Safety Assessment and Improvements in Indian Nuclear Power Plants

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Abstract. Nuclear Power Corporation of India Limited (NPCIL) is the organization responsible for site selection, design, construction, commissioning, operation, maintenance, plant life extension and decommissioning activities of nuclear power plants in India. Presently, NPCIL has 17 Nuclear Power Plants (NPPs) in operation with a total installed capacity of 4020 MWe and is constructing 5 NPPs, which will add 2660 MWe. With rich experience in all relevant fields of nuclear power generation and to meet the emerging regulatory requirements, safety of Indian NPPs is continuously being reviewed and upgraded. The paper details some of the safety improvements and activities for the existing NPPs in India like life extension and safety upgradation of BWR TAPS-1&2, retrofitting of components and Upgradation for old PHWR units and accident management.

1. INTRODUCTION

Nuclear Power Corporation of India Limited (NPCIL) is responsible for site selection, design, construction, commissioning, operation, maintenance, plant life extension and decommissioning of nuclear power plants in India. NPCIL’s mission is to develop nuclear power technology and produce nuclear power as a safe, environmentally benign and economically viable source of electric energy to meet the electricity needs of the country.

Presently, NPCIL has 17 Nuclear power plants (NPPs) in operation with a total installed capacity of 4020 MWe and is constructing 5 NPPs with a total capacity of 2660 MWe.

With NPCIL’s tenet to continuously improve performance of country’s nuclear power plants without compromising safety; the safety assessment and improvements are primarily driven by:

- Evolution of technology
- Feedback from experience, both indigenous and international
- New research findings
- Evolving regulatory requirements

Regulatory Body in India, Atomic Energy Regulatory Board (AERB) follows a multi tier review and assessment for different stages of licensing. For operating NPPs, the Unit Safety Committee (USC), the Safety Review Committee for Operating Plants (SARCOP) and the Board of AERB constitute the multi tier review organs for regulatory control. The USC constituted for every station or a group of stations having NPPs built to the same safety standards, assists SARCOP in the review and assessment function to ensure comprehensive safety review on a regular basis. The decisions of these committees concerning major policy issues and important authorisations require endorsement of the governing Board of AERB. NPCIL also has internal two tier review system, first at the plant site, Station Operation Review Committee (SORC) and second at the head quarters in Safety Review Committee (SRC). Moreover periodic safety report review and operating experience review along with regulatory review give the input
for up gradation and safety improvements. This paper brings out certain safety improvements carried out in older plants.

2. LIFE EXTENSION AND SAFETY UP GRADATION OF BWR TAPS-1&2

Tarapur Atomic Power Station-1&2 (TAPS-1&2) is a twin-unit BWR plant with an installed capacity of 2 x 210 MWe. Commissioned in 1969, this was the first nuclear power station in the Indian sub-continent and is one of the longest serving Boiling Water Reactor (BWR) plant in the world. A comprehensive review of the plant including station-operating performance, ageing assessment & management, design basis & safety analysis and structural integrity studies, after 35 years of operation using relevant current safety standards and practices concluded that the physical condition of the Systems, Structures and Components (SSCs) permits continued operation for several more years. This indeth review identified certain activities to be completed for common safe operation. After successful completion of all the identified activities, the units were restarted in February -2006 after obtaining AERB authorisation for operation of the units for next five years.

2.1. Safety Assessment for License Renewal

2.1.1. Operation Performance Review

During the long operation period in order to improve operating performance, a number of engineering modifications have been incorporated in TAPS-1&2. All these modifications are taken up based on operating experience both domestic and global. Among there some of the significant ones are: [1]

i) Isolation of secondary side of Secondary Steam Generators (SSGs). This was done as tube failures in SSGs were causing operational difficulties and were resulting in major dose consumption for tube plugging and repairs.

ii) Replacement of primary system piping with nuclear grade material to mitigate generic issue of Inter Granular Stress Corrosion Cracking (IGSCC).

iii) Replacement of tube bundles in emergency condenser

iv) Installation of additional air compressor

v) Augmentation of spent fuel storage capacity

vi) Up gradation of station batteries to enhance reliability of Class-I power supply

vii) Installation of a full capacity Station Blackout Diesel Generator

viii) Improvements in fire protection system (e.g. cable segregation of safety related system, coating of critical cables with fire retardant material, installation of fire barriers).

2.1.2. Ageing Management

Among the major critical components, Reactor Pressure Vessel (RPV) being the main component is covered by an exhaustive material surveillance programme. The assessment indicates that vessel material condition is satisfactory and has adequate fracture toughness. The material is fit enough for operation well beyond 60 Effective Full Power Years (EFPYs), as against the design life of 40 EFPYs, whereas these units have seen around 20 EFPYs so far. The good water chemistry ensures negligible deterioration of material of RPV and internals. Additionally, fatigue analysis for RPV has been done. For this analysis the number of pressure and temperature transients seen by the vessel and anticipated cycles for the next 40 years were considered. Examination of containment and civil structures has concluded that these structures are not degraded in any significant manner and match reactors capability for future long-term operation.
Important systems (engineered safe guard systems and other important support systems) are in good condition. The piping, valves, pumps and instruments are inspected, conditioning monitoring done and actions are taken as required. Replacement of equipment is done on the basis of condition monitoring; Salt Service Water pumps, Emergency condenser tube bundles are such examples.

Other critical components like Reactor Relief Valves, Safety Valves, Secondary Steam Generators, Primary Steam Isolation Valves etc. are checked/inspected periodically as per defined programme and are maintained in good condition. Safety related cables are visually checked and their Insulation Resistance measured. A set of power and control cables have been subjected to Residual Life Assessment (RLA) and actions are taken on the basis their findings.

The ageing studies indicate that the main component, the RPV in both the units in good condition to operate the station for many more years. Containment and civil structures condition match reactors capability for future long-term operation. A systematic ageing management program is in place to monitor any degradation in SSCs and to take corrective actions.

2.1.3. Design Basis

The main objective of this review was to compare design basis of systems with respect to the current safety practices and identify and prioritize safety issues and finally to identify and implement corrective/compensatory measures where necessary.

A number of recommendations are made as an outcome of this review to bring TAPS safety provisions at part with the current safety requirements. The salient recommendations include:

i) Up gradation of 3x50 % Emergency Diesel Generators (EDGs) by 3x100% EDGs.

ii) Unit wise segregation of shutdown cooling system and de-linking it from fuel pool cooling system

iii) Segregation of Class-III, II and I power supplies into two zones with a physical barrier and redistribution of supply to redundant loads from separate buses.

iv) The existing shared shutdown cooling system was upgraded to make it independent for each unit by installing additional pump and heat exchangers. The fuel pool cooling system has also been made independent of shutdown cooling system.

v) Rerouting of cable through diverse routes for redundant loads.

vi) Augmentation in fire protection system

vii) Up gradation of electrical power supplies to important loads

viii) Provision of supplementary control room

ix) Installation of strong motion seismic instrument

3. RETROFITTING OF COMPONENTS AND UPGRADATION FOR OLD PHWR UNITS:

Rajasthan Atomic Power Station (RAPS) and Madras Atomic Power Station (MAPS) are the earlier generation Pressurized Heavy Water Reactor (PHWR) in India. The design philosophy of later PHWRs has undergone continuous improvements. While it is not possible to implement all the new concepts, certain modifications and retrofitting are incorporated in these old units to bring their safety standards close to that of new plants.[1] These include:

- Retrofitting of high pressure ECCS,
- Incorporation of supplementary control room,
- Replacement of thermal shield cooling coils,
- Segregation of power/control cables,
- Replacement of coolant tubes and feeders
• Provision of dedicated instrument air supply to safety related valves,
• Installation of Flood Diesel Generator in RAPS-1&2
• Installation of Emergency Diesel generator, Fire water pump and air compressors at higher elevation.

3.1. Retrofitting of high pressure ECCS

The existing Emergency Core cooling System (ECCS) in MAPS-1 consists of low pressure moderator water injection/recirculation into Primary Heat Transport (PHT) system in case of a loss of coolant accident (LOCA). This system could adequately take care of the large breaks in PHT system. However, for small/medium breaks, a longer time is expected for the PHT system to fall down to a value sufficient to enable the moderator injection. During this period, void formation may take place in the channel, leading to high fuel temperature. Hence, the retrofitting of the ECCS system has been carried out to mitigate this situation.

The retrofitted ECCS consists of high-pressure heavy water (D₂O) injection followed by modified low pressure long term moderator water injection /recirculation at low pressure. The retrofitting with high pressure heavy water injection and other modifications are carried out to strengthen the existing ECCS for minimizing the fuel failures during LOCA, covering the entire range of break sizes.

3.2. Provision of Supplementary Control Room (SCR)

Supplementary control room has been provided to ensure safe shutdown of the reactor in case of loss of inhabitability in main control room. It will also help in maintaining reactor in a safe shutdown state, ensuring core decay heat removal for extended period of time by monitoring essential plant parameters. Supplementary control panel has the following provisions.

a) Tripping of the reactor.
b) Opening of 2 nos. of the Atmospheric Steam Discharge Valves.(ASDV)
c) Monitoring of all essential plant parameters like PHT system pressure, temperature, boiler pressure, radiation field in boiler room, boiler room pressure and temperature, calandria level and calandria moderator outlet temperature.
d) Lamp indications for boiler level low/high conditions and reactor building isolation damper positions.
e) Recording of neutronic parameters, Log N and Log Rate N
f) A terminal of plant information system (PIS) to monitor various plant parameters.

SCR is designed in such a way that it is physically and electrically isolated from main control room. Except in the case of neutronic parameters, separate sensors are used for monitoring important process parameters and they are routed separately to SCR.

3.3. Replacement of Thermal Shield cooling Coils

Due to frequent leaks experienced in thermal shield cooling coils, the heat exchanger tubes have been replaced with that having 70% Cu and 30% Ni from 90% Cu and 10% Ni tubes. The ‘U’ bends of the tubes are made corrosion resistant by coating with tin.

3.4. Replacement of coolant tubes

In the earlier design of PHWRs used at RAPS-1&2, MAPS-1&2, NAPS-1&2 and KAPS-1 a zirconium alloy (Zircalloy 2) was used for coolant channels. It was considered the best available material at that
time. However, in pile experience brought out, the requirement of replacement of coolant tubes after 10 to 12 effective full power years in view of the modification of material characteristics especially the reduction in mechanical strength due to hydriding under radiation during service. The job of Enmass Coolant Channel Replacement (EMCCR) at RAPS-2, attempted first time with indigenous technology, was completed successfully in a record time and all the coolant channels were replaced by an improved coolant tube material – Zirconium - 2.5% Niobium. This was for the first time in a developing country and only the second time in the world that such a highly complex technical project was completed. The job was completed using indigenously developed remotely handled tools. Similar EMCCR work been accomplished at MAPS 1&2 & NAPS-1, is underway at NAPS-2 & KAPS-1.

3.5. Replacement of Feeders in the Primary Coolant system.

Flow assisted corrosion leading to thinning of the carbon steel feeder elbows at the outlet of reactor was noticed in some Canadian PHWRs in 1996. A study and assessment was carried out in-house to detect any thinning of feeder pipes in Indian PHWRs. However, significant thinning of feeders was not noticed in Indian PHWRs. The reduced degradation of feeders in Indian PHWRs was attributed to good chemical control and better operational practices in Indian reactors. For the earlier PHWRs RAPS-2 and MAPS-1&2, which had seen a long service life, a decision was taken to replace the feeders to extend their life. Hence at MAPS-1 the Enmasse Feeder Replacement (EMFR) was carried out along with the EMCCR work. This was for the first time in the world that feeders were replaced in a PHWR. EMFR has also been carried out at NAPS-1 & RAPS-2.

3.6. Segregation of control cables

In order to ensure intended performance of safety system even under accidental condition of major fire and also to minimize common mode failure, the existing control and instrumentation cable routes related to safety system on MAPS-2 have been reviewed. Following points are considered for segregation of cables.

a) Three separate channel wise cable trays are provided for routing of all triplicated instrument cables from RB to control equipment room.
b) All triplicated loads are fed by different source of power supplier. (For both AC and DC requirements)
c) All safety related equipments are grouped into two categories.
d) Pneumatic supply for safety related Pneumatic valves are fed from different air manifolds.

3.7. Provision of dedicated instrument air supply to safety related valves

Isolation of instrument air supply to RB becomes essential under the following two conditions.

1) When RB is boxed up in the event of LOCA, the bulk supply of compressed air to the equipment located in the RB is required to be isolated for preventing the continued pressure rise in the containment, on account of air leakage and facilitate depressurization.
2) When all compressors are unavailable during SBO, caused due to extended Class-IV and Class-III failure, the available diesel driven HP compressor capacity is to be conserved for meeting the essential requirements.

Immediately after LOCA, the critical isolation dampers on lines penetrating the containment close automatically on containment isolation logics, in order to box up the activity inside RB. It is desirable to cool down and thereby depressurize the containment to limit ground level release from primary containment. Pressure is brought down gradually by unit coolers. However pressure inside the RB again
rises on account of in-leakages of instrument air. Subsequently RB depressurization system is required to
depressurize the containment, discharging via HEPA and charcoal filters through stack. Frequency of
such depressurization depends on the amount of instrument air in-leakages. To reduce this frequency and
thereby reducing activity release through stack and reducing load on charcoal filter in the depressurization
circuit, it is desirable to reduce instrument air in leakage.

Essential loads requiring instrument air supply after LOCA were identified. The dedicated instrument air
supply system consists of two separate tanks at 4.5 m³ volume always charged with instrument air supply
which is capable to cater to the need of essential loads inside the reactor building during emergency. The
incorporation of new scheme will ensure supply of dedicated instrument air to selected valves for retaining
them in their dedicated position

3.8. Installation of Flood Diesel Generator

The older PHWR units at RAPS and MAPS had fixed their safe grade elevation on certain basis. With
emerging regulatory requirements, it is seen that later plants at these locations have hyper grade
elevations. This is an area where no back fitting is possible. However to ensure continued power supply
for essential functions; one emergency diesel generator at these stations is located at elevation
responding to the later plants at these sites. In addition, in MAPS, one diesel operated fire water pump
and air compressor is also located at higher elevation

4. ACCIDENT MANAGEMENT

Indian PHWRs have event based Emergency Operating Procedures (EOPs) to handle accidents within the
design basis. These procedures are being complemented with the symptom based guidelines, which are
developed to maintain identified ‘safety functions’ - monitored continuously through the identified set of
plant parameters (symptoms) and maintained in the acceptable state.

Standardized Indian PHWR units have large inventory of light water in calandria vault surrounding
calandria vessel. Also unlike earlier PHWR units (RAPS 1&2, MAPS 1&2); these standardized units
always have heavy water moderator inside calandria. The inventories of heavy water moderator in
calandria and light water in calandria vault have an important role in delaying/restricting progression of
severe accidents in PHWRs. These large inventories of water are capable of removing decay heat from the
fuel for a long term after coolant cover over the fuel is lost. The Loss of Coolant Accident (LOCA) with
failure of Emergency Core Cooling System (ECCS) in PHWRs does not lead to failure of coolant
channels as long as moderator cooling is maintained. However, in this scenario, limited fuel damage takes
place and is termed as Limited Core Damage Accident (LCDA). However, together with LOCA and
ECCS failure; if failure of moderator cooling is also postulated then coolant channels cannot be held in
position in side calandria. With the boil off/escape of moderator from calandria, the top coolant channels
start disintegrating and falling on the remaining channels leading to failure of other coolant channels in
succession. Finally, the molten fuel and core debris collect at the calandria bottom and can be contained
there if light water inventory of the calandria vault is maintained. A failure in maintaining water cooling in
the calandria vault will lead to boil off this inventory of water as well leading to failure of calandria and
molten core concrete interaction with subsequent re-pressurization of the containment.

The severe accident progression as described above is without any intervention. As part of the Severe
Accident Management for IPHWRs, for which work is initiated, appropriate intervention in the form of
strategies will be identified to maintain/protect coolant channel integrity, calandria, calandria vault and
containment, which act as ‘barriers’ in severe accident progression. These strategies will
• Exploring full design capability of the plant including possible use of some systems (both safety and non-safety ones) beyond their originally intended function.
• Use of additional temporary system/ad-hoc arrangements to return the plant to a controlled safe state
• Getting support from other units in a multi unit station provided safe operation of the other units is not compromised.
• Identifying instruments that could aid handling of severe accident situations
• Severe Accidents Management Guidelines (SAMGs) are under development for Indian PHWRs. Following strategies are envisaged
  - Inject into PHT system
  - Maintain calandria heat sink
  - Maintain calandria vault heat sink
  - Control Reactor Building condition
  - Reduce Containment Pressure
  - Control Containment Atmosphere flammability and H₂
  - Mitigate fission product release

5. CLOSURE

In India mechanism of review of NPPs is well established, both in Utility and Regulatory body. Operating experience is one of the main factor of this review process. For older structure, special attention is paid to improve their safety standard in line with current regulatory requirements.

REFERENCES
