ANALYSIS OF FUEL LONGTERM EFFECTS USING SUBCHANNEL CODES

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ABSTRACT

Thermohydraulic computer codes such as COBRA3C are now becoming in practical use as part of power plant licensing process.

One of the areas of interest to reactor designer is the effect due to longterm operations and the changes that might occur either due to irradiation (burnup) effects or thermal stresses in the reactor.

The present study is an attempt to predict coolant channel temperature profiles at the core End Of Life. In addition deeper insight is given to the individual single partial contribution to the changes in Begin Of Life and End Of Life profiles. The effects due to fuel swelling, rod bowing, and changes in physical properties are studied in details using COBRA3C code.

The codes COBRA3C and the kinetic code AIREKMOD together a structure routine are joined in a modular system. As a result coupling between the different transient parameters in normal and abnormal operation conditions could be performed.

Results of such coupling are shown to improve the prediction of DNBR.

الخلاصة :

لقد أصبحت المنظومات الحسابية المستخدمة للحسابات الثرموهيدروليكية مثل المنظومة COBRA3C معتمدة علمياً كجزء من عملية ترخيص المحطة النووية.

إحدى الحقول المهمة لمصممي المفاعلات هو التأثر الناتج من التشغيل الطويل الامد والتغيرات التي ربما تحدث نتيجة تأثير التشعيع (الاحتراق) أو نتيجة للاجهادات الحرارية داخل المفاعل .

الدراسة الحالية هي محاولة لاستنباط توزيع درجات الحرارة في قنوات التبريد عند نهاية عمر القلب . بالإضافة إلى ذلك أعطت هذه الدراسة ضوء عميق على التأثير المنفرد لكل عامل على حدة في بداية وفي نهاية عمر القلب .

التأثيرات الناتجة عن تضخم الوقود ، انحناء اقلام الوقود , التغيرات في الخواص الفيزياوية تمت دراستها تفصيلياً بواسطة المنظومة COBRA3C . لقد تم ربط كل من البرنامج AIREKMOD وCOBRA3C بالإضافة إلى روتين يتم فيه الحسابات التركيبية للوقود في نظام موحد . نتيجة لهذا الربط أمكن دراسة تأثير العوامل المتغيرة في الحالات الغير مستقرة.

نتائج الدراسة بالمنظومة الجديدة حسنت من حسابات (DNBR) .

INTRODUCTION:

Thermal hydraulic margins are key determinants of plant operation flexibility and maximum attainable power rating and can therefore impact on plant availability and capacity factor.

Operating limits are established to ensure acceptable fuel conditions under normal as well as during abnormal operational transients or accident events. Examples of parameters that have to be limited to achieve these ends are linear generation rate, departure from nucleate boiling ratio (DNBR). Calculation of these parameters requires a detailed core subchannel thermal hydraulic analysis to provide the flow and fluid enthalpy in various regions of the fuel assembly. The important subchannel analysis computer codes that are in use for licensing calculations are : COBRA-IV,FLICA ,HAMBO, and COBRA3C (C.L.WHEELER. *et.al*, 1976) , (FAJEAU, M. , 1969) , (BOURING,R.W, HAMBO , 1967), (ROWE.D.S. , 1973) . These computer codes have been checked against experimental data and proved to predict accurately the critical parameters required to react or design.

A nuclear reactor operates for a long cycle (almost two years).During this time interval the reactor fuel is subject to drastic changes due to burnup and the presence of severe conditions of mechanical and thermal stresses beside the irradiation effects.

All these longterm effects influence the reactor performance to a large extent.

The longterm reactor behavior gains increasing importance with time. As a result, both experimental and computational efforts are spent to investigate the End Of Life (EOL) behavior because of its connection to reactor safety (PAUL.R, BETTEN, 1986), (HOFMAN *et.al*, 1980).

In recent study, (PAUL.R, BETTEN 1986), some experiments with instrumented fuel assembly were irradiated for 22 months and removed after it reached 8.8% burnup, (30568MWd). A comparison between calculations at the Begin Of Life and End of Life power reduction shows discrepancies between measured and calculated assembly temperature profiles by COBRA-IV. The differences were attributed to bowing in fuel elements.

In the present study the subchannel code COBRA3C is used to simulate a pressurized water reactor PWR fuel assembly. The LOFT reactor is considered as our model reactor. Some of the longterm parameters affecting performance are introduced. The parameters are introduced separately; this way the effect on temperature profile at EOL could be easily identified the parameters studied are: fuel swelling, rod bowing, changes of power distribution due to heterogeneous burnup.

Swelling, (MA .B.M. 1983), results from fission gas release during the fission process as well as fuel densification resulting from solid fission products. The direct effect of swelling is the change in the gap conductance. In case of severe conditions the fuel rod dimensions will increase at the expense of coolant channel dimensions.

Bowing results from temperature gradients which affect the mechanical properties of fuel rods. Bowing affects flow and therefore has great effect on heat transfer between fuel rods and coolant. We are going to investigate the effects of the above factors on the reactor assembly temperature profiles.

In the course of the present work it was found necessary to intercouple thermohydraulic, kinetic and structure calculations. The efficient method of performing such coupling is through a modular system. Brief description of the method and the results obtained are presented in section VI.

- PROBLEM MODELING IN COBRA3C:

The model reactor used in the present study is the LOFT reactor. The LOFT reactor is designed to be representative of a large pressurized water reactor with reduction in power and size.

The reactor operates with a power level 50 MW at 2330 psi pressure, and core temperature of 595°F. The fuel rods are 36 inches in length.

The fuel enrichment is 2.27 and 4.95% of U235, The hottest assembly distribution layout with the section modeled in COBRA3C code are given in Fig. (1a). The assembly is 8×8 rods. In Fig.(1.b) the large number refer to the rod numbering in the code input ,and the small ones refer to subchannel numbers. Details about the LOFT COBRA3C calculation are given in (AL AHMADY, 1988).

The EOL power distribution of the hottest assembly is taken from neutronic calculation (AMIN .E.A. et. al, 1990). In this neutronic calculation a modified version of the diffusion burnup code FEVER-7 has been used. The modifications allowed for longterm behavior effects on fuel elements to be (AMIN .E.A. et. al, 1990) included during the burnup treatment. Fig. (2) gives the EOL power distribution as calculated by compared to that at the begin of life BOL. The rod region with higher power suffers higher burnup and the reduction in power is higher .This leads to a more uniform power distribution than at the begin of life, as is clear in Fig (2) .As a result to the previous effect the average power is closer to the regions of less power and the ratio pin power to average power increases for lower power regions.

- FUEL TEMPERATURE CHANGES DUE TO BURNUP:

The fuel burnup leads to changes in the axial and radial power profiles .Beside the reduction of the overall power, the power peaking tends to reduce with irradiation. The axial power profile after 2000 hours operation is shown in Fig (2) (with control rod out) together with the initial power distribution.

The COBRA3C calculation with power distribution for the Begin Of Life and End Of Life are shown in Figs. (3) and (4). Fig. (4) gives the axial temperature profiles for the hottest pin. It is clear from this figure that the temperature tends to flatten with burnup.

The temperature profi1es presented above the heat flux variation with burnup. The different burnup consequences are not considered in the COBRA3C calculation

BOWING:

The results of reactor physics calculation reported by (C.L.WHEELER. *et. al*, 1976) showed that the maximum pin power is in the low enriched zone with power peaking factor of 1.6 and the highest pin power in the high enriched zone peaking factor of 1.51.

The neutronic calculation in the same reference showed also that the largest rod power gradient occurs in the high enriched rod.

The calculation for bowing prediction was performed as a result for this high enriched rod.

The calculation for bowing assumed that the fuel rod is a simply supported beam acted upon by bending moment. The resulted deflection could then be calculated .The assumption that temperature drop across the fuel pin is linear was also made.

The bending moment due to linear temperature across a plane of the cladding was taken from the formula given in the LOFT design report to be:

$$M_t = \frac{\Delta T E \alpha}{8R_o}(R_o^4 - R_i^4)$$

Where:

M_t = thermal bending moment
α = coefficient of thermal expansion
R_o = outside radius
R_i = inside radius
E = Young's modulus.

The midpoint deflection of the fuel pin simply supported at both ends of the active fuel length was calculated to be 0.0155 inch. ΔT was calculated for the different axial nodes from the COBRA3C calculation.

For the bowing contribution to the HOT-spot, HOT channel-analysis, four rods are assumed to bow toward each other to give center to center dimensions of 0.530 inch.

This value for the effect of bowing was introduced in the COBRA3C input, the resulting radial temperature profiles are is given in Fig. (5).

It can be seen from the Fig. (5) that the effect of bowing is to increase the temperature. Since bowing is a longterm effect, it is most probable near the End of Life, its effect would be flattening of the temperature profile .It could be partially responsible for the power variation from BOL and EOL that is calculated from burnup alone.

Swelling; Effect of Fission Gas Release:

Irradiation swelling induced by inert gases of fission products in the oxide fuel is less severe than that in metallic fuel, yet in both cases it limits the fuel performances, in particular the burnup.

Since the thermal properties of the fission gases are different from helium, the presence of the fission gases affects the-Heat transfer between the fuel and the clad. The COBRA3C calculations were performed once when the gap is filled with helium, in the second the gap was filled with fission gases. Published correlations (HOFMAN *et.al*, 1980), give the amount of fission gas produced as $1.35x10^{-3}$gram-mole/megawatt-day.

The results are shown in Fig. (6).The increase in temperature with the presence of fission gases in gap tends to saturation for high rod power. The maximum temperature increase in the rod central temperature amounts to 230° F. It was not possible in the present calculation to treat partial filled gaps, i.e. all the rods have the same fission gases.

The axial temperature profile for the hottest rod is shown in Fig. (6). The fission gas release and fuel temperature are interconnected, and therefore the hotter rod parts increase in temperature more than colder parts. The rod surface temperature increases by 13°F.

Coupling between thermohyraulic, kinetic and structure calculations:

The most efficient and convenient method to couple thermohydraulic & kinetic and structure calculation can be achieved through a modular system. In the present system the codes

COBRA3C and AIREKMOD together with a structure routine are coupled in a modular system.
The communications between different modules occur either through a main database or through the common area .The overlay programming structure is adopted to minimize storage. Detail description of the system is given by (E. AMIN, 1990). The coupling between the different modules is governed by the module sequence and the input control parameters.
The following variations have been implemented in COBRA3C:

- Restart option.
- Heat flux profile varies with time.
- Channel flow rate variation with time.
- Gap thickness varies with time.

The heat flux profile variation with time is obtained from AIREKMOD calculation, whereas the gap thickness, flow rate and pressure variation are obtained from the structure calculations.

The restart option was found necessary in order to save the information from the previous time step. In this option all informations required are buffered out to the database and then read in the subsequent time step, together with the current information calculated by the other modules.

The AIREKMOD kinetic code calculates the axial power profile variation during the transient event according to the local thermal hydraulic conditions given by COBRA3C and the reactivity coefficients.

The variations in the AIREKMOO code to be implemented in the system are the restart option and the communication procedure with the system through the database and the common area.

Results of calculations using the above coup1ing procedure are discussed in the following:

Transients including partial loss of flow, anticipated transient without scram ATUS for the LOFT reactor is presented, The transient results in the reduction of the coolant mass flow rate from 100% to 90% with no scram occurring was run with and without AIREKMOD coupling.

The normalized total reactor core power and the coolant mass flow rate histories are shown in Fig. (6). The transient was first run using the kinetic coupling; the output is then used in the thermohydraulic module. A second run was made without kinetic coupling; the results are shown in Fig. (7). The DNBR was calculated using the Westinghouse w-3 correlation implemented in COBRA3C. The minimum axial values for both runs are given in Fig. (7).

It is shown in this figure that using coupled calculation improves the prediction of MDNBR i.e. it tends to the conservative side. Fig. (8) gives the relative axial heat flux profile for different transient times. It is clear that the axial heat flux profile is suppressed as the transient progresses in the upper region of the core due to the negative void coefficient which is larger in this part of the core.

The above results show, that using a fixed heat flux profile during the transient in the subchannel analysis codes introduces an inherent conservatism. The application of variable heat flux profile via kinetic codes removes this conservatism.

Another important effect is the gap conductance effect on the fuel and clad surface temperatures.

Results of calculation showing the effect of gap conductance variation on fuel and clad temperatures is shown in Figs. (9) and (10).

Both curves show that using dynamic gap conductance model i.e. allowing for gap thickness and conductance to vary with time gives higher values for both the clad and the fuel temperatures as the course of the transient progresses.

Summary and Conclusions:

In the present work the COBRA3C thermohydraulic subchannel analysis code is used to show the effects of longterm parameters on fuel performance. Although the parameters studied are interconnected, yet for the purpose of identifying the single effects, they were considered to be independent. It has been shown that not only the burnup but the swelling and bowing could have relevant effects on the temperature distribution. In the present treatment each variation in the parameter studied needed new run for the code.

Coupling between thermohydraulic, kinetic and structure codes have been achieved through a modular system. Changes in both the codes COBRA3C and AIREKMOD have been made and a structure routine is written. It was therefore possible to study both steady state and transients. The present study gave quantitatively the effects of bowing and swelling in both cases.

The system developed in the present study is a good tool to obtain the influence of coupling calculation during transients.

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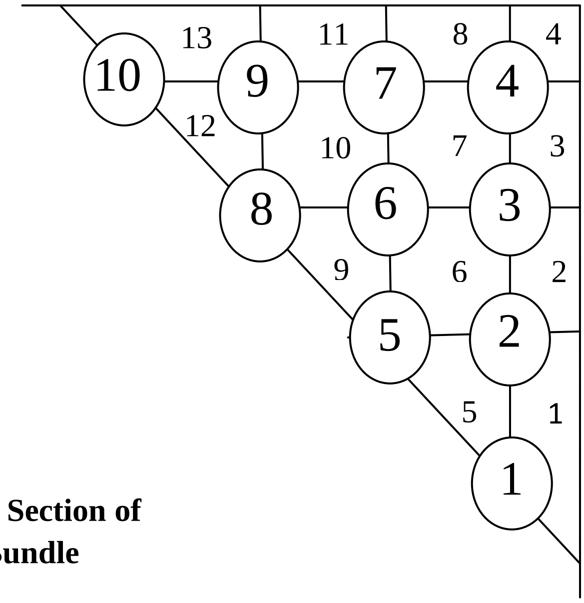


Fig.(1.b) Sub-channel Layout For A 1/8 Section of Symmetry From A 6A – RDO Bundle

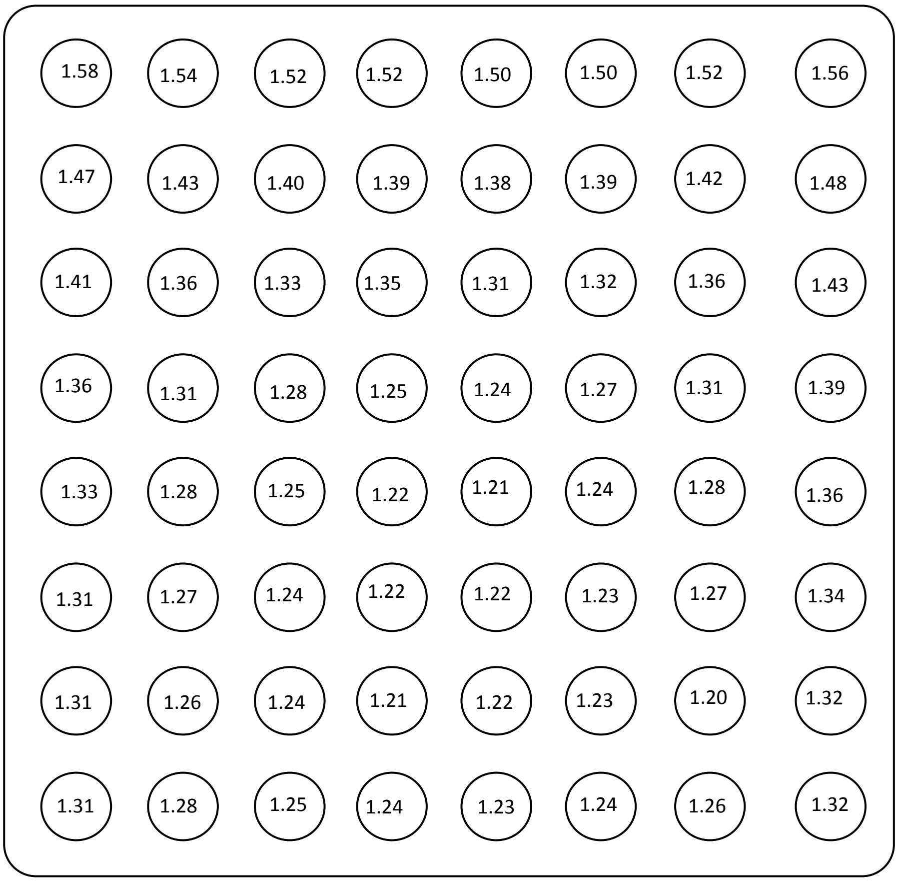


Fig.(1.a) Relative Power Factor For Hottest Assembly

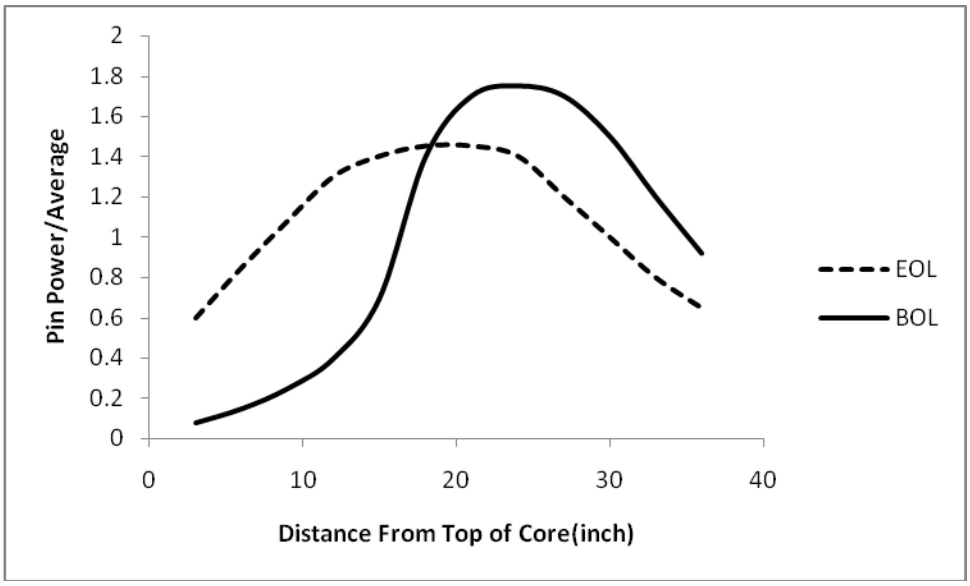
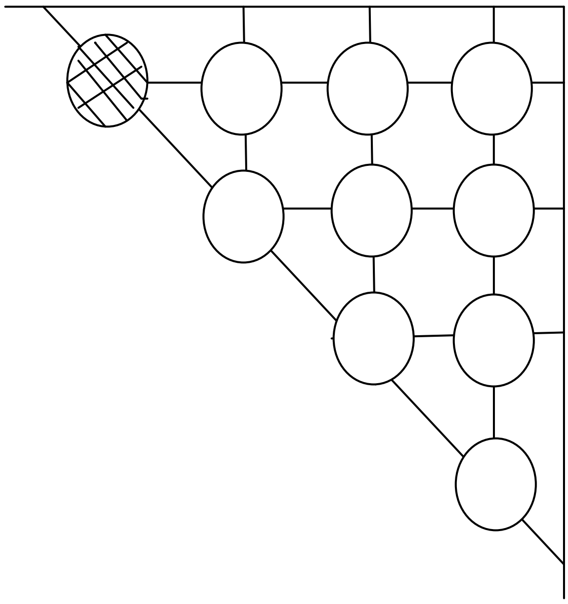


Fig.(2): Axial Power Profile For Begin Of Life and End Of Life (2000EFPH).



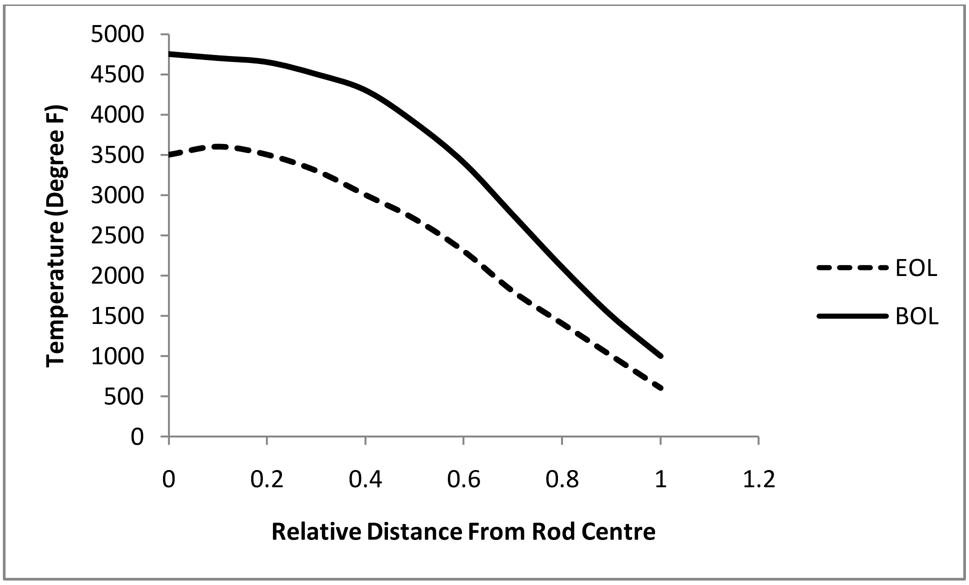
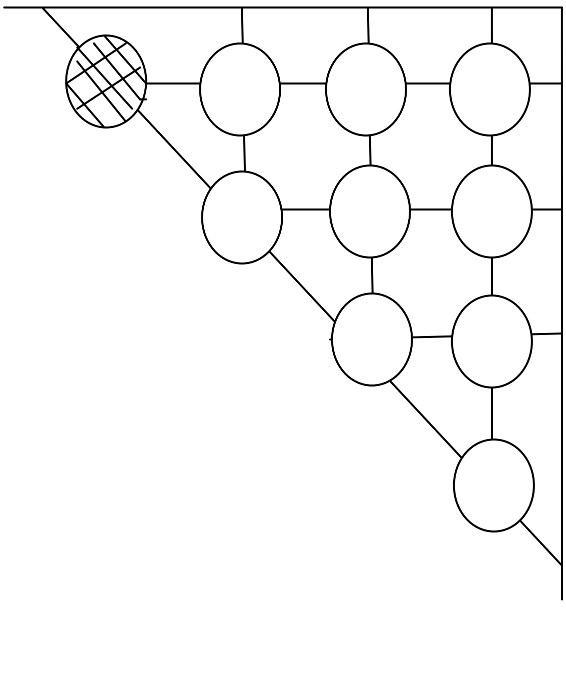


Fig.(3): Radial Temperature Profile and Effect of Burnup for the Hottest Rod.



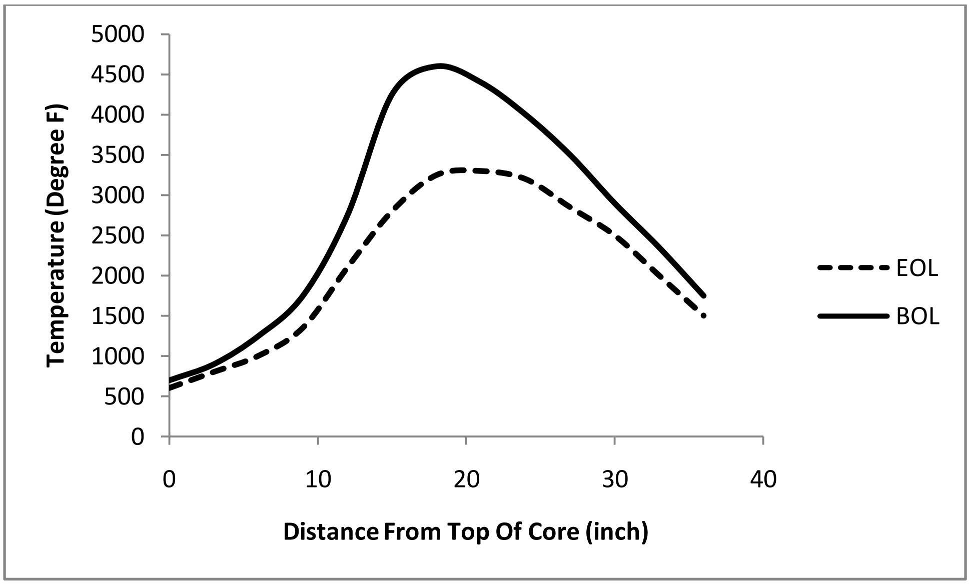


Fig. (4): Effect of Burnup on Axial Temperature Distribution For the Hottest Rod.

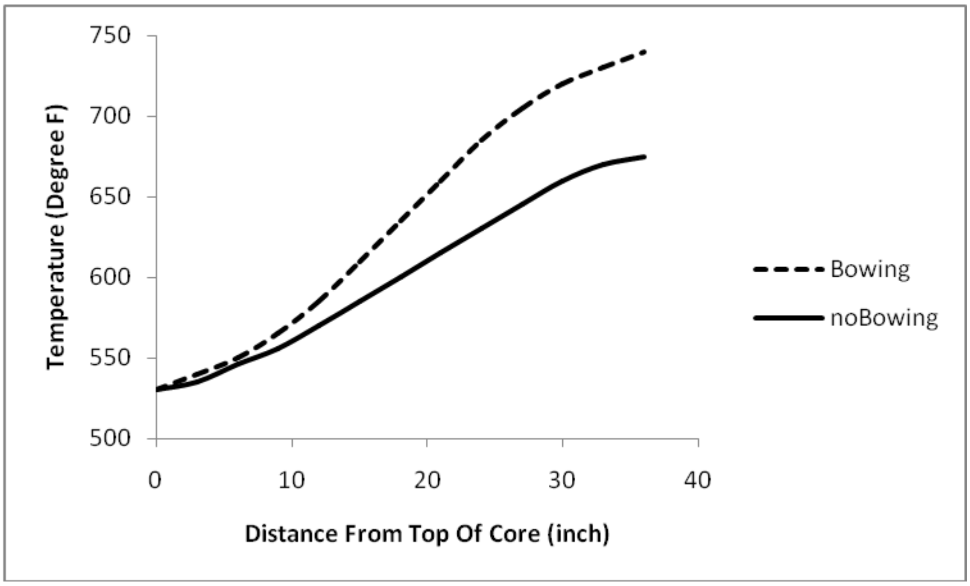


Fig.(5): Hottest Channel Axial Temperature Distribution and The Effect of Bowing.

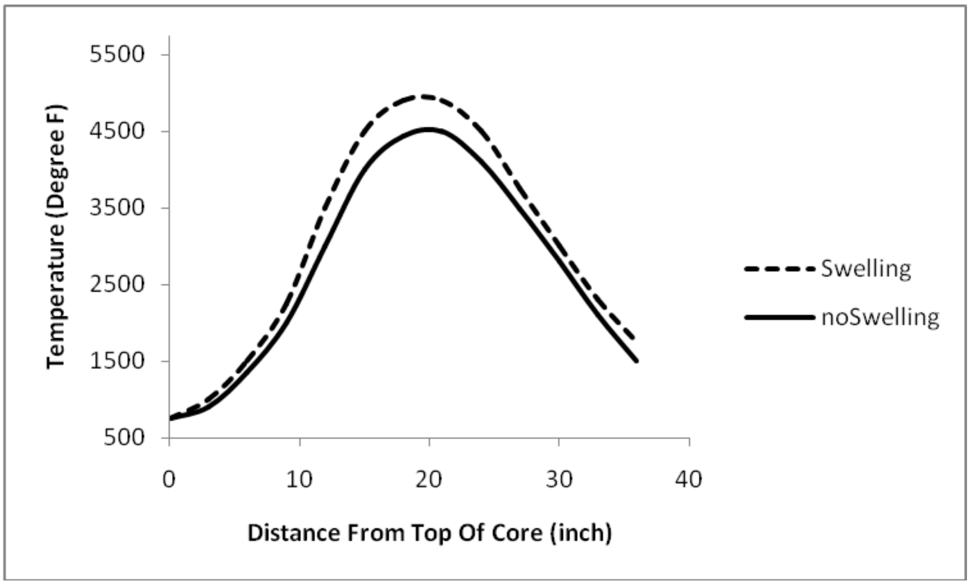


Fig.(6): Axial Temperature Distribution and The Effect of Swelling.

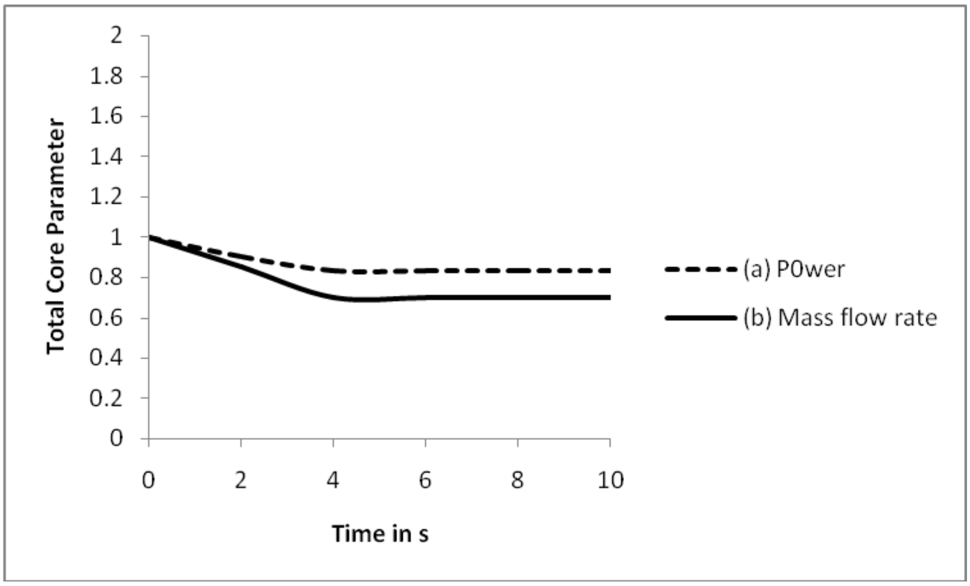


Fig.(7): Partial Loss of Flow ATWS Transient :

(a) Total Core Power

(b) Total Core Mass Flow Rate

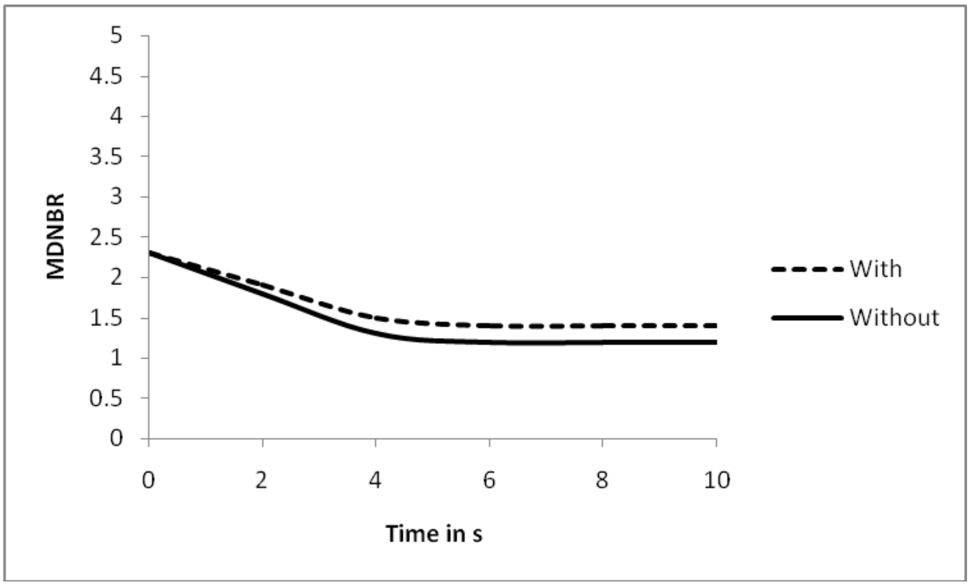


Fig.(8): Partial Loss of Flow ATWS Transient MDNBR for Transient With and Without

Kinetic Coupling.

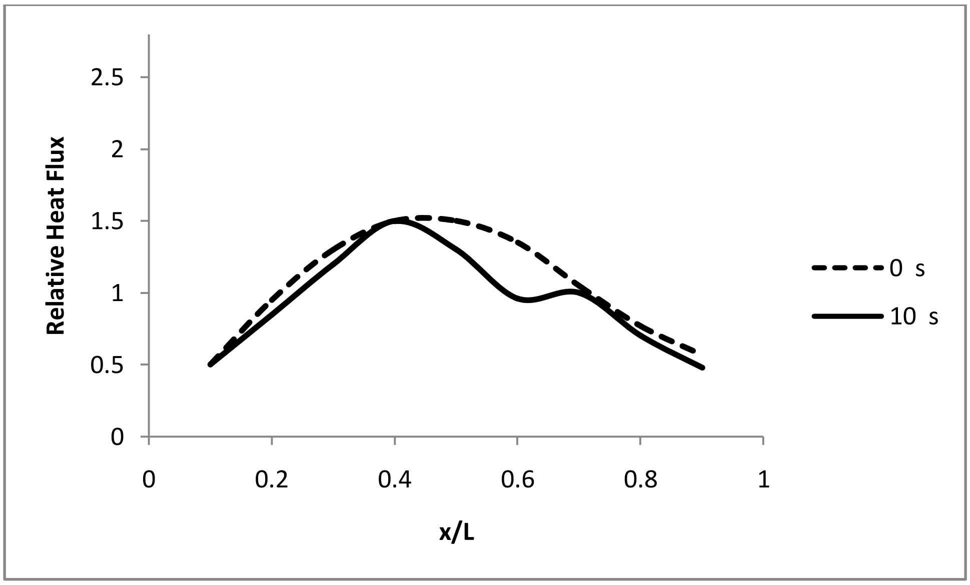


Fig.(9): Normalized Axial Heat Flux Profile at Time Steps 0 and 10 Seconds.

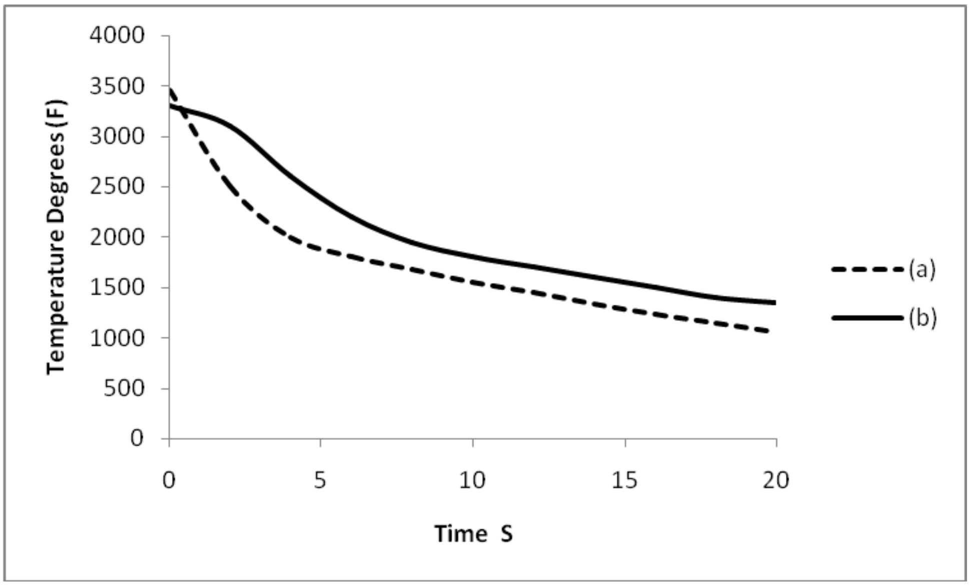


Fig.(10): Transient Peak Fuel Center Node Temperature:

(a) Fixed Gap Conductance.

(b) Dynamic Gap Conductance.