Enhanced safety features of CHASHMA NPP UNIT-2 to encounter selected severe accidents, various challenges involved to prove the adequacy of severe accidents prevention/mitigation measures and to write management guidelines with one possible solution to these challenges.

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Abstract. This paper describes enhanced safety features of Chashma Nuclear Power Plant Unit-2 (C-2), a 325 MWe PWR to encounter selected severe accidents and discusses various challenges involved to prove the adequacy of severe accidents encountering measures and to write severe accident management guidelines (SAMGs) in compliance with the recently introduced national regulations based on the new IAEA nuclear safety standards. C-2 is being built by China National Nuclear Corporation (CNNC) for Pakistan Atomic Energy Commission (PAEC). Its twin, Unit-1 (C-1) also a 325 MWe PWR, was commissioned in 2000. Nuclear power safety with reference to severe accidents should be treated as a global issue and therefore the developed countries should include the people of developing countries in nuclear power industry's various severe accidents based research and development programs. The implementation of this idea may also deliver few other useful and mutually beneficial byproducts.

1. INTRODUCTION

Pakistan and China have been co-operating for a long time in the area of peaceful uses of nuclear energy. China National Nuclear Corporation (CNNC) has already constructed a 325 MWe Pressurized Water Reactor (PWR) for Pakistan. The plant, Chashma Nuclear Power Plant Unit-l or C-1, is located at Chashma site. C-1, which was commissioned in 2000, has now been in operation for about four years.

Prior to this, there was only one nuclear power plant operating in Pakistan, the 137 MWe KANUPP, a CANDU type Pressurized Heavy Water Reactor, supplied by Canada. For almost all of its design life, KANUPP received no vendor support because technical assistance in the area of nuclear technology was denied to Pakistan. However, KANUPP was operated safely for 30 years by PAEC. Upon completion of its design life in 2002, KANUPP carried out many safety retrofits and has been allowed by PNRA to operate at a reduced power – pending additional retrofits required to complete the licensing process for life extension.

To carry out the licensing process of the first unit, Chinese nuclear regulatory authority, National Nuclear Safety Administration (NNSA), cooperated with the nuclear licensing authority of Pakistan which was called the Directorate of Nuclear Safety and Radiation Protection (DNSRP). At that time, DNSRP operated under the auspices of Pakistan Atomic Energy Commission which owned the plant. NNSA provided the necessary consultancy.

The site for C-1 was selected in the early seventies. The first site report, completed in 1984, was for a 900 MWe NPP of French design. However, after signing the contract for the 325 MWe NPP with CNNC, a new site evaluation report was prepared. Both site reports were prepared as per the format and content of Chapter 2 of the PSAR according to the USNRC RG 1.70 (Rev. 3). The safety review was carried out according to the Agency's standards and the USNRC Standard Review Plan (NUREG-800).

The C-1 PSAR, prepared with the assistance of the Chinese vendor and designer, was submitted in 1992. Again, the format and content of USNRC RG 1.70 (Rev. 3) were followed due to unavailability of such guidance from IAEA. During the early period of negotiations for the supply of C-1, the regulatory requirements in China were based on IAEA 50 C-D-Rev. 0, whereas in Pakistan the regulatory requirements were based on the later version, Rev. 1. The Chinese vendor maintained that sufficient guidelines were not available to fully implement the Agency’s requirements. However, after several discussions, consensus was reached on an acceptable solution regarding application of the revision in C-1 design.

The safety analysis reports were reviewed primarily by using the NUREG-800. However, during the review process, DNSRP supported by NNSA and the Chinese designer ensured that IAEA Guide’s intent was generally fulfilled by the design and the safety analysis reports. The industrial standards used were mostly of US origin but Chinese standards were also used. The basic design was also reviewed by an expert team organized by IAEA. The review result was positive.

Since the Chashma site was planned and developed for twin units, some structures and buildings were built to accommodate two units. When the work on C-1 started, negotiations were held between CNNC and PAEC for the second unit, C-2. However, no deal could materialize at that time because the economic situation in Pakistan did not allow another major investment in nuclear power without knowing the final outcome of the first exercise. The successful commissioning and operation of C-1 paved the way for the second unit. Dialogue was reinitiated with CNNC who showed their willingness to supply the second unit but similar in design to the first unit.

In the mean time, an independent licensing body, Pakistan Nuclear Regulatory Authority (PNRA) came into existence in Pakistan. The regulation issued by PNRA in 2002 on the design of NPPs, is based on the IAEA NS-R-1 document. The IAEA standard was relatively new and more demanding, and the lower tier guidelines did not provide adequate information necessary for fully implementing the requirements of these standard. By that time NNSA had not adopted the new IAEA standard. Because of the uncertainties in the implementation of the regulation, the Chinese designer and vendor found it difficult to enter into a commercial contract guaranteeing licensability on the basis of the regulation. This led to difficulties in finalizing the technical details around which the contract could be written. An effort was made to invite PNRA to the technical negotiations with the vendor, but PNRA found it difficult to get involved as no formal submission had been made to them.

The C-2 Contract was ultimately signed by making such design changes in the C-1 design as would meet the intent of the new national regulation according to the understanding of the vendor and the buyer. The practical experience of full compliance to the regulation does not exist in China, Pakistan or elsewhere. Therefore, the licensing process of C-2 will be a challenging exercise and an example of the application of the new IAEA standards based regulation to the existing design.

C-2 design is upgraded on the basis of Qinshan-1 and C-1 experience/feed back. Based on international experience, engineering judgment; analysis of a selected set of severe accident sequences was conducted. On the basis of this analysis, the measures were incorporated in C-2 design in the form of enhanced sefety features. Like many other nuclear power plants of the world, the challenging job of in-depth analysis to verify the adequacy of measures against severe accidents is yet to be done. Furthermore, severe accident management guidelines (SAMGs) to control and mitigate the consequences of severe accidents are also yet to be drafted, analyzed, verified and validated.

2. ENHANCED SAFETY FEATURES OF C-2 DESIGN

The approach adopted by CNNC and PAEC was to take C-1 as a reference plant, review the design, and make modifications based on the new requirements. As a result of a two-year effort, it was agreed between the two parties that the C-2 design would include the following:

(1) Incorporation of more than 170 design changes on the basis of feed back from Qinshan-1 and C-1;
(2) Installation of Loose Part Monitoring System in the Primary circuit;
(3) Use of Probabilistic Safety Analysis to check and balance the plant design with respect to safety;
(4) Severe accident analysis leading to the design of preventive and mitigation measures and preparation of Severe Accident Management Guidelines (SAMGs);
(5) Specific measures for the prevention and mitigation of a selected set of Severe Accident sequences based on international experience, engineering judgment and the result of the above analysis. Typically, the list would include the following:
    (a) Large Break LOCA with High Pressure Injection/Low Pressure Injection failure (during safety injection phase)
    (b) Large Break LOCA with High Pressure Injection/Low Pressure Injection failure (during safety injection recirculation phase)
    (c) Small Break LOCA with High Pressure Injection/Low Pressure Injection failure (during safety injection phase)
    (d) Small Break LOCA with High Pressure Injection/Low Pressure Injection failure (during safety injection recirculation phase)
    (e) Loss of Off-site Power with Auxiliary Feed Water failure
    (f) Steam Generator Tube Rupture with High Pressure Injection and Feed Water failure
(6) Upgrading the Control Room to meet the requirements of human factor engineering;
(7) Safety System Bypass and Inoperable Status Indication System in Control Room; and
(8) Commitment of the designer to follow the principles of defense in depth and ALARA in the design.

Table 1. **Comparison of C-2 with other Two-Loop PWRs**

Plant Characteristic	**ANGRA-1**[1] **Brazil**	**Calvert Cliffs**[2] **USA**	**Qinshan-1**[3] **China**	**C-2**[4] **Pakistan**
Power (MW_{th})	1876	2570	966	998
No. of Steam Generators	2	2	2	2
Pressurizer Volume (m^3)	28.3	42.5	35.0	35.0
Containment Type	Reinforced Concrete	Reinforced Concrete	Pre-Stressed Concrete	Pre-Stressed Concrete
Containment Volume (m^3)	39,600	56,633	49000	49000
Containment Volume/Power	21.1	22.0	50.7	49.0
Containment Design Pressure (psig)	41.4 (2.8 bar)	50 (3.4 bar)	38.22 (2.6 bar)	38.22 (2.6 bar)
Containment Design Pressure/Power	0.0221	0.0195	0.039	0.038
Containment Basemat Thickness (m)	-	3.048	3.0	5.4

Table 2. **Comparison of C-2 Accident Mitigation Countermeasures with those in other PWRs with Large Dry Containment**

Plant Characteristic	**Surry**[5] **USA**	**Robinson**[5] **USA**	**Beznau**[5] **Switzerland**	**Biblis-B**[5] **Germany**	**Tsuruga-2**[5] **Japan**	**C-2**[4] **Pakistan**	**Remarks**
Power (MW_{th})	2441	2300	1130	3750	3441	998	-
Containment type	Reinforced Concrete	Reinforced Concrete	Steel	Steel	Pre-stressed Concrete	Pre-Stressed Concrete	Indicates load capacity, leak tightness
Containment Volume (m^3)	51000	59400	36400	70000	73300	49000	Containment loads, Time scale of accident time budget for Accident Mitigation
Containment Volume / Power	20.83	25.64	32.25	18.51	21.73	49.10	-
Fuel mass (kg)	79000	78900	43500	-	89500	40746	-
Containment Volume / fuel mass	0.65	0.75	0.84	-	0.82	1.20	Indicates Containment loads from Direct Containment Heating (DCH)
Zirconium mass(kg)	16500	16335	12000	29750	19500	10767	-
H_2-mass (kg), with 100% Zr-oxidation	780	718	530	1350	855	472	$Zr+H_2O= ZrO_2+2H_2+Heat$
Average H_2-concentration (%) at 30^0C, dry, 100% Zr-oxidation	15.0	12.1	12.8	19.0	12.4	8.1	Potential for H_2 burn, Containment loads from H_2 burn. Density of hydrogen calculated at design pressure of containment i.e. 2.6 bar
Estimated pressure (bar) due to H_2 burn, 100% Zr-oxidation	9.4	7.6	7.9	11.7	6.7	4.7	Heat of reaction (18°C) = 241750 kJ/kmol K[6]
Containment spray	Yes	Yes	Yes	No	Yes	Yes	Yes

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Hydrogen control	-	-	Containment venting	Igniters/ Recombiner	Combination of recombiners and inertisation of atmosphere	Passive auto-catalytic recombiners and/or igniters	Reduces potential for hydrogen burn
Additional water injection to inside of containment	-	-	External from the fire truck: *backup water source for containment spray *Flooding of containment External from river for cooling of fan coolers	-	Under preparation: water injection from RWST and fire water system	*Provision of connection to outer sources at the suction of containment spray pump *Reactor cavity water injection system for cooling of RPV lower head under severe accident	External vessel cooling for core debris and late containment failure by over-pressurization
Depressurization of RCS for prevention of High Pressure Melt Ejection (HPME), ("primary side bleed")	Transient with loss of all FW	Transients with loss of all FW	Transients SLOCA with loss of all FW	Most events with loss of all FW	Most events with loss of all FW	Most events with loss of all FW with the help of POV/MTV	Reduces chances of DCH.
Estimated containment failure pressure (bar)	9.7	10.4	7.8	8.0	-	6.5	For C-2, it is calculated as 2.5 times the design pressure.
Containment failure pressure/ Power	0.6467	0.8595	0.6094	0.4211	-	0.8025	-
Use of Primary Bleed/Feed in the event of SGTR	-	-	Yes	-	Yes	Yes, by Pilot Operated Valves (POV) /Motor Throttle Valves (MTV)	To reduce release attending SGTR
Filling of SG with water in the event of SGTR	-	-	Yes	Under study	-	Under study	To reduce release attending SGTR

3. A RELATIVE EVALUATION OF THESE ENHANCED SAFETY ASPECTS

The design of C-2 is in essence based on C-1, which was based on Qinshan-1. The design philosophy of Qinshan-1 was similar to the Westinghouse type plants and the safe operation of Qinshan-1 for more than ten years has demonstrated the safety and reliability of the Chinese design. Thus, the vendor has gained sufficient experience on a design which was robust to begin with. Because of the successful Q-1 and C-1 experiences, and because of the safety enhancement features added to the C-1 design which already has large safety margins built into it, the design of C-2 will now be as safe as or even better than most of the PWR type nuclear power plants operating in the world.

Table-1 compares some inherent safety features of C-2 with some of the older two-loop PWRs which are still in operation. The enhanced system capacities (containment volume, basemat thickness) tend to increase the safety margins allowing the plant a better chance to cope with accidents. Use of PSA to balance the plant design will further result in the enhancement of plant safety. Sufficient PSA experience already exists with the PAEC because full power Level-1 PSA has already been carried out for its Karachi Nuclear Power Plant and that of C-1 is almost ready. A Pre-IPSART mission is likely to be invited next year to review the C-1 PSA.

In the area of Severe Accidents (SA), emphasis has been given to all three aspects, i.e., prevention and detection of accidents and mitigation of accident consequences.

The following measures will be taken in C-2 to prevent severe accident:

(1) Motor Throttle Valve (MTV) will be used in addition to power operated valves and safety valves of the Pressurizer during normal and abnormal conditions. In addition, this valve will also be used to depressurize the primary system to allow emergency coolant injection to the primary system in case of accident;
(2) The anti-dilution mechanism or interlocks will be implemented during conditions of reactor cold shutdown and Reactor Coolant Pumps tripped; and
(3) In addition to two Emergency Diesel Generators on independent trains, one diverse Non-1E diesel generator will be provided to withstand the events of Station Black-Out (SBO) which may lead to the seal LOCA conditions

To increase the accident detection capability, the following specific measures will be taken:

(1) Provision of wide range hydrogen concentration monitoring system
(2) Provision of instruments with their limiting capability to meet the SA environment

Several steps will be taken to mitigate the consequence of severe accident and to reduce the challenge to the containment integrity. These include:

(1) Primary System depressurization with Motor Throttle Valve to prevent high pressure melt ejection
(2) Reactor Cavity Cooling Water Injection system
(3) Passive Hydrogen Recombination Facilities

Containment is the final barrier to the release of radioactivity to the environment. Concerted efforts will be made to prevent this release by strengthening the containment boundary including the penetrations, namely:

- Equipments hatch
- Personnel airlock
- Fuel transfer compartment
- Process penetrations

- Electric penetrations
- Isolation valves inside the containment
- Sleeve of gate valves of containment recirculation sump
- Ventilation valves of the containment
- Isolation valves of fire protection for the containment

Event sequences that may result in a containment by-pass will also been taken care of by:

(1) Quick primary depressurization with Motor Throttle Valve in case of Steam Generator Tube Rupture; and
(2) Increase in the design pressure of Residual Heat Removal System piping.

In addition, PAEC and CNNC have entered into a contract to jointly develop the severe accident management guidelines.

Table-2 compares plant/containment design characteristics and provisions for accident detection/prevention/mitigation measures in C-2 with some of the operating PWRs with Large Dry Containment.

However, the effort to keep the plant safe is a continuing process and does not end with a safe design. PAEC, as the owner of the plant, is committed to take measures to ensure safe operation of the plant. Such measures would include personnel training, development of a safety culture and an effective surveillance program. A continuing process of dialogue between the regulators and the utility has also been set up and this will greatly help in the accomplishment of the safety goals.

4. VARIOUS CHALLENGES INVOLVED TO PROVE THE ADEQUACY OF MEASURES AGAINST SA AND TO WRITE SAMGs AND ONE POSSIBLE SOLUTION

Pakistan is not a vendor country nor does it have much experience with the design of nuclear power plants. Besides C-1, which has been operating for about 4 years, PAEC's nuclear experience in the field of managing nuclear power plants over the last 30 years has been limited to essentially one NPP which is basically a first generation PHWR – the Karachi Nuclear Power Plant. When this plant was designed and built forty years ago, there were very few safety requirements and the relevant experties (both at the national and the at the international level) with refernce to analyze and combat the severe accident scenarios did exist.

TMI and Chernobyl's severe accidents leave a strong message that nuclear power safety with reference to severe accidents is a global issue and older safety believes, concepts and practices should be replaced with some more safe and secure safety culture. Therefore, now we should believe that nuclear power safety with reference to severe accidents is no more regional issue and people must join hands to save the world from the threats and consequences of severe accidents. The abandoning of hydrocarbons with consistent price hike has given birth to the revival of nuclear energy. Nuclear energy is thought to be the one of the best alternative sources of energy for underdeveloped countries like Pakistan having limited number of other options.

Nuclear power industry's various severe accident based phenomena's like hydrogen production/combustion rate are yet to be completely known. Therefore, a serious challenge does exist to confirm the adequacy of various encountering measures against severe accidents. A live and ittirative research/technical collaborations among the nuclear power producing countries can provide an appropriate solution to such challenges.

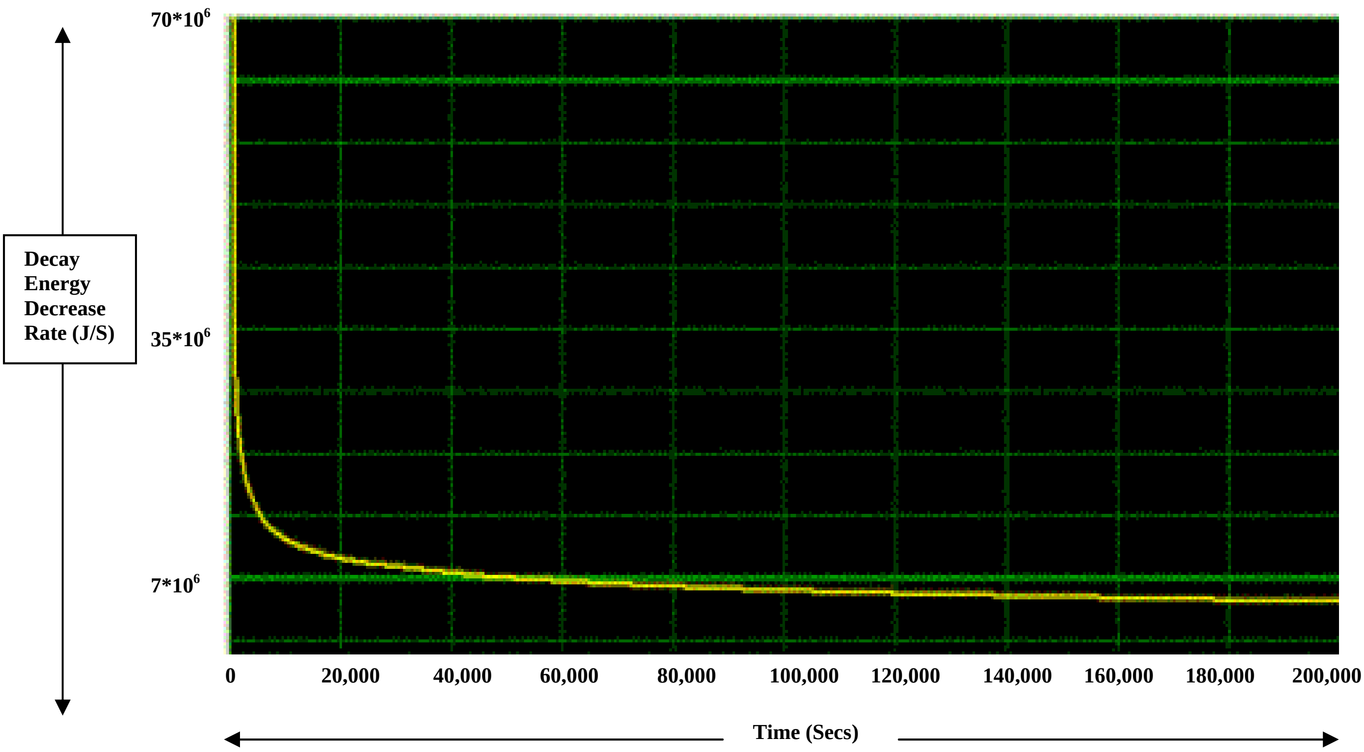
For example, reactor containment is consider as the ultimate safety barrier or last radioactive confinement option. Therefore, we would like to check and confirm the adequacy of containment heat removal system to save the reactor containment in case of severe accident.

Assuming a worst severe accident scenario; when significant core meltdown occurs, shutdown rods shut the reactor under gravity, a complete external/internal power failure makes the ECCS and heat exchangers unavailable for a postulated period of $2*10^5$ Secs. We supposed to ensure the in-vessel retention of molten fuel through removing the plant decay heat inside containment through borated water feed and bleed using the water of accumulators (35.4 m^3 each) and refueling water storage tank (RWST, 1995 m^3) under gravity (C-2 RWST design heat load is $4.675*10^6$ Joules/Sec). We Suppose to continue with this until ECCS and heat exchangers become available (after $2*10^5$ Secs). Proceeding sections contain the conservative based rough estimate of total energy added into the C-2 containment.

As C-2 core is about 1000 MWt; therefore in-vessel retention is the most significant and effective severe accident controlling and consequences mitigating measure/job required to be completed appropriately as well as adequately. C-2 containment free volume is about 49,000 m^3 and design pressure is 2.6 bar ($2.6*10^5$ pascals per square meter). Therefore, up to (maximum) $12.74*10^9$ joules energy can be safely stored inside the C-2 containment.

Decay heat is the heat produced by the decay of radioactive fission products after a nuclear reactor has been shut down. Decay heat is the principal reason of safety concern in Light Water Reactors (LWR). It is the source of 60% of radioactive release risk worldwide. The amount of decay heat that will be present in the reactor immediately following shutdown will be roughly 7% of the power level that the reactor operated at prior shutdown. Decay heat exponentially decreases and it may decrease to about 2% of the pre-shutdown power level within the first hour after shutdown and to 1% within the first day. Failing to cool the reactor after shutdown results in core heatup and possibly core meltdown (i.e. Three Mile Island 2). Emergency Core Cooling Systems (ECCS) and heat exchangers achieve decay heat removal. These highly redundant systems are designed to provide sufficient amount of makeup water for several days without operator intervention

Immediately after shutdown (at time = 0 sec), 7% of the total 1000 MWt was present in the C-2 core in form of decay heat, and then it exponentially decreases as per following graph (Ref-8).



Initial (at time 0 sec) energy addition rate (power) into C-2 containment = $70*10^6$ Joules per sec (70 MW); Then it exponentially decreases as per above graph upto $2*10^5$ Secs. The conservative based total energy added into containment (the area under curve) = $1400*10^9$ Joules. This mean containment heat removal system with design capacity ≥ $7*10^6$ joules/sec or in general ≥ 0.7 % of the core thermal power will be adequate to save the reactor containment.

The significant thing is that without inclusion of hydrogen factor (hydrogen production/combustion rate) no one shall accept/validate this calculation. Worldwide research is yet underway to completely understand the hydrogen factor/phenomena and the severe accident analysis tools are also under-development and validation stage. Therefore serious challenges do exist to verify the adequacy of severe accidents encountering measures like the in-vessel retention in C-2 case. Similar challenges do exist to draft, verify and validate C-2 severe accidents management guidelines (SAMGs) or the execution procedures with reference to conrol and mitigate the consequences of various probable severe accidents. This is one example from C-2 and nuclear power industry might have many other similar nature unsolved problems of the newly but fast emerging severe accident field.

Assuming nuclear power safety with reference to severe accidents is no more regional issue, then an active international collaboration or establishment of a joint research forum having clear mandate aimed at making and promoting the more safer and secure nuclear power can provide appropriate solutions to such unresolved problems. The various challenges involved in severe accidents analysis, codes preparation/validation and management guidelines drafting, verification and validation can also be addressed/resolved through such collaborations or joint platform. Such collaborations or joint research forum may also deliver few other useful and mutually beneficial byproducts.

5. CONCLUSION

In conclusion, it must be said that the new national regulations lay the foundation of a useful exercise to introduce and promote a new safety culture. This new safety culure may significantly improve the safety of existing and future nuclear power plants of the country. However, the application of this new safety culture does involve some serious challenges. Conceding that C-2 is not based on advanced design concepts, PAEC, through collaboration with the vendor/designer or international forums, is willing to put her best aimed at enhancing the plant safety. Nuclear power safety with reference to severe accidents should be treated as a global issue, therefore, it will be pertinent to give consideration to establish technical collaborations among the nuclear power industry itself and the entire nuclear power producing countries for the promotion of more safer and secure nuclear power, espacially with reference to design and evaluate effective severe accidents encountering measures. The possible implementation of this idea is the inclusion of people from underdeveloped countries in various ongoing and futurus severe accidents based research and development programs of developed countries like severe accident codes preparation/validation, severe accident analysis, and management guidelines preparation. This may also deliver few useful and mutually beneficial byproducts.

REFERENCES

[1] A PRACTICAL APPROACH TO APPLYING THE NEW IAEA SAFETY STANDARDS BASED NATIONAL LICENSING REQUIREMENTS TO CHASHMA NPP UNIT-2, By Ansar Parvez, Beijing, China, 2004.

[2] ANGRA NUCLEAR POWER STATION, Preliminary Facility Description and Safety Analysis Report, Furnas Centrais Electricas S.A.

[3] CALVERT CLIFFS NPP, Final Safety Analysis Report, Baltimore Gas and Electric Company

[4] SNERDI Reference Documents

[5] CHASHMA NUCLEAR POWER PROJECT UNIT-2, Technical Description, China National Nuclear Corporation

[6] ORGANIZATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT, Level 2 PSA Methodology and Severe Accident Management, OCDE/GD (97) 198, Paris.

[7] Walter J Moor, Basic Physical Chemistry, Prentice Hall of Pvt. Limited, 1989

[8] Internet search engines like “google.com”, http://decay-heat.tripod.com etc.