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Tarapur Atomic Power Station Units-1 and 2 Design features, operating experience and license renewal

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Abstract

Tarapur Atomic Power Station (TAPS), a twin-unit BWR station is the oldest NPP in India. These units are operating for the last 30 years with impressive operation performance indicators and in this period their operating license was periodically extended by the Atomic Energy Regulatory Board. Certain modifications and improvements have been made in these units based on operating experiences and to further improve the safety standards. Testing of surveillance samples of the reactor vessel material confirmed that the vessel could operate well beyond 60 Effective Full Power Years (EFPYs), whereas till 2003, these units have completed 18 EFPYs. In order to extend the operating license of these units a fresh comprehensive review involving review of plant operating performance, ageing management and design basis and safety analysis was carried out. These units were reviewed against the current safety practices and based on this review some recommendations are made to bring the safety levels of these vintage units on par with the recently authorized plants. This paper gives a general description and describes the operating experience of TAPS. Also, safety review of TAPS carried out for license renewal is covered.

1. Introduction

Tarapur Atomic Power Station (TAPS), located on the west coast of India about 100 km north of Mumbai, is a twin-unit BWR plant with an installed capacity of 2 × 210 MWe. Commissioned in 1969, this reactor plant was the first nuclear power station in the Indian sub-continent and is one of the longest serving BWR plants in the world.

TAPS was built by the International General Electric Co., USA on a turnkey basis. Equipment for the plant was manufactured by General Electric's Atomic Power Equipment Department and Combustion Engineering, USA, supplied the reactor vessels. The Bechtel Corporation was the engineer–constructor.

TAPS is a BWR-1 design plant with pre-mark-1 containment having suppression pool. The Nuclear Steam Supply System is of second generation BWRs. Among its sister plants of similar vintage built in 1960s, TAPS is the only serving power plant and its performance indicators continue to be comparable to world median on all counts. A recently conducted review of the plant including station-operating performance, safety analysis, ageing assessment and management and structural integrity studies shows that the physical condition of the station is good enough for continued operation for several more years.

This article on Tarapur Units-1 and 2 is covered in three sections. The first section gives a brief description of the general layout and design philosophy. The second section comprises the operating experience including the design changes and modifications done in the plant. This section also deals with performance to highlight the problems related to ageing as experienced at TAPS and the corrective actions and modifications undertaken to keep plant operation within the specifications of safety and regulatory requirements. The third section covers the highlights of a review of design basis and safety analysis carried out for license renewal of these units.

2. General description

2.1. Civil structures

In Tarapur Atomic Power Station, many facilities and services are common to both units, making for considerable compactness

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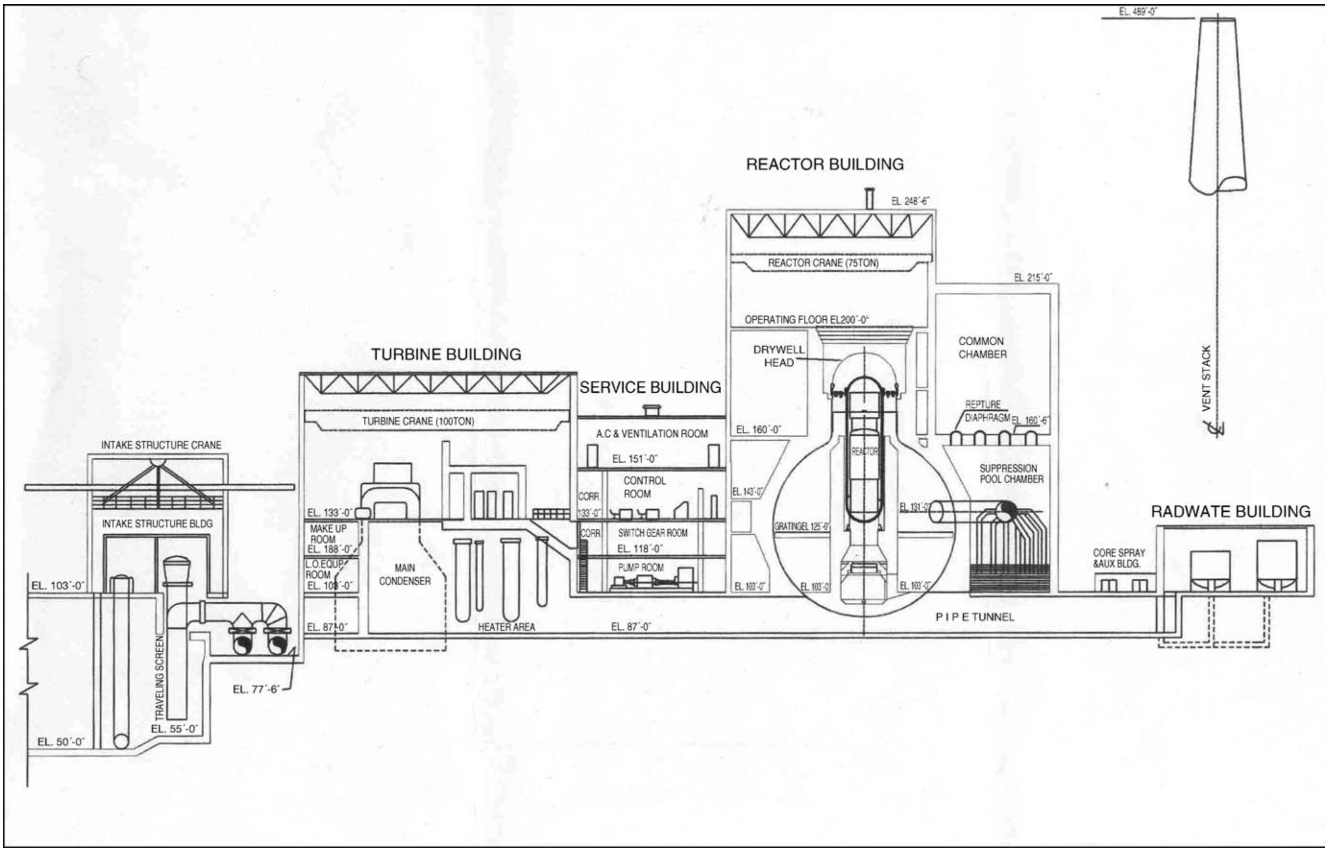

Fig. 1. Key plan longitudinal section.

and economy. The reinforced concrete seismically qualified civil structures housing the plant (Fig. 1), consist of a turbine building, reactor building with suppression chamber, service building, containment cooling building, radwaste processing building, intake pump house and stack. The control room, which is common to both the reactors, is located in the service building. Two separate drywells house each reactor within the common reactor building. Administrative building is located to the north of the service building.

Each reactor vessel (Fig. 2) and its recirculation system (Fig. 3) are contained inside the drywell of a pressure suppression primary containment system (Fig. 4). The primary containment system consists of a drywell, a suppression pool and a common chamber (Fig. 5). The reactor building, which is the secondary containment, encloses the drywell. The reactor building is designed as a controlled leakage structure.

2.2. Design features

The reactor is a forced circulation boiling water reactor producing steam for direct use in the steam turbine. The fuel consists of uranium dioxide pellets contained in Zircaloy-2 tubes. Water serves as both the moderator and coolant. The reactor design parameters are listed below.

Core design data	
Thermal output	660 MWt (dual cycle)/210 MWe
Re-rated thermal output (operating on single cycle since 1984 after isolation of secondary steam generators)	530 MWt (single cycle)/160 MWe
Operating pressure	6.89×10^6 Pa (1000 psig)
Total core flow	2897.78 kg/s (23×10^6 lbs/h)
Primary steam flow	261.11 kg/s
Fuel assemblies	
Number of fuel assemblies	284
Fuel rod array	6×6
Fuel rod pitch	0.018 m (0.701 in.)
Active fuel length	3.66 m (12 ft)
Fuel rod, cold	
Fuel material	UO_2 (pellets)
Average fuel enrichment	2.44% ^{235}U
Fuel clad	Zircaloy-2
Discharge burn-up/fuel bundle	21,600 MWd/t
Control system	
Number of movable control rods	69
Shape of control blade	Cruciform
Control rod pitch	0.25 m (10 in.)
Control rod poison material	B_4C granules in 0.479 cm (0.185 in.) o.d. $\times$ 0.0635 cm (0.025 in.) thick SS tubes

Reactor pressure vessel	
Reactor vessel inside diameter	3.66 m (12 ft)
Reactor vessel overall length	16.41 m (53 ft–10 in.)
Reactor recirculation system	
Number of loops	02
Pipe size	0.61 m (24 in. dia.)
Pump capacity	7404.26 m^3/h (32,600 gpm each)
Pump head (total)	50.29 m (165 ft)
Turbine	
Name plate rating	210,000 kW
Exhaust pressure	8465.97 Pa (2.5 in. Hg abs.) (1.23 psi)
Makeup	0.5%
Type	TC dual admission single flow
Last stage bucket length	0.89 m (35 in.)
Speed	1500 rpm

2.3. System description

High purity water enters from the bottom of the reactor core and flows upwards through the fuel element assemblies where boiling produces steam. Mechanical steam separators located within the reactor vessel separate the steam water mixture. The steam passes through the steam lines to the turbine. Seawater is used for condensing the steam in the condenser. An elaborate system prevails for monitoring and maintaining purity of the water that enters the reactor.

Reactor recirculation water flows from the reactor through two recirculation pumps to the tube side of two secondary steam generators and then to the reactor core. The secondary steam generators are kept isolated since 1984 due to tube leaks. The system operation on single cycle has since been proved to be more stable.

The bottom entry cruciform control rods provide Safety Control Rods Accelerated Movement (SCRAM) as well as reactor control function. Each drive has its own separate control and scram, devices. A control rod reactivity-limiting device called Rod Worth Minimizer (RWM) limits the reactivity addition in the reactor and assures that any reactivity addition is as per predetermined programme.

In addition to the control rod drive system, a stand by Liquid Poison System (LPS) is provided to manually initiate injection of a neutron poison (liquid sodium pentaborate solution) into the reactor to make the reactor sub-critical.

A passive decay heat removal system "The Emergency Condenser System" is provided for each reactor to remove the decay heat when the reactor is isolated from the main condenser. The system is initiated automatically.

One train of decay heat removal on long-term basis is common for both reactors. This consists of closed loop shutdown cooling system coupled with reactor building cooling water system and salt service water system. The sea serves as the ultimate heat sink.

An Emergency Core Cooling System (ECCS) (Fig. 6) is designed to pump water directly from the suppression pool into the reactor vessel and drywell under the postulated loss of coolant accident conditions. The system design is such that there is minimal fuel damage and containment structures are not challenged even in the worst accident scenario. This system has inbuilt redundancy to take care of any component failure.

2.3.1. Containment

The pressure suppression system (Figs. 4 and 5) is similar to that at Humboldt Bay 3. It consists of vertical ducts from a 20 m (66.6 ft) diameter drywell discharging into a suppression chamber pool. The Mark I (inverted light bulb and torus) pressure suppression systems came later.

The chamber volumes above the pool of each reactor are interconnected by rupture diaphragms to a common chamber, which helps to relieve the pressure build-up.

The Reactor Building (RB) housing both reactors provide secondary containment during reactor operation, when the drywells are closed and in service. When the drywell is open, as during refueling, the reactor building provides primary containment. The emergency ventilation and clean-up system is meant to filter and exhaust the RB atmosphere to the stack 110 m (360.9 ft) during secondary containment isolation conditions, with a minimum release of radioactive materials to the environs.

2.3.2. Safety systems

The ECCS (Fig. 6) has a sparger of spray nozzles above the reactor core, which under LOCA conditions sprays water 415.8 m^3/h (6930 l/min) over the fuel elements immediately after actuation by low water level or high drywell pressure to prevent fuel meltdown. Either one of the two core spray loops is sufficient to provide cooling to the core to prevent clad melting.

The relief valves of the automatic depressurization system enable the core spray system to provide protection against a small break, in the event of the feed water system becoming ineffective. The following signals are used in the protection system: core spray low water level, low reactor water level, high drywell pressure and low feed water flow. A timer set at 2 min is provided to allow the feed water system time to perform its function.

The core spray system was designed to deliver about 136.2 m^3/h (2270 l/min) in each loop, with a backpressure of 0.55 MPa (80 psi) in the reactor vessel. A sodium pentaborate solution can be injected for emergency control.

The reactor system has provisions for isolation from the turbine condenser in case of emergency, by high integrity, frequently tested isolation valves. Both primary steam lines of each reactor are individually monitored by two radiation monitors each to keep a continuous watch on the radiation levels. When radiation level increases to 10 times the normal value, the reactor scrams automatically and steam line isolation valves close. This prevents release of radioactivity to the environment.

In the event the reactor is isolated from the main condenser an emergency condenser is provided which removes the decay heat during post-reactor scram/shutdown period. The emergency condenser system is a passive system, which provides heat sink with capability to remove decay heat for a period of 8 h without makeup. The system consists of tank located at 60.96 m (200 ft) elevation in reactor building having two full capacity parallel condensing tube bundles connected to the primary system. Each

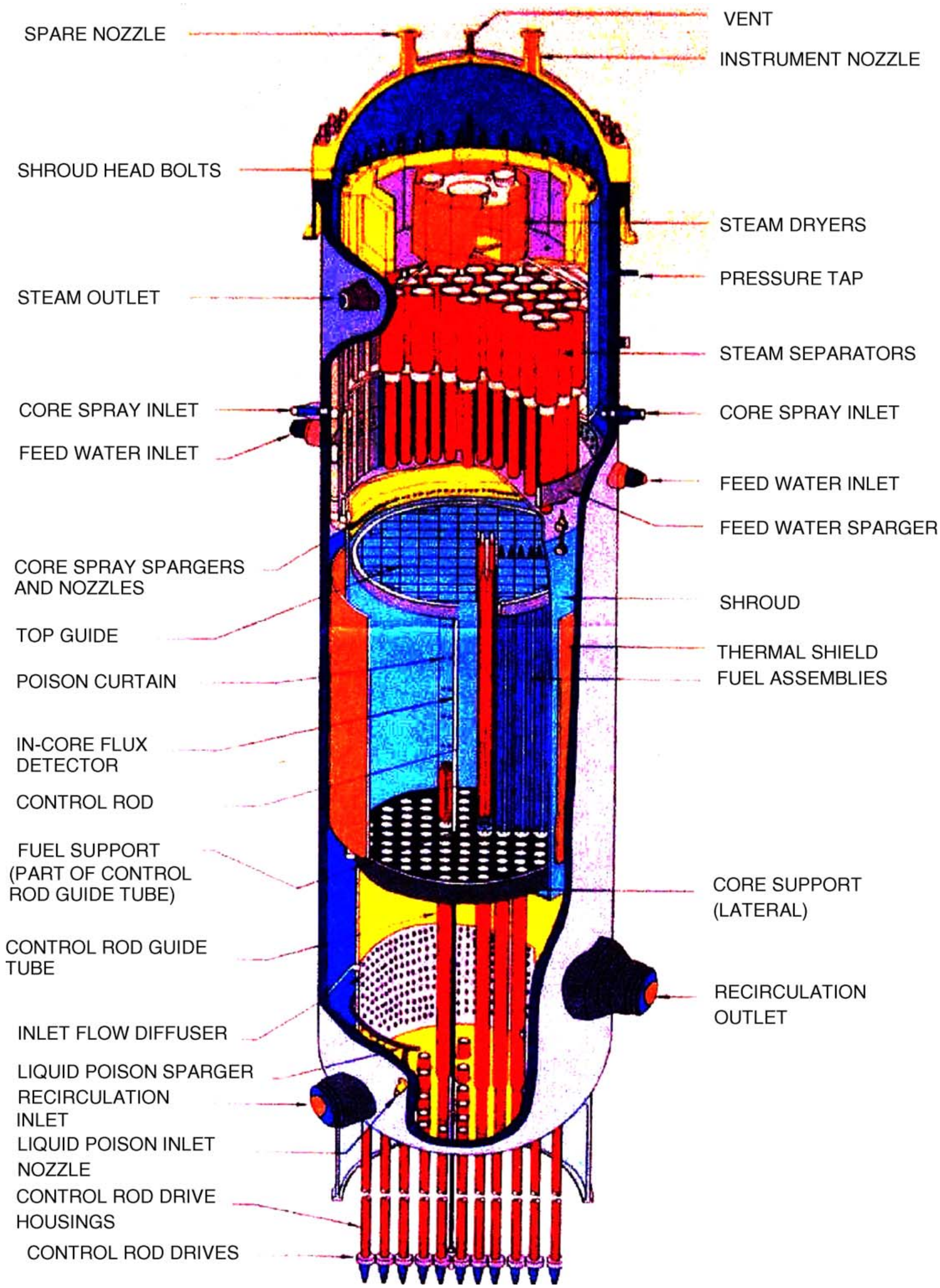

Fig. 2. TAPS BWR cross-section.

loop is having two inlets and one outlet Motor Operated Valves (MOV) and a common outlet MOV for both the loops.

The reactor vessel of TAPS has been built to house 368 fuel assemblies (FAs) but contains 284 fuel assemblies to deliver the rated power. The provision of additional volume helps in reducing the fluence intensity thereby adding the safety margin to the design life. As a result the core average power density is relatively low 32 kW/l as compared to 50 kW/l for current BWRs. Since the operating power is low with respect to the contemporary reactors, the containment volume to core power ratio is high which is $35\,m^3$/MWt ($1236.01\,ft^3$/MWt) against $13\,m^3$/MWt ($459.09\,ft^3$/MWt) of contemporary BWRs. In the event of accidents, these features would result in lower containment peak pressure, lower hydrogen concentration, and better pressure withstanding capability of containment and consequently would provide better accident management capability.

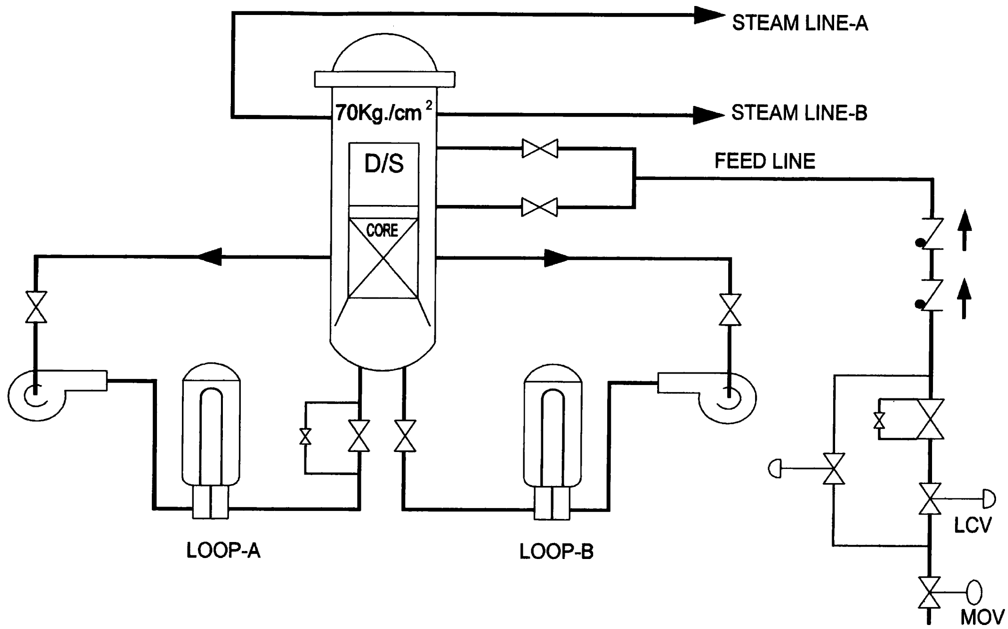

Fig. 3. Reactor vessel and recirculation loop arrangement.

2.3.3. *Instrumentation and control*

The normal BWR control methods (control rods and recirculation flow) are employed. The units are equipped with a Travelling Incore Probe (TIP) system.

2.4. *Electrical distribution systems*

The two power generating units of the station are arranged on a unit basis so that each is essentially independent and self-sufficient for normal operation. The electrical systems of each unit consist of a generator, main transformer, unit auxiliary transformer and auxiliary power system. Two start-up transformers are provided to supply part of the auxiliary loads. Power is generated at 12 kV and stepped up through the main power transformer to the transmission voltage of 230 kV. TAPS switchyard supplies power to Maharashtra and Gujarat state power systems through four 230 kV feeders. Unit auxiliary transformers step down auxiliary power of each unit from 12 to 3.3 kV. Start-up transformers provide start-up auxiliary power from 230 kV transmission systems by stepping down to 3.3 kV.

Station distribution system consists of medium voltage (3.3 kV) and low voltage (415 V) switchgears, 415 V motor control centers, 415 V power panels/lighting panels and 120 V control panels. Three emergency diesel generators are provided for supplying essential station auxiliaries in the event of a loss of electrical power to the station. These diesel generators are backed-up by a station black out diesel generator installed at a separate location. A 250 V DC system with two battery banks is provided to assure reliable DC control power supply.

3. Operating experience of TAPS-1 and 2

Commercial operation of TAPS began in the year 1969 and the station has successfully completed 33 years of operation till date. Some of the historical and operational data of the station are highlighted in Table 1. During the initial operations many teething problems were faced, which were critically reviewed and overcome by suitable design changes. In

Table 1
Salient features of the station

Salient dates	
Site selection	August 1960
Approval of site selection	August 1960
Global tenders issued	October 1960
Start of construction	May 1964
Unit #1 first critical on	February 1, 1969
Unit #2 first critical on	February 27, 1969
Unit #1 synchronized on	April 1, 1969
Unit #2 synchronized on	May 8, 1969
Commercial operation on	November 1, 1969
Dedicated to nation on	January 19, 1970
Unit #1 completed 2,00,000 online hours on	December 12, 2000
Unit #2 completed 2,00,000 online hours on	September 15, 2001
Completion of 33 years of commercial operation	November 1, 2002
Completion of 34 years of operation by Unit #1	April 1, 2003
Completion of 34 years of operation by Unit #2	May 8, 2003
Operational highlights	
Total station generation as on 31st March 2003	62.89 TWh
Longest continuous run of the Unit #1	269 days (1998–1999)
Longest continuous run of the Unit #2	209 days (1999–2000)
Number of refueling outages of Unit #1	17
Number of refueling outages of Unit #2	17
Shortest outage of Unit #1	20 days (in 2003)
Shortest outage of Unit #2	27 days (in 2002)
Station highest generation	2.54 TWh (2002–2003)
Station highest capacity factor	91% (2002–2003)
Unit #1 highest generation	1.30 TWh (1998–1999)
Unit #1 highest capacity factor	92.58% (1998–1999)
Unit #2 highest generation	1.31 TWh (2001–2002)
Unit #2 highest capacity factor	93.75% (2001–2002)

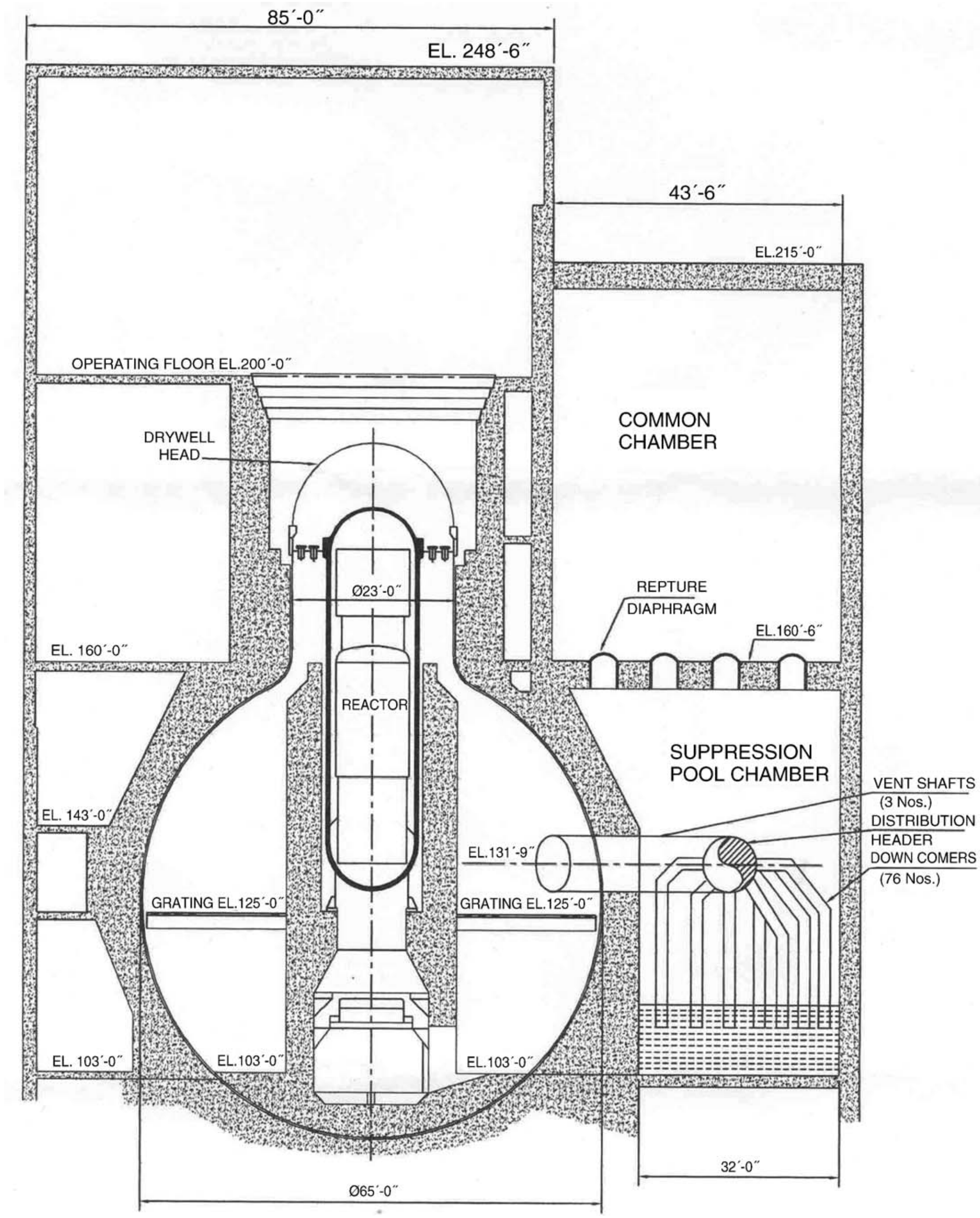

Fig. 4. Schematic sketch of TAPS-1 and 2 containment.

order to improve the station's performance several engineering modifications have been done and incorporated. In addition to these, station has adopted various inspections and testing methodology as per the latest ASME codes and standards. Some of the significant modifications to the system incorporated as a result of operating experiences are brought out in this section.

3.1. Secondary steam generators (SSG)

Initially, the station was operated on dual cycle but right from the initial stage, tube failures of secondary steam generators were experienced. This caused difficulty in operation and posed a major load on radioactive dose consumption for tube plugging and repairs. In the year 1985, both the SSGs secondary sides were isolated and primary side continued as part of recirculation system. The reactor thermal power was re-rated to 530 MWt and electrical power output to 160 MWe as a result of single direct cycle operation. Both reactors have been satisfactorily operating in this mode since then.

3.2. Intergranular stress corrosion cracking (IGSCC) failure on nuclear piping

According to the original design, reactor system piping at TAPS was made of SS 304 or SS 316. IGSCC failures in such

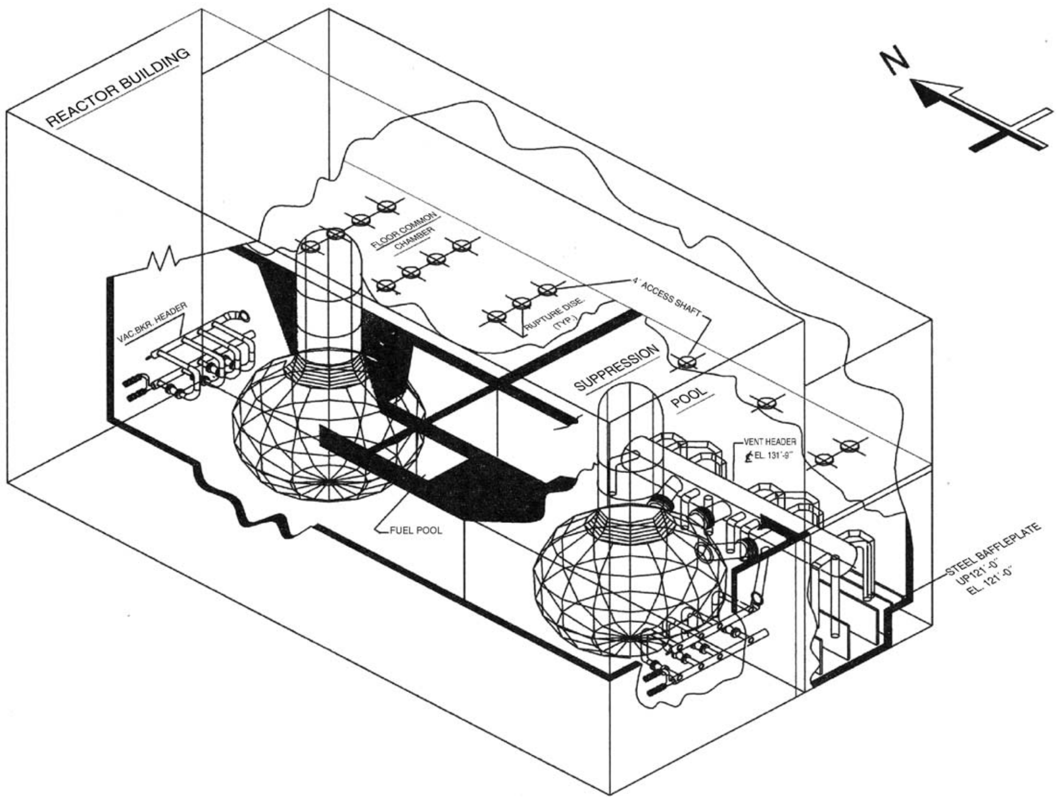

Fig. 5. Drywell, suppression pool and common chamber.

material were observed in overseas reactors. Based on this information the surveillance activities on stainless steel pipelines, especially the weld joints of reactor systems, were intensified. The first IGSCC failure at TAPS was witnessed on emergency condenser steam line weld joints. In subsequent years, cracks were detected on reactor recirculation discharge valve bypass piping, reactor recirculation equalizing piping, clean-up piping and poison system piping.

As a corrective measure, the affected piping were replaced with SS 316 LN (nuclear grade stainless steel) corrosion resistant material to mitigate/eliminate the generic problem of IGSCC. Condition monitoring of the critical areas is being followed as per the structured In-Service Inspection programme.

Mitigative methods adopted to tackle this included ‘Last past heat sink welding’ for improving upon the residual stresses; reinforcement of welds by overlay technique, corrosion resistant cladding, etc.

3.3. Feed water spargers and nozzles

The feed water spargers and nozzles are components of the reactor that are subjected to maximum cyclic loading and are potential candidates for fatigue failure. Cracks in feed water spargers and nozzles were detected in overseas reactors and this was a subject of concern for TAPS. Since 1977 the nozzles were subjected to ultrasonic examination and the spargers examined by underwater camera. No indication on spargers was observed but the nozzle blend radii did give ultrasonic indications at a few locations. As a result replacement of feed water spargers and inspection of inner blend radii of feed water nozzles were undertaken in 1987 for Unit 1. This was a major engineering challenge as it involved working from inside the reactor. The entire action plan for assessment, design and replacement of feed water spargers of TAPS was carried out indigenously. This was a major experience in modification of vital reactor internal components.

However, contrary to what was reported in BWRs abroad, at TAPS there were no indications/cracks in the nozzle blend radius in both nozzles, except for a minor indication in the clad region in one of the nozzles. Liquid Penetrating Test (LPT) did not reveal any surface cracks on the nozzle cladding nor on the base metal of the nozzle. Feed water spargers were replaced with spargers of modified design to eliminate flow through the leakage path.

3.4. Failure of mechanical seals of reactor recirculation pumps

The TAPS units are equipped with two reactor recirculation pumps per unit with a flow capacity of approximately 3055.55 kg/s each. Frequent seal failure of reactor recirculation pumps started in the late 1970s causing high drywell leaks and at times leading to unit shutdown.

Detailed investigations were made to identify the root cause and corrective action including modification to the seal cooling arrangement and replacement of critical components of the mechanical seals. Subsequently, the performance of these seals has vastly improved.

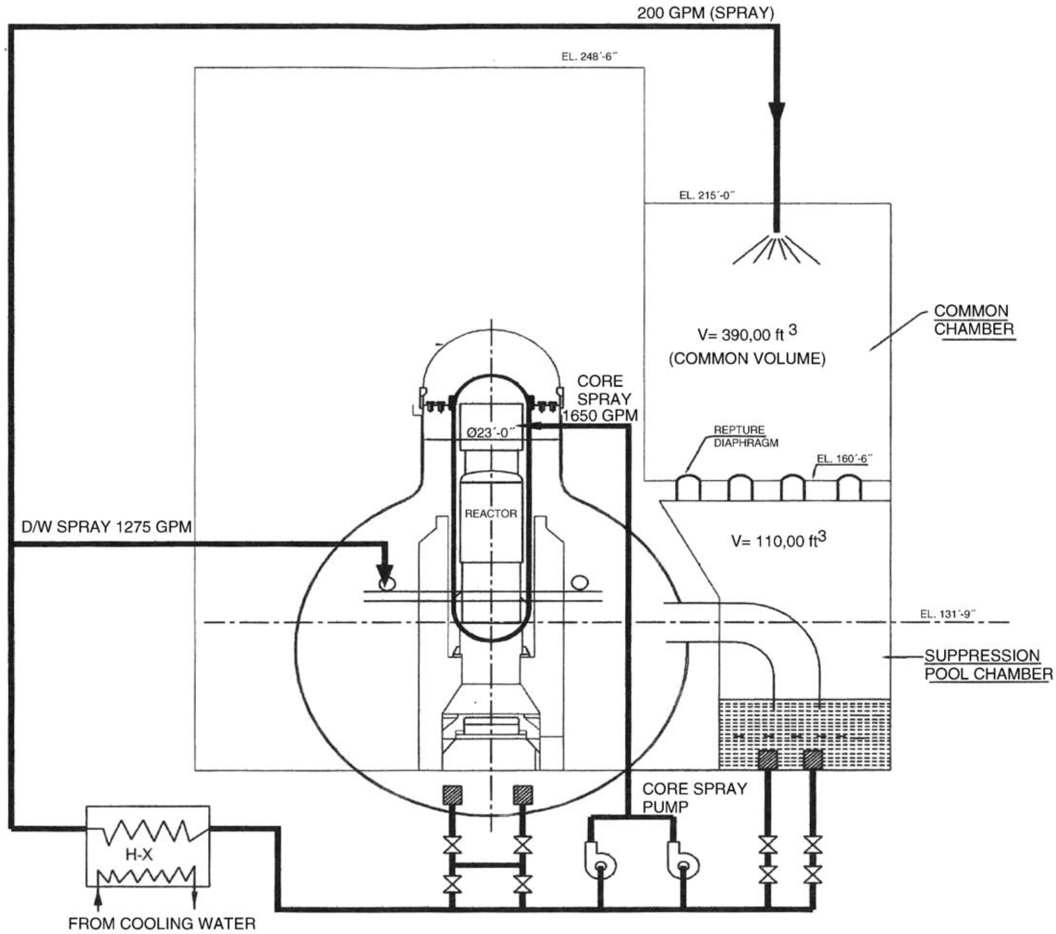

Fig. 6. Schematic of TAPS-1 and 2 containment compartments with spray provision.

3.5. Reactor internals

Two reactor internal components, i.e., the fuel spacer and fuel orifice, had posed operational problems during refueling. The spacer arm used to lift up and the spacer would get unlatched from the top grid. Similarly fuel orifices would get dislodged from their location in the guide tube. These problems were overcome by simple but effective modifications. The spacers were replaced with spacers designed for latching firmly in the grid notch. As regards the fuel orifice, the design was changed from a grid-mounted type to a fuel mounted type. TAPS also developed in-core neutron monitors indigenously and replacement of these core components are completed.

Critical reactor internals like, control blades, local power range monitors, in core neutron monitor dry tubes, fuel channels, fuel channel fasteners, shroud head bolts, control blade guide tubes, etc. have been developed by Indian nuclear industry ensuring availability of these critical components.

3.6. Core Shroud inspection

The reactor Core Shroud is a cylindrical structure inside the reactor and serves the function of supporting the core and separating the core flow from the feed water inlet flow. Cracks were reported in the weld areas of the Core Shroud in overseas reactors. From 1995, extensive examination campaigns of the Core Shroud of both reactors were carried out using indigenously developed inspection manipulators and techniques. One hundred percent ultrasonic and visual examination of all accessible welds of the Core Shroud of both reactors has been completed and these structures are observed to be free of any defects.

Detailed analytical study has also been carried out and it is concluded that no safety function is impaired even with a cracked shroud subjected to postulated accidents including a seismic event. Inspection of the Core Shrouds continues to be a part of the regular In-Service Inspection programme of the station.

3.7. Fire protection system

Though the operating performance of the station's fire protection system was satisfactory, the system was critically reviewed in early 1990s with respect to current standards. Based on the study and analysis carried out, a number of engineering changes were implemented to enhance the system capability and reliability. These included:

(a) Cable segregation of safety related systems.
(b) Coating of critical cables with fire retardant material.
(c) Installation of fire barriers.
(d) Use of modern fire detection system to improve the detection capability.
(e) Replacement of under ground liquid fire protection piping with new piping and fittings.
(f) Enhancing the surveillance on overall fire protection systems.

These engineering changes have not only improved the system availability but also increased reliability.

3.8. Engineering modifications and upgradations

Apart from the experiences listed above, TAPS has taken up major engineering modifications and upgradations to improve the plant availability as well as the system reliability. Over 300 modifications in various systems have been taken up so far out of which 100 modifications pertains to electrical systems, 150 modifications in mechanical systems and about 50 modifications in instrumentation and control systems. All these modifications were taken up based on operating experience, both domestic as well as global. Some of them are listed below.

3.8.1. Electrical systems

The major renovation and refurbishment in electrical systems are:

(a) Upgradation of station batteries to enhance the reliability of Class-1 power supply system.
(b) Provision of a second start-up transformer to enhance availability of reliable start-up power supply.
(c) Installation of a full capacity station black out diesel generator to enhance the emergency power reliability in the event of prolonged Station Black Out (SBO).
(d) Replacement of safety related cables inside primary containment to counter their ageing.
(e) Replacement of Air Blast Circuit Breaker (ABCB) with Sulfur Hexafluoride 6 (SF6) gas circuit breaker in the 220 kV switchyard, to improve overall system components reliability.

All these modifications have been done indigenously after detailed study and analysis. The development of these critical equipments to meet the latest standards was a difficult but challenging task. All these equipments were indigenously designed, developed and manufactured without compromising quality and these are presently operating satisfactorily.

3.8.2. Mechanical systems

The degradation mechanisms in equipment and supports of extraction and feed water system has been identified. The major degradation mechanism in the process piping (carbon steel) has been identified as "erosion–corrosion". Failure of components has resulted in forced outages of the units on a number of occasions. This problem was overcome by replacing the affected piping and fittings with chrome–moly (1% Cr–1/4% Mo) material, which is resistant to this phenomenon. This has not only improved the plant availability but also reduced the forced outages and thermal energy loss. The performance of carbon steel pipe lines are being monitored by systematic In-Service Inspection program.

At TAPS the primary systems connected to reactor pressure vessel is having material of construction as stainless steel grade either 304 or 316 in some cases (except steam line piping and feed water piping). These are classified as Class-1 as per ASME Section-XI. The generic issues related to failure of stainless steel piping were identified as IGSCC. To mitigate this the piping was replaced with corrosion resistant material, such as nuclear grade piping SS-316LN and SS-304L/LN in some cases. TAPS has now completed the replacement of all the vulnerable system piping with nuclear grade material.

In addition to this some of the piping had shown Trans Granular Stress Corrosion Cracking (TGSCC) due to presence of high chloride content in the conventional insulation. In view of this all the Class-1 piping insulation has been replaced with controlled chloride content of less than 200 ppm. In addition to this conventional insulation is being replaced by either metallic type mirror insulation or mineral wool filled jacketed type insulation in a phased manner. The insulation material was developed and procured to meet the requirement of relevant ASTM standards. This has not only improved the general condition of the area but also reduced manrem consumption in removal and re-installation during In-Service Inspection. The healthiness of these piping is being checked through regular ISI, as per station's ISI manual and ASME Section-XI.

The degradation mechanism noticed in the feed water heaters was erosion–corrosion due to steam impingement and tube failures due to fretting. This has resulted in forced unit outages and also reduced cycle efficiency and capacity factors. In view of this, all the feed water heaters have been replaced with new ones, in both the units. Many design improvements have been incorporated such as heater's shell was lined with stainless steel to take care of resistance against impingement shell erosion and baffle plate design changed to improve the resistance against flow-induced vibrations. These heaters were designed and manufactured indigenously and their performance is satisfactory.

In brief following critical mechanical systems have undergone major renovation and upgradations. These are:

(a) Replacement of reactor primary system piping with nuclear grade material to mitigate generic issues of IGSCC.
(b) Replacement of tube bundles in the emergency condensers.
(c) Replacement of high and intermediate pressure feed water heaters.
(d) Installation of additional air compressor to augment compressed air system.
(e) Augmentation of reactor water clean-up system by providing additional pumps to improve system availability.
(f) Augmentation of condensate polishing system by adding additional demineralizer bed to improve system availability and performance.

(g) Replacement of reactor clean-up system heat exchangers and pumps.

In order to ensure the effective performance of these equipments, system performance evaluation is conducted at regular intervals as per the structured program.

3.8.3. Control and instrumentation

The field of instrumentation and control is the one, which faces the maximum challenge of obsolescence. This aspect could be overcome indigenously by developing vendors to supply instruments, which were getting out dated. Indigenous development of instrumentation could be achieved for in-core neutron monitoring detectors, probabilistic disturbance analyzer, rod worth minimizer, control room recorders, etc. The upgradation of fire safety was done by addition of addressable type fire detection system to meet current safety standards. State of the art instrumentation was introduced to monitor turbine-generator performance. A dedicated turbo-supervisory instrumentation was installed which has many features such as data logging and analysis with respect to behavior of turbine-generator during various operating conditions.

All these refurbishment activities involved detailed engineering, following appropriate codes and standards, qualification, maintaining stringent quality control for fabrication and erection and regulatory clearance as required.

3.9. Spent fuel storage capability augmentation

The initial spent fuel storage capacity in the station was for 750 spent fuel assemblies. The original intent was that long-term storage of spent fuel would not be there and the fuel would be removed from the pools and sent back to USA. However, with unexpected changes in the global political scene it became evident that no shipment of spent fuel would take place from TAPS fuel pools. The fuel is under international safeguards and India has taken a principled stand not to reprocess this fuel. Hence it became essential to store all the discharged fuel in a safe manner. As the station generates around 100 spent fuel assemblies each year of operation, it became imminent to expand the fuel storage capacity to continue plant operations. The storage capacity was augmented to 5000 in a phased and multi pronged manner by first replacing the existing aluminum fuel storage racks in TAPS fuel pools with high density stainless steel fuel racks, developing dry storage racks and construction and commissioning of a modern, stand alone Away From Reactor (AFR) Spent Fuel Storage Pool. Today the station has adequate spent fuel storage capability for the next 2 decades of operation.

3.10. Overall station performance

The operating life of a reactor is governed by the cumulative fast neutron flux that the vessel material experiences during its operating history. An exhaustive material surveillance programme covers the TAPS reactor vessels. Testing of surveillance samples of the reactor vessel material have confirmed that the vessel material is in good shape and fit for operation for well beyond 60 Effective Full Power Years (EFPY). The reactors have experienced less than 18 EFPY till year 2003. Throughout the operating history, a stringent chemistry control has been exercised on both reactors and this has eliminated the probability of corrosion related degradation of the reactor vessels and their internal structures.

Examination of the containment and civil structures has concluded that these structures have not degraded in any significant manner during the last 33 years and they match the reactor's capability for future long-term operation. Continuous refurbishments of plant equipment have kept the plant in excellent condition and timely development of critical components have ensured that the station is not found wanting for spares.

Major analytical studies have been carried out for the station to confirm that the station meets the latest requirements of safety and reliability. Probabilistic Safety Analysis (Level-1) study concluded recently that the Core Damage Frequency (CDF) due to various initiating events for TAPS reactors is of the order of 10^{-5} per reactor year which is as per International Nuclear Safety Guide (INSAG) guidelines for earlier generation reactors.

The station has continuously upgraded itself with advancement in technology and a major change that is evident is the application of computers in plant operations. Computerized Maintenance Management System (CMMS), Material Accounting and Information System (MAIS), Local Area Networking (LAN), etc. are some of the systems that have been developed and are in use for efficient operation of the station.

Human resources development has played a pivotal role in the continual performance improvement of the station. Systematic Approach to Training (SAT) technique, regular upgradation of knowledge, latest techniques like Stop, Think, Act and Review (STAR), peer check, peer review, pre-job briefing and post-job critique, operating experience review and mobile training are a few of the methods in practice at the station for human performance improvement. The benefits of these methods are evident from the trend of station performance for the past few years.

The station's capacity factors have improved considerably and have reached a value of 92% comparable to the best plants in the world. The radioactive liquid waste discharges from the station have reduced to a fraction of the earlier values and the trend continues. TAPS has zeroed-down the forced outages for the past 2 decades and has made considerable progress in reducing the period of planned outages. Aggressive planning, preparation, resource mobilization and effective supervision have brought out a dramatic reduction in the refueling outage period. The outage period of an average 100 days has been continuously improved on and the last refueling outage of Unit #1 in 2003 was completed in less than 20 days. This is comparable to the best in the world.

The performance on industrial safety and fire safety front has been excellent. There has been no fatal accident in the last 25 years or any major fire in the plant since it was commissioned. The Incident Rate and Severity Rate of the station are comparable to the best in the world.

4. Safety review of plant for license renewal

4.1. Design basis review to enhance safety levels

The twin units of TAPS have been operating since 1969 with periodic renewal of authorization. In 2000, Atomic Energy Regulatory Board (AERB) desired a fresh comprehensive review involving review of plant operating performance, ageing management and the review of design basis and safety analysis in order to authorize continued operation beyond May 2002.

The review of operating performance and ageing management has been dealt in Section 2 while the review of design basis of systems is covered here.

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

The review of design basis was carried out for 26 systems and components relevant to safety and contributing to safe operation of plants. The review included following systems:

- Reactor coolant system and components including reactor vessel and internals and coolant systems and components.
- Reactor protection system including control rod drive hydraulic system, safety system power supply system and stand by liquid poison system.
- Residual heat removal system including shutdown cooling and head spray system, emergency condenser system and fuel pool cooling system.
- Engineered safety features including primary containment system and emergency core cooling system.
- Waste management system.
- Instrument air system.
- Station power supply system:
  - AC/DC electrical power supply system including cable layout.
  - Emergency diesel generator system.
- Ultimate heat sink:
  - Salt service water system.
  - Reactor building cooling water system.
- Fire protection.
- Control and instrumentation systems.

For the identified systems, design details were collected. This was an important activity since the safety analysis report for this plant (written in the late 1960s) does not give adequate information as required by today's reviews, and therefore collection of design information from the variety of documents became an essential task. Documents identified for this purpose include design basis reports, technical specifications, process and instrumentation diagrams, electrical schematic diagrams, design modification notes and operating procedures and practices.

The review was performed based on guidelines of USNRC Standard Review Plan (NUREG 800, 1984). The Standard Review Plan identifies the USNRC requirements as specified in General Design Criteria (GDC) of 10 CFR 50 (USNRC, 1986), to be met for specific system based on the safety functions it performs. The areas, which were reviewed, included the current safety requirements such as conformance with single failure criteria/redundancy, defense-in-depth, physical and functional separation of components and resulting common cause failure vulnerabilities, as applicable. Attention was given to common cause vulnerabilities like fire, flood, etc.

With respect to components/equipment of the identified systems, the review was performed with regard to their meeting the design parameters, and applicable design codes and standards.

The outcome of this review, in certain cases brought out the non-conformances to current requirements of design and safety practices. The safety significance of these non-conformances was reviewed based on guidelines of IAEA Safety Report Series 12 (IAEA, 1998).

Assessment of acceptability or otherwise of the non-conformances would include consideration of the available/proposed compensating factors, e.g., performance of the affected safety function by other engineered features or operational practices. Insights from the results of the Level-1 Probabilistic Safety Analysis (PSA) of TAPS-1 and 2 were also made use of in this assessment.

While performing the above review, the relevant other outstanding/pending safety issues arising from earlier safety reviews by utility/regulatory authorities and generic safety issues (IAEA TECDOC 1044, 1998) were also considered and their safety significance placed in the overall perspective.

The net outcome of the review was a set of recommendations for upgradation. Majority of the recommendations pertains to upgradation of station electric power supply system, separation of power supply sources to redundant equipment and segregation of power and control cables catering to redundant equipment. Additionally, strengthening of surveillance and testing of the systems and equipment has been recommended in some areas.

4.1.1. Recommendation of systems modification/upgradation

The salient recommendations of system modifications are as follows:

- Upgradation of existing $3 \times 50\%$ capacity emergency diesel generators by $3 \times 100\%$ emergency diesel generators.
- Segregation of electrical distribution system for Classes-III, -II and -I supplies into two zones with physical barrier.
- Redistribution of supplies to redundant loads from separate buses.
- Cable re-routing through diverse routes for redundant loads.
- Augmentation and unit wise segregation of emergency feed by addition of one pair of control rod drive pumps.
- Unit wise segregation of reactor shutdown cooling system and delinking from fuel pool cooling system.
- Upgradation of compressed air system by providing additional dryer, relocation of compressors to avoid common cause failure and powering of compressor by Class-III power supply.
- Installation of strong motion seismic instruments.

- Major recommendations on fire protection:
  - Electric power supply system modification.
    - Segregation of power supply sources into independent trains.
    - Diverse routes for cables to redundant equipment.
  - Augmentation of fire detection system: addressable type smoke detection system with optical, flame and heat detection covering all vital areas.
  - Fire barriers (3 h rating) for cable penetration between control room and cable spreading room.
  - Fire resistant false ceiling in control room.
  - Fire doors, fire dampers.
  - Fire retardant coating on cables.
  - Emergency Operating Procedure (EOP) for cold shutdown in event of fire in control room.

As stipulated by AERB implementation of the recommended upgradation will be completed by 2005.

4.2. Safety analysis

The original safety analyses for TAPS was carried out in late sixties by GE, USA and was reported in the final safety analysis report, design basis report NEDO reports, etc. Additional safety analyses for the plant had been carried out over the years. The regulatory approach had been to demonstrate the safety aspects prior to attempting plant modifications. Department of Atomic Energy (DAE) agencies, i.e., Bhabha Atomic Research Centre (BARC), Nuclear Power Corporation of India Limited (NPCIL) and consultants had carried out the in-house safety analysis. With advancement of the analytical methodologies and certain modifications carried out in plant configuration over a period of 30 years, the regulatory requirement was to carry out a fresh analysis prior to extending the operating license of the plant. In this context, the fresh analysis covering selected enveloping DBA's have been identified and analysis has been performed using the current analytical methodologies and current state of the art computer codes. Identified analyses broadly covered the following:

1. Safety analysis of DBAs including LOCA analyses for various break sizes in recirculation line and steam line (Malhotra et al., 2002).
2. Containment analysis.
3. Fatigue analysis of the Reactor Pressure Vessel (RPV) for integrity assessment.
4. Probabilistic Safety Analysis, Level-1.
5. Seismic assessment.

4.2.1. Vessel integrity analysis (Kumar et al., 2002)

The RPV for TAPS units were designed as per ASME Boiler and Pressure Vessel code, Section III and Section VIII (1962) and related code cases. In the fresh analysis done, RPV is qualified to meet the requirements of ASME, Section III, its sub-section NB (2002) for the various loads experienced by the vessel. The reactor pressure vessel has been analyzed using finite element analysis wherein it is subjected to cyclic loading due to the temperature and pressure variation that are experienced during the reactor operation and upset conditions of the plant for which fatigue analysis is carried out.

The calculated stress intensities at various critical locations of reactor pressure vessel for different service levels for the vessel nozzles are less than their allowable stress limits.

For fatigue analysis, the number of pressure and temperature transient cycles includes the cycles already seen by the vessel and also the anticipated number of these cycles for the next 40 years. Based on this, the cumulative usage factor ($\sum n/N$) for all the service levels is found out to be well within the codal requirements of being less than one.

Table 2
Dominant sequences

	Sequence/failure of	Contribution to CDF (%)	Main contributors
1.	Class-IV, Class-III (station black out), emergency condenser	21.6	• Diesel generators common cause failure • Emergency condenser TDR for motorized operating valve opening
2.	Loss of main feed water system, emergency feed system and core spray system	21.4	Mainly Class-III power
3.	Inadvertent opening of relief valve, loss of main feed water system and auto blow down system	11.8	Mainly instrument air
4.	Loss of Class-IV, emergency condenser, auto blow down system	11.0	
5.	Loss of Class-IV, emergency condenser, post-incident cooling system	6.6	
6.	Loss of main feed water system, emergency condenser, auto blow down system	6.6	
7.	Medium size LOCA, auto blow down system	5.1	

Core damage frequency (internal events) 6E−5/Rx/y.

4.2.2. Reactor pressure vessel embrittlement

The original design life of the reactor pressure vessel is estimated as 40 EFPYs based on embrittlement due to fast neutron fluence. So far the reactor has experienced 17 EFPYs. The fracture toughness of reactor pressure vessel was assessed based on surveillance coupons installed on various locations inside RPV, which were drawn on regular interval periodically.

Surveillance specimens used for charpy V-notch impact assessment were fabricated from the same material, which was used for base, weld and Heat Affected Zone (HAZ) of reactor pressure vessel. These specimens kept at the wall and shroud locations were evaluated in 1983 after 6.5 EFPYs and 59.7 EFPYs, respectively, of reactor operation and assessment was done on basis of USNRC regulatory guide 1.99 Revision-2. The assessment assured the integrity of the vessel till the End of Service Life (EOL) of 40 years.

In order to re-confirm the assurances of the integrity of reactor pressure vessel additional impact surveillance specimens were evaluated in 1998 after 13 EFPYs.

4.2.3. Probabilistic safety studies (Daniel et al., 2001)

PSA Level-1 has been carried out concurrently to determine the core damage frequency and to identify the weak links in affecting plant safety. Based on these studies, the weak areas requiring improvements in design to enhance safety levels and to increase plant availability have been identified. Cognizance of level-1 PSA has been taken in suggesting the modifications while carrying out the review of the design basis of the systems based on current safety standards and practices.

Table 2 indicates the dominant failure sequences and their contribution to CDF. It is seen that failures of station electrical power supply (Class-III power) and instrument air have major contributions in core damage sequences.

4.2.4. Seismic re-evaluation of TAPS-1 and 2

The seismic re-evaluation was performed by a combination of analysis and walkdown. Qualification of safety related civil structure and piping was performed by analysis, while walkdown approach was used for safety support systems and other systems, control and instrumentation/electrical items and interactions.

The findings of the seismic evaluation were that the safety related civil structures and piping qualify the seismic requirements.

A few systems/components/equipment were required to be upgraded which were as follows:

- Unanchored equipment (to be anchored or removed).
- Strengthening of anchorages of some equipment (S/D cooling heat exchangers, condensate storage tanks).
- Relocation of certain supports of cable trays.
- New stands for battery banks.

5. Conclusion

The twin BWR units at TAPS are operating for the last 30 years. The gross capacity factors for the year 2001 are 84.8% and 93.8% for Units-1 and 2, respectively. Their continued safe operation is the main driving force for seeking the extension of the operating license of these oldest operating units in India. For this purpose, an extensive exercise involving review of plant operating performance, ageing management and review of design bases and safety analysis has been carried out as required by AERB. The design basis review has identified certain improvements in the existing design to meet the currently followed safety principles and practices.

In overall perspective, the plant performance over the past 3 decades has been good; the technological challenges faced have provided a wealth of operating experience and development of indigenous technology. The station has matured and is cruising towards a long run of safe and efficient power generation.

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