Pramāṇa, Vol. 24, Nos 1 & 2, January & February 1985, pp. 211–226. © Printed in India.

Engineering development and safety back-up for nuclear power programme

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Abstract. This article reviews the engineering development and safety aspects that are relevant to the nuclear power programme being pursued in the country. Some of the important aspects have been discussed in detail bringing out the current status and also the directions for further work.

Keywords. Nuclear power programme; engineering development; safety back-up; pressurised heavy water reactor; emergency core cooling system; thermosyphon; coolant tubes

PACS No. 28·45; 28·40

1. Introduction

It is now well known that nuclear power is the only source of energy for bulk power generation which can meet our future requirements. Our resources of fossil fuel which form the mainstay for bulk electricity production at present would tend to get depleted with time and it is necessary to develop the alternate nuclear source to a level where it can take over from the fossil fuel sources while maintaining the growth of electricity generation necessary to sustain the national development. The three-stage nuclear power programme envisaged by the Department of Atomic Energy is aimed at achieving this objective. As of now the emphasis has been mainly towards mastering of the nuclear technology and to demonstrate that nuclear power stations can be built within the country with nuclear electricity available at a cost which is competitive with the conventional electricity. Time has now come when this effort needs to be accelerated to enhance the installed nuclear capacity. The nuclear power programme of 10,000 MWe by the year 2,000 outlined by Dr Ramanna is our immediate objective. In this phase, not only one would increase contribution of nuclear electricity but also enable production of sizable quantities of plutonium which, with the available uranium resources, can sustain a much larger fast breeder programme. It is envisaged that through the use of fast breeder reactors an installed capacity to the tune of 350,000 MWe can be realised with available uranium resources. Subsequent to this, further generation of nuclear electricity would be sustained by large reserves of thorium resources.

Realisation of the objective of 10,000 MWe capacity by the year 2,000 is envisaged to be achieved by installing a number of 235 MWe and 500 MWe power units of PHWR type. In order to realise installation of such a large number of units in a small time, many aspects such as availability of finance, manpower, industrial capacity, organisational setup, safety regulation etc have to be looked into in detail. The purpose of this article is to review the engineering development and safety back-up that would be necessary to support the proposed power programme.

2. The PHWR system

Our power programme is based on pressurised heavy water type reactors. We are now at a stage when for the 235 MWe units we have gone through a full cycle of design, construction, commissioning and operation. Although the experience with operation of RAPS-I unit, which has been the first in the PHWR series of reactors, did reveal some inadequacies, the performance of RAPS-2 and MAPS-1 has been highly encouraging. Even for RAPS-1 the lessons have been learnt and it should be possible to bring this reactor back on stream quite soon. It is worth mentioning at this stage (Strachan and Brown 1984) that similar stations in Canada have shown excellent performance. As a matter of fact, in terms of lifetime capacity factors upto end of 1981, seven out of 8 CANDU units, find a place in the first ten positions of the world's most reliable operating plants of capacity greater than 500 MWe. The eighth one has 18th position. The design of reactor units beyond Narora considers new features that have been incorporated in Canadian design as well as our own experiences. There is thus no doubt that the PHWR units that are being added would show even better performance. There is however, always a need to look for further improvements on a continuing basis. For the 500 MWe units which are now under design this need is even greater.

For realisation of improved performance as well as safety characteristics of the reactor units it is necessary to evolve new systems and generate adequate data to understand and predict their behaviour under a variety of conditions that could be expected during service. We would be reviewing some of these areas in this article.

Figure 1 shows a schematic of pressurised heavy water reactor power plant. Reactor core is housed in a cylindrical calandria which contains heavy water moderator and reflector at low pressure. A large number of (306 in 235 MWe units and 392 in 500 MWe units) fuel channels pass through the calandria. These channels contain fuel and also flowing coolant at high pressure thus permitting heat removal from the core. The primary coolant circulating pumps circulate the coolant through the core and the steam generators. The heat extracted from the core by the coolant is transferred in the steam generators to raise steam which drives the turbo generator, thus producing electricity. A number of other important systems also have to function. A moderator

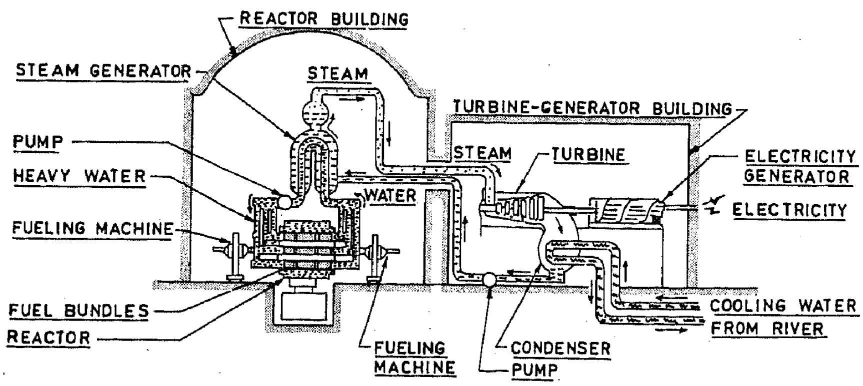
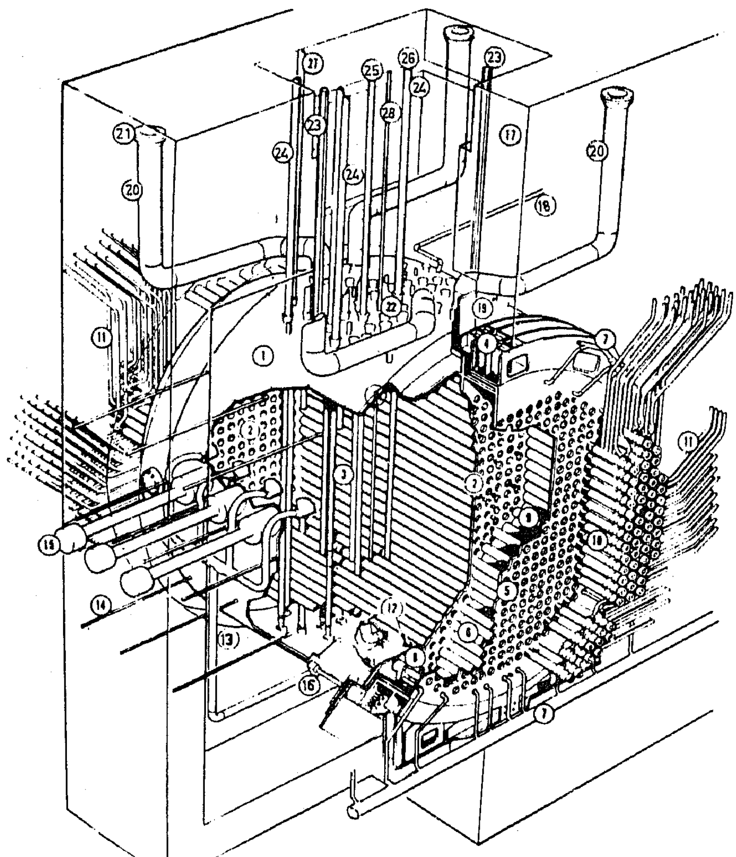


Figure 1. Pressurised heavy water reactor power plant.

heat removal system, on-power fuel handling system and reactor regulating system are some of the important systems. In addition, a number of safety systems are provided to protect the whole system, plant personnel and members of public. A view showing more details of the reactor is shown in figure 2.

The PHWR system has been an ideal choice for our power programme. It is based on natural uranium and heavy water. Technology for both has now been developed and demonstrated on industrial scale. Manufacturing technology for various equipment has also been successfully mastered. The system has a number of safety features not available in other reactor types. These are: (i) The actuators for reactor protection act in low pressure environment unlike the popular PWR or BWR reactors. Thus PHWRS are immune to accidents that can be caused by ejection of neutron absorber from the core. (ii) The excess reactivity of the core is not required to be high since on-power refuelling is available to compensate for the fuel burn-up at any time. (iii) Cold moderator is



1. CALANDRIA
2. CALANDRIA SIDE TUBESHEET
3. CALANDRIA TUBES
4. EMBEDMENT RING
5. FUELLING MACHINE SIDE TUBESHEET
6. END SHIELD LATTICE TUBES
7. END SHIELD COOLING PIPES
8. INLET OUTLET STRAINER
9. STEEL BALL SHIELDING
10. END FITTINGS
11. FEEDER PIPES
12. MODERATOR OUTLET
13. MODERATOR INLET
14. HORIZONTAL FLUX DETECTOR UNIT
15. ION CHAMBER
16. EARTHQUAKE RESTRAINT
17. CALANDRIA VAULT WALL
18. MODERATOR EXPANSION TO HEAD TANK
19. CURTAIN SHIELDING SLABS
20. PRESSURE RELIEF PIPES
21. RUPTURE DISC
22. REACTIVITY CONTROL UNIT NOZZLES
23. VIEWING PORT
24. SHUT OFF UNIT
25. ADJUSTER UNIT
26. CONTROL ABSORBER UNIT
27. ZONE CONTROL UNIT
28. VERTICAL FLUX DETECTOR UNIT

Figure 2. Pressurised heavy water reactor.

always available in the core. In extreme situations this heat sink comes into play in removing heat from the fuel, thus limiting maximum temperature to a lower value. (iv) High pressure reactor pressure vessel is eliminated.

In contrast the channel type design of PHWR presents some safety considerations that are different from those of other designs. These are: (i) provision of adequate cooling to all channels under accident situations (ii) effects of the possibility of fuel coming in contact with the pressure boundary, (iii) introduction of positive reactivity in an unlikely loss of coolant accident event.

The safety criteria to be used for assessment of the system therefore have to consider the above characteristics which are distinct and specific to PHWR. Need for additions or improvements has also to be judged based on these characteristics.

3. Evolution of safety approach

The safety approach has to encompass a number of situations. During normal operation the permissible maximum individual dose limits as well as population dose limits have to be satisfied. In this category one would also include all off normal situations of rather frequent occurrence. Next we have to take care of various accident situations. Clearly both failure probability and consequences have to be considered. The known difficulties with regard to certain accident situations like stagnation breaks, anticipated transients without scram and common cause effects of earthquake have led to the philosophy of two group concepts with each function being provided for by independent systems from each group (see figure 3). With this philosophy it is possible to categorise accident situations in two categories (Atchison *et al* 1983). One category consisting of what are called serious process failures (single failure) and the other consisting of events which are dual failures *i.e.* failure of the process equipment along with failure of a safety system. In the later case accident consequence is mitigated by the second safety system.

The next step is to assign maximum frequencies and the reference dose limits for each of the above two categories and to ensure that possibility of triple failure *i.e.* failure of process as well as safety systems in both groups is made acceptably small *i.e.* around 10^{-7}/yr.

Talking in engineering terms, the following guidelines evolve. (i) The design, construction and operation of all components, systems and structures essential to

	Main control room Group - I	Aux. control room Group - II
Process system	- Reactor regulation - All process systems except aux. moderator cooling	
Safety system	- Shut down system-I - ECCS	Shut down system - II Containment Aux. moderator cooling Emergency water and power supply

Figure 3. Two-group concept for safety system.

safety of the reactor will follow the best applicable codes, standards or practices and confirmed by an independent audit. (ii) The quality and nature of the essential process equipment will be such that the total of all serious process failures should not exceed a specified number. A serious process failure in this context is a failure of process system or equipment that, in the absence of safety system action, could lead to fuel failure or the release of radioactive material to the environment. (iii) The safety systems belonging to two groups will be physically and functionally separate from the process system and from each other and (iv) each safety system will be readily testable and will be tested at a frequency that demonstrates its unavailability to be less than a specified number.

Consideration of various dual failures also leads to definition of performance requirements for safety systems. For example, a loss of coolant accident (LOCA) plus failure of emergency core cooling system will lead to release of fission products from the fuel (the "source term") that must be accommodated by the containment system. Similarly, a LOCA with impaired containment system sets the effectiveness required of the ECCS. Some of the concerns which are not satisfactorily answered following the single/dual failure approach are:

(a) inability to consider large variation in frequency and consequences of different single/dual failures. To overcome this difficulty the event frequency vs permissible dose limit acceptance curve is divided into a larger number of steps;

(b) problems relating to safety system support features such as electrical, air or water supplies whose failure could result in failure of both process as well as safety system;

(c) need to ensure continued operation of safety system after an accident and

(d) need to include consideration of accidents like earthquake which could damage both process and safety system.

A more comprehensive approach to safety evaluation is thus needed. This leads to the next stage of evolution—which is based on use of safety design matrix (SDM). It is essentially a systematic record of visual inspection by the analyst of all selected events of potential safety concern. Extensive use of fault tree analysis and event sequence diagram enables identification of possible causes for the event and also the various consequences. SDM enables better understanding of the system behaviour and has the potential to identify desirable design modifications. This approach can be extended to predict the risk posed by any postulated event sequence.

Progress is being made towards application of probabilistic risk assessment (PRA) techniques. With development of this technique and also appropriate data base it appears that PRA would have a strong influence on licensing decisions in future.

4. Engineering development

Engineering development is a process of continuous evolution. The availability of new technologies, evolution of new safety criteria, experience with design, construction, commissioning and operation of power reactors, availability of new information with regard to design techniques, material behaviour, equipment reliability, understanding of accident behaviour etc all have an impact on the new engineering development effort. As mentioned earlier the PHWR system is expected to perform well both in terms of safety and in terms of its performance as a power unit. However, there is always a scope to improve the system through new development. Let us now review some of the important areas in greater detail.

5. Emergency core cooling system

From the point of view of performance of emergency core cooling system the distinctive features of PHWR cooling system in contrast to the more popular PWR systems should be recognised. There are: (i) The length of large diameter piping is quite small in PHWRS. In contrast the small diameter piping is far more extensive in PHWR. The PHWRS have thus a larger probability for small break LOCA. (ii) The coolant tubes which contain and also support the fuel are horizontal. They can deform into different modes depending on the pressure temperature history. (iii) The primary heat transport system is arranged in a figure of eight configuration (figure 4). Certain combinations of size and location of break can cause stagnation conditions in the core. (iv) Voidage of core introduces positive reactivity.

It is thus clear that the system behaviour in case of LOCA in a PHWR is likely to be quite different from the pressure vessel type reactors for which most of the published literature is relevant. Thus while the design of ECCS goes on based on the best available calculational procedures, an extensive experimental programme to study various effects

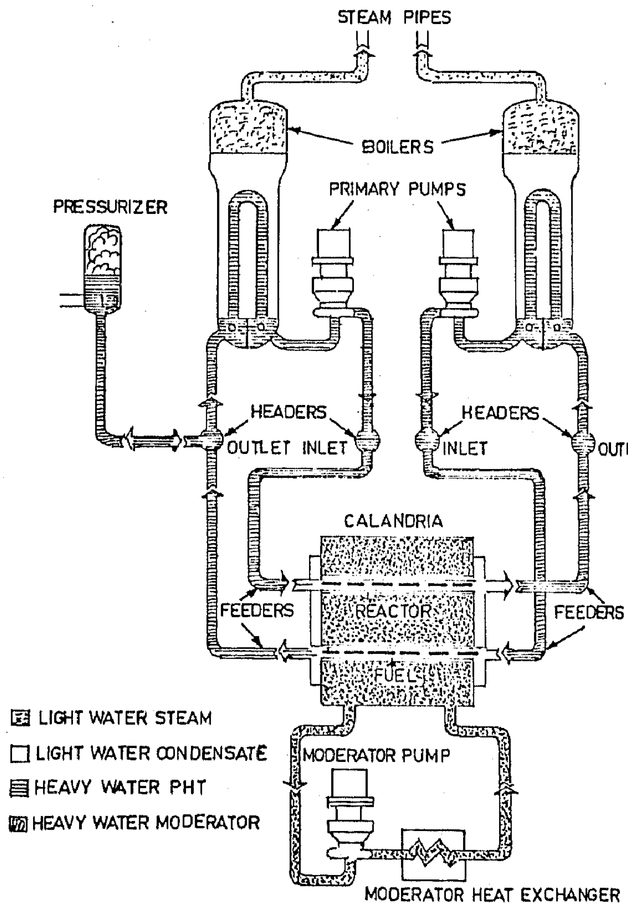


Figure 4. Figure of eight configuration.

that would be manifested during various stages following LOCA is also being pursued. This work can be divided into two broad areas. One relating to the blowdown phase and the second relating to emergency core cooling phase.

In the blowdown phase the main feature that distinguishes the blowdown in PHWR is the fact that the fuel channels in PHWR are horizontal. There is thus need to understand through experiments the various phenomena that are involved in blowdown from horizontal channels. Apart from rigging up the experimental set-up, this also involves development of techniques for measurement of blowdown rates with a variety of break sizes. In parallel to the experimental programme, analytical studies aimed at development of a computer program to calculate transient two-phase flow and heat transfer during blowdown from horizontal pipes starting with subcooled condition and with heat addition need to be pursued (Venkat Raj *et al* 1983).

In case of PHWRs this analysis needs to be extended to analyse the effects of failure of ECCS (Hancox 1981). In such an event the build-up of temperature would lead to deformation of coolant tube which could sag into contact with the calandria tube if the coolant pressure is already low. Alternatively if the pressure has not sufficiently dropped down, the coolant tube can balloon into contact with the calandria tube. The two possibilities are pictorially depicted in figure 5.

Once the contact with the calandria tube is established heat can get dissipated to moderator and the maximum fuel temperatures get decided by the resistances in this heat transfer path.

In the ECCS phase, the specific areas of concern to PHWRs are: (i) problems caused by the injection of cold emergency coolant in voided header such as condensation induced

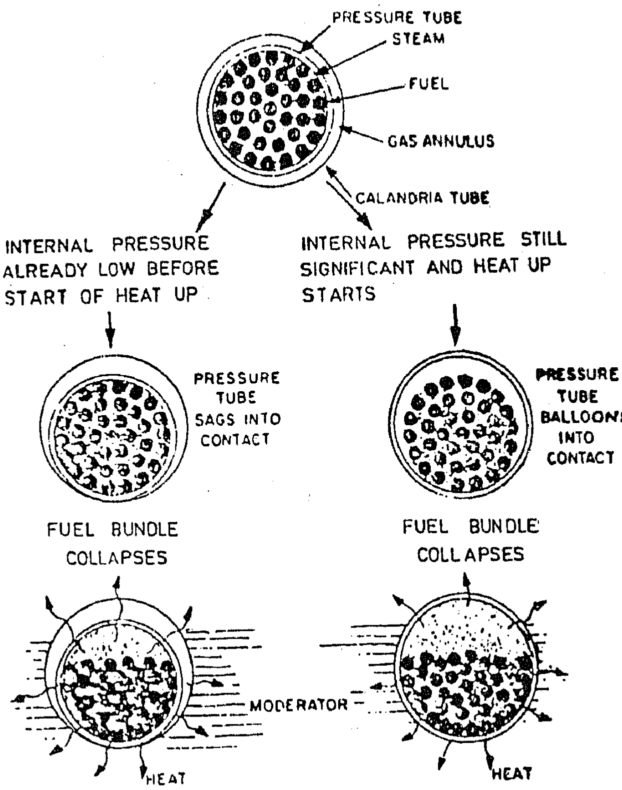


Figure 5. Schematic diagram of the stages of fuel channel heat up and deformation.

flow oscillations; (ii) flow distribution between parallel channels under ECCS conditions. (iii) Rewetting of hot horizontal channels.

Here again extensive experimental data are required to enable analysis of conditions in the core during LOCA. Depending upon the temperature of the fuel in various channels, it is possible that a larger share of the flow injected in the header may get preferentially diverted to the cooler channels contrary to the actual requirements. Model studies are therefore required to investigate this phenomenon and to ensure that all channels receive at least the required quantity of coolant flow.

Considerable work has been done to understand the rewetting phenomenon in horizontal rectangular as well as annular ducts (Venkat Raj 1984). A typical experimentally determined rewetting transient starting with an initial temperature of 500°C is shown in figure 6. Comparison of traces CB, CM and CT indicates an inclined rewetting front. Significant stratification, flow chugging and oscillations in pressure drop have also been observed. A comparison of experimental data on rewetting velocity with theoretical prediction for an annular horizontal duct is given in figure 7. The data collected during such tests can be used to work out a more realistic prediction of the

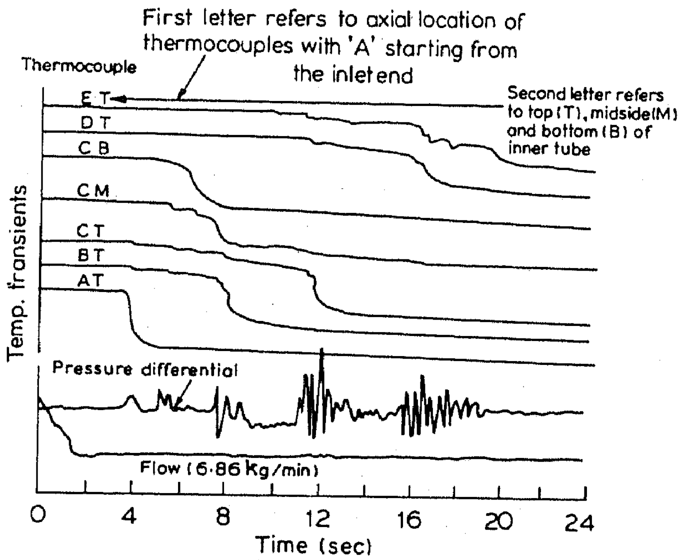


Figure 6. Rewetting transients (initial temperature – 500°C).

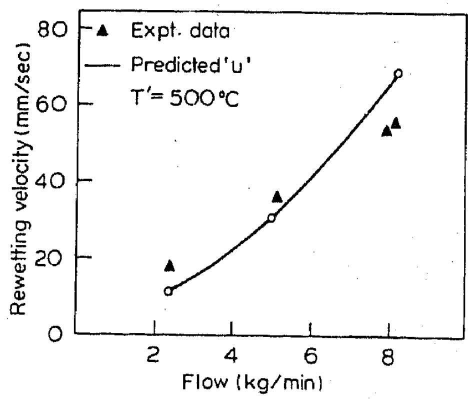


Figure 7. Comparison of predicted rewetting velocity for Ants-2 with experimental data.

integrated system behaviour following a LOCA and the performance of the ECCS. Such a behaviour can also be studied on specially designed experimental set-ups.

6. Thermosyphon

The current designs of PHWRS depend on the thermosyphon for maintaining the circulation through the core under conditions when the main circulating pumps are not running and the system pressure and temperature do not permit bringing in the standby heat removal system (pumps and heat exchangers). While for the new reactors this situation can be avoided by incorporating a standby heat removal system capable of working at full pressure and temperature, a study of thermosyphon phenomenon would be useful both for existing reactor systems as well as future reactor systems. Figure 8 shows a plot of flow (in terms of temperature difference) vs time determined on an experimental set-up simulating the PHWR primary system layout. This plot indicates both initial establishment of flow as well as stability of flow under single phase flow conditions.

Experimentally determined data for channel pressure drop over the full range of Reynolds numbers are shown in figure 9. Similar data need to be generated for the coolant channel of 500 MWe PHWR. Such data are essential for evaluation of safety in a number of situations when flow is less than normal flow.

Study of thermosyphon in the PHWR system under two phase flow conditions also needs to be pursued.

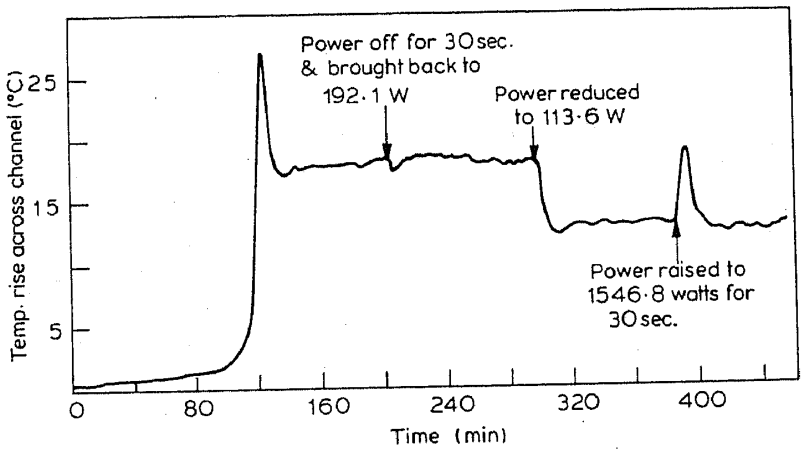


Figure 8. Stability test at different power levels.

7. Zr-2$\frac{1}{2}$ % Nb coolant tubes

Our PHWRS so far use zircaloy-2 as the material for the coolant tubes. This material was also used by Canadians in their earlier reactors. With the development of Zr-2$\frac{1}{2}$ % Nb material which has better strength and creep resistance Canadians switched over to this new material starting from pickering-3 unit. The primary objectives have been to extend the creep life of coolant channels and effect greater neutron economy.

A number of problems have been faced by Canadians in the use of Zr-Nb tubes. Most of them have been caused by delayed hydride cracking caused by faulty rolling

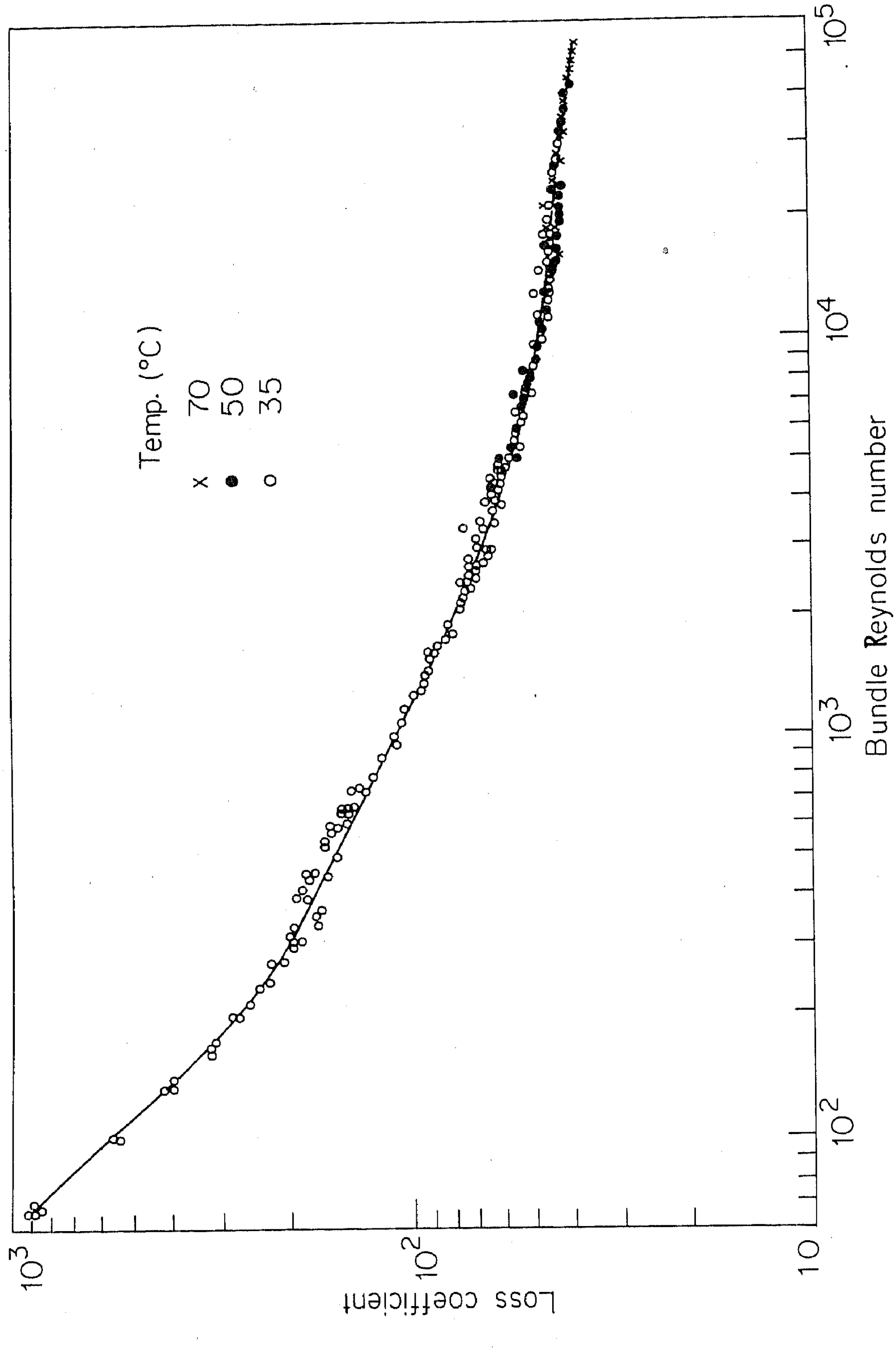


Figure 9. Fuel channel pressure drop data.

procedure. Although all such tubes have been subsequently stress-relieved, it is anticipated that some initial cracking might have been caused during the operating period between faulty rolling leading to high residual stress and the actual stress relieving operation. The cracks initiated during this period would progressively grow and tube failures on this account are expected even in future for initial reactors tubed with Zr-Nb tubes. The mechanism of delayed hydrogen cracking has now been understood well and the new rolling procedures have been developed which minimise the residual stress thereby eliminating delayed hydride cracking and also making the stress relief operation unnecessary.

A new mechanism has come to light consequent to failure of zircaloy-2 pressure tube in Pickering-2. Displacement of garter spring support between the coolant tube and the calandria tube led to sagging of coolant tube in contact with the calandria tube. The cold spot so developed on the outside of coolant tube developed blisters of hydride almost along the line of contact. A fairly long partial thickness crack got generated through cracks in these blisters. When the crack exceeded the critical dimension there was a sudden unstable propagation of crack leading to gross failure of the coolant tube. This event has led to violation of the belief held till then that any failure in the coolant tube would be of leak before break type. On the positive side of this event were the facts that the calandria tube did not rupture in this event and also that the reactor was shut down in an orderly manner demonstrating that such events can be safely handled. A large amount of literature is being published indicating that the hydrogen pick-up rates in Zr.Nb tubes are much less as compared to zircaloy-2 tubes.

In view of the serious safety implications this matter has been very carefully reviewed. A decision has been taken to adopt Zr-Nb coolant tubes for future 235 MWe units as well as the 500 MWe units. Needless to say that additional features like increase in number of garter springs, use of gartersprings immune to unwanted displacement etc have also to be looked into at the same time.

An extensive programme is already underway to develop and evaluate Zr.Nb coolant tube production at the Nuclear Fuel Complex, Hyderabad, and to develop a coolant channel based on this new material. Clearly the most crucial activity in this work is the development of rolled joint which would not pose any delayed hydride cracking problem.

8. Structural safety

Failure of piping and other important structures could have significant safety implication. For example LOCA is caused by a sudden failure of piping or some other component in the primary heat transport system. Development of an identifiable leak well before sudden fracture is a preferable situation as it generates an early warning to enable corrective action. Dependable in-service-inspection of components to identify development of cracks or defects and their early repair is even more desirable. Apart from development of techniques for leak detection, non-destructive examination and *in situ* repair we need to be able to assess areas prone to development of defects, assess growth of defects and the associated safety margin. This requires work to be pursued in area of stress analysis particularly with regard to residual stresses, fracture mechanics, fatigue and life assessment etc. Some examples of work done in this area are shown in figures 10 to 12. Figures 10 and 11 show the stress transients during longitudinal welding of a cylinder and its subsequent annealing. The residual stress distribution after welding and its reduction due to annealing operation can also be seen in these figures. Effect of existence of residual stress on the component life is depicted in figure 12 where it can be seen that with existence of tensile residual stress, the cyclic life gets considerably reduced (compare case 1 with case 2). Further with compressive residual

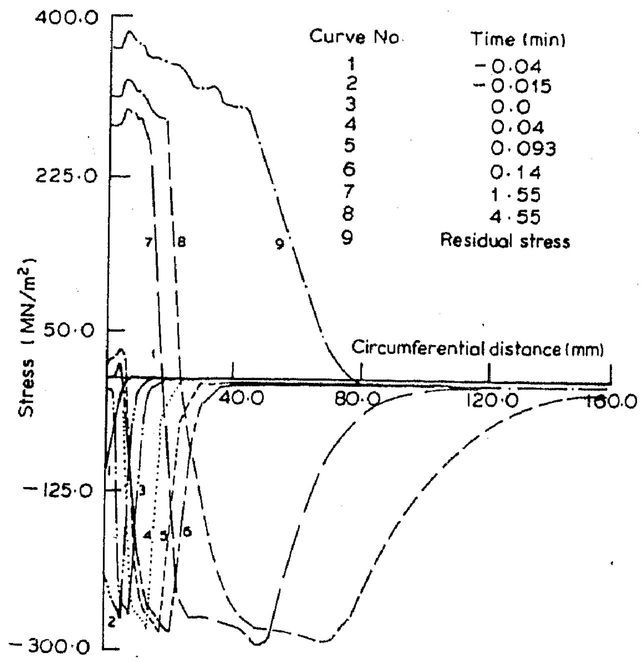


Figure 10. Stress transients during cylinder welding.

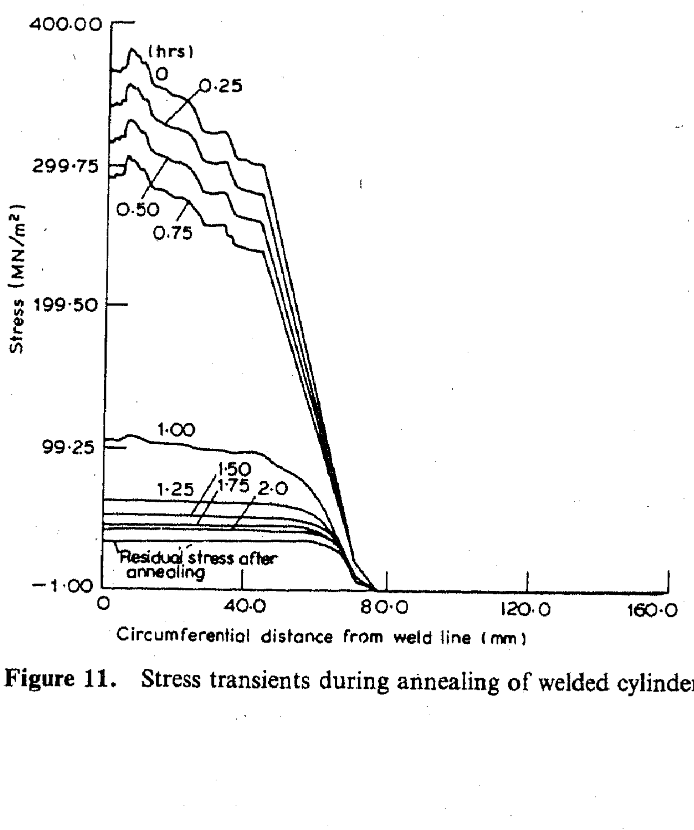


Figure 11. Stress transients during annealing of welded cylinder.

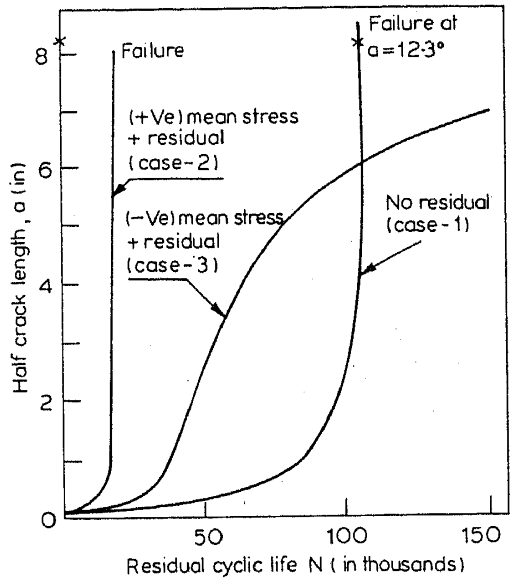


Figure 12. Effect of residual stress on cyclic life.

stress the life can be increased (compare case 1 with case 3). This forms the basis of development of various techniques like induction heating for stress improvement, heat sink welding, mechanical pipe lock etc, which are aimed at delaying the component failure.

Even assessment of crack growth and the critical crack length becomes considerably difficult with complications in geometry. Considerable work has been done in this area in view of the assessment need with regard to embrittlement of end-shields in RAPS-I, II and MAPP-I, as well as the feed water nozzles in TAPS.

Another aspect of structural safety pertains to the behaviour of structures to seismic excitations. Figure 13 shows a typical structural model of *Dhruva* pile block used for seismic analysis. A particular deformed mode shape is shown in figure 14. Although sufficient capability exists in this area, work is required to be pursued to enable nonlinear analysis and water sloshing studies.

9. Secondary shutdown system

The current design of PHWRS starting from Narora reactor incorporates a fast acting secondary shutdown system. The system consists of a number of verticle tubes running through the reactor through which a liquid poison is injected. The number of positions available for this purpose is limited due to severe congestion of various nozzles on top of the calandria. The maximum negative reactivity worth of such a system is thus restricted.

An alternate system which can inject the liquid poison in moderator is therefore considered attractive and needs to be developed. With such a system the tubes running through the core for poison injection can be kept horizontal thereby permitting effective spatial separation between the two shutdown systems (see figure 15). This system is expected to be much simple and not so severely limited in negative reactivity worth.

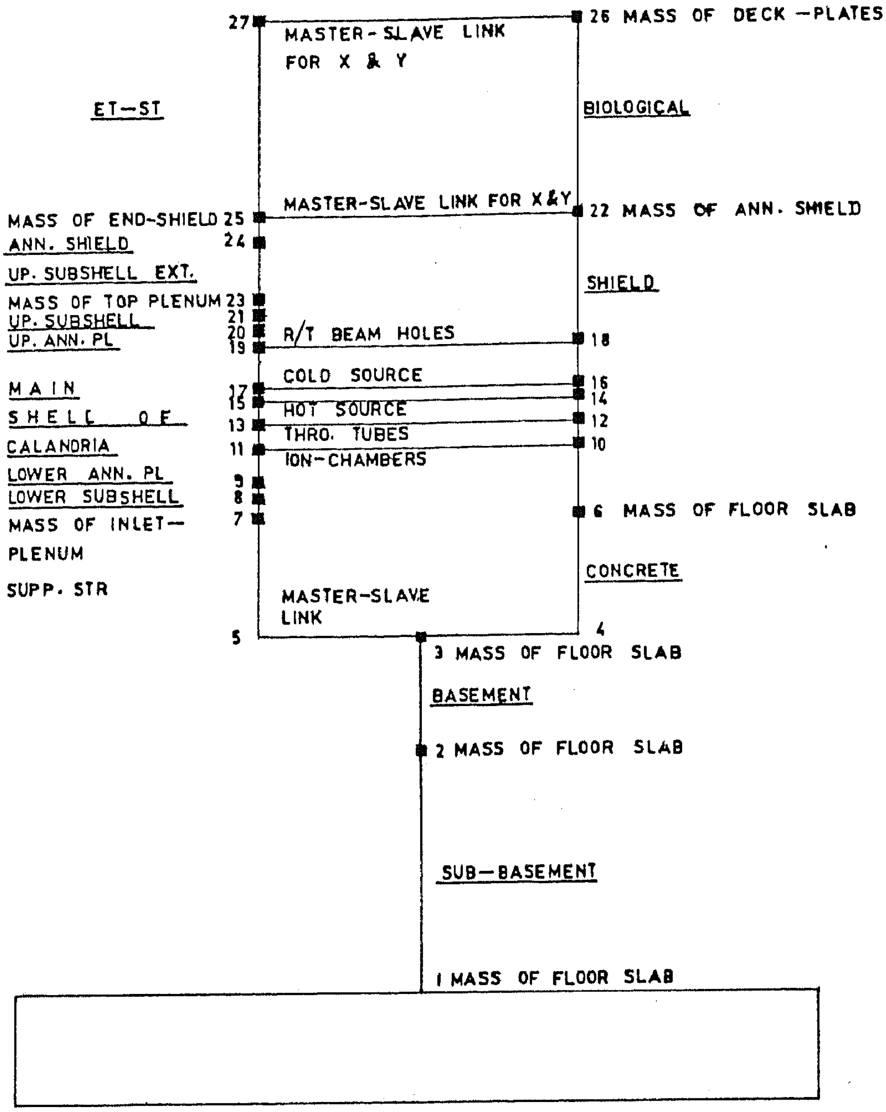


Figure 13. Model of *Dhruva* reactor for earthquake analysis.

10. Pressuriser

Pressure control in current PHWRs is maintained by a feed and bleed system. This system can cope with transients to a limited extent and also requires steam side pressure to be regulated depending on the reactor power to minimise the swell/shrinkage on the primary side. The severe fluctuations that are possible with our grids could lead to unacceptable pressure fluctuations causing reactor trip. A need has therefore been felt to incorporate a pressuriser which would damp out the pressure fluctuations and maintain them within the desirable limits. It is expected that incorporation of a pressuriser would lead to more reliable performance of the reactor unit.

We have reviewed various items of engineering development which could lead to improved safety characteristics of the system. The coverage is by no means complete. Further, there are a large number of areas of development which are aimed at improving the performance. Some of the important ones are: (i) improved fuel with

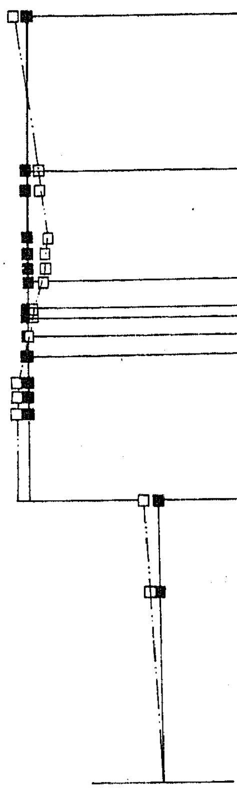


Figure 14. Plot of fourteenth mode of model of *Dhruva* reactor.

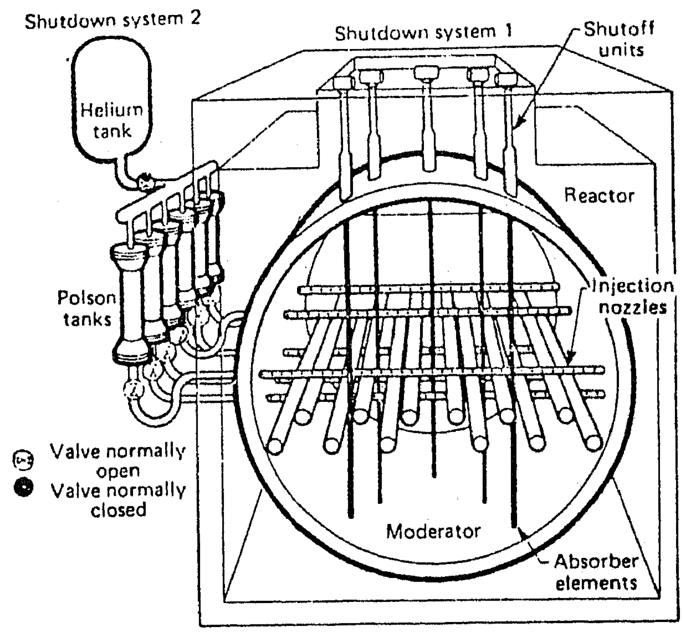


Figure 15. Secondary shutdown system.

graphite lubrication which would minimise pellet clad interaction thereby improving fuel performance particularly under transients; (ii) fuel handling system which is one of the most sophisticated systems and key to successful performance of the reactor plant and (iii) partial boiling in channels which would enable larger power to be extracted through the same reactor unit.

With accumulation of more and more reactor years of operation the need to carry out maintenance as well as some of the planned activities like channel adjustment, channel replacement, steam generator tube plugging, repair on piping system etc would grow. Since these activities require a lot of man-rems it is necessary to develop methods which can enable performance of these activities in minimum time and with minimum of man-rem expenditure.

In conclusion one can say that the existing R & D base can match the needs of the identified power programme. The efforts should be continued to enable further improvements. Some of the thrust areas particularly having a bearing on safety have been identified. These are considered important for realisation of better PHWR system.

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