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Study on the Evaluation and Simulation of Steady State Behavior and Reactor Safety Concept for Integral Pressurized Water Reactor

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Abstract: In this study, a research has been carried out on the normal operational state concerning with the reactor under different load changes as well as simulation of pool type reactor concept by the use of thermal hydraulic system code Relap5/Mod3.4 for Integral Pressurized Water Reactors (IPWR's). These reactors belongs to the class of small and medium sized nuclear reactors (SMR's) and are designed in such away that most of the primary loop components are housed in a single Reactor Pressure Vessel (RPV). In this study, the reactor under study is the Inherent safe uranium zirconium hydride nuclear power reactor INSURE-100. Since, the conceptual design study phase of this reactor has been completed so the current research focuses on the normal operation which is inter related with different load characteristics as well as for the safety concept the considered reactor is simulated as a pool type. The effect of load changes on the different components of the reactor have been studied and are figure out in the graphical as well as in a parametric way. The current study also focuses on the simulation of pool type reactor concept, for this case the pressurized containment, which is partially filled with water and acts as a suppression pool. Finally the load percentage results and the steady state results obtained from the pool type reactor concept have been extracted from the graphical approach and are depicted in a parametric way and are sorted out in a table. The purpose of conducting the research is to evaluate the steady state behavior as well as safety analysis of the reactor.

Key words: Normal operational conditions, different load percentages, pool type reactor concept, relap5/Mod3.4, graphical explanation, parametric analysis

INTRODUCTION

Integral Pressurized Water Reactors (IPWR's) are basically Small and Medium Sized Nuclear Reactors (SMR's) that have been characterized by the addition of components like pressurizer, steam generators, pumps etc which are located in the same Reactor Pressure Vessel (RPV). These reactors are most efficient with reduced cost and are well suited for developing countries especially for the marine applications (Kuznetsov, 2005) whereas large nuclear power plants cannot accommodate the electrical grid crises (Abu-Khader, 2009; Khan *et al.*, 2010). The reactor under study has the power output of 100MW and named as Inherent Safe Uranium Zirconium Hydride Nuclear Power Reactor INSURE-100 (Shulin *et al.*, 1994) except the inclusion of Passive Residual Heat Removal System (PRHRS) (Gou *et al.*, 2009) which is composed of water tank and is situated outside the Reactor Pressure Vessel (RPV) and has the function of absorbing core residual heat in case of transient conditions. The conceptual design study of this reactor has been finished but it is not subjected to normal operational, so in the current research the study focuses on the normal operational of the reactor under the load changes from 100% FP to 10% FP and in order to get more safety concept the considered reactor has been subjected to make the pool type reactor which is based upon the Multi Application Small Light Water Reactor (MASLWR) (Fisher *et al.*, 2003). This all work is done by the use of thermal hydraulic system code Relap5/3.4 (Nuclear Safety Analysis Division, 2001). For this purpose, first the study is conducted on the effect of varying load percentages upon the different components of the reactor such as core, pressurizer, Once Through Steam Generator (OTSG). These components are interlinked with the mass flow rate, inlet, outlet, average temperature and pressure conditions of the primary and secondary system. Concerning with the OTSG the major parameters discusses are the temperature distribution, void fraction and saturation temperature of

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the secondary side of the OTSG. The above mention parameters have been studied by using the thermal hydraulic system code i.e., Relap5/Mod3.4 and are well explained by considering the mathematical and graphical approach that lead to extract the parameters which are presented in the study.

REACTOR DESCRIPTION

In this study, research on the Integral Pressurized Water Reactor (IPWR) that has the power output of 100MW has been performed as shown in Fig. 1. In this reactor, OTSG's are placed in the annular space between core barrel and the reactor pressure vessel and twenty OTSG's have been modeled and the pressurizer is placed at the top doom. The description of the reactor is such that feed water is entering the reactor and collected in a small water tank in the annular space between the core barrel. The reactor pressure vessel from where it move towards the secondary tube of the OTSG where it exchanges heat from the primary side of OTSG and steam is going outwards to the secondary loop as shown in Fig. 1. The key feature of this reactor is that the water flows without the use of pump. It works on the phenomenon of natural circulation. The primary water flows downward and collected in the down comer where it enters the core and the temperature of the water increases and then it rises upward through the riser above the core from where it moves towards the pressurizer and then go back towards the OTSG. In this reactor, Passive Residual Heat Removal System (PRHRS) is modeled which is composed of water tank and the heat exchanger so that residual heat can be compensated easily incases of accident vulnerabilities. For the simulation of pool type reactor RPV is submerged in a pressurized suppression pool, which is partially filled with water and it, consists of annular space bounded by exterior surface of RPV and inner surface of containment walls. And it is integrated to the long-term removal of decay heat during blowdown conditions (depressurization). This whole arrangement is then inserted into the large pool of water, which serves as ultimate heat sink as shown in Fig. 2. At the top of the pressurizer doom there is connected an Automatic depressurization system valve which can infer postulated pressure transients. The sudden increase in the RPV results in opening of ADS valve that will make the reason of opening the sump recirculation valve, which is connected at the bottom of the RPV. The main advantage of this suppression pool is that it quickly maintained the pressure in the RPV and the containment thus eliminating the blowdown condition effectively. As soon as the pressure in both containment becomes equalized the

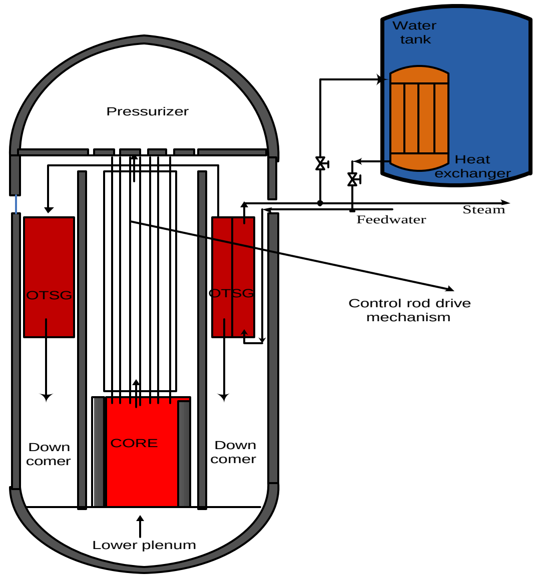

Fig. 1: Schematic diagram of IPWR

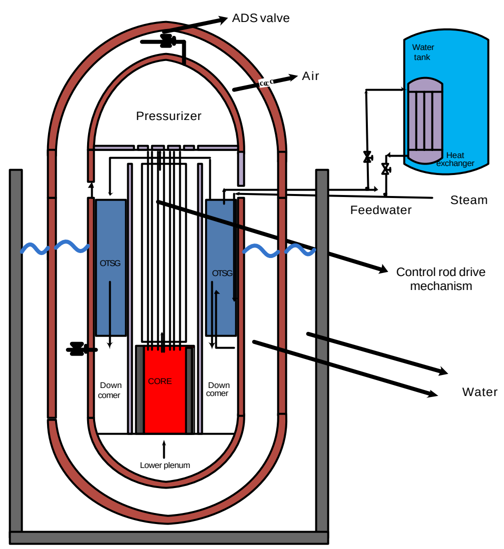

Fig. 2: Schematic diagram of pool type IPWR

natural circulation flow path is established, the sump valve opens and descend the water to the lower portion of the RPV which then rises through the core and finally comes out through the ADS valve as saturated vapors. From the ADS valve the water returns to the sump due to the condensation through the containment walls and thus completing the natural circulation.

RELAP5/MOD3.4

More and more attentions have been given to the thermal hydraulic systemic codes from the last few decades for the analysis of transient and accident conditions in nuclear power plants. In this study, normal operational as well as the steady state results are obtained by using the thermal hydraulic system code i.e., Relap5/Mod3.4 which is used for transient simulation of Light Water Reactor's (LWR's) during postulated accidents. This code has been developed at the National Engineering Laboratory (NEL) for the US Nuclear Regularity Commission (NRC). This is highly generic code and can be used for the simulation of wide range of thermal hydraulic phenomenon including both nuclear and non nuclear system such as steam, water or non condensable gases etc.

NODALIZATION DIAGRAM EXPLANATION OF REACTOR

The reactor is modeled by making its nodalization diagram first as shown in Fig. 3 and 4. The reactor is designed to be composed of four loops. For the simulation of pool type reactor there is connected one additional loop representing the suppression pool and ultimate heat sink. The concept of the reactor is that the secondary component water i.e., coolant coming from feed water pump enters the RPV through the time dependent volume TDV-208 as the name implies, it totally depends upon time. It then moves towards the down comer pipe 204P and collected into the branch 202B which is considered to be a big tank from where it enters the secondary side of OTSG through annulus 200A. The down comer is divided into ten equal parts so that it can simulate easily in the same way steam generator has been divided into twenty equal parts. When the coolant is passing through the annulus region it will transfer heat through the pipe components 134P and 136A, respectively, which are primary down comer components. The steam is ejected to the 232B component and then transfers to turbine components in the secondary loop through time dependent volume 236TDV for generating electricity. The coolant goes down from the 134P and 136P

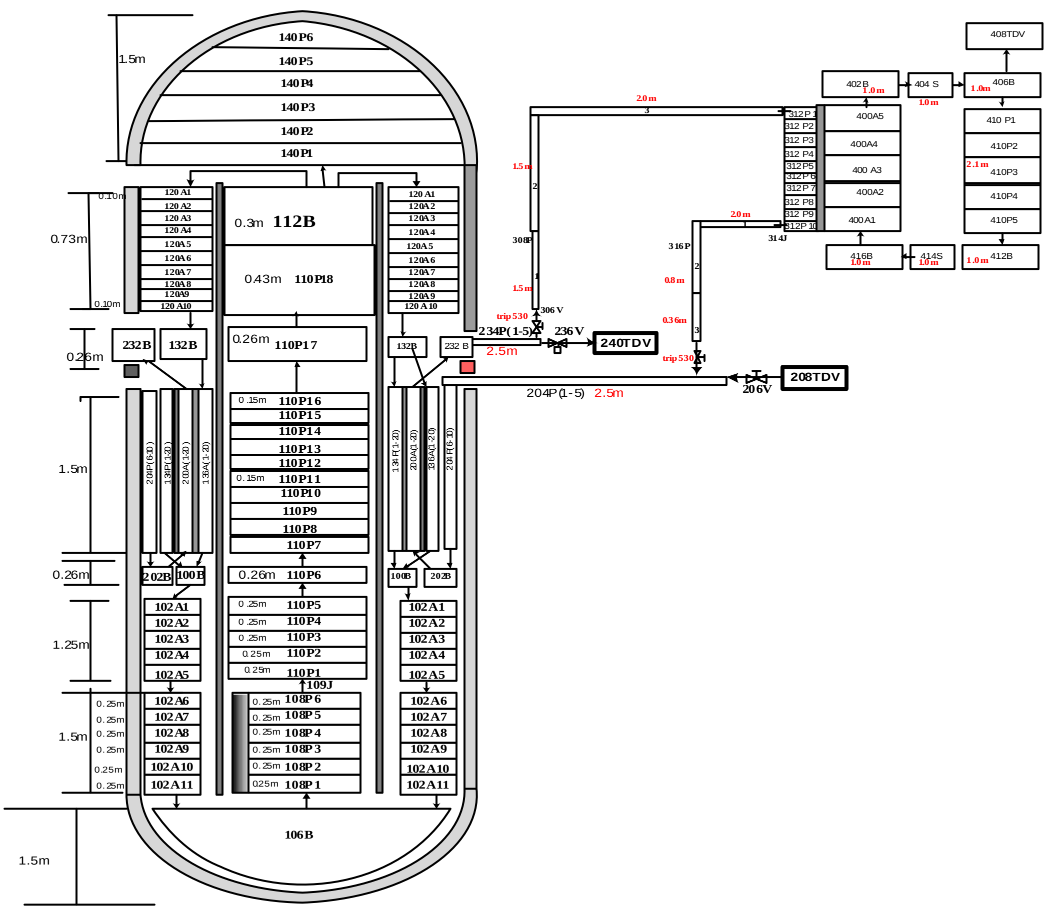

Fig. 3: Relap5 Nodalization diagram of IPWR

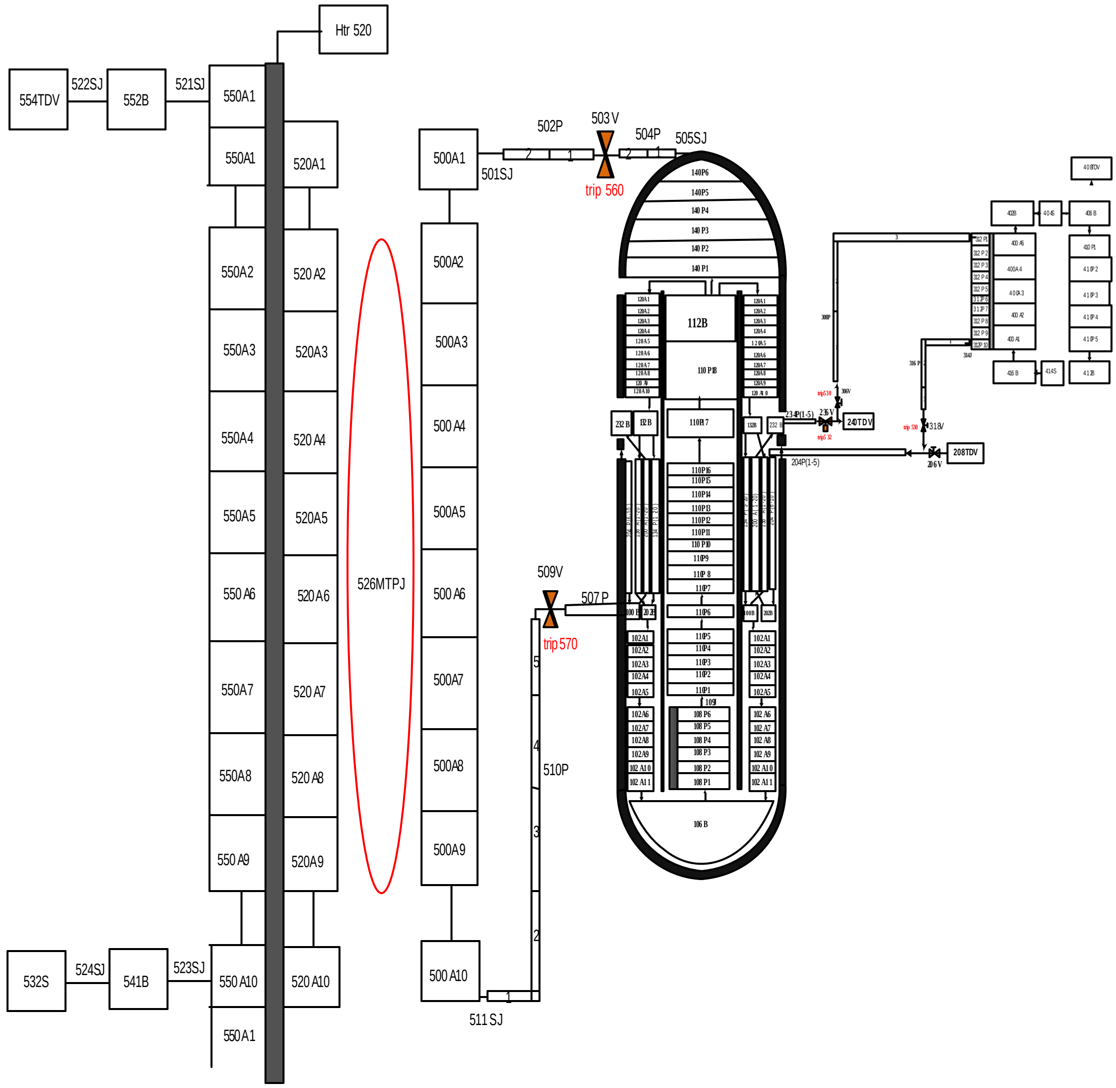

Fig. 4: Nodalization diagram of Pool type reactor

to the branch component 100B. Where it flows towards the lower region via the annulus component 102A and enters the lower plenum 106B. From there, the coolant will enter the core region which is 108P component and after heating it up, it flows out of the core and rises towards core upper region. The core upper region is also subdivided into pipe components for simulation. After reaching at the top, the coolant will make two paths first it goes towards pressurizer component which is 140P and the next one is to flow back via the component 120A. And it is flowing downward to reach at the 132B components and then flow down towards steam generator primary components. In order to simulate the pool type reactor there is connected a Pipe 504P. And 502P is connected with the pressurizer at the top of the reactor pressure vessel and a valve component 503V with trip 560 is inserted between the two components that has the function of operating under high pressure conditions. There is connected 500A and 520A which implies inner and outer vessel components of the container that is inter linked via Multiple junction 526. There is another valve 509V which is connected between component 510P and 508P that will operate when the trip560 ON allowing the pressure to release. The natural circulation conditions is established when the trip 570 ON which is connected with the sump valve 509V at the bottom. The pipe component 550A along with the heat structure Htr 520 represents the outer containment, which is actually the pool of water.

PASSIVE RESIDUAL HEAT REMOVAL SYSTEM (PRHRS)

In order to explain the PRHRS again consider the nodalization diagram as shown in Fig. 3. When the steam is going towards the turbine from the component 232B via 240TDV there is connected pipe components 308P via valve 306V connected with a trip530 so that if there is any accident happens then reactor trips and this valve is opened and the steam is going towards the water tank which is situated outside the reactor. In PRHRS heat exchanger has been submerged in a water pool in order to absorb core residual heat through the water and condenses the steam and this again will go towards the feed water pipeline coming from TDV208. Here two natural circulation conditions have been achieved as one inside the reactor and other in the PRHRS outside the reactor (Gou *et al*., 2009).

MATHEMATICAL EXPLANATION

In order to explain the steady state analysis of the reactor under load variations consider a mathematical model. As the concern reactor works on the natural circulation phenomenon so using Boussinesq assumptions for a single-phase natural convection system since the fluid is incompressible except for the case of gravitational term in momentum equation. The conservation laws can be taken for the one-dimensional formulation as given below:

- Continuity equation

$$u_i = \frac{a_o}{a_i} u_r \tag{1}$$

- Integral momentum equation

$$\rho \frac{du_r}{dt} \sum_i^n \frac{a_o}{a_i} l_i = \oint \rho g \sin\theta dz - \frac{\rho u_r^m}{2} \sum_i^n \left(\frac{fl}{d} + k\right)_i \left(\frac{a_o}{a_i}\right)^m \tag{2}$$

- Fluid energy equation for ith section

$$\rho c_p \left\{\frac{\partial T}{\partial t} + u \frac{\partial T}{\partial z}\right\} = \frac{4h}{d}(T_s - T) \tag{3}$$

- Solid energy equation for ith section

$$\rho_s c_{ps} \frac{\partial T_s}{\partial t} + k_s \nabla^2 T_s - q_s = 0 \tag{4}$$

The boundary condition between the ith section fluid and structure is given by:

$$-k_s \frac{\partial T}{\partial y} = h(T_s - T) \tag{5}$$

In the above equations, u_r is the representative velocity of the system corresponding to the velocity of the section having cross sectional area a_o and natural circulation flow resistance is given as:

$$\Delta P_{fric+form} = \frac{\rho u^m}{2} \sum_i^n \left(\frac{fl}{d} + k\right)_i, m \in (1,2]$$

where, Subscripts o and r denotes reference constant value and the representative variable of a system respectively whereas the subscript st denotes the steady state condition. Note that the ith component are denoted by the I and energy equation for solid as s (Ishii and Kataoka, 1982).

STEADY STATE ANALYSIS

From the Integral Momentum Equation, we get:

$$\oint \rho g \sin\theta dz = \frac{\rho u_r^m}{2} \sum_i^n \left(\frac{fl}{d} + k\right)_i \left(\frac{a_o}{a_i}\right)^m \tag{6}$$

For the temperature difference in hot source and cold sink that might be taken as the core inlet and outlet respectively which is denoted by l_{hc} are given as:

$$\oint \rho g \sin\theta dz = \beta \rho g l_{hc} \Delta T_{st} \tag{7}$$

Where:

$$\begin{aligned} \Delta T_{st} &= T_h - T_c \\ \beta \rho g l_{hc} \Delta T_{st} &= \frac{\rho u_r^m}{2} R \\ R &= \sum_i^n \left(\frac{fl}{d} + k\right)_i \left(\frac{a_o}{a_i}\right)^m \end{aligned} \tag{8}$$

From Eq. 3, 4 and 5 the Energy and Momentum Equation reduces to:

$$a_o \rho \bar{c}_p u_r \Delta T_{st} = \iiint q_s dv = Q \tag{9}$$

where, a_o, Q, β, c_p are the flow reference area , the power of heat source, thermal expansion coefficient and average heat capacity, respectively.

From the Eq. 8 and 9, we obtain:

$$G_{st} = c_1 Q^{\frac{1}{m+1}} \tag{10}$$

$$\Delta T_{st} = c_2 Q^{\frac{m}{m+1}} \tag{11}$$

Where:

$$G_{st}=a_o * u_r$$

$$c_1 = a_0\left(\frac{2\beta g l_{hc}}{\rho R \bar{c}_p}\right)^{\frac{1}{m+1}}$$

$$c_2 = \left(\frac{R}{2\beta g l_{hc}}\right)^{\frac{1}{m+1}}\left(\frac{1}{a_o \rho \bar{c}_p}\right)^{\frac{m}{m+1}}$$

Equation 10 and 11 are the steady-state solutions of single-phase natural circulation in closed loop. In a certain natural circulation system, a_o, a_i, g are constant; while in single-phase flow, l_{hc}, c_p, ρ, β, $(fl/d+k)_i$ fluctuate slowly in a narrow range with the changes of operating conditions. As a result, R, c_1, c_2 also change slightly in a narrow range.

Equation 10 and 11 is also be approximated by the relationship between steady state mass flow rate of natural circulation and power of heat source Q.

GRAPHICAL EXPLANATION OF LOAD VARIATION

After getting the steady state results at 100% Full Power the variation of different parameters with load changes have been studied as shown in Fig. 5 to 10. It has been concluded that in a certain loop structure steady state characteristics of single phase natural circulation governing parameters such as mass flow rate, temperature differences of the core inlet and outlet etc can be determined by the power of the reactor. According to the changes in the power, there are two mentionable dissimilar regions of the steady state characteristics and the distinction between the two can be determined by the eigenvalue m. This can be notifying in the mathematical explanation, which gives the relationship between steady state characteristics and the power of the reactor. The buoyancy, which is generated by the density difference of the hot and the cold fluid, is the only driving force for the natural circulation conditions. It ultimately changes the power level, since different power level produces different driving forces and generates different flow pattern, hence the steady state characteristics of single-phase natural circulation changes. Figure 9 gives the relationship between temperature difference of the core and the Fig. 10 shows the relationship between the mass flow rate under the load changes and it has been cleared that the Relap5 calculations matches the numerical calculations well. Figure 11 to 14 gives the relationship between the different power level and steady state characteristics of the OTSG. Temperature distribution of

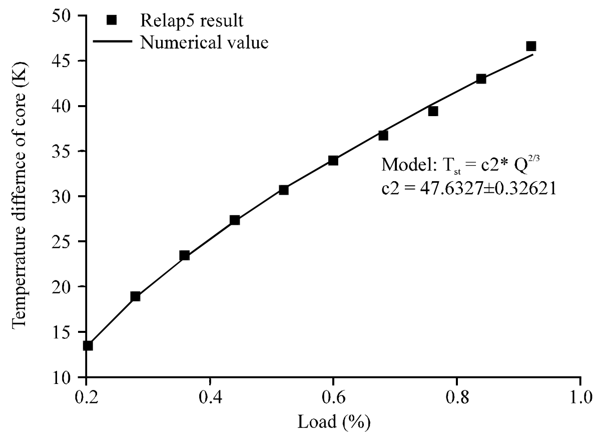

Fig. 5: Core Temperature trends varies with load percentages

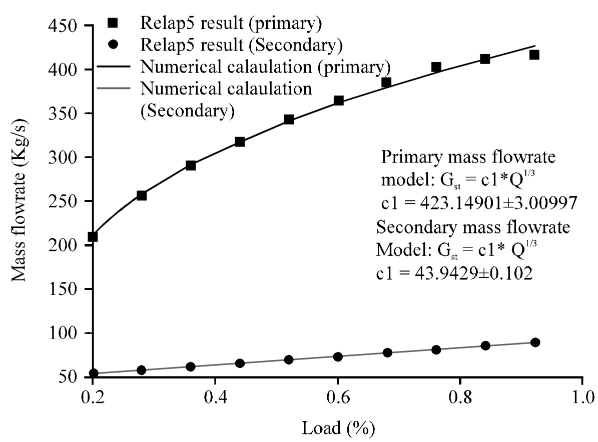

Fig. 6: Mass Flow rate trends varies with load percentage

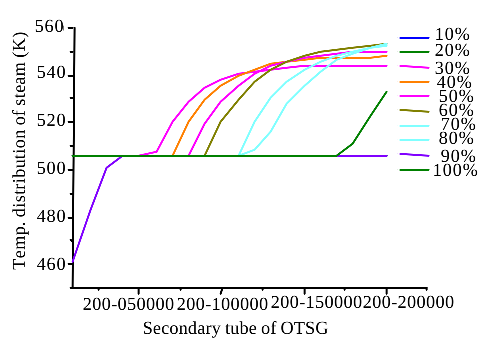

Fig. 7: Temperature distribution of steam in OTSG tube

the steam, water and the void fraction in the secondary tube of the OTSG is shown in the graphs. It has been shown that the OTSG tube section can be divided into three sections i.e., preheat section (super cooling water), evaporation section (saturated water) and super heat section (super heated steam). The scenario is such that the super cooled water enters the preheat section which changes the fluid temperature and enthalpy gradually and the fluid temperature rises to saturation value which continues to be heated to reach at the evaporation stage and finally becomes superheated steam is generated.

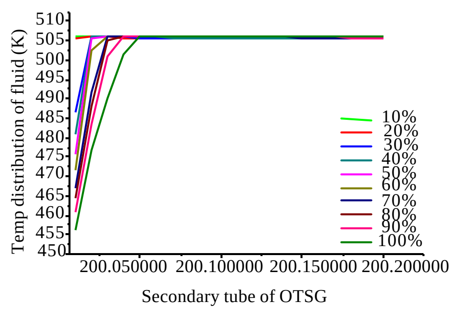

Fig. 8: Temperature distribution of OTSG tube

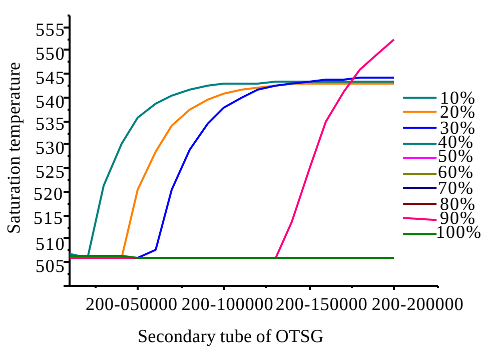

Fig. 9: Saturation temperature of OTSG tube

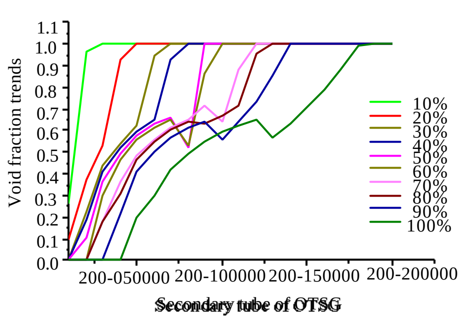

Fig. 10: Void fraction of OTSG tube

PARAMETRIC ANALYSIS OF IPWR

It has been shown that with the variation of load, the related parameters have different values and are depicted in Table 1. The table shows the different trends for power (MW) of the reactor, Core Inlet/Outlet and average temperature, secondary inlet outlet temperature, primary and secondary mass flow rate under different load variational state i.e., from 100% FP to 10% FP. All the values depicted in the table has been derived from the Relap5 code and are well suited for the steady state behavior of the reactor at any nominal power.

GRAPHICAL EXPLANATION

Figure 11 to 14 show the normal steady state results of the Pool type IPWR reactor concept which have been extracted from the Relap5code. The graphs reveal the pressure, temperature and the mass flow rate of the primary as well as secondary component of the IPWR. From the graphical explanation it has been verified that

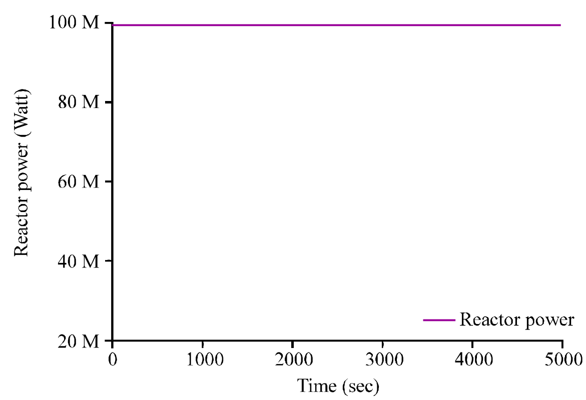

Fig. 11: Nuclear power steady state trends

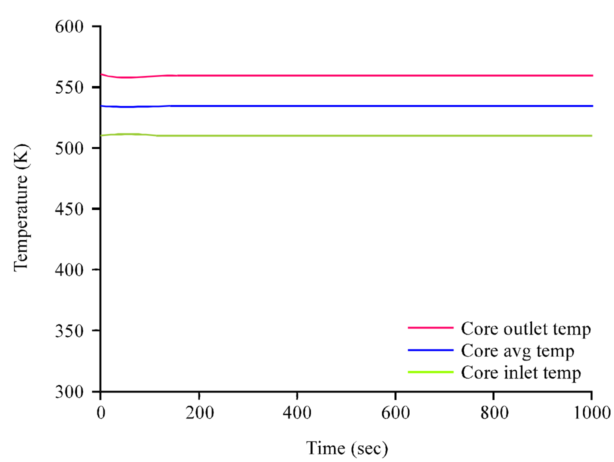

Fig. 12: Core temperature steady state trends

Table 1: Parametric analysis of IPWR

	Percentage load at full power(FP)Load									
Parameters	10%	20%	30%	40%	50%	60%	70%	80%	90%	100%
Core power(MW)	10	20	30	40	50	60	70	80	90	100
Core inlet temp(K)	533.403	527.538	524.041	523.798	522.939	522.852	520.137	520.557	518.868	514.349
Core outlet temp(K)	543.389	543.188	544.232	547.881	550.488	553.509	553.741	556.784	558.307	557.912
Core average temp(K)	538.396	535.363	534.1365	535.8395	536.7135	538.1805	536.939	538.6705	538.5875	536.1305
Secondary inlet temp(K)	423.122	423.15	423.123	423.122	423.123	423.123	423.123	423.127	423.124	423.124
Secondary outlet temp(K)	543.296	543.088	544.099	547.605	550.039	552.693	551.796	553.113	552.108	532.076
Primary mass flowrate(kg sec^{-1})	179.56	231.27	269.23	300.03	326.93	350.71	374.07	394.16	407.88	412.98
Secondary mass flowrate(kg sec^{-1})	4.32	8.7	13.05	17.35	21.71	25.98	30.15	34.62	39	44.5

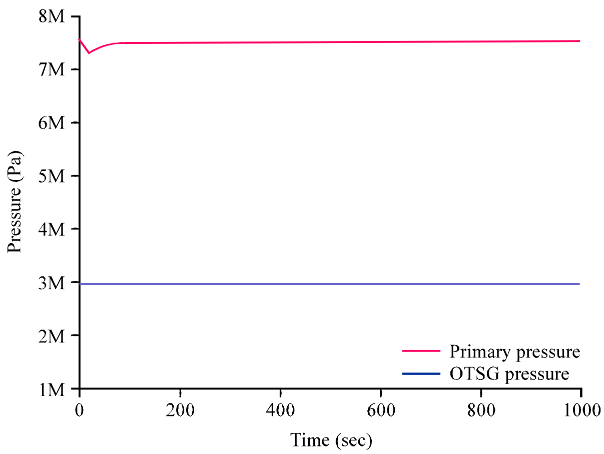

Fig. 13: Pressure Steady state trends

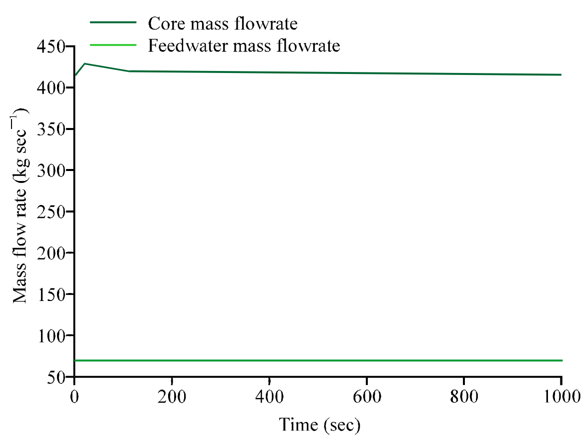

Fig. 14: Mass flow rate trend varies with time

further research on postulated accidents such as the simulation results from are in good agreement for inadvertent opening or closing of ADS valve.

PARAMETRIC ANALYSIS OF THE POOL TYPE IPWR

Table 2 shows the parameters concerning with the power of the reactor, the core parameters including inlet/outlet temperature as well as mass flow rate and other simulated parameters especially for the pool type reactor. All these parameters have been extracted from the graphical approach by using Relap5 code.

Table 2: Parametric analysis of pool type reactor

Parameters	Values	Units
Reactor power	100	KW
Primary pressure	7.55	MPa
Pressure of OTSG	2.96	MPa
Water level of pressurizer	0.948	M
Core inlet temperature	510	K
Core outlet temperature	559	K
Core average temperature	534	K
Temperature of steam	536	K
Feed water mass flow rate	42	kg sec^{-1}
Core mass flow rate	412	kg sec^{-1}
Temperature of suppression containment	375	K
Temperature of water pool	303	K
Pressure of suppression containment	4.2	MPa
Pressure of water pool	0.2	MPa
PRHRS water tank temperature	298	K
PRHRS outlet temperature	295	K
PRHRS water tank pressure	0.1	K

CONCLUSIONS

In this study, a research has been carried out on the normal and the steady state results of different components of the reactor along with the simulation of pool type reactor by using thermal hydraulic system code Relap5/Mod3.4 under different load percentages i.e., from 100% FP to 10% FP. Every time we got exceptional results that have been presented in this study. The purpose of this study is to check out whether the concern reactor can be operated well in different power level. And what are the circumstances or the problems that may happens during the load variations. But as simulation results are in good agreement with the numerical and the design values so it has been verified that the reactor has attained the safety criteria so that more research can be done further on the accidental conditions.

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