (NACs) in IAEA Response and Assistance Network (RANET). Being the NCA designated under Early Notification and Assistance Conventions, PNRA, with the consent of GoP, offered assistance to Japan through IAEA in areas of radiation monitoring, source search and recovery, environmental measurements and assessment and advice on emergency response.

After the Fukushima accident, PNRA started air sample collection at seven sampling stations in the country to detect any releases in the environment due to this disaster. These stations were established at Islamabad, Karachi, Lahore, Kundian, Peshawar, Bahawalpur and Quetta. One hundred and fifteen (115) samples were collected up to 15th June, 2011 and no anthropogenic radio nuclides were detected.

To further ensure public protection, PNRA in coordination with Pakistan Customs restricted import of edible goods from Japan without radiation free certificate issued either by PNRA or Japan.

PNRA briefed the NDMA, EPA, NGOs, representatives of Prime Minister Secretariat and other concerned Government departments about potential consequences of Fukushima accident on Pakistan. PNRA also briefed Journalists and media, as and when contacted.

3.5.5.2 Actions taken for improvement

PNRA has directed the licensee to re-evaluate and strengthen emergency preparedness and response arrangements considering unavailability of necessary infrastructure (bridges, roads, communication means, etc.) due to severe natural disasters and demonstrate implementation of emergency plans specially the evacuation aspect by involving public. PNRA also required licensee to re-evaluate Emergency Planning Zones (EPZs). The licensee has submitted action plans for the re-assessment of emergency preparedness and response arrangements. The process for finalization of NREP has also been expedited after the Fukushima accident.

3.5.5.3 Schedule and milestones to complete the planned activities

Task	Target Date
Review of revised C-1 Onsite Emergency Plan	30-04-12
Review of revised CHASHMA Nuclear Power Generating Station (CNPGS) offsite Emergency Plan	30-06-12
Revision of PNRA Response Procedure for Incident and Emergencies	31-08-12
Finalization of NREP	31-12-12

3.5.5.4 Preliminary or final result of these activities, including proposals for further actions

Revised C-1 Onsite Emergency Plan has been reviewed and found acceptable.

3.6 International Cooperation

3.6.1 Status of International Conventions / treaties relevant to nuclear power technology

Presently, Pakistan is party to the following conventions related to nuclear safety:

S/N	Convention	Date of Accession
1.	Convention on early notification of a nuclear accident	12 October 1989
2.	Convention on assistance in the case of a nuclear accident or radiological emergency	12 October 1989
3.	Convention on nuclear safety	29 December 1997

As a party to the convention on assistance, Pakistan is part of IAEA's Response and Assistance Network (RANET). Under this network, Pakistan has registered its capabilities (radiation monitoring, source search & recovery, environmental measurements, assessment & advice) to provide assistance to other member states as needed. Pakistan also offered to provide assistance under RANET to Japan after the Fukushima accident.

3.6.2 Mechanisms for communicating with neighboring countries and the international community

Under the convention on Early Notification of a nuclear accident, state parties commit that in case of a nuclear accident that may have trans-boundary radiological consequences; they will notify the IAEA and the countries that may be affected. Pakistan has identified a National Warning Point (NRECC) to which a notification can be directed as well as a Competent Authority (Chairman, PNRA) which is authorized to send notifications. Pakistan is also exploring the possibility of concluding bilateral agreements for early notification of nuclear emergencies with its neighboring states.

Pakistan has established links with international organizations like the World Health Organization for mutual cooperation in the time of crisis. The country-level technical cooperation between World Health Organization (WHO) and the Government of Pakistan was launched in the early 1950s and formalized in 1960 with the establishment of the WHO Country Office. WHO is providing assistance to Pakistan in the area of "Emergency Preparedness and Humanitarian Action". PAEC and PNRA personnel with relevant experience participate and contribute with their expertise in many international activities related to nuclear safety (Expert Missions, Technical Committee Meetings, Conferences, Peer Reviews etc.) organized by IAEA, WANO, and other international bodies and professional organizations.

Experts from PNRA participated in various IRRS missions, which also included modules on lessons learnt from Fukushima. PNRA is evaluating the feedback of these missions for implementation within its regulatory activities.

3.6.3 Use of IAEA Safety Standards and strengthening of the global Nuclear Safety Regime

The regulatory framework of Pakistan is mainly based on the IAEA Safety Standards. Even before the formation of the PNRA, the then regulatory body conducted licensing of K-1 and C-1 according to IAEA safety standards and other international practices. PNRA will continue to incorporate new developments in IAEA Safety Standards into its regulatory framework.

Pakistan actively contributes in the development of IAEA Safety Standards. PNRA participates in the proceedings of various IAEA committees for the development of safety standards, such as the Nuclear Safety Standards Committee (NUSSC), Transport Safety Standards Committee (TRANSSC), Waste Safety Standards Committee (WASSC), Radiation Safety Standards Committee (RASSC), and the Committee on Safety Standards (CSS). Pakistan has also recently gained membership of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR). In the operational safety regime of IAEA, experts from Pakistan provide consultancy in the development of IAEA documents e.g. TECDOCs, Safety Reports, Technical reports etc.

Immediately after the Fukushima accident, Pakistan conducted an international seminar on "Nuclear Safety and Security challenges of the 21st century" from 21-23 April, 2011 at Islamabad in collaboration with IAEA at which feedback from Fukushima and challenges to Nuclear Safety and Security were discussed in detail.

3.6.4 Use of International Peer Review Services

Pakistan invites international Review Missions such as IAEA Design Safety Review Service, Operational Safety Review Team (OSART), International PSA Review Team (IPSART), Integrated Regulatory Review Team (IRRT), Peer Review of Operational Safety Performance Experience (PROSPER), World Association of Nuclear Operators (WANO) from time to time. Pakistan has again invited an IAEA IRRS Mission to be conducted in 2013 with preparatory mission in 2012.

Pakistan shares information with the international community through Nuclear Event Web based System (NEWS), Incident Reporting System (IRS) and WANO Operating Experience Program.

C-1 and C-2 share operating experience feedback with the Designer (SNERDI), Reference Plant, i.e., Qinshan Nuclear Power Company (QNPC) etc. K-1 interacts with CANDU Owners Group (COG) and Atomic Energy of Canada Limited (AECL) to exchange Operating Experience Feedback.

3.6.5 Actions taken for improvement

Pakistan has again invited an IAEA IRRS Mission. A WANO Peer Review was conducted at C-1 in April, 2012.

3.6.6 Schedule and milestones to complete the planned activities

Task	Target Date
WANO Peer Review at C-1	April, 2012
IRRS Mission at PNRA	2013

3.6.7 Preliminary or final result of these activities, including proposals for further actions

Final recommendations arising from the WANO Peer Review have not been received yet.

PART IV

Tabulated Summary of Activities to be conducted /carried out by Operator and Regulator

4. Summary Table of Activities undertaken

4.1 Summary Table of Activities undertaken by Operator in response to Fukushima accident

Topic 1 : External Events			
K-1			
Earthquake Hazard			
Task	**Target**	**Activity Status**	**Results Available**
Surveillance walk-down of seismic supports to identify any weaknesses or potential new hazards to critical equipment, and their resolution	30-06-12	Study Completed, Implementation is in progress	Yes
Study to incorporate auto shutdown if SSE is detected	31-01-12	Completed	No
Detailed visual inspections of all the structures, especially the structures important to safety	29-02-12	Completed	Yes
Re-evaluation of seismic capacity of all structures for the revised seismic input of 0.2 g	31-12-12	In Progress	No
Confirmation of seismic qualification of existing diesel fuel tanks & mounting platform	31-12-12	In Progress	No
Re-assessment of earthquake hazard based on the new IAEA methodology (without field work)	31-03-12	In Progress	No
Tsunami / Flooding Hazard			
Task	**Target**	**Activity Status**	**Results Available**
Fresh Estimate of Tsunami hazard for K-1	31-03-12	Completed	Yes
Identification of the equipment vulnerable to unprecedented tsunami/flooding and feasibility to improve resilience	30-06-12	In Progress	No
Feasibility of tsunami wall on sea side of the plant or around vulnerable equipment / systems	31-03-12	Completed	Yes

Fresh study of Hazards due to cyclones and flooding due to maximum probable precipitation	30-06-12	In Progress	No
Storms/ Cyclones Hazard			
Task	**Target**	**Activity Status**	**Results Available**
Study of Cyclones including effects of different categories of Cyclones	29-02-12	Completed	Yes
Re-assessment of structures for wind loading	31-12- 12	In Progress	No
C-1/C-2			
Earthquake Hazard			
Task	**Target**	**Activity Status**	**Results Available**
Seismic walk-down of SSCs to identify any non-conformances and potential new hazards to critical equipment from external natural hazards	30-06-12, 31-01-13(for Rx area	In Progress	Preliminary Results available
Enhance seismic structures surveillance program	31-08-12	In Progress	No
Feasibility study of auto shutdown if SSE is detected	31-08-12	In Progress	No
Tsunami / Flooding Hazard			
Task	**Target**	**Activity Status**	**Results Available**
New study of flooding hazard potential for Chashma Site using updated historical information and considering Chashma Barrage break as a result of upstream dam break and other potential new scenarios	30-09-12	In Progress	No
Re-assessment of Emergency Control Centers (ECCs) robustness of both C-1 and C-2 against flooding and earthquake	30-04-12	In Progress	No
Other External Hazards			
Task	**Target**	**Activity Status**	**Results Available**
Reassessment of vulnerability against hazards like storms, tornados, etc.	15-04-12	Completed	Yes

Topic 2 : Design Issues			
K-1			
AC Electrical Power			
Task	**Target**	**Activity Status**	**Results Available**
Development of Integration scheme for 300 kW, 400V mobile DG with Essential power supply	30-10-12	In Progress	No
Provision for energizing the plant essential buses through FIJW-DG1/2	31-08-12	In Progress	No
Installation of a new 80 kW, 400V DG set and its integration scheme	30 -10-12	In Progress	No
Shifting of emergency lighting from 230 VDC to 220 V AC (UPS)	30-10-12	In Progress	No
Provision of alternate power supply to essential valves of ECCS	30-10-12	In Progress	No
Feasibility study to arrange trolley mounted DGs from other organizations	29-02-12	Completed	Yes
Feasibility to increase diesel fuel storage capacity onsite	31-03-12	In Progress	No
Identification of external sources of Diesel Fuel	31-03-12	In Progress	No
Feasibility of use of natural gas in one of the Diesel generators for diversity	31-01-12	Completed	Yes
Procedure for conserving DC power to prolong its availability	30-06-11	Completed	Yes
Emergency Core Cooling			
Task	**Target**	**Activity Status**	**Results Available**
Feasibility study to determine the need of increasing pumping heads of EFW system	29-02-12	Completed	Yes
Provision of additional points for fresh water injection and use of fire fighting system (as a last resort) for emergency core cooling	31 -10-12	In Progress	No
Feasibility study of passive cooling such as natural circulation after shutdown and in case of SBO	31-03-12	Completed	Yes
Feasibility of steam-driven pumps to feed the boilers in extreme case	31-12-11	Completed	Yes
Containment Integrity			
Task	**Target**	**Activity Status**	**Results Available**
Assessment of hydrogen hazard	31-01-12	Completed	Yes

Feasibility and need for (passive) hydrogen recombiners and hydrogen igniters	31-01-12	Completed	Yes
Identification of measures that can be taken in the worst case scenario to ensure containment integrity	31-01-12	Completed	Yes
Review adequacy of DSW for severe accident	31-01-12	Completed	Yes
Feasibility of operating motorized relief valves (dampers) manually when power is not available	31-01-12	Completed	Yes
Feasibility of installing system for relieving containment pressure	31-10-12	In Progress	No
Ultimate Heat Sink			
Task	**Target**	**Activity Status**	**Results Available**
Demonstration of Passive cooling (in cold condition) such as natural circulation	31-03-12	Completed	Yes
Integrity Assessment of Intake Bay Structure	30-06-12	In Progress	No
Spent Fuel Cooling			
Task	**Target**	**Activity Status**	**Results Available**
Re-assessment of Spent Fuel Pool (SFP) seismic design	31-03-12	In Progress	No
Assessment of safe dry times of spent fuel	29-02-12	Completed	Yes
Provision of measures against loss of cooling or drainage of SFP	30-10-12	In Progress	No
Estimation of source term of spent fuel when water is lost or configuration is disturbed in SFP	31-03-12	In Progress	No
Study of criticality hazard of enriched (10%) fresh booster fuel assemblies stored at new fuel storage area in case of tsunami	31-01-12	Completed	Yes
Study of criticality hazard of SFP in case of earthquake & tsunami due to presence of used enriched booster fuel	31-03-12	In Progress	No
C-1/C-2			
AC Electrical Power			
Task	**Target**	**Activity Status**	**Results Available**
Preparation of conceptual proposal for providing additional AC Power Source covering extreme natural hazards, all NPPs, interconnections of all installations, provision of trolley mounted	30-06-12	In Progress	No

small DGs, hookup of individual essential loads/buses, cables to remote connection points, resources available with the other organizations			
Study to increase the storage capacity of Diesel fuel	31-12-11	Completed	Yes
Arrangements for supply of Diesel fuel in case of natural disaster	31-12-11	Completed	Yes
Feasibility for increase in DC Power capacity	31-12-12	In Progress	No
Emergency Core Cooling			
Task	**Target**	**Activity Status**	**Results Available**
Exploration of possible additional points for water injection into steam generator using temporary pumping sources in extreme case	30-06-12	Completed	Yes
Feasibility study of interconnecting SAF system with Safety Injection system (SIS)	31-12-12	In Progress	No
Revisit procedure for use of Fire Fighting system for emergency cooling through steam generators	30-06-12	In Progress	No
Feasibility of installing Cavity flooding system at C-1	31-12-12	In Progress	Preliminary results available
Containment Integrity			
Task	**Target**	**Activity Status**	**Results Available**
Installation of PARs at C-1	31-07-14	In Progress	Preliminary results available
Feasibility of installing SRC Pressurizer Throttle Valve at C-1	31-12-12	In Progress	Preliminary results available
Feasibility of Filtered Venting System for C-1 and C-2	31-12-12	In Progress	No
Ultimate Heat Sink			
Task	**Target**	**Activity Status**	**Results**

			Available
Re-assessment of the consequences of Loss of Ultimate Heat Sink	31-12-11	Completed	Yes
Improvement in design of Essential service water pumping station entrance to prevent inundation in case of extreme flooding	30-09-12	In Progress	No
Spent Fuel Cooling			
Task	**Target**	**Activity Status**	**Results Available**
Estimation of source term for spent fuel if water is lost or configuration is disturbed, in SFP	30-06-12	In Progress	No
Determination of safe dry time (Fuel un-covered) of SFP	31-12-11	Completed	Yes
Study of measures against SFP loss of cooling or drainage	30-06-12	In Progress	No
Provision of SFP cooling with fire water	31-12-12	In Progress	No
Topic 3 : Severe Accident Management			
K-1			
Task	**Target**	**Activity Status**	**Results Available**
Revisiting the existing SAMGs	30-06-11	Completed	Yes
Availability of all necessary equipment / gears for implementing EOPs, SAMGs	31-03-12	Completed	Yes
Review / ensure functionality of TSC in accident conditions on the basis of Fukushima experience	31-03-12	Completed	Yes
Development of a reserve force (ERT) of workers for coping with severe accident consequences	31-03-12	Completed	Yes
Supporting Analyses for SAMGs	31-12-12	In Progress	No
Revision of SAMGs based on supporting analyses	31-12-13	In Progress	No
C-1/C-2			
Task	**Target**	**Activity Status**	**Results Available**
Increase in minimum inventory of Boric Acid to cover potential emergencies at C-1	31-12-11	Completed	Yes
Preparation of C-1 SAMGs	31-12-13	In Progress	No

Preparation / Enhancement of onsite Emergency Plan Implementation Procedures (EPIPs) to address external natural hazards in light of Fukushima accident	31-08-12	In Progress	No
Enhancement of capability of Technical Support personnel for severe accidents	30-06-12	In Progress	No
Development of a reserve force of workers to cope with severe accident consequences at C-1/C-2	31-12-12	In Progress	No
Preparation of proposal for common alternate ECC/ resource center for Chashma site	31-07-12	In Progress	No
Availability of necessary equipment / gears for implementing SAMGs	31-12-13	In Progress	No
Topic 4: National Organizations			
PAEC			
Task	**Target**	**Activity Status**	**Results Available**
Review of progress of all NPPs on FRAP	31-01-12	Completed	Yes
Follow-up of the implementation of FRAP	Quarterly	In Progress	No
Topic 5: Emergency Preparedness and Response and Post-Accident Management (Off-Site)			
K-1			
Task	**Target**	**Activity Status**	**Results Available**
Reassessment of EPZs and corresponding emergency response actions	30 -06-12	In Progress	No
Revision of Emergency Plans	31-03-12	Completed	Yes
Exploration of alternate communication links at plant and AECC	30-06-12	In Progress	No
Revisit / enforce / support improvement of National Disaster Management Authority (NDMA) / Nuclear Emergency Management System (NEMS) plans	30-06-12	In Progress	No
Provision of Post Accident Monitoring equipment	31-12-13	In Progress	No
C-1/C-2			
Task	**Target**	**Activity Status**	**Results Available**

Improvement of EPPs	30-04-12	In Progress	No
Exploration of alternate communication links at MCR and ECC at C-1 /C-2	30-06-12	In Progress	No
Reassessment of EPZ	30-12-11	Completed	Yes
Assessment of possible additional access routes to the site	30-06-14	Completed	Yes
Up-gradation of personnel de-contamination facility in local hospital	30-04-12	Completed	Yes
Topic 6 : International Cooperation			
PAEC			
Task	**Target**	**Activity Status**	**Results Available**
WANO Peer Review at C-1	April 2012	Completed	No

4.2 Summary Table of Activities undertaken by Regulator in response to Fukushima accident

Topic 1 : External Events			
Task	**Target**	**Activity Status**	**Results Available**
Initial Review of Regulation PAK/910	31-12-11	Completed	Yes
Detailed Review of Regulation PAK/910	31-12-13	In progress	No
Topic 2 : Design Issues			
Task	**Target**	**Activity Status**	**Results Available**
Initial Review of Regulation PAK/911	31-12-11	Completed	Yes
Detailed Review of Regulation PAK/911	31-12-13	In progress	No
Initial Review of Regulation PAK/912	31-12-11	Completed	Yes
Initial Review of Regulation PAK/913	31-12-11	Completed	Yes
Detailed Review of Regulation PAK/913	31 -12-13	In progress	No
Topic 4: National Organizations			
Task	**Target**	**Activity Status**	**Results Available**
Initial review of PNRA regulations related to safety of nuclear power plants	31-12-11	Completed	Yes
Revision of PNRA regulations in the light of feedback from Fukushima accident	31-12-13	In progress	No
Topic 5: Emergency Preparedness and Response and Post-Accident Management (Off-Site)			
Task	**Target**	**Activity Status**	**Results Available**
Review of revised C-1 On-site Emergency Plan	30-04-12	Completed	Yes
Review of revised CNPGS Off-site emergency plan	30-06-12	In progress	No
Revision of PNRA Response Procedure for Incident and Emergencies	31-08-12	In progress	No
Finalization of NREP	31-12-12	In progress	No

Topic 6 : International Cooperation			
Task	**Target**	**Activity Status**	**Results Available**
IRRS Mission at PNRA	2013	Planned	No