# EFFECT OF FUEL STRING RELOCATION ON THE CONSEQUENCES OF POSTULATED INLET HEADER LBLOCA IN KANUPP REACTOR

**IQBAL AHMED**
Karachi Nuclear Power Plant, P. O. Box: 3183, Paradise Point, Karachi 75400, Pakistan

**HANK C. CHOW and MOHAMED H. YOUNIS**
Atomic Energy Canada Ltd., 2251 Speakman Drive, Mississauga, Ontario, L5K 1B2, Canada

**NGA T. LE**
Ontario Hydro, 700 University Avenue, Toronto, Ontario M5G 1X6, Canada

## ABSTRACT

*An investigation aimed at determining the effect of fuel string relocation on reactivity excursion and power pulse following a hypothetical Large Break Loss of Coolant Accident in KANUPP reactor is reported. The assessment of reactivity insertion was performed making use of global (reactor) core analysis computer code RFSP. The reactor kinetics module *CERBERUS of the of RFSP code and the SOPHT (thermal-hydraulics code) were subsequently employed for the neutronic transient analysis.*

*The effect was evaluated in context of determining the adequacy of moderator dump shutdown system. Because of the presence of the gap between the inlet shield plug and the fuel string, the fuel bundles may shift in such a manner that low-irradiated fuel is moved towards the core centre. This represents an additional reactivity increase to be accounted for in the analysis. The reactivity excursion, however, is alleviated by an earlier reactor trip. The net impact is that the energy deposited in the maximum rated fuel pencil is increased from 56% of the 960 kJ/kg fuel-centre-line melting limit to 63%. The result demonstrated the adequacy of the shutdown system against the maximum credible accident event.*

## 1. INTRODUCTION

Karachi Nuclear Power Plant (KANUPP) is a 137 MWe CANDU, PHWR. It is generating electricity for the Karachi metropolis since 1972. The reactor consists of a tubed calandria vessel which contains the heavy water moderator/reflector and 208 horizontal coolant tube assemblies. The 8.285 cm I.D. and 518 cm long zirconium-niobium alloy (Zr-2.5% Nb) coolant tubes, containing fuel bundles and heavy water coolant, pass through concentric calandria tubes. The coolant tubes extend into end-shields and terminate in stainless steel end-fittings equipped with removable shield-plugs which permit access for refuelling. The fuel is natural uranium in the form of sintered uranium di-oxide pellets sheathed in thin zirconium alloy (Zr-4) tubes to form solid fuel elements about 49 cm long by 1.522 cm diameter. Nineteen such elements are assembled between zirconium alloy end supports to form fuel bundles 8.131 cm in diameter. Eleven fuel bundles reside in each of the 208 coolant tubes. The coolant flow in adjacent channels is in opposite directions. The fuelling which is performed against the flow is accordingly also bi-directional.

A fuel latch is installed at the downstream end of the fuel channel to position the fuel string against the hydro dynamic force of the coolant flow. The shield-plugs located in the end-fittings are equipped with a locking device which prevents their removal except by the proper mechanism provided in the fuelling machines. In order to account for thermal expansion and to provide adequate dimensional tolerance, the fuel channels are provided with a gap between the inlet shield-plug and the fuel string. Irradiation induced axial creep increases the size of this gap with time. Following a break upstream of the channel, the resulting pressure differential established across the fuel string can result in rapid relocation of the fuel bundles towards the inlet shield-plug. Because the KANUPP fuel channels are fuelled opposite to the direction of normal coolant flow, the irradiation distribution along the channel is such that the least irradiated fuel resides nearest the downstream end. Therefore in the event of fuel string displacement towards the upstream end of the channel, fuel with lower irradiation would move into the higher flux region. This would result in rapid insertion of positive reactivity.

Besides the insertion of positive reactivity, the fuel string relocation may, depending upon the size of the available gap also manifest itself in additional effects of impact and axial expansion forces due to sudden movement of massive fuel string [1]. These effects may reveal themselves within the same period of time as the positive reactivity insertion. The smaller the gap, lesser is the impact force. The compressive force may lead to fuel element deformation and ultimate shield-plug and/or pressure tube failure. The work presented in this paper includes only the effect of reactivity insertion on the power pulse. The other two effects have not been addressed.

The KANUPP reactor is equipped with a single, moderator dump shutdown system (SDS). The heavy water moderator in the calandria is supported by differential helium gas pressure. This arrangement permits reactivity control by moderator level variation and provides for rapid shutdown by dumping the moderator into the dump space below the calandria, following gas pressure equalization. The shutdown system is actuated upon the detection of an unsafe process condition that causes at least one trip parameter in each of two-out-of-three SDS logic channels to exceed its set point. When a trip is called for, the protective system shuts down the reactor by opening the dump valves, causing the moderator to drain from calandria. The effect of a hypothetical loss of coolant accident involving a large break in the inlet header in a reactor such as KANUPP would besides the expected power excursion on account of voiding, compound as a result of fuel string relocation (FSR) in the broken pass. In view of the fact that moderator dump shutdown system is characterized by comparatively long actuation time of the dump valves and longer still dump time, it was imperative to evaluate the effect of hitherto unknown phenomenon of FSR on the consequences of LBLOCA and assess the adequacy of moderator dump SDS afresh.

The work presented in this paper was part of the project, KANUPP Final Safety Analysis Report (KFSAR) Update, carried out in collaboration with AECL and Ontario Hydro under a commercial contract. Besides the evaluation of special safety systems, the analysis was performed in general to assess, to what extent the old safety report remained valid when current analytical methods, safety standards and knowledge were taken into account [2]. The conducted analyses not only took the phenomenon of FSR into consideration but also fully accounted for coolant down grading, boron in the moderator as well as the 20% uncertainty that currently is deemed to exist in the calculation of void reactivity of PHWRs.

## 2. FUEL STRING RELOCATION (FSR) REACTIVITY EFFECT

The magnitude of positive reactivity insertion as a result of fuel string relocation is dependent upon the size of the gap and number of channels affected.

### 2.1 Assessment of Gap Size

The design of inlet shield plug is such that it could accommodate a fuel bundle up to the orifice plate to a maximum of 8.8 cm. On the basis of average bundle length of 49.53 cm, a string of 11 fuel bundles during normal operation already has a overhang of about 1.3 cm into the shield plug [3]. The maximum a string could move forward would therefore be 7.5 cm (Fig. 1).

KANUPP employs heat treated Zr-2.5% Nb, 8.285 cm I.D pressure tubes. The elongation due to creep has recently been measured in eight representative pressure tubes to indicate an average length increase of about 3 mm only which is consistent with the previous data for heat treated Zr-2.5Nb, and is much less than that measured for cold worked Zr-2.5Nb pressure tubes [4]. In view of insignificant elongation of 0.6%, this has not been taken into consideration.

### 2.2 Number of Channels Affected

The design of KANUPP incorporates two pass, single HTS loop. The fuel string relocation would accordingly take place in half of the core involving only 104 fuel channels.

### 2.3 Magnitude of Reactivity Insertion

In a preliminary assessment of the reactivity effect of FSR, the fuel string relocation was treated as a simultaneous re-fuelling operation in half of the channels in the core. The reactivity decay rate of -0.324 mk/FPD and fuel consumption of 5.28 bundles/FPD for currently existing operating conditions (moderator and coolant isotopics of 99.56 wt % and 98.8 wt % respectively) were used [5]. The reactivity effect was estimated to be 0.129 mk/cm of movement. For a displacement of 7.5 cm, the positive reactivity insertion was accordingly assessed to be 0.97 mk.

A more accurate calculation was subsequently done using the RFSP computer code [6]. Based on the time average model, an instantaneous snapshot of fuel burnup was created using the *INSTANT module of RFSP with a random channel age map. In a perturbed case, the fuel string relocation was treated as simultaneous refuelling operation in half of the channels (104). The fuel channels were modelled as consisting of 21 half bundles, each of length 24.765 cm. The reactivity effect of 104 channels, simultaneously being shifted one half-bundle length was calculated to be approximately 2.97 mk. In a second perturbed case a different model with 40 pseudo bundles in each channel was created. Each pseudo bundle was 12.3825 cm long (a quarter of length of a real bundle). An instantaneous refuelling of 104 channels in the same pass gave a reactivity increase of 1.47 mk. Making use of these results and applying linear interpolation, it was deduced that instantaneous shift of the fuel strings by 7.5 cm in half of the channels will lead to a reactivity increase of 0.9 mk.

## 3. ASSESSMENT OF FUEL STRING RELOCATION TIME

The timing of fuel string relocation following a large break LOCA was predicted making use of the SOPHT-RFI code [7]. The calculation model accounted for the various forces acting on the bundle due to reverse flow, which included the pressure differential between two ends of the fuel string besides the drag and friction forces. The main parameters which affected the bundle movement were, location and size of the break, channel coolant properties and channel geometry. The analysis was carried out for the break that resulted in greatest challenge to the effectiveness of the shutdown system. In this case it was 100% reactor inlet header break which led to largest power pulse. The fuel string movement as a function of time after LBLOCA is given in Table-I.

## 4. CERBERUS MODELLING

The *CERBERUS [8] module of RFSP was used to simulate the 100% RIH break LBLOCA event. The fuel string was modelled as a group of pseudo reactivity devices spanning the core. The incremental absorption cross section of these reactivity devices at a given time was adjusted to produce the reactivity inserted by the fuel string movement at that time. A total of 10 pseudo devices as depicted on the core map (Fig. 2) were used to model FSR. Nine of these devices represent parallel columns of the core and were given the same incremental cross section as a function of time. The 10th device is located only in channel G-12 (which is defuelled) and was used to offset the effect of FSR in that channel. The incremental cross section of this device is equal to that of the other devices but with an opposite sign.

Table-II shows the time variation of the device incremental cross section assumed in CERBERUS analysis of the postulated break. It can be seen that the effect of FSR was assumed to occur somewhat earlier in the analysis than predicted by the SOPHT-RFI code. For example after 14 ms of the break, SOPHT-RFI predicted no fuel string movement yet. However as shown in the referred Table, one quarter of the total incremental cross section (and hence one quarter of fuel movement) was conservatively assumed in the analysis at that time. Similarly at 55 ms, three quarters of total movement was assumed in the analysis instead of the predicted one half.

Table-II also exhibits the calculated reactivity effect of the FSR. It is shown that the reactivity transient associated with fuel string relocation occur sooner in the CERBERUS model than it would have been had the actual fuel string movement depicted in Table-I was used.

## 5. ANALYSIS METHODOLOGY

Analysis for the postulated RIH guillotine break loss of coolant accident, both with and without fuel string relocation was carried out making use of SOPHT [9] and RFSP/CERBERUS system of computer codes in a coupled manner. Table-III lists the assumptions and input data used as initial conditions for the presented study.

To establish the initial core state, thermal-hydraulic data of fuel and coolant temperatures and densities in various zones (thermal-hydraulic) were used as input to CERBERUS for dynamic assessment of core neutronics. The resultant power distribution in turn was used in the thermal-hydraulic calculation. After ~3 iterations, convergence of such parameters as core reactivity, bundle and channel powers was achieved.

Having obtained the steady state convergence, the initial part of power excursion was simulated by iterations between thermal-hydraulic and neutronic calculations. The thermal-hydraulic calculations started with an assumed power transient. The SOPHT generated input for reactor kinetics calculations in the initial part of the transient was broken down into smaller intervals ($\leq$ 100 ms), The subsequent CERBERUS calculated powers in various zones were in turn used in revised thermal-hydraulic simulation. The converged flux transient was used to determine the high neutron power and high rate-log trip times. The time delays due to electronic signal processing and time required for the dump valves actuation were accounted for to determine the start of (moderator) dump

With the knowledge of hierarchy of trips (the second neutronic trip was credited), subsequent iterative transient calculations were re-started from the time of start of moderator dump. The moderator level as a function of time was modelled according to the measured moderator drop characteristics (Fig. 3) The neutronic time steps were chosen on the basis of moderator level matching successive lattice pitches. The power was quickly turned over and reduced to decay heat level. The coupled SOPHT - CERBERUS calculations in the above described iterative procedure continued to be carried out until the desired level of convergence in the values of reactivity, powers and deposited energy was achieved [10].

## 6. RESULTS

For the postulated 100% RIH guillotine break, the dynamic reactivity and the neutron flux transients, with and without FSR, are shown in Fig. 4 and 5 respectively. With FSR there is a sharp increase in power due to fuel string relocation starting at 14 ms from the time of accident. The string takes up the available gap of 7.5 cm in 69 ms. The initial step reactivity addition of 0.225 mk (as modelled in CERBERUS) results in reaching both the neutronic trip set points very quickly. The second reactor trip signal from the log rate N trip is received at 253 ms and moderator dump valves open at 753 ms (valves actuation time: 500 ms). A peak transient reactivity ~4.9 mk due both to FSR and voiding has been inserted until then resulting in power increase at a fast rate of over 350% / s. The initiating of moderator dump succeeds in arresting the increase of power at 1.05 s. The reactor power peaks at 3.9 times the steady state operating power. The negative reactivity insertion due to moderator dump makes the reactor sub-critical at 1.7 s into the accident at which point in time the moderator is just over 6 lattice pitches up from the core centre at a height of ~ 406 cm (160 inches). The reactor power at this stage is still however 2 times the operating power.

With increasing acceleration of moderator drop the power rundown continues. At 5.3 s when it has dropped to the level of half core, inserting over -70 mk of reactivity, the reactor power reduces to below 10%. With the lowering of moderator the maximum channel power shifts downward. While the upper channels having been uncovered by the moderator, enter the decay heat regime, the channels covered by the moderator still have some contribution from fission power. With the moderator at half core level, the maximum channel power shifts to J-09 which is still producing ~600 kW of heat. The maximum integrated power of 285 kJ/kg during the transient up to 5.3 s is seen by the outer elements of bundle at position 6 in channel L-09. This bundle is assumed to be operating at the licensing limit of 496 kW at the time of the accident and its outer element an additional 8% higher power.

A summary of CERBERUS results for each of the two cases studied is given in Table - IV. It is seen that although the FSR affects a reactor trip 48 ms earlier due to sharper increase of power, yet the maximum transient reactivity was 4.89 mk compared to 4.15 mk for the case without FSR. The corresponding reactor power peaking is 3.9 and 2.8 times initial power respectively. The maximum channel power of 11.55 MW occurred in channel K-08 compared to 8.28 MW in channel J-09 for the case without FSR. Similarly maximum bundle power was 913 kW for a bundle at position 5 of channel K-08 with FSR and 656 kW for the bundle at identical position in channel J-09 for the case without FSR. The hot element enthalpy increased by only 11.25 % from 542 to 603 kJ/kg and still remained substantially less than the limit of 960 kJ/kg for fuel centre line melting.

## 7. DISCUSSION

At the time KANUPP was commissioned more than two decades ago, the phenomenon of possible fuel string relocation in a LOCA was not known. It was neither mentioned nor analyzed in the final safety analysis report. The updating of KFSAR afforded the opportunity to investigate its effect on the safety of the reactor under postulated inlet header LBLOCA conditions. There are several reasons for the effect of FSR in KANUPP reactor being relatively insevere. Besides the fact that the available gap for the movement of fuel string is much smaller, the use of heat treated Zr - 2.5% Nb, pressure tubes and only ~ 8 Full Power Years of operation that KANUPP so far has experienced, would stand out as possible valid arguments.

The irradiation induced creep in KANUPP pressure tubes has been found to be insignificant compared to noticeably large elongation found in more mature CANDUs. Again, the fact that linear power rating of 41.27 W/cm of 19 element KANUPP fuel bundles is smaller than that for the 37 element bundles, employed in larger CANDUs has also allowed greater margin to centre line melting.

In addition to the FSR, the present analysis also took into account other possible conditions (e.g., coolant down grading and concentration of boron in moderator) which could enhance coolant void reactivity. These were not considered in the original safety analysis report. The results established the effectiveness and adequacy of the shutdown system against the most hypothetical LBLOCA scenario. The updating of KFSAR has presented an opportunity to bring the KANUPP reactor at par with other CANDUs in terms of safety analysis methodology and the use of up-to-date analysis tools. Such efforts will contribute to the continued safe operation of the plant in the years ahead.

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**TABLE - I**

**FUEL STRING RELOCATION AS A FUNCTION OF TIME AFTER LBLOCA (SOPHT - RFI CALCULATION)**

| Time After Break (ms) | Fuel String Movement (cm) |
|---|---|
| 0.0 | 0.0 |
| 14.0 | 0.0 |
| 55.0 | 3.75 |
| ≥ 83.0 | 7.50 |

**TABLE - II**

**INCREMENTAL CROSS SECTION OF PSEUDO REACTIVITY DEVICES AND ASSOCIATED REACTIVITY CHANGE AS A FUNCTION OF FUEL STRING DISPLACEMENT (CERBERUS MODEL)**

| Time After Break (ms) | String Movement (cm) | $\Sigma_{abs}^{th}$ ($10^{-5}$ cm$^{-1}$) | Reactivity (mk) | | Reactivity (Introduced by FSR) (mk) |
|---|---|---|---|---|---|
| | | | Without FSR* | With FSR | |
| 0.0 | 0.0 | 0.000 | 0.0 | 0.0 | 0.00 |
| 14.0 | 0.0 | - 0.110 | 0.012 | 0.236 | 0.22 |
| 55.0 | 3.75 | - 0.330 | 0.153 | 0.824 | 0.67 |
| ≥ 83.0 | 7.5 | - 0.440 | 0.360 | 1.254 | 0.89 |

* From Coolant Void Only

## TABLE - III

## ANALYSIS ASSUMPTIONS AND INPUT DATA

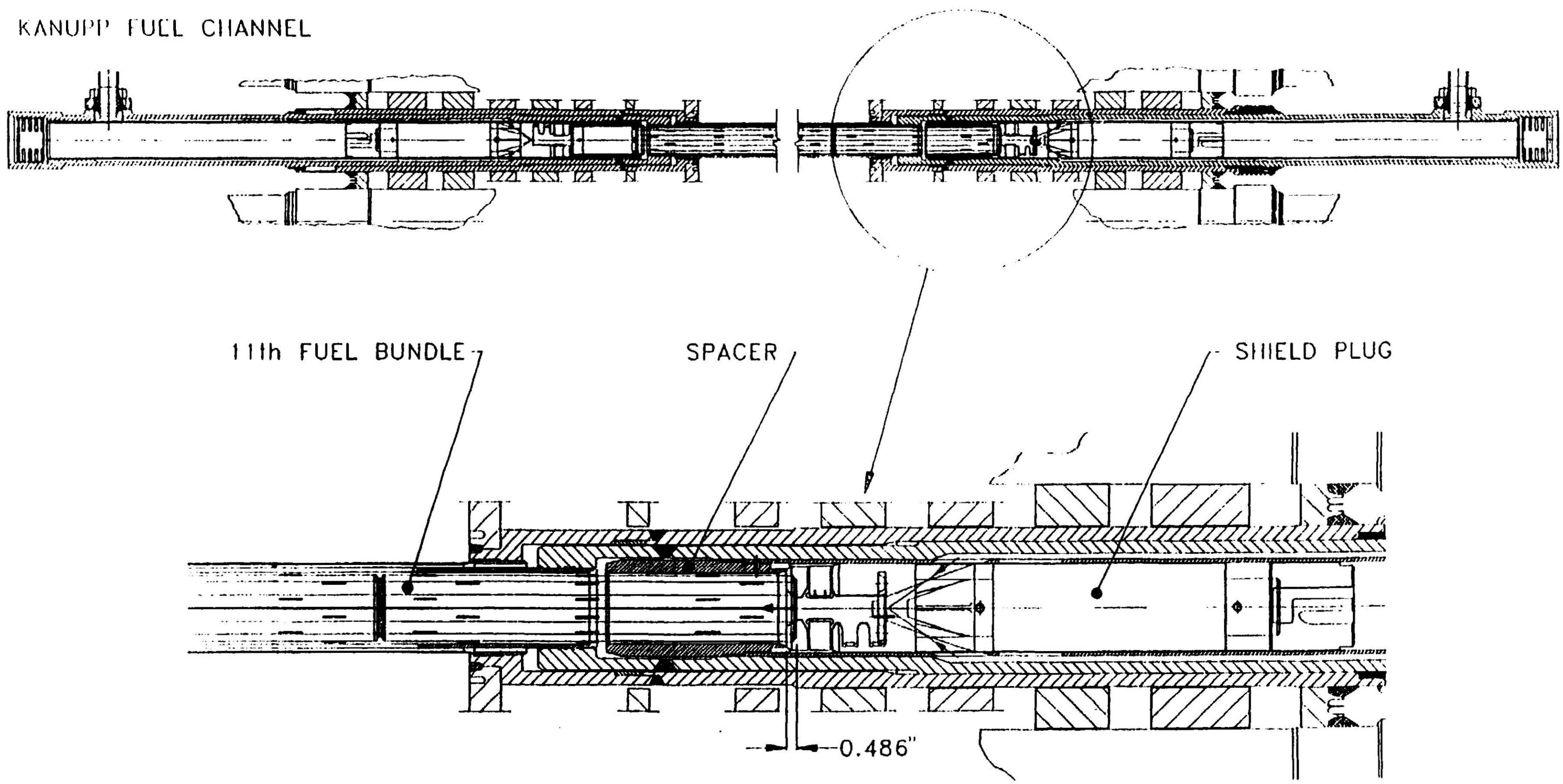
| PARAMETER | VALUE |
|---|---|
| Steady State Reactor Power | 100 % FP |
| Moderator ($D_2O$)Isotopic Purity | 99.5 wt % |
| Coolant ($D_2O$) Isotopic Purity | 97.5 wt % |
| Position of Control Absorber Rods | 3 Lattice Pitches Inserted |
| Moderator Poison | 0.5 ppm (~ 4.0 mk) |
| Void Reactivity Uncertainity | + 1.376 mk |
| Bundle Power Licensing Limit | 496 kW |
| Initial Moderator Level | 477.5 cm (188 inches)<br>(Max. of Operating Band) |
| Linear N (Neutron Power High) Trip | 119 % FP<br>(includes 6.5% Instrument Uncertainity) |
| Rate Log N (Log Rate High) Trip | 15 %<br>(Includes 5% Instrument Uncertainity) |
| Low PHT Pressure Trip | 9.775 MPa (1417.7 psig)<br>(Includes 9.3 psi Instrument Uncertanity) |
| Fuel String Relocation Distance | 7.5 cm |
| Channel Elongation Due to Creep | Negligible (Ignored) |
| Energy Required for Fuel Centre Line Melting | 960 kJ/kg |

## TABLE - IV

### SENSITIVITY OF NEUTRONIC POWER TRANSIENT TO FUEL STRING RELOCATION FOLLOWING 100% RIH RUPTURE (SUMMARY OF CERBERUS RESULTS)

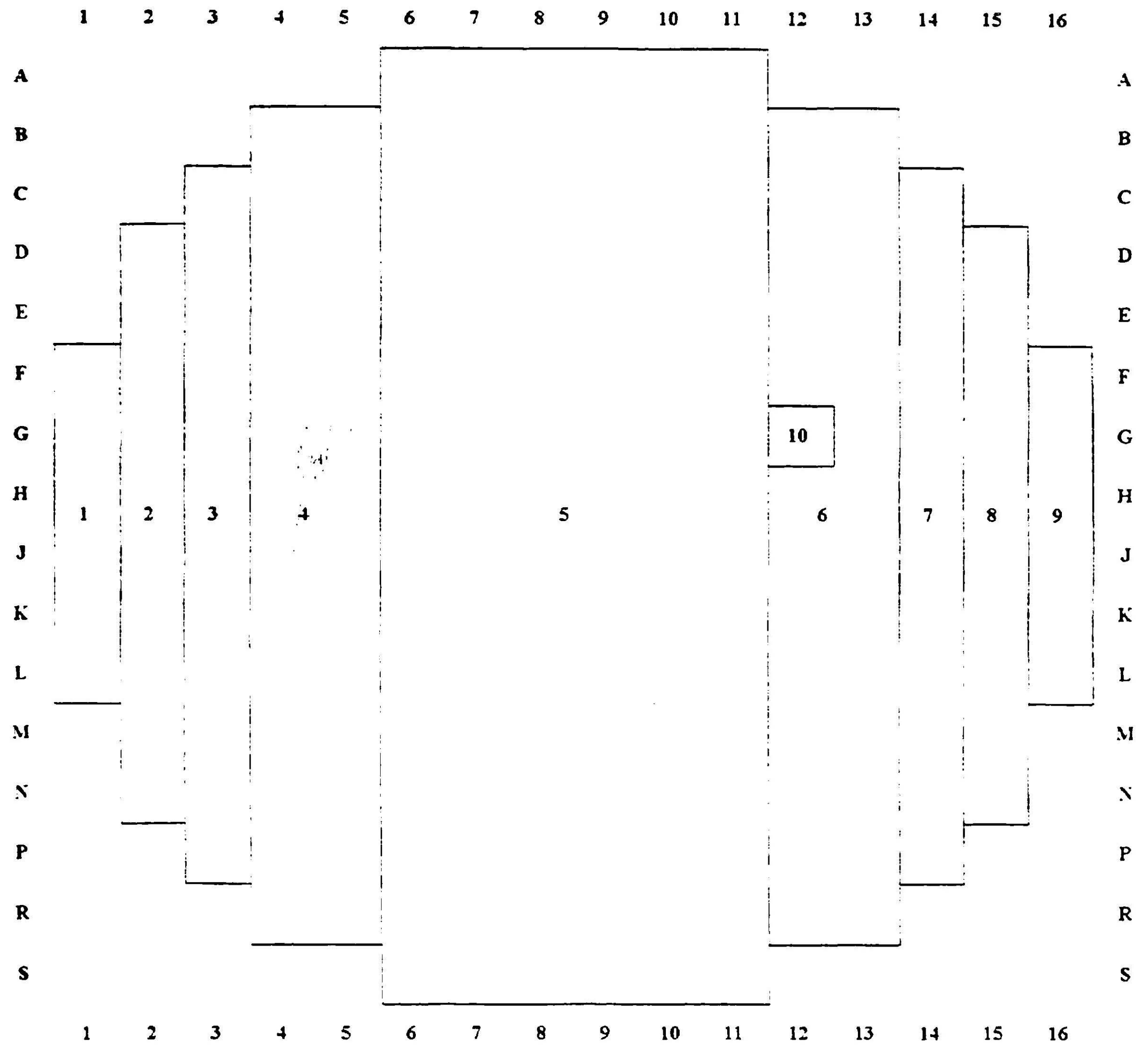
| PARAMETER | WITH FSR | WITHOUT FSR |
|---|---|---|
| Neutron Over Power Trip (ms) | 235 | 300 |
| Rate Log High Trip Signal (ms) | 253 | 301 |
| PHT Pressure Low Trip Signal (ms) | 433 | 433 |
| Reactor Tripped On (assumed in analysis) | Rate Log High | Rate Log High |
| Peak Reactivity (mk) | 4.89 | 4.15 |
| Peak Power (MW) | 1670 | 1221 |
| Transient Relative Neutron Flux Amplitude | 3.971 (at 1050 ms) | 2.883 (at 982 ms) |
| Maximum Channel Power (MW) | 11.6 (K-08) | 8.28 (J-09) |
| Maximum Bundle Power (MW) | 1825 (K-08/5) | 1312 (J-09/5) |
| Stored Energy (kJ/kg) in Hot Element (at t=0) | 318 | 318 |
| Heat Added (kJ/kg) to Hot Element ~ 5s into Accident | 285 | 224 |
| Hot Element Enthalpy (kJ/kg) | 603 | 542 |

Fig. 1 AXIAL GAP BETWEEN THE DOWN STREAM FUEL BUNDLE AND ORIFICE PLATE OF INLET SHIELD PLUG.




OVERHANG OF 11th BUNDLE IN LINER SPACER ASSEMBLY BASED ON MAXIMUM BUNDLE LENGTH OF 19.530" (ASSUMING ALL BUNDLES ARE EQUAL LENGTH)

**Fig.2 Pseudo Reactivity Devices Representing Fuel String Relocation (CERBERUS Model)**




**KANUPP Core Map**

Fig. 3

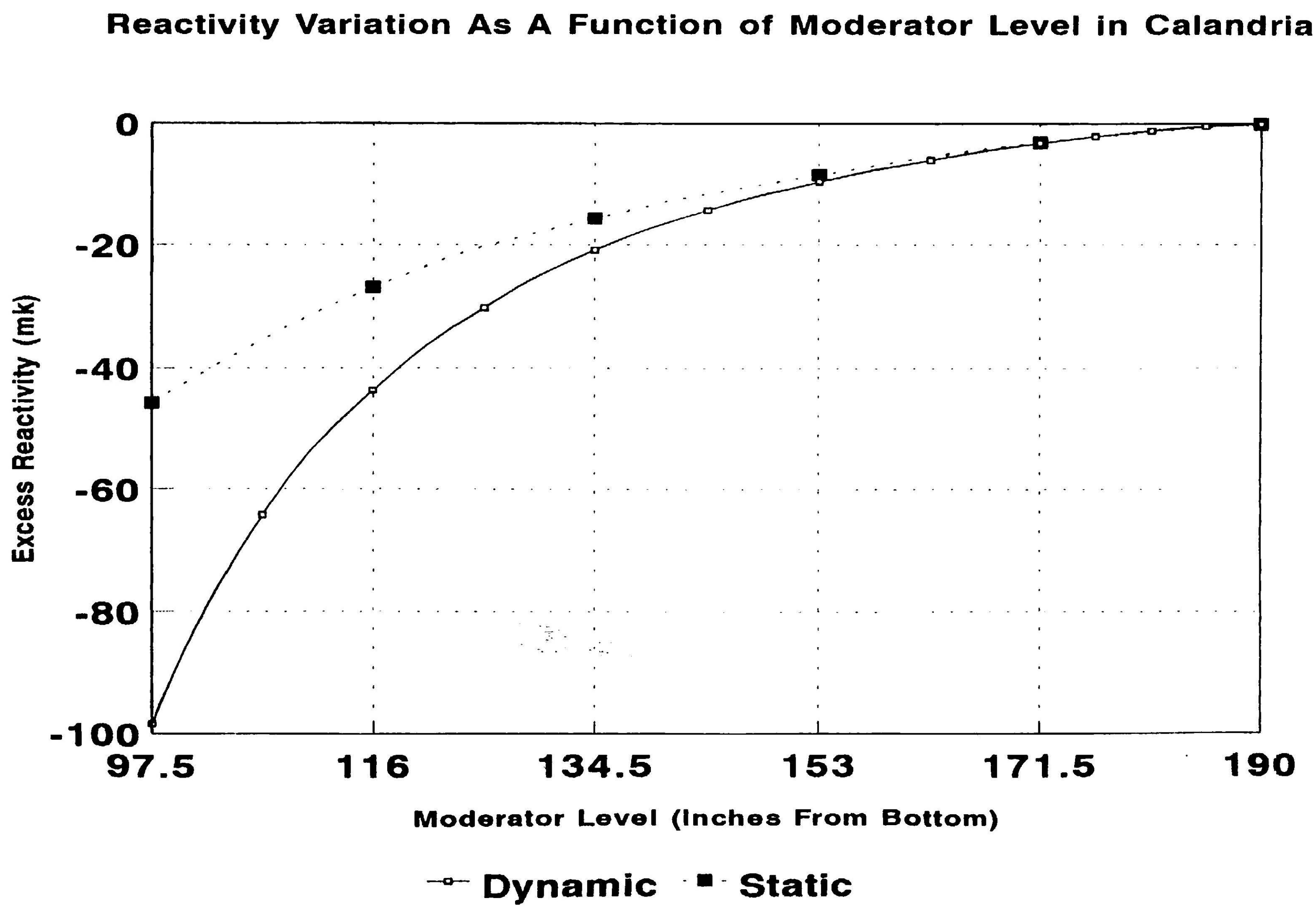

Reactivity Variation As A Function of Moderator Level in Calandria
Excess Reactivity (mk)
0
-20
-40
-60
-80
-100
97.5
116
134.5
153
171.5
190
Moderator Level (Inches From Bottom)
Dynamic
Static

Fig. 4

# RIH Guillotine Break LOCA

## Effect of FSR on Dynamic Reactivity Variation

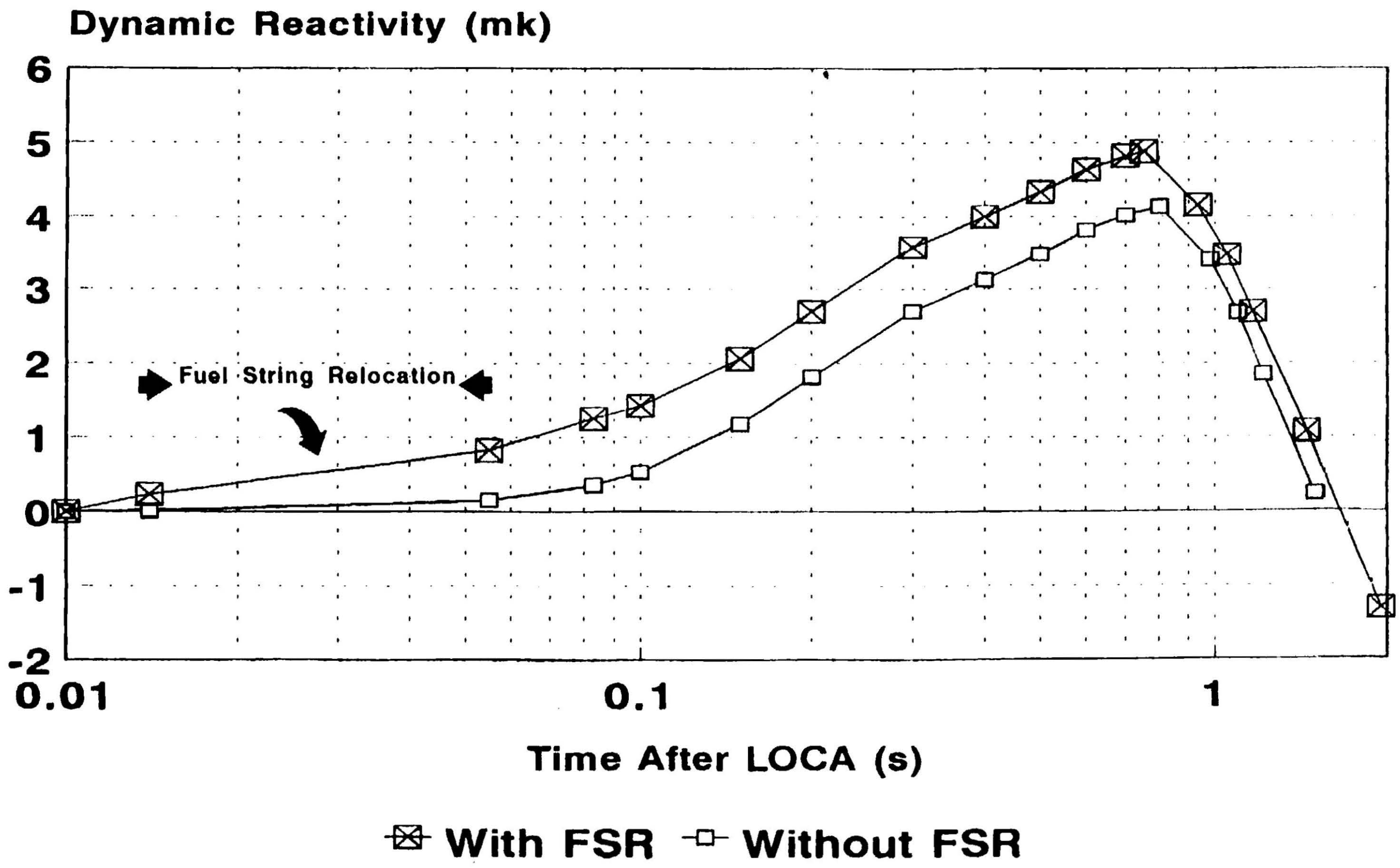
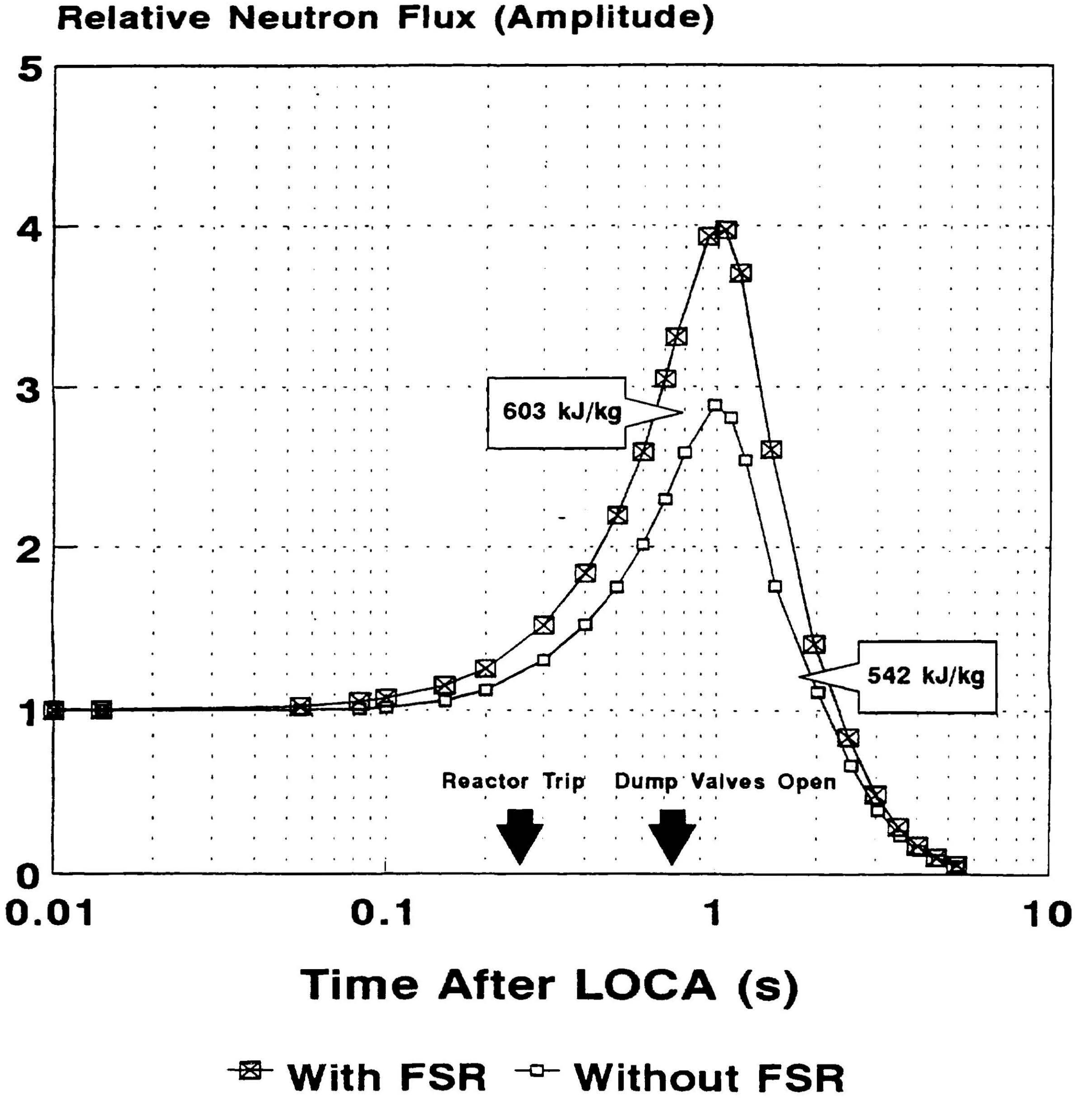


Fig. 5

## Neutronic Power Transient Following RIH Guillotine Break LOCA
### Sensitivity to Fuel String Relocation




**Note:**

The indicated Trip and Dump Times relate only to the case with FSR