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A Computer Model for the Transient Analysis of Compact Research Reactors with Plate Type Fuel*

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A COMPUTER MODEL FOR THE TRANSIENT ANALYSIS OF COMPACT RESEARCH REACTORS WITH PLATE TYPE FUEL

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ABSTRACT

A coupled neutronics and core thermal-hydraulic performance model has been developed for the analysis of plate type U-Al fueled high-flux research reactor transients. The model includes point neutron kinetics, one-dimensional, non-homogeneous, equilibrium two-phase flow and heat transfer with provision for subcooled boiling, and spatially averaged one-dimensional heat conduction. The feedback from core regions other than the fuel elements is included by employing a lumped parameter approach. Partial differential equations are discretized in space and the combined equation set representing the model is converted to an initial value problem. A variable-order, variable-time-step time advancement scheme is used to solve these ordinary differential equations. The model is verified through comparisons with two other computer code results and partially validated against SPERT-II[1] tests. It is also used to analyze a series of HFIR[2] reactivity transients.

I. INTRODUCTION

The mechanics and thermal-hydraulics of rapid core transients for compact research reactors with plate-type highly enriched aluminum-based fuels are unique and quite different from those encountered in power reactors. Among these are ORNL's High Flux Isotope Reactor (HFIR) and Advanced Neutron Source (ANS) reactor concept, BNL's High Flux Beam Research Reactor (HFBRR), and INEL's Advanced Test Reactor (ATR). Many existing computer codes that have been developed for commercial power reactors have limited applicability for these research reactors. It is often necessary to modify the codes for flow and heat transfer in narrow rectangular channels with high coolant velocities, high heat fluxes, and for highly subcooled core conditions. Furthermore, commonly used numerical schemes are designed primarily for the analysis of mild system transients with relatively long periods. During a typical rapid high-flux research reactor transient, however, the time to initiation of boiling may be of the order of milliseconds which causes significant pressure spikes in the coolant channels. Moreover, poor heat transfer for the voided regions may lead to plate melting in less than a second.[3] Therefore, the need exists for a transient computer model that is valid for highly specialized compact reactor geometries and operating conditions for the analysis of rapid core transients with strong feedback.

In this work, a model which incorporates neutronics, two-phase thermal-hydraulics, and heat conduction for U-Al fueled research reactors is developed and applied to the analysis of HFIR reactivity transients. A fully implicit scheme is utilized for solution of the model equations which avoids numerical instabilities typically encountered due to semi-implicit coupling between the thermal, hydraulic, and neutronic modules. The analysis of HFIR transients is also supplemented with a study for consequences of local hot-plate melting by examining initial fuel disruption modes at the onset of melting and potential for molten fuel entrainment at expected hot-channel conditions.

II. MATHEMATICAL MODELS

The point kinetics approximation is used for the neutronics which is quite adequate for a small compact core with a smooth flux shape that is not expected to distort significantly during the transients. The fluid dynamics model used can be characterized as one-dimensional, non-homogeneous, equilibrium two-phase flow with a provision for subcooled boiling in the coolant channels. Since the coolant flow in compact research reactor cores is often a collection of single channel parallel flows that are coupled through fuel plates and by common bulk boundary conditions on each end, the conservation equations can be reduced to one-dimension

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as follows:

$$\frac{\partial}{\partial t}[(1-\alpha)\rho_l] + \frac{\partial}{\partial z}[(1-\alpha)\rho_l v_l] = -\dot{\Gamma}_g \quad (1)$$

$$\frac{\partial}{\partial t}(\alpha\rho_g) + \frac{\partial}{\partial z}(\alpha\rho_g v_g) = \dot{\Gamma}_g \quad (2)$$

$$\frac{\partial}{\partial t}[(1-\alpha)\rho_l H_l] + \frac{\partial}{\partial z}[(1-\alpha)\rho_l H_l v_l] = (1-\alpha)\left(\frac{Dp}{Dt} + v_l F_w\right) + \dot{q}_{vl} + \dot{q}_{sl} \quad (3)$$

$$\frac{\partial}{\partial t}(\alpha\rho_g H_g) + \frac{\partial}{\partial z}(\alpha\rho_g H_g v_g) = \alpha\left(\frac{Dp}{Dt} + v_g F_w\right) + \dot{q}_{vg} + \dot{q}_{sg} \quad (4)$$

$$\frac{\partial}{\partial t}[(1-\alpha)\rho_l v_l + \alpha\rho_g v_g] + \frac{\partial}{\partial z}[(1-\alpha)\rho_l v_l^2 + \alpha\rho_g v_g^2] + \frac{\partial p}{\partial z} + F_w - [(1-\alpha)\rho_l + \alpha\rho_g]g = 0 \quad (5)$$

In the momentum equation, the wall friction loss, F_w, is represented by the Darcy formula[4] in which the single phase friction factor is expressed by the Waggener correlation[5] in terms of channel relative roughness. For the two-phase friction multiplier, the Jones correlation[6] which includes high flow rate effects is preferred. The flow split between the liquid and vapor phases is effectively calculated using the Bankoff-Jones correlation[7] based on the local void fraction.

The equations representing the fluid dynamics model are transformed to a more convenient set of differential equations by expanding the time derivatives and employing various approximations including assumption of thermal equilibrium between the phases.[3] In terms of mass flux, pressure, and enthalpy, these equations are:

$$\frac{\partial G_m}{\partial t} = -\frac{\partial}{\partial z}\left(\frac{G_m^2}{\rho_m^*}\right) - \frac{\partial p}{\partial z} - F_w + \rho_m g \quad (6)$$

$$\frac{\partial p}{\partial t} = \frac{c^2}{\rho_m}\left[-\rho_m\frac{\partial G_m}{\partial z} - \frac{\partial \rho_m}{\partial h_m}(C_{mech} + e_c C_{therm})\right] \quad (7)$$

$$\frac{\partial h_m}{\partial t} = \frac{c^2}{\rho_m}\left[-\frac{1}{e_c}\frac{\partial G_m}{\partial z} + \frac{\partial \rho_m}{\partial p}\left(\frac{C_{mech}}{e_c} + C_{therm}\right)\right] \quad (8)$$

where

$$G_m = (1-\alpha)\rho_l v_l + \alpha\rho_g v_g \quad (9)$$

$$\rho_m = (1-\alpha)\rho_l + \alpha\rho_g \quad (10)$$

$$h_m = \frac{(1-\alpha)\rho_l H_l + \alpha\rho_g H_g}{\rho_m} \quad (11)$$

$$C_{mech} = \left(\frac{G_m}{\rho_m}\right)^*\left(\frac{\partial p}{\partial z} + F_w\right) \quad (12)$$

$$C_{therm} = (h_m - h_m^*)\frac{\partial G_m}{\partial z} - G_m\frac{\partial h_m^*}{\partial z} + \dot{q}_v + \dot{q}_{sl} + \dot{q}_{sg} \quad (13)$$

$$\rho_m^* = \frac{G_m^2}{(1-\alpha)\rho_l v_l^2 + \alpha\rho_g v_g^2} \quad (14)$$

$$h_m^* = \frac{(1-\alpha)\rho_l H_l v_l + \alpha\rho_g H_g v_g}{G_m} \quad (15)$$

$$\left(\frac{G_m}{\rho_m}\right)^* = (1-\alpha)v_l + \alpha v_g \quad (16)$$

A one-dimensional, spatially averaged heat conduction model through the fuel plates is coupled to the fluid dynamics model via the heat flux terms, $\dot{q}_{sl}$ and $\dot{q}_{sg}$, at the fuel element outer surface. The heat conduction model is based on a three region (two for Al cladding, one for fuel meat) flat fuel plate geometry as shown in Fig. 1.

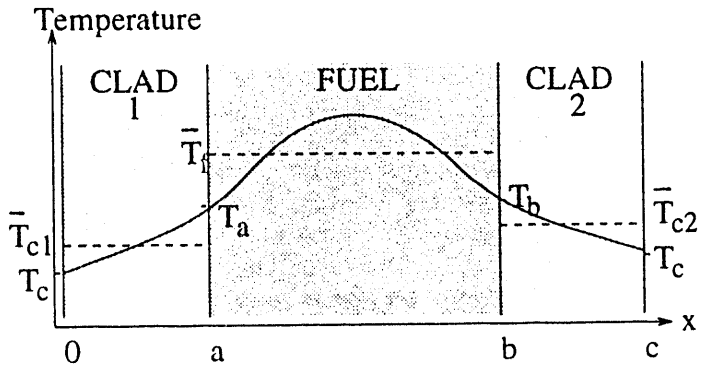

Fig 1: Typical temperature profile and average temperatures within the fuel plates.

Since the reactivity feedback from the fuel plates is calculated using average temperatures and the detailed power profile within the plates is generally not known, a differencing scheme is not essential. Therefore, the model developed in this work uses an analytical approach to obtain average temperatures based on integration of the heat conduction equation in each region:

$$\rho_c c_{pc} \frac{d\bar{T}_{c1}(t)}{dt} = \frac{1}{a}\left[(k_c \frac{\partial T_{c1}}{\partial x}|_a - (k_c \frac{\partial T_{c1}}{\partial x}|_0\right] + <Q_c> \quad (17)$$

$$\rho_f c_{pf} \frac{d\bar{T}_f(t)}{dt} = \frac{1}{b-a}\left[(k_f \frac{\partial T_f}{\partial x}|_b - (k_f \frac{\partial T_f}{\partial x}|_a\right] + <Q_f> \quad (18)$$

$$\rho_c c_{pc} \frac{d\bar{T}_{c2}(t)}{dt} = \frac{1}{c-b}\left[(k_c \frac{\partial T_{c2}}{\partial x}|_c - (k_c \frac{\partial T_{c2}}{\partial x}|_b\right] + <Q_c> \quad (19)$$

The evaluation of surface and interface heat flux terms in these equations involve surface temperatures, T_0 and T_c, and temperatures at material interfaces, T_a and T_b. In general, these temperatures can be expressed in terms of average cladding, average fuel, and coolant bulk temperatures as follows:

$$T = c_0(t) + c_1(t)\bar{T}_f(t) + c_2(t)\bar{T}_{c1/c2}(t) + c_3(t)T_l(t) \quad (20)$$

This definition of surface and interface temperatures is formally exact as long as the weight factors c_0 to c_3 are calculated properly as a function of time. In the model, a steady-state approximation for these coefficients is assumed applicable at each time step for the thin fuel plates with high thermal conductivity. The coefficients c_0 to c_3 and, therefore, the surface and interface temperatures are still time dependent since both material properties and the heat generation rate are updated at each time step.

This analytical treatment limits the number of equations to be advanced in the time domain and allows implicit coupling of the heat conduction model with the rest of the equation set without a significant compromise in computation time. In the analysis, volumetric heat sources and material properties are assumed piecewise constant in fuel and cladding. The material properties are considered as functions of temperature and updated at each time step through interpolations between the tabulated data.

A boiling curve is used in the model to govern the selection of the heat transfer mode from fuel plate surface to the coolant in each boiling regime. For single-phase liquid flow, the Gambill-Bundy correlation[8] is preferred for the heat transfer coefficient which takes high velocity and narrow coolant channel effects into account properly. For single-phase vapor flow, the Dittus-Boelter correlation for super-heated steam[9] is used. And in the bulk boiling regime, Thom's correlation[10] which is modified version of Jens-Lottes correlation[11] is used for heat transfer coefficient.

The highly subcooled, high velocity coolant flow typically delays the boiling incipience, reducing the margin between the *incipient boiling* and *departure from nucleate boiling* heat fluxes. And, the burnout condition is often experienced in the subcooled boiling regime during rapid high-flux research reactor transients at full flow and full power. Subcooled boiling is essentially formulated in this work as an approximation to the surface heat-flux term in the single-phase flow regime by considering boiling, non-boiling, and condensation heat flux components. The approximation is based on the assumption that vapor enthalpy remains constant and uses the following equation for the vapor generation rate:

$$\dot{\Gamma}_g = \frac{\dot{q}_B - \dot{q}_l}{H_{fg}} \quad (21)$$

where $\dot{q}_B$ and $\dot{q}_l$ are boiling and condensing heat flux components. To express condensation of vapor, the Hancox-Nicoll correlation[12] is used. In this regime, Levy's approach[13] is adopted to formulate the bubble departure dynamics which considers both thermal and hydrodynamic factors.

After the critical heat flux is reached, vapor blanketing of the surfaces causes the heat transfer to deteriorate significantly, resulting in a rapid and substantial increase in fuel temperatures. The Bowring correlation[14] which employs four optimized pressure parameters is used for marking this transition to film boiling. This correlation is preferred since it has a wide range of applicability in terms of mass flux. The Tong and Young model[15] is adopted to simulate film boiling heat transfer conditions which allows the liquid phase to be subcooled. Post-dryout heat transfer may also be encountered during low-power, low-flow reactor transients. This regime is modeled by single-phase vapor flow.

In addition to these basic models, a provision is made to include the safety system response and the feedback from core regions other than the fuel elements by providing the capability to evaluate generic simultaneous algebraic and ordinary differential equations. In application to HFIR core, for example, the feedback from the target and reflector regions is included by employing a lumped parameter approach with a one-node representation of the metal, and a two-node representation of the coolant in these non-fueled regions. The HFIR safety system response is also incorporated by approximating the movement of the safety quadrants with a second order differential equation.

III. METHOD OF SOLUTION

A direct numerical solution of the multiple point finite-difference approximation to the equations representing this core dynamics model has been employed in a fully-implicit fashion. The non-linear partial differential equations of fluid dynamics are discretized in space along the flow direction leading to a large initial value problem of ordinary differential equations. Similarly, three components of heat conduction along the thickness of the fuel plates are divided into finite difference cells in axial direction.

Both discretized fluid dynamics and heat conduction equations, when organized in a vector form, constitute sparse, band structured matrices whose properties can be exploited mathematically to improve computational efficiency of the time advancement algorithms. Sequential iteration schemes used in large nuclear systems analysis codes perform iterations over the components of the global solution vector at a fixed point in time. Even though the solution is implicit within an individual component, data exchange between the modules are explicit in those schemes with time levels of feedback being dictated by the order of block execution. This semi-implicit solution method, however, often causes numerical instabilities due to strongly non-linear coupling between modules for the analysis of rapid thermal transients which involve large reactivity changes and strong feedback. In particular, an explicit treatment of the data exchange between thermal-hydraulics and point kinetics effectively reduces the reactivity feedback to a linear approximation.

Therefore, a fully implicit numerical solution scheme is utilized in this model for the solution of the whole equation set describing the point kinetics, fluid dynamics, heat conduction, other core components, and safety system response models. The whole set constitutes a sparse but irregular matrix for which a simultaneous block iteration scheme should be performed. Although this fully-implicit treatment is more rigorous and avoids numerical instabilities, it still requires a significant amount of computer time.

A variable-order, variable-time-step time-advancement scheme based on the forward and backward Euler method (FEBE) is used to solve this initial value problem.[16] To be effective, a solution algorithm must be able to cope with severe stiffness originating from the real and complex eigenvalues of the Jacobean matrix. Most widely used stiff-solvers have a narrow stability margin due to the size of the matrices involved (a typical simulation requires simultaneous solution of as many as 500 ODE's). The time advancement scheme used in this work tackles this problem by using a generally low order algorithm whose accuracy is offset by the use of an extrapolation technique in addition to classical implicit iterations.[17] By combining extrapolations and iterations, FEBE takes advantage of better economy compared to purely iterative methods, and improved robustness compared to the standard extrapolation methods. The introduction of an extrapolation scheme reduces error contributions from the discretization of the time derivatives and the linearization of the matrices. In addition, information from extrapolation tables provides important information for time step size control.

IV. RESULTS

The model described above has first been verified by benchmarking with two existing validated computer codes for a series of transients. Partial validation using SPERT-II tests has also been performed. In the second part, an illustrative analysis of HFIR reactivity transients has been conducted using the verified model .

A. Summary of Model Verification Efforts

The model developed has been benchmarked against the RELAP5/Mod2 and REACC codes.[18,19] Partial comparisons with RELAP5 have been performed for a series of transients including a range of flow and heat transfer regimes. Since theRELAP5 model used for these comparisons did not explicitly include neutronic behavior and corresponding feedback models, only the verification of the basic thermal-hydraulic schemes has been possible by providing the power history as input. Good agreement is obtained between the model and RELAP5 results for all variables in the single-phase region.[3] Three sets of comparisons have been performed in the boiling range covering subcooled, bulk, and film boiling heat transfer conditions. Despite some discrepancies regarding the initiation of various boiling regimes due to modelling differences, the agreement between the results for the key state variables has been satisfactory. For the two-phase flow comparison cases, RELAP5 results have exhibited a high frequency oscillatory behavior for coolant pressure and velocity for the short time frame in which the boiling incipience is realized. This discrepancy is attributed to the two-momentum equation formulation of RELAP5/Mod2. The amplitude of these oscillations is marginal and, other than the transitory region, results are in good agreement in general.

The neutronic behavior of the model and the feedback mechanism were also compared to a previous HFIR core transient model, REACC, which is a lumped-parameter model with point neutron kinetics.[19] A

moderate reactivity transient was chosen for comparisons in order to avoid boiling which the REACC code is not designed to handle. Good agreement was obtained for the calculated power and all other parameters except coolant temperatures and heat fluxes.[3] Recognizing the inherent differences between the two models, the overall agreement between the results has been considered satisfactory.

And finally, in an attempt to validate the overall coupled neutronic and thermal-hydraulic behavior of the model, a series of comparisons were performed against the SPERT-II experiments. In order to facilitate these comparisons, several components of the model developed for the HFIR core such as various feedback mechanisms are removed and only a single-channel analysis is performed. Therefore, only a partial validation of basic thermal-hydraulic and point kinetics modules of the model is possible for single phase flow. Several comparisons with experimental data are performed for transients with different reactivity insertion rates, coolant inlet enthalpies, and inlet flow rates. The results of these limited comparisons have also been satisfactory.[3] As an example, the effect of coolant inlet flow on the calculated and measured power is presented in Fig. 2.

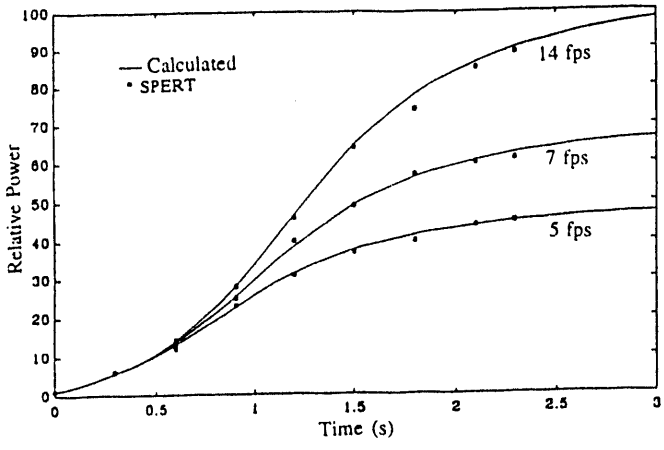

Fig. 2: SPERT-II comparisons for calculated and measured power with different inlet flow rates.

B. Analysis of HFIR Reactivity Accidents

The analysis of rapid HFIR core transients has been conducted to demonstrate the capabilities of the model and its applicability to safety analysis accident sequences of compact research reactors. The model developed has been used to describe the transient behavior of the HFIR core for both nominal and hot conditions of the inner and outer fuel elements, both at the beginning and at the end of the fuel cycle.[20] An average HFIR channel and a fuel plate for both inner and outer elements have been used to model the nominal conditions which is defined by a core operating at nominal power and heat flux, with no uncertainties in the operational characteristics. A nine channel hot-assembly model with two hot-channels and a hot-plate has been used to describe hot conditions and to examine fuel damage propagation. The dynamic behavior of the reactor has been conservatively assumed to be dependent only on nominal conditions and, therefore, the feedback from hot-assembly has been neglected.

The model has been used to analyze a series of HFIR reactivity transients associated with malfunction of the reactor control system, formation of voids in the regions with positive reactivity coefficient, and pump start in cold loop. Each transient has been studied for four operating states of the reactor which are characterized by different initial conditions and safety system settings. A summary of the sequences considered and the results of the analysis are presented in Table 1.

Table 1: Summary of HFIR Transients Analyzed in the Order of Decreasing Expected Frequency.[21]

Transient Description	Classification	Results
Protected cold water accident	Moderate	Non-damaging
Protected uncontrolled shim withdrawal	Infrequent	Non-damaging
Unprotected cold water accident	Limiting fault	Non-damaging
Unprotected uncontrolled shim withdrawal	Limiting fault	Nominal plate melt
Shim-regulating cylinder ejection	Limiting fault	Non-damaging
Single shim-safety quadrant drop	Limiting fault	Non-damaging
Accident associated with optimum target voids	Limiting fault	Partial hotplate melt

Due to substantial amount of positive reactivity involved, some of the unprotected uncontrolled shim withdrawal sequences are expected to result in significant fuel damage and whole core involvement in all operating states considered. Other than that, only a hypothetical

optimum void accident is expected to result in partial hot-plate melting in Mode-1, at full-power and full-flow, and at the end of fuel cycle.

The analysis of the optimum void accident has been the most challenging among all transients considered due to the amount of reactivity involved (1.8$) and the short time frame it is assumed to be inserted (30 msec) to the core. The flux trap region of the HFIR core is highly over-moderated, and introduction of voids to this region through formation of flow vortex or blockages results in a positive reactivity contribution. The analysis of this sequence indicates that the time to boiling is about 50 msec which leads mild pressure spikes in the coolant channels. Immediately following the inception of boiling, film blanketing of the heat transfer surfaces leads to critical heat flux while power is still rising.

With the transition to film boiling, the sudden change in sonic velocity at the boiling boundary and the discontinuities in heat transfer impose extra problems on the stability of the numerical solution, forcing the time-advancement scheme to use uncharacteristically small time steps. Partially due to these numerical problems, the hot channel pressure and velocities exhibit chaotic behavior at the inception of film boiling as shown in Fig. 3.

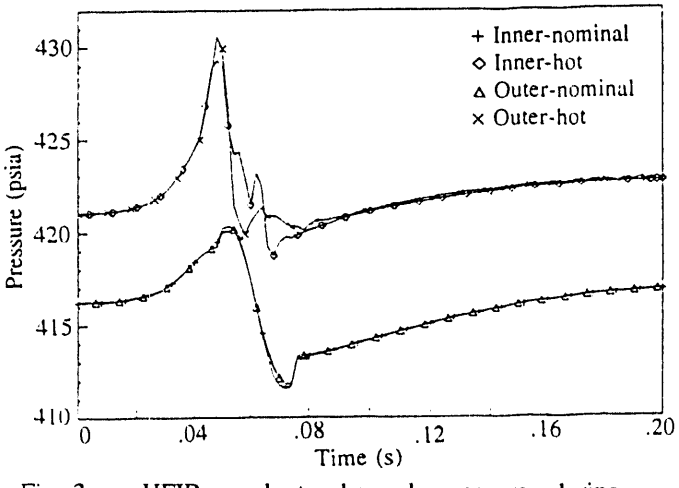

Fig. 3: HFIR coolant channel pressure during optimum void accident at axial mid-plane.

The thermal variables, however, remain largely unaffected. With the sudden decrease in heat transfer, hot-plate temperatures for the nodes with high power density keep increasing (Fig. 4). Although temperatures do not reach the generally accepted Al melting point, the difference is marginal and it is assumed that the hot-plate contains sufficient heat to initiate melting.

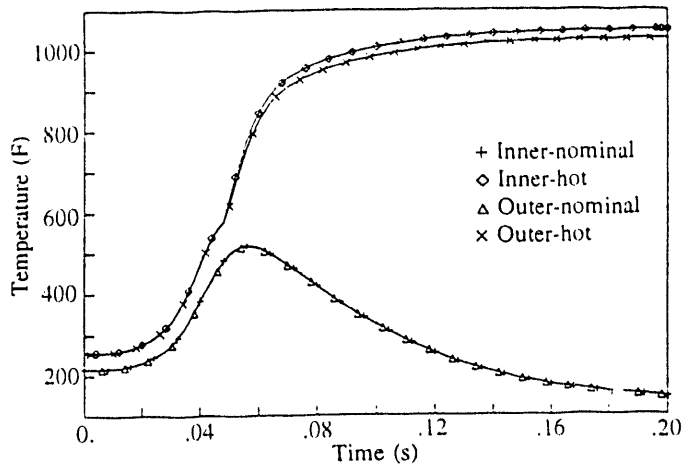

Fig. 4: Average fuel temperatures during optimum void accident above core axial mid-plane.

Since design requirements and transient acceptance criteria require that the nominal fuel structural integrity and a coolable geometry to be maintained during such an accident, the analysis is also supplemented with a study for the consequences of local hot-plate melting. Scoping calculations are performed to assess the potential for additional melting beyond the hot-plate.[3] The initial disruption modes of the fuel plates due to thermal and mechanical stresses, U_3O_8-Al chemical interactions, and internal fission gas pressure are examined at the onset of fuel melting. The molten fuel entrainment at the expected hot-channel conditions and the consequences of partial flow blockages are also addressed by examining the dynamics of molten fuel accumulation.

In summary, once the poor heat transfer in regions with film boiling initiates melting, the fuel is expected to expand in a foam like form and partially fill the coolant channels. Correlations for the entrainment inception velocities for saturated steam[22] suggest that the high pressure drop across the HFIR core will force this molten fuel out of the core under the calculated channel conditions. Nevertheless, possible flow blockages are also modeled as a sudden reduction in the coolant flow area with varying size. In Fig. 5, the inner fuel element hot-assembly temperatures are shown (for z/L=0.7) during optimum void accident followed by a flow blockage in which a 75% reduction in flow area is realized. As shown in Fig. 5, the hot-plate (#5) and two neighboring plates (#4 and #6) experience film boiling at heat transfer surfaces; however, neighboring plate temperatures do not reach solidus until complete channel blockages are realized. In the final analysis, it is concluded that the damage will most likely be limited to melting in a portion of the hot-

plate, and the HFIR can withstand the optimum void accident with only a minor release of fission products to the primary coolant.

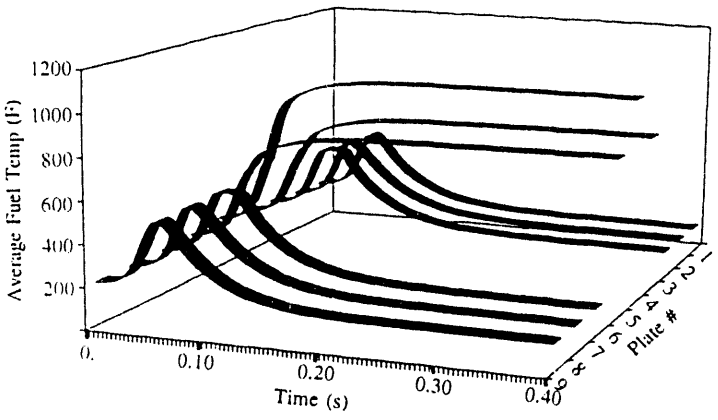

Fig. 5: Hot-assembly temperatures during optimum void accident followed by a flow blockage.

V. SUMMARY and CONCLUSIONS

A dynamic model for the analysis of rapid core transients of U-Al dispersion fueled research reactors has been developed. The model includes point neutron kinetics, two-phase thermal-hydraulics, and a one-dimensional heat conduction for average fuel temperatures. The flow and heat transfer regimes considered are single-phase liquid flow, subcooled boiling, bulk boiling, film boiling, and single-phase vapor flow. Appropriate correlations are described for friction factor, heat transfer coefficient, void fraction, incipient boiling, and critical heat flux.

The model developed is benchmarked against the RELAP5/Mod2 and REACC codes for verification of the thermal-hydraulics and neutronics/feedback behavior, respectively. A series of comparisons with SPERT-II experiments are made to validate the coupled neutronic and thermal-hydraulic performance. The model is also applied to the HFIR core; specifically, a series of reactivity transients are analyzed related to control system malfunctions, pump starts in cold loop, and voids in regions with positive reactivity coefficient. The results indicate that the worst credible protected reactivity accident for the HFIR is due to optimum amount of void formation in the target region. For this case, a partial hot-plate melting is predicted in Mode-1, at full power, and at the end of fuel cycle conditions. However, additional scoping calculations suggest that fuel damage is limited to the hot-plate only and the mechanical integrity of the rest of the fuel elements is maintained. These calculations demonstrate that the model developed in this work can be an effective tool in safety analysis and design of compact plate type fueled research reactors.

A major limitation of this model observed in the analysis of HFIR transients is the 1-D heat conduction inside the fuel plates. The lack of axial conduction results in unrealistically high local temperatures for the axial segments which experience film boiling while the neighboring segments are in much lower temperatures. In order to circumvent this situation, axial heat conduction can be introduced to the model in an explicit fashion. In other words, instead of using classical 2-D heat conduction, two 1-D conduction problems can be synthesized. Since this formulation would require few additional equations, it would retain the advantages of the current model with the ability of calculating the effects of axial heat transfer.

Also, the model does not deal with radial variations in power density and flow which may be significant in thin fuel plates and narrow coolant channels. Therefore, local fuel segregation/blister effects inside the fuel plates and local small flow blockages cannot properly be analyzed with this model. The analysis of this kind of local power peaking (hot-spots) and undercooling events would require more rigorous 3-D heat conduction and 2-D fluid flow models.

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NOMENCLATURE

c	sonic velocity
c_p	heat capacity
e_c	energy conversion factor
F_w	wall friction term
g	gravitational acceleration
G	mass flux
H	enthalpy
p	pressure
q_s	surface heat transfer rate per unit volume
q_v	volumetric heat generation rate
Q	volumetric heat generation rate in plates
t	time
T	temperature
v	velocity

z axial flow direction
α void fraction
Γ_g vapor generation rate
ρ density

Subscripts

c Al cladding
f fuel meat
g vapor phase
l liquid phase
m mixture

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