REFURBISHMENT, CORE CONVERSION AND SAFETY ANALYSIS OF APSARA REACTOR

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ABSTRACT

Apsara, a 1 MWt pool type reactor using HEU fuel has been in operation at the Bhabha Atomic Research Centre, Trombay since 1956. In view of the long service period seen by the reactor it is now planned to carry out extensive refurbishment of the reactor with a view to extend its useful life. It is also proposed to modify the design of the reactor wherein the core will be surrounded by a heavy water reflector tank to obtain a good thermal neutron flux over a large radial distance from the core. Beam holes and the majority of the irradiation facilities will be located inside the reflector tank. The coolant flow direction through the core will be changed from the existing upward flow to downward flow. A delay tank, located inside the pool, is provided to facilitate decay of short lived radioactivity in the coolant outlet from the core in order to bring down radiation field in the operating areas. Analysis of various anticipated operational occurrences and accident conditions like loss of normal power, core coolant flow bypass, fuel channel blockage and degradation of primary coolant pressure boundary have also been performed for the proposed design.
Details of the proposed design modifications and the safety analyses are given in the paper.

INTRODUCTION

Apsara, India's first research reactor was commissioned in 1956. In view of the long service period of the reactor extending over more than 40 years, it is now planned to carry out extensive refurbishment of the reactor, so as to extend it's useful life as also to upgrade it's safety features in line with the current safety standards. Bhabha Atomic Research Centre has recently developed the design of a LEU fuelled, light water cooled and moderated, heavy water reflected pool type Multi- Purpose Research Reactor (MPRR) in the power range of 5~10 MW. Heavy water reflector contained in an annular tank surrounding the core, provides a good thermal neutron flux over a large radial distance from the core, thereby enhancing the neutron flux irradiation volume product of the reactor considerably. In the process of refurbishing Apsara, it's basic design will be suitably modified by incorporating certain salient design features of MPRR such that the modified reactor also serves as a demonstration reference reactor to provide a cost effective technological simulation of the MPRR design. Since the HEU fuel presently loaded in Apsara reactor has not reached the targeted exit burn up, the refurbishment and core conversion programme will be implemented in two phases. In the first phase, the existing HEU fuel (U- Al alloy) will be used along with 4-5 Pu-Al alloy booster assemblies to constitute the core and reactor power will be restricted to 500 Kw. In the second phase the core will be loaded with LEU Silicide fuel assemblies and power will be raised to 1 MW to yield a thermal neutron flux of approximately 1.3×10^{13} n/cm^2/sec. All the design provisions to permit reactor operation at 1 MW will be incorporated in the first phase itself so that second phase can be implemented without involving any further system modifications or pool drainage.

The modified reactor will offer facilities for beam tube research, neutron radiography, neutron activation analysis, neutron transmutation doping of silicon, testing and calibration of neutron detectors etc.

SALIENT DESIGN MODIFICATIONS

Reactor Core

The existing core is laid in a 7x7 array at a square pitch of 76.8 mm. The modified core will be similar to MPRR-10 core except that due to constraints of space in the pool, the reflector thickness will be about 10 cm less than the envisaged reflector thickness of 60 cm in MPRR-10. The core will be laid in a 5x5 array at a lattice pitch of 84.8 mm. There will be 3 added on positions on each side of the square grid (Figure-1).

The core will be surrounded by an annular heavy water filled reflector tank. The gap between the inner shell of the annular reflector tank and core lattice assemblies will be filled with graphite fillers for better coupling with the reflector and neutron economy. Since the existing fuel has already seen about 20 atom % (U^{235}) burn up and the fuel lattice pitch is being increased from 76.8 mm to 84.8 mm, the phase-I core will be constituted with existing Apsara fuel along with 4 to 5 Pu-Al booster assemblies to compensate for reactivity losses. In phase-II, the entire core will be fuelled with LEU (19.7 % W/W U^{235}) dispersion type silicide fuel. Each standard fuel assembly will consist of 18 fuel bearing plates and two inert aluminium end plates. Each fuel plate has meat thickness of 0.6 mm and is clad with 0.4 mm thick aluminium alloy 6061.

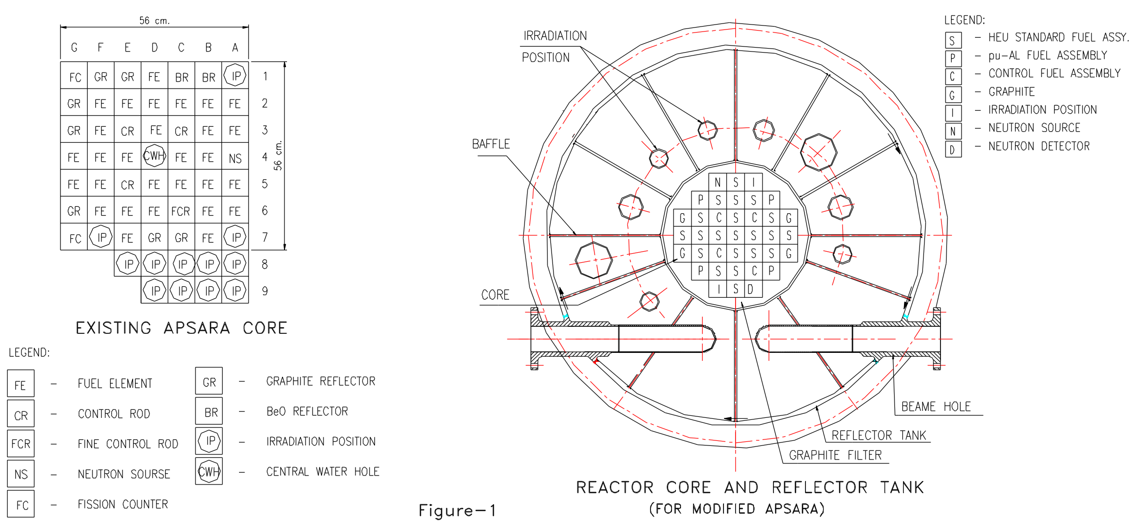
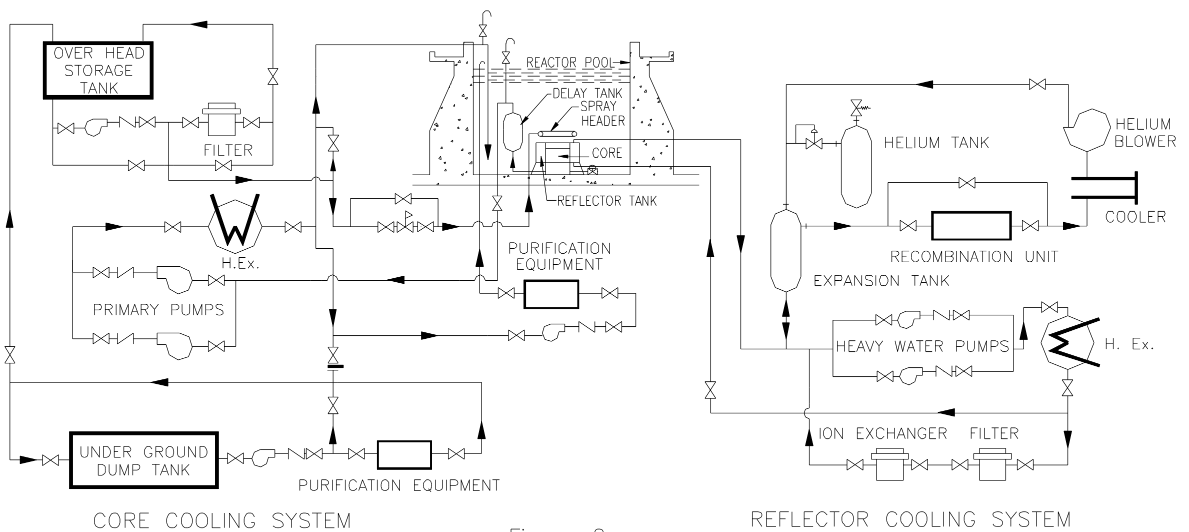


Figure-1

The reactor core comprising of standard fuel assemblies, control fuel assemblies and graphite fillers are supported on a 150 mm aluminium grid plate. The grid plate and the surrounding reflector tank are supported on a common support structure anchored to the pool floor. The reflector tank is designed to provide a heavy water reflector thickness of 500 mm around the core and incorporates the structural elements of horizontal beam tubes and vertical experimental/irradiation thimbles.

Core Cooling System

At present Apsara reactor core is cooled by demineralised water flowing from bottom to top across the core. After picking up the heat from the core, the hot coolant mixes with the pool water. Coolant pumps draw hot water (at a rate of 1000 lpm) from the pool and is sent to a heat exchanger where the heat is transferred to the secondary coolant which in turn rejects the heat to atmosphere through a cooling tower. In the modified reactor, the flow direction across the core will be reversed with a view to minimize the radiation levels at the pool top. Water at rate of 2500 lpm will be drawn from the core outlet at the bottom so that coolant flow in the core is from top to bottom and will be led to a delay tank located in the reactor pool itself before joining the primary pump suction header in the process equipment room. The delay tank is adequately sized to provide a delay time of 1 minute for N^{16} activity to decay sufficiently to reduce the radiation field in the process equipment room. The heat picked up by the coolant is transferred to the secondary coolant (cooling tower water) in a plate type heat exchanger provided at the primary coolant pump discharge. Heat is eventually rejected to atmosphere through a cooling tower (Figure-2).



Figure–2

In the event of non-availability of coolant pumps, two hydraulically operated fail safe flapper type valves installed at core outlet will open automatically when the pump discharge pressure falls below a preset value to establish natural convection cooling flow through the core. Opening of one flapper type valve is adequate to meet the core cooling requirements when the reactor is shutdown.

The system piping is laid in such a manner that in the event of postulated coolant pipe breakage, a minimum core submergence of 2.5m is ensured. In the highly unlikely event of core getting uncovered due to drainage of pool water, provision is made to cool the core by demineralised water stored in an overhead storage tank through a spray header provided just above the core.

Reflector Cooling System

Since heavy water is introduced in the modified design, an appropriate system consisting of a pump, a heat exchanger, purification circuit and associated piping has been engineered to facilitate recirculation, cooling and chemistry control of the heavy water (Figure - 2). Pressure of heavy water in the reflector tank will be maintained at a value slightly higher than the surrounding pool water pressure. Similarly, the pressure of heavy water in the heat exchanger will be maintained at a value higher than that of the secondary coolant. This differential is maintained to avoid degradation of heavy water due to ingress of light water in case any leak develops in the reflector tank or the heat exchanger.

Helium will be used as cover gas for heavy water in the system. Provision has been made to recirculate helium through a recombination unit consisting of palladium catalyst to limit the concentration of radiolytic decomposition product D_2+H_2 and O_2 in the system.

Reactor Regulation and Protection

In order to enhance the reliability and availability of the existing system, some of the control & Instrumentation features of MPRR will be incorporated in the modified design. Accordingly, existing single channel regulating system of Apsara will be upgraded to a triplicated system. For improving the response and stability characteristics, the existing On/Off mode of control making use of neutron power signal alone will be modified to proportional control mode using both power and period signals. As an operator aid, incorporation of a computerized data acquisition and information retrieval system is also planned.

The design envisages four control rods, of which three (shim rods) are used for coarse control and one for fine control. The shim rods of higher worth are moved manually in steps while the fine control rod of lower worth is adjusted incrementally by a stepper motor.

For reactor start-up, two fission counter based pulse channels will be provided. For power range operation, the detectors of three independent nuclear channels will be placed in the reactor pool about 120 degrees apart, adjacent to the reflector tank outer wall to obtain a closer representation of the flux. The mean of the triplicated sensor output will be compared with the set-point, and the error fed to the servo system for driving the fine control rod to obtain the set power level. Apart from regulation action the system also generates reactor trip on high linear power, high log rate, linear power disagreement, log power disagreement and log rate disagreement.

A Multi Range DC Channel will also be provided for monitoring the reactor power in entire power range. This channel has an independent detector and also generates a trip at 110% of each range.

The protection system makes use of various parameters, generated from neutronic and process instrumentation signals, for providing necessary protection action, by tripping the reactor, through fast insertion of gravity assisted shim rods. In addition, the fine control rod will also be driven down automatically on a reactor trip signal.

The trip parameters have been segregated in two distinct groups each forming an independent bank, in order to provide two diverse chains for protection action. The segregation has been carried out in a manner to ensure that each postulated initiating event is covered by two independent parameters. These two parameters which act as a back-up to each other, will be wired separately in two banks.

The important neutronic and process parameters are triplicated and the trip function for these parameters will be generated by local coincidence logic. The output of the local coincidence logic will be fed to one of the two banks in the protection system.

In addition to the trips generated from reactor regulating system, two independent Linear Safety Channel trips will be provided in the reactor protection system.

SAFETY ANALYSIS

The safety analysis has been carried out in line with the requirements of "Code on the Safety of Nuclear Research Reactor : Design (35-S1)" to demonstrate that operational limits and conditions are satisfied for the normal operation of the plant. Various postulated initiating events as appropriate for the plant design were characterized and the event sequences resulting from them were analyzed. The results of the analyses were compared to the design limits and radiological acceptance criteria and found acceptable.

The PIE's as listed in Code on the Safety of Nuclear Research Reactor Design (35-S1) as applicable to the modified Apsara Reactor were categorised into Anticipated Operational

Occurrences (AOO’s) and Accident Conditions on the basis of frequency of occurrence and their potential radiological consequences. Since it is neither practical nor necessary to include every event sequence in the safety analysis, only those PIE’s that are likely to produce boundary cases for safety design are considered in the analyses. Some of the PIE’s analysed for modified Apsara Reactor (for PHASE - I) and presented in this are given below.

Anticipated Operational Occurrences

i) Primary pump failure
ii) Primary coolant flow reduction due to flow bypass
iii) Fuel channel blockage

Accident Conditions

i) Primary coolant boundary rupture

Acceptance Criteria

Parameter	Normal operation	Anticipated operational occurrence	Postulated accident condition
Reactor power (kW)	<550	<600	-
Fuel clad temp. (^{o}C)	<100	<120	<400
Pool water level (m)	>7.3	>6.8	>4.5
DNBR	>1.5	>1.5	-

Analytical Tool

Thermal hydraulic analysis of modified Apsara Primary Coolant System was carried out using the computational codes PLTEMP-3 and RELAP4/MOD6.

1. PLTEMP-3

The PLTEMP-3 code is designed to carry out steady state thermal hydraulic analysis of any research reactor with plate-type fuels. This code is developed by Research Reactor Institute of Kyoto University (KURRI) and Argonne National Laboratory (ANL) to carry out the thermal-hydraulic analysis of the Kyoto University Reactor (KUR) for core conversion with LEU (<20% enrichment) fuels. For the analysis of modified Apsara core, PLTEMP-3 Code was suitably modified to incorporate segment wise distribution of coolant temperature along the direction of flow as also heat transfer correlations for laminar flow. DNB correlations were also changed considering low flow, low pressure and low temperature situations prevailing in the modified Apsara core. The modified PLTEMP-3 code was validated by analysing two typical calculations given in IAEA TECDOC-233.

2. RELAP4/MOD6

The RELAP4/MOD6 code is developed to describe thermal hydraulic conditions attendant to postulated transients in light water reactor systems. This code is developed at the Idaho National Engineering Laboratory for the Nuclear Regulatory Commission (NRC).

RELAP4/MOD6 applies so-called one-dimensional node and junction method, in which fluid conservation of mass, momentum and energy are integrated over a mathematically defined control volume called node and the resulting set of simultaneous equations are linearised and advanced for small time increment by fully implicit numerical technique.

RESULTS AND DISCUSSIONS

Failure of Primary Recirculation Pumps

During normal operation coolant is circulated through the core from top to bottom by the primary recirculation pump. Failure of this pump will result in continuous reduction in coolant flow through the core depending on the coast down characteristics of the primary coolant system and its pump motor assembly. Reduction in core flow will result in the increase of fuel, clad and coolant outlet temperatures. During the pump coast down, as the pump discharge pressure falls below the preset value, the natural circulation valves will open to provide a flow path for natural convection cooling of the core.

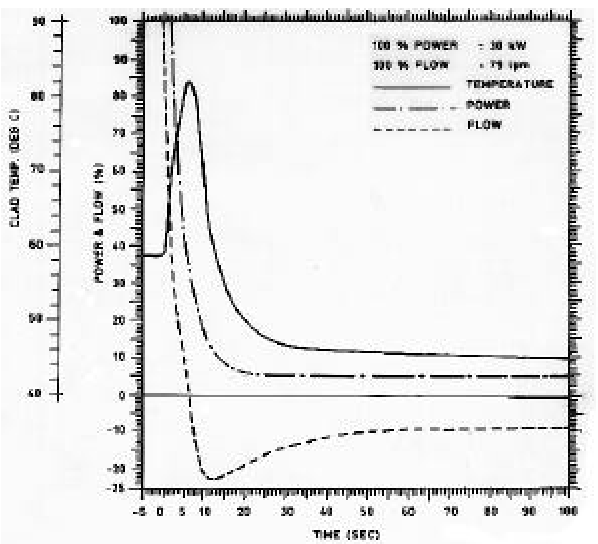


Fig. 3. Flow coast down transient of Pu-Al fuel assembly during failure of primary recirculation pumps

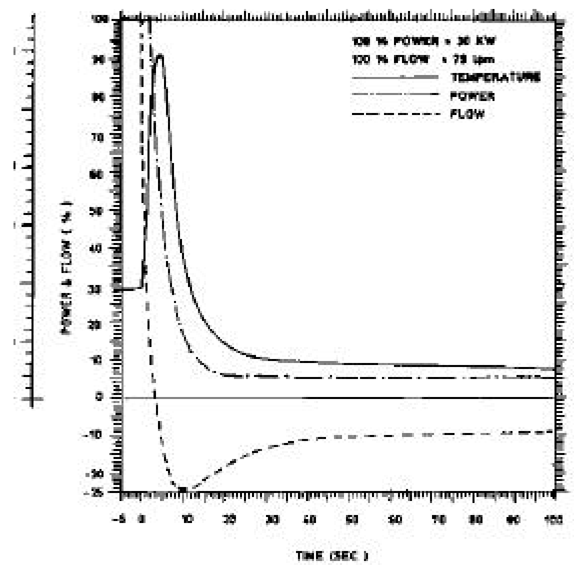


Fig. 4. Flow coast down transient of Pu-Al fuel assembly during abrupt failure of recirculation pumps

In the analysis, reactor trip initiation has been taken 2 seconds after the pump trip. This time delay of 2 seconds has been taken with the conservative assumption that reactor fails to trip on class IV power failure and the trip is registered by back up trip on primary coolant flow low (at 85% of normal flow) which is initiated in about 500 milli-seconds after the pump trip. After initiation of the reactor trip there can be a maximum delay of about 300 milli-seconds for shim rod actuation. Allowing shim rod drop time of 500 milli-seconds, reactor will be in safe shut-down state well within the period of 2 seconds.

Maximum clad temperatures of standard fuel assembly, control fuel assembly and special (Pu-Al) fuel assembly are found to be 77.5°C, 78°C and 82°C respectively which are well bellow the acceptable limits.

Abrupt Failure of Primary Recirculation Pumps without Coast Down

Since flow coast down under pump seizure is faster than that during normal coast down, the rate of temperature rise is faster for each fuel assembly. Maximum clad temperatures of standard fuel assembly, control fuel assembly and special fuel assembly are found to be 102.4°C, 99°C and 99.3°C respectively which are well within acceptable limits.

Flow Blockage of Sub-Channel in Fuel Assembly

Blockage of a sub-channel between the fuel plates of any fuel assembly can be caused by small object falling into the pool inadvertently and being lodged at the fuel top (inlet) or can also be caused by any floating material getting lodged in the coolant passage of the fuel assembly. Most of the blockages are expected to cause only partial blocking of a sub-channel. However in the analysis it is considered that flow is totally stopped in the blocked sub-channel. Analysis indicates that the clad temperature of the affected fuel plates will go up to 78.6^{o}C from normal value of 58.7^{o}C.

Coolant Flow Reduction due to Opening of Natural Circulation Valve

Inadvertent opening of natural circulation valve creates a flow path parallel to core during normal operation. This will result in reduction in core flow and consequent increase in fuel and clad temperatures. Analysis indicates that on full opening of one of the natural circulation valves, flow through the hottest channel fuel assembly reduces to 75% of normal flow and clad temperature increases to 80^{o}C with reactor operating at full power. The corresponding ONBR and DNBR values are 2.1 and 4.1 respectively. However, to ensure that reactor operation shall not be continued when natural circulation valves are not fully closed, reactor trip has been provided on non-closure of any one of the natural circulation valves.

Loss of Coolant due to Primary System Pipe Failure

Apsara coolant system is a low pressure, low temperature system. For low enthalpy system USNRC (NUREG-800) has recommended analysis based on crack area of Dt/4, where D is the pipe diameter and t is the pipe wall thickness. The same approach has been adopted here.

In this analysis leakage through a circular area equivalent to Dt/4 in the pump suction line is considered. It is observed that initial leakage flow is 105 lpm (3.5% of pump flow) which drops slowly with time as the pool level drops. On initiation of the leakage, pump flow drops by 0.5% and is maintained. Therefore, initially there is a net increase in core flow with consequent reduction in clad temperature of the fuel assemblies. Subsequently the core flow will start reducing slowly as the pool level drops resulting in increase of clad temperature. Due to reduction in flow through the primary side of the heat exchanger as a result of reduction in pump flow, heat transferred to the secondary side will decrease. This will also lead to small increase in primary side temperature as also of the fuel clad temperature. The clad temperature reaches a value of 60^{o}C by the time pool level drops by 10% in three hours and trips the reactor on pool low level. This results in reduction of temperature of all fuel assemblies (Figure -5). The recirculation pump will continue to operate till the pool level reaches 4.6m at which pool gets isolated because of siphon break at the pool outlet line and pump loses suction due to stoppage of flow from the pool. Further removal of core decay heat is achieved by natural convection.

Loss of coolant analysis was also carried out considering the unlikely situation of catastrophic failure at the suction of recirculation pump. The pump is assumed to have gas locked immediately after the pipe failure. The leakage flow goes beyond normal pump flow at the point of break, where the pressure actually reduces to atmospheric pressure after break. This will result in an initial increase in flow rate through the fuel assemblies and consequent reduction in clad temperature. The reactor will trip on low pump flow resulting in a reduction of clad temperature at a faster rate. Subsequently, as the natural circulation valves open due to drop in pump discharge pressure, part of the flow gets bypassed through these valves. Du to this the core flow reduces resulting in a slower drop of clad temperature (Figure - 6). The leakage flow reduces slowly with time as the pool level drops and stops as the pool level reaches 4.6m. This happens 25 minutes after the initiation of the incident. Subsequent core cooling is achieved by natural convection cooling.

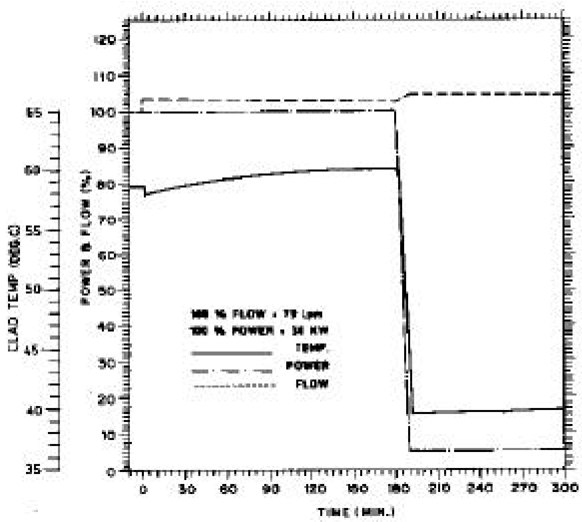


Fig. 5. Flow & Clad temp. variation of Pu-Al fuel assembly during Small Break LOCA

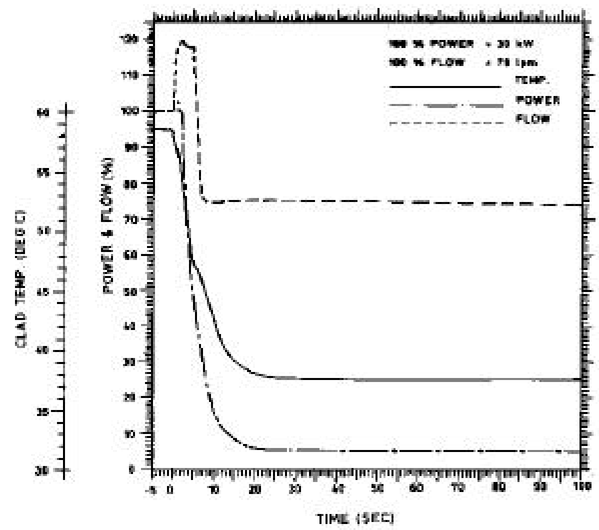


Fig. 6. Flow & Clad temp. variation of Pu-Al fuel assembly during LOCA (Catastrophic failure)

Concluding remarks

The design modifications planned to be carried out to the existing Apsara reactor will enhance the reactor safety significantly while extending its service life and increasing its utility. The modified reactor meets all the acceptance criteria as demonstrated by the analysis of various postulated initiating events as stipulated in the IAEA Code on the 'Safety of Nuclear Research Reactors : Design' (35 - S1).

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