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REACTIVITY REQUIREMENTS AND SAFETY SYSTEMS FOR HEAVY WATER REACTORS

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Abstract

The natural uranium fuelled pressurised heavy water reactors are currently being installed in India. In the design of nuclear reactors, adequate attention has to be given to the safety systems. In recent years, several design modifications having bearing on safety, in the reactor processes, protective and containment systems have been made. These have resulted either from new trends in safety and reliability standards or as a result of feed-back from operating reactors of this type. The significant areas of modifications that have been introduced in the design of Indian PHWR's are : sophisticated theoretical modelling of reactor accidents, reactivity control, two independent fast acting safety systems, full double containment and improved post-accident depressurisation and building clean-up.

This paper brings out the evolution of design of safety systems for heavy water reactors. A short review of safety systems which have been used in different heavy water reactors, of varying sizes, has been made. In particular, the safety systems selected for the latest 235 MWe twin reactor unit station in Narora, in Northern India, have been discussed in detail. Research and Development efforts made in this connection are discussed. The experience of design and operation of the systems in Rajasthan and Kalpakkam reactors has also been outlined.

INTRODUCTION

The pressurised heavy water reactors in the 220/235 MWe sizes are currently being built in the Indian nuclear power reactor programme. Starting with twin units of 220 MWe each of Rajasthan Project, and then two 235 MWe units each in Madras and in Narora projects, several design modifications have been made in the safety systems in order to improve the safety and reliability of operation. The most significant changes made are in the reactor shut off systems of Narora, where two shut off systems are being provided. There are major changes again in the containment systems of Madras and Narora reactors, where modified vapour suppression systems are incorporated in preference to the dousing systems of RAPS. The philosophy underlying these changes are brought out in this paper and our current thinking for the thermal research reactor project (100 MWt research reactor) and the future 500 MWe reactor are also brought out.

SHUT-OFF SYSTEM

General

Dumping of moderator has been the primary shutdown mode in some of our heavy water power reactors. This is however not an effective device against fast reactivity transients due primarily to its slow reactivity removal rate. In recent designs, insertion of solid and liquid absorber into the reactor core is employed as fast acting safety systems.

For reactors presently under construction, solid shut off rods constitute the primary shut-down mechanism. Rope and pulley arrangement is adopted for these devices. The rod is tied to a steel rope which is wound on a drum. When the reactor is operating the rod is parked above the core and the drum is held in position by means of an irreversible gear system through a magnetic clutch. On a trip signal the clutch is deenergised and the rod drops under gravity.

The secondary shutdown system is based on a different principle of operation from the primary system. This system injects absorber solution (poison) into a set of tubes fixed inside the core vessel. Since the secondary system is actuated only in the event of malfunction of the primary system, this is made faster in operation.

Long term shutdown for these reactors is achieved by either moderator dumping or addition of suitable amount of poison in the moderator.

The safety instrumentation for these reactor follows a two out of three coincidence logic to strike a balance between safety and serviceability. The evolution of shut-off systems for various heavy water reactors in India is presented in Table I.

Physics Calculations

Heavy water moderated natural urnaium fuelled reactors fall midway between two extremes namely, reactors with inherent power limiting capability through strong negative reactivity coefficients and those with positive power coefficients.

So far as the worth of a safety system is concerned there are two requirements to be met. First, any rising power transient resulting from an accidental introduction of positive reactivity must be terminated within a time period small enough to keep the net energy release within safe limits. Second, even after the immediate power surge is terminated a large shutdown margin should be maintained so that the possibility of further danger is avoided. It is not necessary that both these functions are met by the same device.

The physics calculations involved in this study of the safety systems are (a) for obtaining the static reactivity worths of the systems and (b) for studying transients for various reactivity inputs preferably including the spatial effects.

In case of the 235 MW(e) NARORA atomic power unit, the state of the reactor with maximum possible positive reactivity corresponds to the cold clean reactor with fresh fuel at room temperature when boron free moderator fills the calandria. In this state the reactor will have an excess reactivity of 100 mk. However, it is not possible for this large reactivity to be added into the system in a short time. The two shut down devices (Figure 1) viz. (a) 14 shut off rods (b) 12 liquid poison tubes have been designed to independently shut down the reactor in the event of the maximum credible accident viz. that of rapid voiding of coolant from the core ($\sim$ 2 secs.). In this case, the shut off system should be able to compensate for the positive reactivity addition coming from (a) the coolant voidage effect a maximum of 9 mk with fresh fuel (b) the power coefficient effect when the chain reaction is stopped whose maximum value is again 9 mk with fresh fuel. In addition there should be adequate safety margins to account for a couple of rods failing to drop.

The physics model used for obtaining the static reactivities consists of standard two group diffusion theory equations being solved in 3 dimensions with a smeared additional thermal absorption cross section $\delta\Sigma_a$ for the fuel bundles (in the cell homogenised representation together with their associated coolant, pressure tube and moderator) adjacent to the shut off rod or liquid poison tubes. The value of $\delta\Sigma_a$ was chosen so as to reproduce the reactivity worth of a rod along the central axis of the reactor as calculated by Nordheim-Scalletter method(1). The rod was assumed to be black to thermal neutrons and completely transparent to epithermal ones. Thus it is hoped that our model will err on the safer side. With this model, a number of patterns for primary and secondary shut down systems were examined, which were consistent with the space requirement for housing the drive mechanisms for shut off rods and other reactivity devices on the top of the reactor. The reactivity worth was found to be 32 mk for the primary system of 14 shut off rods while 12 liquid poison tubes had a worth of 36 mk.

In order to study the effect of some of the rods failing to drop on the worth of the remaining rods, we divided the shut off rods in three groups (Fig. 1) (i) Group A consisting of 4 rods lying 60 cm away from the central plane (perpendicular to axis of the cylinder) on either side; (ii) the groups B and C consisting of 5 rods each, clustered around 120 cm. from the central plane on either side. The worth of group A rods alone is 10 mk. while that of groups B and C individually is 7.5 mk and together (in the absence of Group A) is 17 mk. Similarly, the combined worth of groups A and B is 18.5 mk., while that of 7 rods on one side of the central plane is 11 mk. Thus due to flux peaking effects, away from the absorbers, the combined worth of all the rods is significantly higher than that of individual groups. However no two rods failing to drop (and thus leading to an asymmetric arrangement of shut off rods) is likely to account for more than 7-8 mk of reactivity and the remaining rods will adequately compensate the positive reactivity addition from two effects mentioned above.

In order to study the transients and their spatial effects we have analysed the loss of coolant accident quenched by various shut of systems. The results are presented for the moderator dump which is the slower than either of the two shut off systems available in NAPP reactor and thus leads to the maximum over power transient. We analysed this accident using a point kinetics model NARD (without feedback option) and the one dimensional space time kinetics code RAUMZT with two groups diffusion equations for the spatial mode. The code RAUMZT does not have feedback option. While it is true that to study the spatial effects one must employ three dimensional diffusion equations and include feed back effects, our comparison does show that point kinetics equations are over conservative. Thus we have found that at no point the over power reached in RAUMZT is more than that predicted by NARD (for the average power)

The reactor was simulated by one dimensional slab along the central diameter of the reactor. The positive reactivity insertion is done uniformally over the core by increasing $\nu\Sigma_f$ in a linear ramp adding 5 mk in 1 sec. The dump is initiated with 300 m. sec delay from the time reactor power reaches 110% of initial value. As dumping proceeds, the transverse buckling increases initially slowly but later steeply. However, the total power falls to below 90% of the initial power after 2.75 secs of the initiation of the transient at a water level of 480 cm when the buckling has increased by 10%. Even so, we have assumed the transverse buckling remains constant, thus making the spatial analysis rather conservative. The flux profiles are shown in the Fig. 2. One notices that at 1.2 secs when the power peaks, NARD over predicts the power all along the reactor because it does not take account of the increased transverse leakage due to dumping.

CONTAINMENT DESIGN

The evolution of containment designs for the Indian PHWRS from RAPS to MAPS and on to Narora and the concepts being evolved for the 500 MWe reactor are presented in this section. See Table II.

The RAPS reactor building which consists of a 1.2m thick cylindrical perimeter wall of R. C. C. and a hemispherical dome of prestressed concrete enclosing a volume of nearly 43000 m^3 has a design pressure of 0.42 kg/cm^2. The specified leakage rate from the building is 0.1% of the contained volume per hour at the design pressure. The pressure suppression during the course of LOCA is effected by means of a dousing water system. This system consisting of a tank of 1.82×10^6 litres capacity located at the highest elevation in the reactor building, when set off by a high reactor building pressure signal, establishes a curtain of water. It is estimated that about 33% of the energy of the flashed steam would be absorbed by the dousing water leaving the remaining 67% amounting to nearly 5.1×10^6 Kcal to be taken care of by building cooling and post-accident depressurisation devices. The post-accident depressurisation systems operate on a reliable class III power supply and are together able to reduce the building pressure to nearly atmospheric over a period of 2-3 hours.

In the Madras atomic power station, significant design changes have been introduced in reactor containment. Detailed analysis indicated that the dousing system of RAPS needs improvement due to overall lower reliability and there is a finite probability of spurious operation causing flooding of the reactor building basement resulting in diminished operational reliability of the station. The dousing system of RAPS is replaced in MAPS and future Indian PHWRs by a modified vapour suppression system. The reactor building is divided into two accident based volumes V1(drywell) and V2 (wetwell) separated by leaktight walls and floors. Also, the depth of the basement has been increased to accommodate an additional floor below the lowest working floor. This floor is filled with 2.4 meters deep suppression pool water. Volumes V1 and V2 are connected by means of a vent shaft/distribution header system via the suppression pool water.

In the event of LOCA, the pressure rise in V1 (which houses the high enthalpy steam and PHT circuitry) causes steam-air mixture to be driven down the vent shafts and discharged into the suppression pool 1.2 m below the water surface. All the steam gets condensed here, while the cooled air rises into volume V2 areas.

The improvements in the design of suppression pool of Narora reactors are shown in Table III. It may be seen that the drywell to wetwell volume ratio is reduced from 2.4 in MAPP to 1.4 in NAPP. A further improvement in this respect is envisaged for the containment of future

500 MWe reactors, where the ratio will be as low as 0.1. This improvement is reflected in greater fraction of the flashed coolant and energy getting absorbed in the suppression pool water, the figure being as high as 91% for the proposed 500 MWe reactors.

Another significant improvement in the containment design is in the double envelope concept for reducing net leakage to the off-site. In case of RAPS, there is only a perimeter pressure cum shield wall specified for 0.1% per hour leak rate. In MAPP, there is an outer envelope also with an annulus air space between the two walls. However, the outer envelope covers about 80% of the containment surface. The annulus space is maintained at a slightly negative pressure of 0.35 cm H_2O by means of exhaust blowers which will discharge via filters to stack. Under such conditions, it is expected that the maximum leakage rate though the containment will not exceed 0.02% per hour at the design pressure of 1.16 kg/cm^2, which is 20% of the specified leak rate for RAPS. The use of prestressed concrete inner wall is a significant improvement in containment designs.

In Narora the concept of double containment is extended to cover the entire surface of the reactor building (base slab excepted). The design pressure of the inner envelope is 1.15 kg/cm^2. The leakage rate specified for inner envelope is 0.1% of the containment volume per hour. For the outer wall of reinforced concrete the specified leak rate is 0.1% per hour at 1 psig. But since the entire containment has a double envelope, which will be purged as in MAPP, the net leakage will be reduced by a very large factor, an improvement of a factor of about 100 over RAPS, and improvement of about 20 over MAPP.

For the 500 MWe reactor, the layout proposed is such that the wetwell will completely envelope the drywell. Thus, in effect, the drywell will have a double envelope while the wetwell will have a single envelope. The net result is that we are able to obtain effectively an extremely leak tight containment without going in for costly steel vessels.

Containment system experiments have been planned to validate the computational models. Phase I experiments have been completed and the assumption of 100% steam condensation for our design conditions is established. Phase 2 experiments in which the simulated rupture of the PHT system will also be attempted, are under way. It is expected that vapour suppression type containment will be standardised for the future Indian PHWRs.

TABLE I

SHUT OFF SYSTEMS FOR INDIAN HEAVY WATER REACTORS

Reactor	Primary System	Secondary System
CIRUS	6 solid B_4C shut off rods - gravity fall worth 50 mk	Moderator dump upto 67% level worth 40 mk
R-5	9 solid B_4C shut off rods - gravity fall worth 90 ml	20 Liquid poison tubes 25000 ppm Gd $(No_3)_2$ worth 100 mk Back up - Moderator dump upto 40% level 130 mk
RAPS	Full moderator dump	--
MAPP	-do-	--
NAPP	14 solid shut off rods – gravity fall – worth 32 mk	12 Liquid poison tubes Boric acid in D_2O - 20 gm/lt. Worth - 36 mk Long term shut down - manual addition of suitable boron amount.

TABLE II

SALIENT DESIGN PARAMETERS OF THE CONTAINMENTS

Parameter	RAPP	MAPP	NAPP	500 MWe
Design pressure, $kg/cm^2 g$	0.42	1.16	1.15	0.9
Test pressure, $kg/cm^2 g$	0.53	0.91	1.14	1.13
Peak temperature, °C (Accident condition)	77	96	120	113
Leak rate, % per hour	0.1	.02	.001	.001
Estimated reliability (unavailability, yrs/yr)	2×10^{-2}	10^{-2}	10^{-2}	10^{-2}
Means of pressure suppression	Dousing Tank	Vapour Suppression	Vapour Suppression	Vapour Suppression

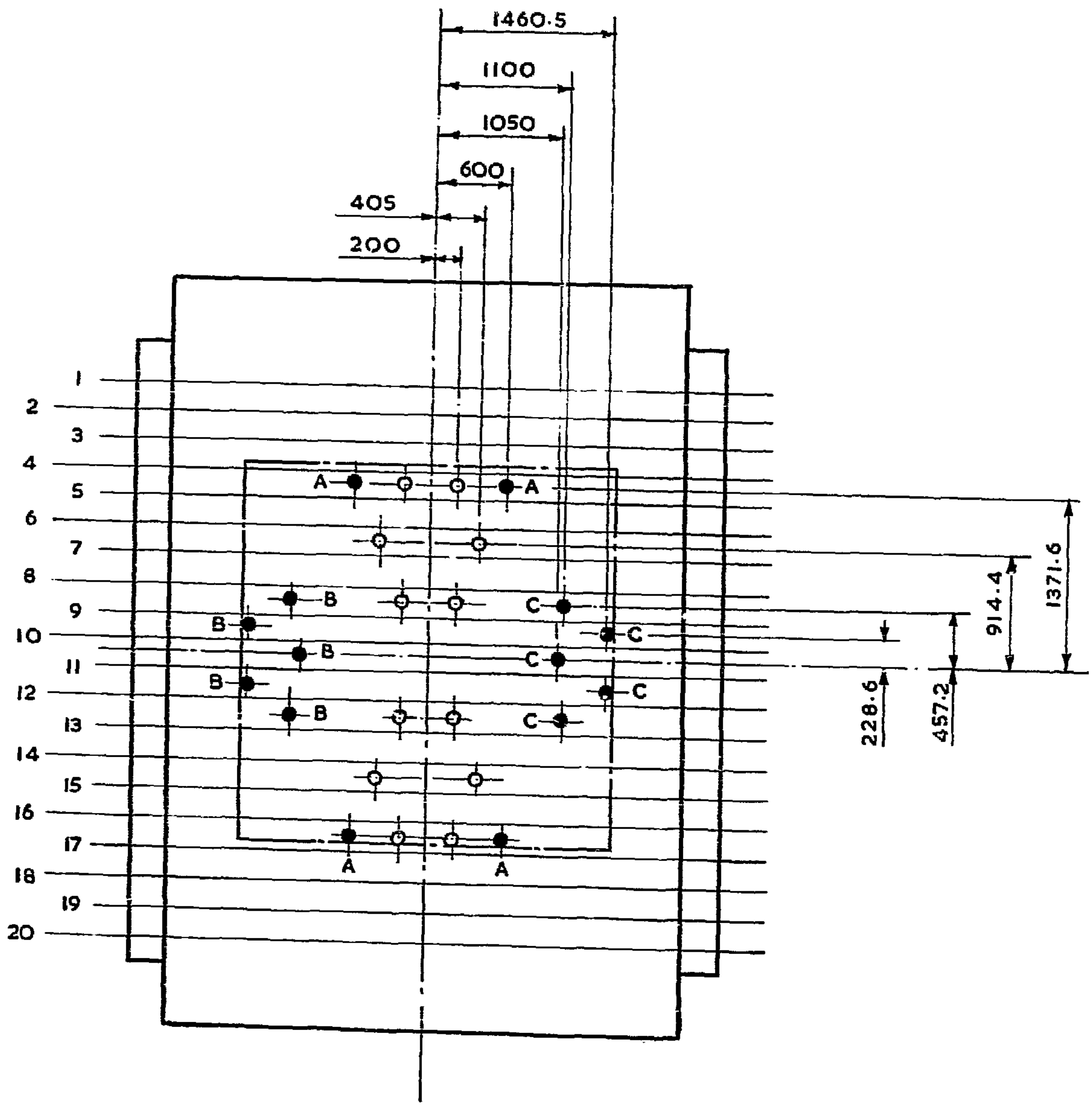
TABLE III

COMPARISON OF VAPOUR SUPPRESSION SYSTEMS OF INDIAN PHWR'S

Parameter	MAPP	NAPP	500 MWe
Dry well volume (V_1), M^3	34, 000	18, 800	4, 500
Wet well volume (V_2), M^3	14. 100	13, 400	40, 000
Dry well/wet well ratio	2. 4	1. 4	0. 1
Energy flashed with PHT coolant kCal	9.1×10^6	12.3×10^6	25.9×10^6
% of flashed energy absorbed in pool water	14. 7	30. 8	91. 5
Peak pressure in Dry well, kg/cm^2 g	0. 73	1. 07	0. 9
Peak pressure in wet well, kg/cm^2 g	0. 7	1. 1	0. 3

NARORA SHUT DOWN SYSTEMS

(CALANDRIA PLAN VIEW)



● SHUT OFF RODS 14

○ LIQUID POISON TUBES 12

FIG - 1

INSTANTANEOUS POWER PROFILES AT DIFFERENT TIMES.

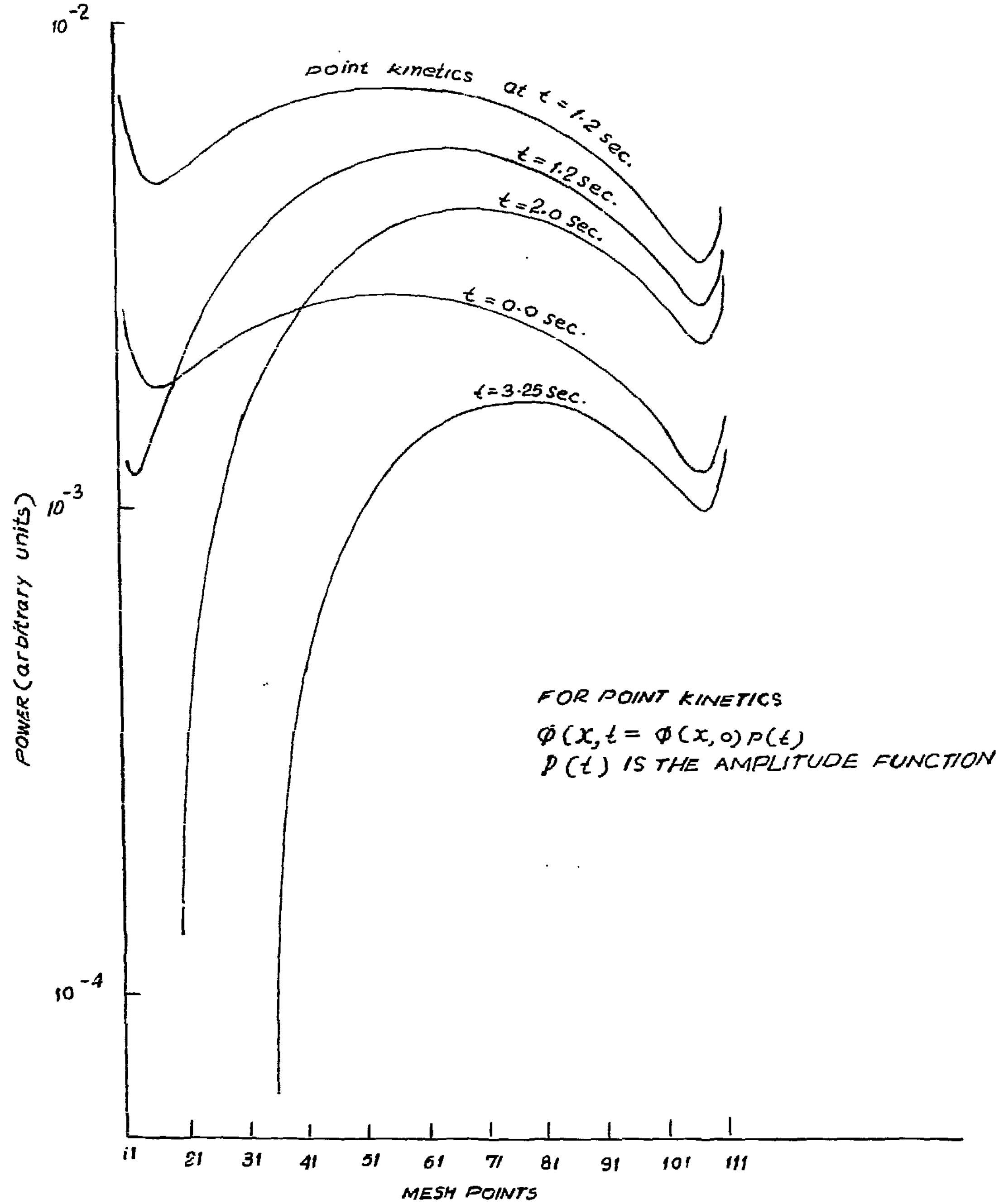


FIG - 2