THERMAL HYDRAULIC AND SAFETY ANALYSES FOR PAKISTAN RESEARCH REACTOR-1

by

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To be presented at:
22nd International Meeting on
Reduced Enrichment for Research and Test Reactors (RERTR)
4-8 October, 1999, Budapest, Hungary

ABSTRACT

Thermal hydraulic and safety analysis of Pakistan Research Reactor-1 (PARR-1) utilizing low enriched uranium (LEU) fuel have been performed using computer code PARET. The present core comprises of 29 standard and 5 control fuel elements. Results of the thermal hydraulic analysis show that the core can be operated at a steady-state power level of 10 MW for a flow rate of 950 m^3/h, with sufficient safety margins against ONB (onset of nucleate boiling) and DNB (departure from nucleate boiling). Safety analysis has been carried out for various modes of reactivity insertions. The events studied include: start-up accident; accidental drop of a fuel element in the core; flooding of a beam tube with water; removal of an in-pile experiment during reactor operation etc. For each of these transients, time histories of reactor power, energy released and clad surface temperature etc. were calculated. The results indicate that the peak clad temperatures remain well below the clad melting temperature during these accidents. It is therefore concluded that the reactor can be safely operated at 10 MW without compromising safety.

INTRODUCTION

Pakistan Research Reactor-1 (PARR-1) is a swimming pool type research reactor. The reactor has been in operation since December 1965. The reactor had been operated at steady-state power level of 5 MW using highly enriched uranium (93%). The coolant flow, which is established by gravity, was 540 m^3/h. The reactor has been converted to use less than 20% enriched uranium at an upgraded power level of 9 MW. The criterion for fixing the maximum power level is to avoid Onset of Nuclear Boiling (ONB) at: (i) when the reactor power level approaches the overpower trip set point of 115% of the normal (steady-state) value and simultaneously (ii) the coolant flow approaches the low flow trip set point of 90% of the normal value. The limiting parameter for reactor power is the maximum achievable flow rate through the core (or the primary outlet pipe). The first high power core was assembled in May 1992 to run at 9 MW. Using the above mentioned criterion this power level was assessed (Ref.1) to be the maximum allowable, with a coolant flow rate equal to 950 m^3/h. The reactor has been running satisfactorily at this power level. Various core changes (shuffling and/or additions of fuel elements) have been carried out in order to get to the equilibrium core, which *is loading No. 94* (see Fig.1).

Many experimental requirements and radio-isotopes demand higher neutron flux levels. Also, most of the instrumentation and controls have already been designed for a power level up to 10 MW. The equilibrium core is a bigger core, consisting of 29 standard and 5 control fuel elements. Thus possibility of raising power level needs to be explored without compromising on reactor safety. The present paper presents the analysis of the core for enhanced power level of 10 MW. Computers codes and standard correlations have been employed to calculate different parameters: coolant velocity distribution in various channels of the core, critical velocity, pressure drop, saturation temperature, temperature distribution in the core, heat fluxes at onset of nucleate boiling, onset of flow instability, departure from nucleate boiling and the corresponding safety margins. Also, detailed accident analysis was performed for this core configuration and its response to the anticipated reactivity insertion accident was studied. The accidents analyzed included:

i) Start-up Accident;
ii) Accidental Drop of a Fuel Element;
iii) Beam Tube Flooding;
iv) Movement of Core Towards Thermal Column;
v) Removal of an In-pile Experiment.

DESIGN PARAMETERS AND CORE CONFIGURATION

The design parameters are presented in Table 1. One side of the core is reflected by two rows of Graphite elements. The remaining sides are surrounded by light water. The fuel elements are of plate type and can be arranged in any configuration on a 6 x 9 grid plate. Each standard fuel element contains 290 g of U^{235} uniformly distributed in 23 straight plates. On the other hand, each control fuel element contains 163.9 g of U^{235} uniformly distributed in 13 straight fuel plates and having a rectangular passage for the movement of oval shaped control rod. The equilibrium core comprises of twenty-nine standard and five control fuel elements.

COMPUTER CODES EMPLOYED

For the determination of pressure drop, velocity distribution and flow rate through different channels of the core, effective and bypass flow, computer code DP (Ref. 2) was employed. The code computes velocities through an iterative procedure after converging to the same pressure drop across the core for each channel.

Computer code PARET [Ref. 3] was employed to carry out steady-state thermal hydraulic analysis. The code was originally developed for power reactors for the analysis of SPERT-III experiments (Ref. 4), which was later modified (Ref. 5) to include library of various parameters suitable to research reactors. PARET is basically a coupled neutronic-hydrodynamic heat transfer code employing point kinetics, one-dimensional hydro-dynamics and on-dimensional heat transfer. The code supports a selection of heat transfer correlations. For the current analysis, in order to have conservative estimates, Dittus-Boelter (Ref. 6), McAdams (Ref. 7) were selected for the single phase and two phase heat transfer, respectively. For the determination of onset of nucleate boiling, Bergles and Rohsnow (Ref. 8) was used. The heat flux at onset of flow instability has been computed using both Forgan (Ref. 9) and CEA (Refs. 10,11) correlations. The value of bubble detachment parameter was taken to be conservative 48 (Ref. 12). For critical flux determination, Labunstov (Ref. 13) and Mirshak (Ref. 14) were employed. For conservative results, correlations were extrapolated with zero sub-cooling (Ref. 15).

METHODOLOGY

For the analysis, two channel model was utilized in the code. One assumes to have the hottest plate and associated flow channel and other being an average plate and flow channel. Axial power distribution has been represented by 21 equi-distance mesh points having peak to average ratio of 1.303. The radial peaking factor of 2.228 was determined by neutronic calculations. To account for the uncertainties, an engineering hot channel factor was incorporated using the conservative multiplicative method. This factor is the product of three components: (i) a factor 1.2 for the coolant temperature rise due to manufacturing tolerances in the coolant channel spacing. (ii) a factor of 1.2 for the film temperature rise due to uncertainties in the heat transfer coefficient and inhomogeneities in U235 distribution etc. and (iii) a factor of 1.1 for uncertainties in the calculated power distribution. It has been assumed that about 90% of the total fission energy is deposited in fuel, about 4% is produced in moderator, about 1% is produced in other reactor materials and remaining 5% is carried away by neutrinos. Kinetic parameters and reactivity feedback coefficients calculated for the core are listed in Table 2. All the calculations have been done with coolant inlet temperature of 38°C, inlet pressure of 1.712 bar which corresponds to the static height of water from core top to a point 15 cm below normal level of the pool (low level set point). It has been assumed that all the protection and safety circuits fail except the ‘overpower trip at 11.5 MW. A time delay of 27 ms has been taken between attainment of trip level and start of shutdown reactivity insertion. It has been assumed that control rods fall from fully out position thus requiring 0.541 s to be fully inserted. The melting temperature of cladding (LT-24 Aluminium alloy) has been taken to be 600 ºC.

RESULTS AND DISCUSSION

Steady-State Thermal Hydraulics

Results of the study are presented in Table 3. The figures computed are rather conservative since, to incorporate the uncertainties, the multiplicative method was used. This method is somewhat unrealistic since it assumes that all the worst conditions occur simultaneously at the same point. On the other hand, use of statistical method shows that reactor could be operated at much higher power levels with the probability of ONB in 1.4 cases per 1000. The statistical method recognizes that all of the worst conditions do not occur at the same time and same location. Since not much information is available about the use of statistical method, therefore, the conservative approach of multiplicative method was adopted. Based on this approach, maximum operating power, corresponding to maximum flow rate of 950 m3/h has been assessed. Results show that at reduced flow rate (90%), nucleate boiling will commence in the core when the reactor power surpasses 12.8 MW. Therefore, it is concluded that with full flow rate (100%), the core can be safely operated at 10 MW with an overpower trip set point of 11.5 MW. The maximum clad surface temperature at steady-state power level of 10MW, will be 102 ºC, which is about 23 °C below than needed to commence nucleate boiling. This gives a safety margin of 1.4 against ONB. The peak clad temperature is about 44.3 cm from top of the plate. Due to small meat thickness and good thermal conductivity of the fuel, plate peak temperatures at the center line of the meat are only about 2 °C higher than the respective peak clad temperatures. The coolant velocity in the inner standard channel is 2.46 m/s. The critical velocity at which hydraulic vibrations can result in deflection of fuel plates, causing local overheating and possibility of a complete blockage of the coolant channel has been calculated to be 10.5 m/s. This is well above the coolant velocity thus provides higher safety margin against the critical velocities.

Flow oscillations are undesirable for various reasons. They may cause undesirable mechanical vibration of components. They may cause system control problems. Also, these oscillations may cause changes on heat transfer characteristics. Peak heat flux at OFI calculated by Forgan (Ref. 9) correlation is slightly lower and is considered conservative. This gives a safety margin of 1.7

Critical heat flux computed by using Mirshak correlation is conservative, which gives the safety margin of 3.1

Safety Analysis

Various reactivity insertion accidents analyzed have been discussed in the following:

Start-up Accident: In this accident, it is postulated that due to circuit malfunctions, during start-up of the reactor, all of the control rods are withdrawn simultaneously, from their most sensitive position at maximum rate of travel with the reactor initially critical at a power level of 1 W and 10 MW. Using the maximum rod withdrawal speed of 102 mm/min and the reactivity versus rod position curves for the core, the maximum reactivity insertion rate has been estimated to be 0.048% $\Delta k/k/s$. Results of the study are provided in Table 4. The results indicated that for an accident starting from 1 W, trip level is reached much later as compared with the case of initial power equal to 10 MW. In spite of the fact that higher power level is reached, the energy released is smaller. On the other hand, for transients starting from 10 MW, less time is available to reach trip level. So the minimum period achieved is much higher and peak power just exceeds the trip level. Because the reactor remains in higher power range for larger time, energy released is much higher. Peak temperature at the clad surface reached is well below the melting point. No boiling occurred in the core. This shows that the core has a large safety margin against the worst foreseeable start-up accident.

Accidental Drop of Fuel Element: According to the approved operating policies at PARR-1, fuel loading during reactor operation is not permitted therefore, probability of occurrence of such an accident is negligible. However, this hypothetical accident has been analyzed. In the accident it is assumed that due to an operator's error during fuel loading a fresh fuel element is dropped on the core when the reactor is initially critical at 1 W. Since the spare insertion holes are always plugged, the insertion would be only partial and a reactivity of 0.902% $\Delta k/k$ will be inserted. From the data on fuel element drop test [13] time required by the fuel element to reach grid plate, after covering a distance of 0.45 m, has been calculated to be 0.375 s. Results of the study are presented in Table 5. Peak clad temperature reaches 107.4 ºC, which is far below the melting point.

Beam Tube Flooding: PARR-1 has six radial beam tubes. When not in use, these beam tubes are plugged with shielding blocks and filled with demineralized water. However, when an experiment is to be setup, water is drained and plugs are removed. If it is assumed that the drained (air filled) beam tube having maximum reactivity worth is filled with water. The transition from the air filled to water filled state adds a positive reactivity into the core. This will add a reactivity of 0.25%$\Delta k/k$. It has been assumed that this reactivity is added within 0.25 s while the reactor is operating at 10 MW. The results have been given in Table 6. In this case the peak power was 12.1 MW and the corresponding peak clad temperature 106 ^{o}C.

Movement of Core Towards Thermal Column: When the reactor core is moved from a position in which it is completely surrounded by water into the stall operating position, a portion of its water reflector is replaced by graphite thermal column. This adds a reactivity of about 0.728 % $\Delta k/k$ into the core. In order to prevent the initiation of excursions through the rapid movement of the core in the vicinity of the thermal column, a micro-switch has been installed in the bridge drive assembly, which scrams the reactor when the crank controlling bridge movement is engaged. Assuming, however, that the bridge scram interlock fails. The intensity of the transient would depend upon the speed with which the reactivity is added. The maximum speed at which the reactor bridge can be moved by an average person is about 13 cm/s. If it is conservatively assumed that the effect of thermal column on the core reactivity occurs only in the last 8 cm (extrapolation length) of the motion. The 0.728% $\Delta k/k$ would therefore, be added into the core in about 0.615 s. Transient response of the core to this ramp insertion has been studied. Results of the analysis are presented in Table 7. Peak power reached is 11.6 MW. Peak temperatures at fuel center line and clad surface are 108 ^{o}C and 106 ^{o}C, respectively. Peak clad temperature remains below the melting point.

Removal of In-Pile Experiment: The experiments which are placed inside the reactor represent a potential means of imparting a sudden increase in reactivity which can be inserted by removal of an experiment while the core is critical. Therefore reactivity of a single in-pile experiment has been limited to be less than 0.5% $\Delta k/k$. Transient response of the core to such situation has been investigated assuming that the experiment is removed through pneumatic rabbit system at a speed of 13 m/s. Under such conditions total reactivity of 0.5% $\Delta k/k$ will be added in 0.023s. This transient has been analyzed for an initial power of 10 MW. Results of the study are listed in Table 16.8. For the transient the maximum clad surface temperature reached about 120 ^{o}C, which is far below the clad melting temperature.

CONCLUSION

Results of the study show that while operating at 10 MW, the peak clad temperature in the core is about 102 °C, which is 23 °C below the temperature at which nucleate boiling will commence. The core will have sufficient safety margins against onset of nucleate boiling, onset of flow instability and departure from nucleate boiling. The core is also safe against reactivity induced accidents. It is therefore concluded that using the equilibrium core, reactor power can be operated at 10 MW without comprising on reactor safety.

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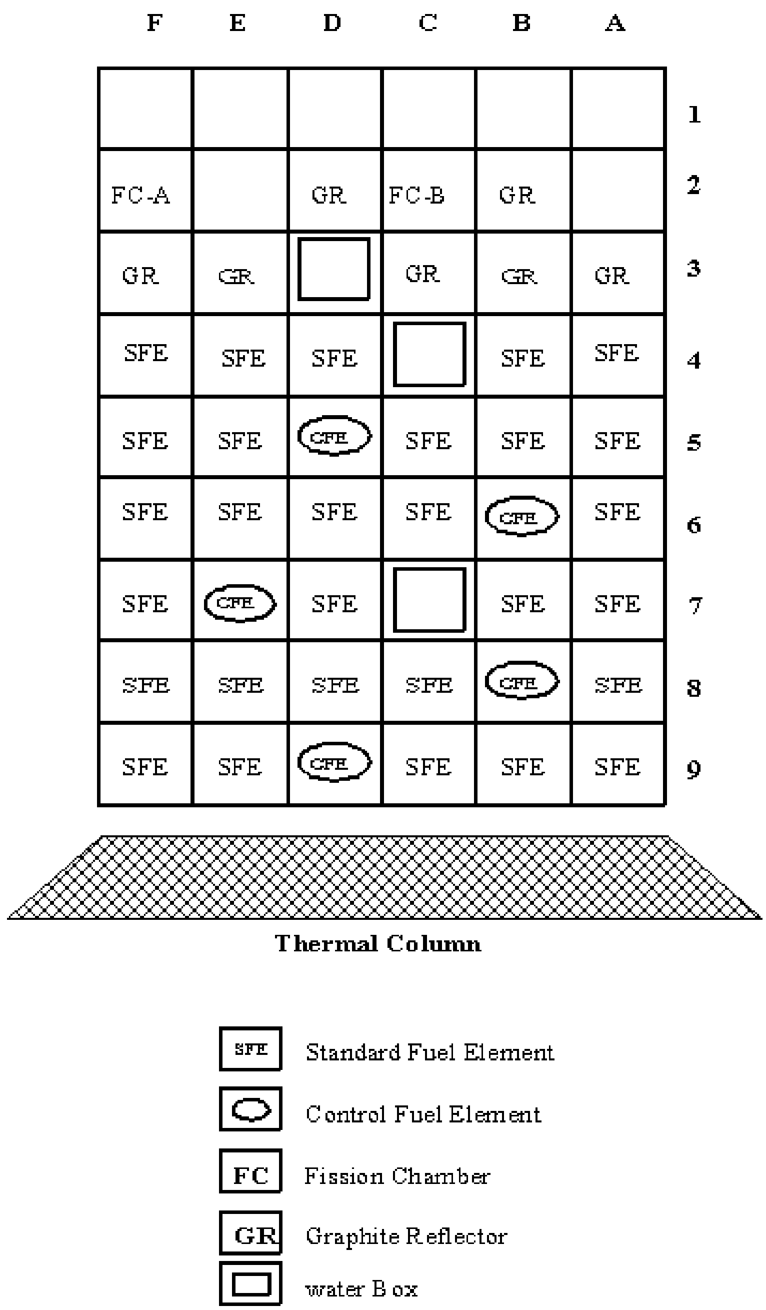

Fig. 1: Configuration of Equilibrium core (Loading No. 94)

Table 1: Design Parameters of the Equilibrium Core of PARR-1

Fuel	U_3Si_2-Al
U^{235} Enrichment (%)	19.99
Cadding Material	Aluminum
Coolant	H_2O
Moderator	H_2O
Fuel Element Dimensions (mm)	79.63 x 75.92
Lattice Pitch (mm)	81 x 77.11
No. of Fuel Elements: . Standard . Control	 29 5
U^{235} Content (g) . Standard . Control	 290 163.9
No. of Fuel Plates/Fuel Element: . Standard . Control	 23 13
Water Channel Thickness (mm)	2.10
No. of Dummy Plates: . Standard . Control	 Nil 2
Shape of Fuel Plate	Straight

Table 2: Kinetic Parameters and Reactivity Feedback Coefficients of the Equilibrium Core of PARR-1

Prompt Neutron Generation Time (s)	41.00
Effective Delayed Neutron Fraction (β_{eff})	0.0072754
Water Temperature Coefficient (-%Δk/k /^{o}C)	1.415 x 10^{-2}
Void/Density Coefficient (-%Δk/k /%void)	0.32
Doppler Coefficient (-%Δk/k /^{o}C)	2.11 x 10^{-3}

Table 3: Results of Thermal Hydraulic Analysis of the Equilibrium Core of PARR-1

Operating Power (MW)	10
Overpower trip level (MW)	11.5
Total flow rate (m^3/h)	950
Effective flow rate (m^3/h)	855
Coolant velocity (m/s)	2.46
Critical velocity (m/s)	10.5
Power peaking factors:	
- Axial	1.303
- Radial	2.228
- Engineering	1.584
- Total	4.598
Pressure at core top (kPa)	171
Pressure at the end of active region (kPa)	161
Saturation temperature at the end of active region (°C)	113.5
Steady-state temperatures (°C):	
- Coolant temperature rise across	
. Average channel	9.47
. Hot channel	33.65
. Core (including bypass flow)	8.55
- Peak clad surface temperature	102.5
- Peak centerline temperature	104.6
Average heat flux (W/cm^2)	18.1
Peak heat flux (W/cm^2)	83.4
Onset of nucleate boiling (ONB)	
- Average heat flux (W/cm^2)	25.52
- Peak heat flux (W/cm^2)	117.34
- Location of ONB from top(cm)	44.3
- Peak temperatures (°C):	
. Fuel centerline	128.5
. Clad surface	125.5
. Coolant exit	85.4
Onset of Flow Instability (OFI):	
- Peak heat flux (W/cm^2)	
. Forgan	138
. CEA	170
Departure from Nucleate Boiling (DNB)	
- Critical heat flux (W/cm^2)	
. Labunstov	326
. Mirshak	257
Safety margins:	
- Margin to ONB	1.4
- Margin to OFI	
. Forgan	1.7
. CEA	2.0
- Margin to DNB	
. Labuntsov	3.9
. Mirshak	3.1

Table 4: Transient Response to Start-Up Accident

Initial Power, MW	$1x10^{-6}$	10
Reactivity Inserted, % Δk/k	0.761(15.86)*	0.096(1.994)
Minimum Period, s	0.158(14.90)	11.59(1.96)
Trip Time, s	15.86	1.995
Peak Power, MW	11.68(15.89)	11.51(2.02)
Peak Temperatures, ^{o}C		
. Fuel Center Line	105.8(15.89)	112.7(2.02)
. Clad Surface	103.4(15.89)	110.2(2.03)
. Coolant Outlet	69.3(15.93)	75.3(2.045)
Energy Released at Time to Peak Power, MW-s	6.83	21.66

* The quantities in parenthesis are the times (in seconds) at which the corresponding values occurred.

Table 5: Transient Response to Accidental Drop Of Fuel Element

Initial Power, MW	1×10^{-6}
Reactivity Inserted, % Δk/k	0.902(in 0.375)*
Minimum Period, s	0.022 (0.38s)
Trip Time, s	0.638
Peak Power, MW	39.8(0.68)
Peak Temperatures, ^{o}C	
. Fuel Center Line	110.4(0.70)
. Clad Surface	107.4(0.70)
. Coolant Outlet	58.16(0.86)
Energy Released at Time to Peak Power, MW-s	1.19

* The quantities in parenthesis are the times (in seconds) at which the corresponding values occurred.

Table 6: Transient to a Beam Tube Flooding Accident

Initial Power, MW	10
Reactivity Inserted, % $\Delta k/k$	0.25(in 0.25s)*
Minimum Period, s	0.425 (0.100s)
Trip Time, s	0.076
Peak Power, MW	12.13(0.100)
Peak Temperatures, ^{o}C	
. Fuel Center Line	108.8(0.115)
. Clad Surface	106.5(0.115)
. Coolant Outlet	71.9(0.115)
Energy Released at Time to Peak Power, MW-s	1.09

* The quantities in parenthesis are the times (in seconds) at which the corresponding values occurred.

Table 7: Transient Response to Movement Towards Thermal Column

Initial Power, MW	$1x\ 10^{-6}$
Reactivity Inserted, % $\Delta k/k$	0.728(in 0.615s)*
Minimum Period, s	0.066 (0.615s)
Trip Time, s	3.285
Peak Power, MW	11.63(3.312)
Peak Temperatures, ^{o}C	
. Fuel Center Line	107.9(3.32)
. Clad Surface	105.5(3.32)
. Coolant Outlet	71.3(3.34)
Energy Released at Time to Peak Power, MW-s	10.45

* The quantities in parenthesis are the times (in seconds) at which the corresponding values occurred.

Table 8 Transient Response to Removal of In-Pile Experiment.

Initial Power, MW	10
Reactivity Inserted, % Δk/k	0.5(in 0.023s)*
Minimum Period, s	0.056 (0.035s)
Trip Time, s	0.0094
Peak Power, MW	24.1(0.040)
Peak Temperatures, ^{o}C	
. Fuel Center Line	123.5(0.065)
. Clad Surface	120.5(0.065)
. Coolant Outlet	74.9(0.155)
Energy Released at Time to Peak Power, MW-s	0.66

* The quantities in parenthesis are the times (in seconds) at which the corresponding values occurred