
A Time-Dependent Neutron Transport Model and its Coupling to Thermal-Hydraulics

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Abstract: A new neutron transport code for time-dependent analyses of nuclear systems has been developed. The code system is based on the well-known Discrete Ordinates code DORT, which solves the steady-state neutron/photon transport equation in two dimensions for an arbitrary number of energy groups and the most common regular geometries. For the implementation of time-dependence a fully implicit first-order scheme was employed to minimize errors due to temporal discretization. This requires various modifications to the transport equation as well as the extensive use of elaborated acceleration mechanisms. The convergence criteria for fluxes, fission rates etc. had to be strongly tightened to ensure the reliability of results. To perform coupled analyses, an interface to the GRS system code ATHLET has been developed. The nodal power densities from the neutron transport code are passed to ATHLET to calculate thermal-hydraulic system parameters, e.g. fuel and coolant temperatures. These are in turn used to generate appropriate nuclear cross sections by interpolation of pre-calculated data sets for each time step. Finally, to demonstrate the transient capabilities of the coupled code system, the research reactor FRM-II has been analysed. Several design basis accidents were modelled, like the loss of offsite power, loss of secondary heat sink and unintended control rod withdrawal.

1. INTRODUCTION

Routine transient analyses for large nuclear power plants are nowadays mainly performed employing nodal coarse-mesh diffusion methods, usually in two energy groups only. Although these calculations have been proven to be sufficient for many accident scenarios and for a lot of different types of reactors, there are clearly situations, where the few-group diffusion theory approach is expected to yield results, which could be improved by applying neutron transport methods. However, the computational effort to solve the transport equation for a given system is usually orders of magnitude larger than solving the same problem using the much simpler diffusion approximation. Hence it is not very surprising, that even today transport theory is almost exclusively used for steady-state investigations. This includes e.g. fuel assembly calculations, reactor design studies or shielding analyses, to name only a few of the diverse applications of neutron transport codes.

In this document it will be shown, that despite the complexity of reactor problems (especially when coupled to thermal hydraulics), it has meanwhile become possible to perform accident analyses for some realistic nuclear systems using transport theory. For this purpose we employed a fully implicit, unconditionally stable time discretisation scheme and the classical Discrete Ordinates method, as it is implemented in the well-known computer code DORT (Oak Ridge National Laboratories)[1]. The decision to use a deterministic code instead of a

Monte-Carlo program was relatively easy, since only the deterministic approach can provide the numerical accuracy required in time-dependent calculations. Stochastic methods, in contrast, always suffer from inherent uncertainties, which would make the analysis of e.g. slow reactivity transients unfeasible.

It should further be emphasised, that the use of a fully implicit scheme in conjunction with transport theory is by now rather unique and is essentially free from any additional approximations, as opposed to the popular quasistatic approaches. The basic assumption of the generic quasistatic method is the slow variation of the spatial shape or form function. The time evolution of the system is then determined by a set of extended point kinetics equations. While the assumption of a slowly varying flux shape may be excellent in many slow transients, it can fail for severe accident conditions and for fast or spatially inhomogeneous reactivity insertions. In any case, such an approximate solution should always be compared to a numerically "exact" solution, which can only be provided by an unconditionally stable implicit method. In this sense, the code system presented in the following sections may well serve as a reference tool for transient analyses for a wide range of nuclear systems and accident conditions.

2. THEORY

The time-dependent neutron transport equation (the symbols and abbreviations are well-known and standard in the literature):

$$\begin{aligned}
 \underbrace{\frac{1}{v(E)} \frac{\partial}{\partial t} \psi(\vec{r}, \vec{\Omega}, E, t)}_{\text{Time Variation}} &= \underbrace{-\vec{\Omega} \cdot \nabla \psi(\vec{r}, \vec{\Omega}, E, t)}_{\text{Leakage Term}} - \underbrace{\Sigma_t(\vec{r}, E) \psi(\vec{r}, \vec{\Omega}, E, t)}_{\text{Interaction Term}} + \underbrace{\int dE' \int_{4\pi} d\vec{\Omega}' \Sigma_s(\vec{r}, E' \rightarrow E, \vec{\Omega}' \cdot \vec{\Omega}) \psi(\vec{r}, \vec{\Omega}', E', t)}_{\text{Scattering Term}} \\
 &+ \underbrace{q_{\text{extern}}(\vec{r}, \vec{\Omega}, E, t)}_{\text{external Source}} + \underbrace{\chi_p(E)(1-\beta) \int dE' v \Sigma_f(\vec{r}, E') \phi(\vec{r}, E', t)}_{\text{Prompt Fission}} + \underbrace{\sum_{l=1}^6 \chi_d^l(E) \lambda_l C_l(\vec{r}, t)}_{\text{Delayed Fission}}
 \end{aligned}
 \tag{1}$$

together with the usually six precursor equations:

$$\frac{\partial}{\partial t} C_l(\vec{r}, t) = -\lambda_l C_l(\vec{r}, t) + \beta_l \int_0^{\infty} dE' v \Sigma_f(\vec{r}, E') \phi(\vec{r}, E', t) \quad l = 1 \dots 6
 \tag{2}$$

constitutes a system of partial differential equations, which describes the behaviour of any nuclear system. However, the vast majority of transport codes can only handle the simpler steady-state equation:

$$\underbrace{\bar{\Omega} \cdot \nabla \psi(\bar{r}, \bar{\Omega}, E)}_{\text{Leakage Term}} + \underbrace{\Sigma_t(\bar{r}, E) \psi(\bar{r}, \bar{\Omega}, E)}_{\text{Interaction Term}} =$$

$$\underbrace{q_{\text{extern}}(\bar{r}, \bar{\Omega}, E)}_{\text{External Source}} + \underbrace{\int dE' \int_{4\pi} d\bar{\Omega}' \Sigma_s(\bar{r}, E' \rightarrow E, \bar{\Omega}' \rightarrow \bar{\Omega}) \psi(\bar{r}, \bar{\Omega}', E')}_{\text{Source Term}} + \underbrace{\frac{1}{k_{\text{eff}}} \chi(E) \int dE' v \Sigma_f(\bar{r}, E') \phi(\bar{r}, E')}_{\text{Fission Term}}$$

(3)

Here the time-derivative and all time-dependencies have been omitted and a separate treatment of the neutron precursors is no longer necessary. The effective multiplication factor k_{eff} may be dropped, if an external, fixed source is present and the system itself is subcritical (otherwise no solution can exist). In what follows we will show, that the solution of the time-dependent equation 1 can be reduced to a series of solutions of the simpler steady-state type equation 3.

We impose the implicit discretisation scheme and approximate the time derivative in equation 1 as follows (H denotes the full transport operator):

$$\frac{1}{v} \frac{\partial}{\partial t} \psi(\bar{r}, \bar{\Omega}, E, t) \approx \frac{\psi(t + \Delta t) - \psi(t)}{v \Delta t} =: \frac{\psi^{(n+1)} - \psi^{(n)}}{v \Delta t} = H^{(n+1)} \psi^{(n+1)}$$

(4)

A similar relation holds for the six precursor equations. Inserting this expression in equations 2 and 3 and sorting by indices n and $n+1$ yields:

$$-\bar{\Omega} \cdot \nabla \psi^{(n+1)}(\bar{r}, \bar{\Omega}, E) + \underbrace{\left(\Sigma_t + \frac{1}{v \Delta t} \right)}_{\Sigma'_t} \psi^{(n+1)}(\bar{r}, \bar{\Omega}, E) = \int dE' \int_{4\pi} d\bar{\Omega}' \Sigma_s(\bar{r}, E' \rightarrow E, \bar{\Omega}' \rightarrow \bar{\Omega}) \psi^{(n+1)}(\bar{r}, \bar{\Omega}', E') +$$

$$\underbrace{\left[\chi_p(E)(1 - \beta) + \sum_{l=1}^6 \chi_d^l \lambda_l \gamma_l \beta_l \right]}_{\chi'(E)} \int dE' v \Sigma_f(\bar{r}, E') \phi^{(n+1)}(\bar{r}, E') + \underbrace{\sum_{l=1}^6 \chi_d^l \lambda_l \gamma_l \frac{1}{\Delta t} C_l^{(n)}(\bar{r})}_{q'} + \frac{1}{v \Delta t} \psi^{(n)}(\bar{r}, \bar{\Omega}, E)$$

(5)

This is nothing else but the steady-state transport equation for the fluxes $\psi^{(n+1)}$ with a modified total cross section, a modified fission spectrum and a "time source" term, which comprises of the fluxes and precursors of timestep n . These quantities have already been calculated in the previous time step and are thus known at timestep $n+1$.

Therefore, the extension of a steady-state transport to time-dependence is rather straightforward. In principle, the only changes to be made are to build a fixed-source term for each timestep, to modify the total cross section and fission spectrum appropriately and call the transport code for each time step over and over again. Additionally, feedback effects can be accounted for by allowing the cross sections to be explicit functions of time, i.e. by varying them at each time step as well. The main drawback of this implicit scheme is the fact, that each timestep will be as expensive to solve as the steady-state problem. If already the steady-state equation of the system under consideration is difficult to solve, a transient with possibly thousands of time steps will normally take an unacceptable amount of computing time.

3. IMPLEMENTATION OF THE CODE DORT IN A TIME-DEPENDENT SCHEME

The last section described the formal background of extending a steady-state code to treat time-dependent problems. However, the methods developed to solve the transport equation are numerous, and it is a priori not obvious, which of those methods is the most promising for the application in transient analyses. Several available computer codes were checked: it turned out, that the classical Discrete Ordinates (S_N -) approach still compares superior to such codes, which e.g. implement integral transport methods or the P_N -formulation of the transport equation. This statement is certainly not a general one, but it is apparently valid for many thermal systems with regular geometry, e.g. research reactors like the HFR, which have been extensively checked for later applications (cf. section 5).

We finally decided to choose the code DORT from ORNL, since this program performed slightly better than its Los Alamos counterpart TWODANT [2] in a couple of test calculations. We give a short summary of DORT's main features:

- Arbitrary S_N -order, arbitrary P_N - (Legendre-) expansion of cross sections.
- Arbitrary number of energy groups
- 2D-Geometries available: x-y, r- ϕ • r-z
- Solution algorithm is the usual inner-outer iteration scheme
- Acceleration of inner iterations by the Coarse-Mesh-Rebalance method (CMR)
- "Error mode extrapolation" for acceleration of outer iterations
- Additional diffusion module available in code package
- Upscatter acceleration by the so called "Upscatter-Rebalance" method
- Code offers a broad parameter choice to optimise the solution progress

The code DORT was implemented as a subroutine in an existing multigroup 2D-diffusion code. This was advantageous, since large parts of input- and post processing routines, the thermal hydraulics interface to the GRS code ATHLET and several other useful procedures could directly be taken over to DORT.

Diffusion and transport calculations are now available as two different options in a single computer code system. To achieve better convergence and higher accuracy, DORT was also adapted to 64-bit architectures and is now running in double precision mode on a large variety of UNIX-platforms, including IBM-AIX, COMPAQ/Digital-Unix and Linux.

As already mentioned above, the time-dependent application of DORT requires several modifications to cross sections and source terms, which have to be performed for each single time step. The modifications of total cross section and fission spectrum are easily implemented and can be done during cross section generation. To build an appropriate "time source" term turned out to be more complicated. The usual way to pass a fixed source to DORT is via a moment representation, i.e. a function expansion of the source term in Spherical Harmonics. This is sufficient for the isotropic precursor terms in q' (equation 5), but not for the flux values of the previous time step, which have to be passed in their explicit angular representation to the next timestep (cf. equation 5). This option is commonly not provided for in transport codes and had to be added to the DORT code.

To improve the convergence rate of the outer iterations, several actions had to be taken. First of all, the rather ineffective upscatter rebalance scheme in DORT was replaced by an upscatter "cycle". Instead of inverting the group transport operator only once per energy group and per outer iteration, a loop over all thermal groups is executed within one outer iteration. This ensures, that all groups receiving upscatter are properly converged. Without this change, a clean error decay in systems with a pronounced thermal spectrum cannot be guaranteed. But a clean error decay is an essential requirement for an effective acceleration of the outer iteration cycle and is hence vital to the performance of the code. This is demonstrated in **Erreur! Argument de commutateur inconnu.**, where the error decay of the fission density in a steady-state calculation (13 energy groups, four of which receive upscatter) is shown.

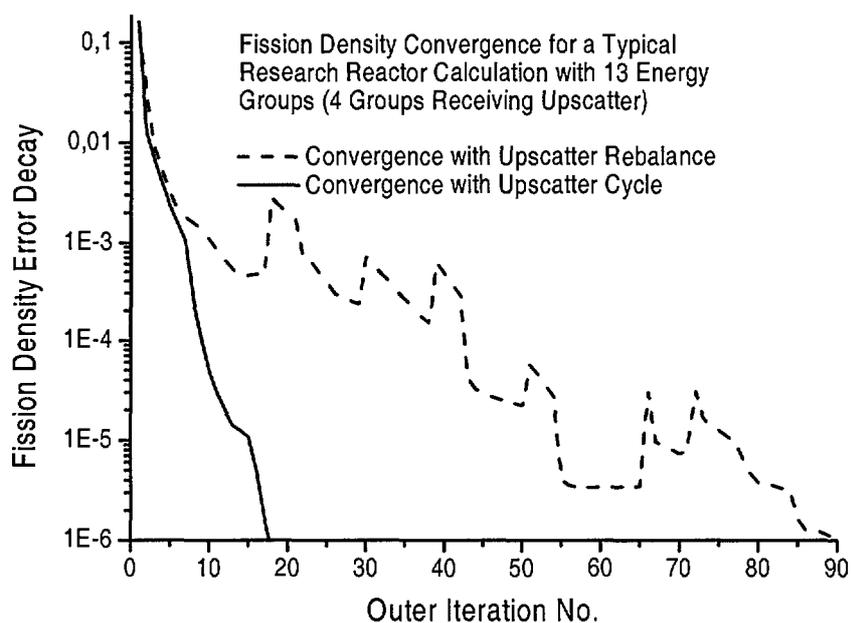


Figure **Erreur! Argument de commutateur inconnu.**: Effectiveness of upscatter cycle scheme compared to the upscatter rebalance as implemented in DORT

In this example, a typical research reactor system was chosen. The standard DORT code, employing the upscatter rebalance scheme, required almost 100 outer iterations to bring the fission density to a convergence of better than 10^{-6} . The error mode extrapolation additionally caused the numerous spikes, giving evidence of the insufficient error decay. Although only 4 thermal groups were used here, the overall-convergence of the problem is very bad. The application of the upscatter cycle reduces the number of outer iterations to less than 20.

To further enhance the convergence properties of the code, advantage was taken from the use of the Chebyshev polynomial method to perform a fission density extrapolation. The corresponding routine can optionally replace the error mode extrapolation, as it is implemented in standard DORT. The effectiveness of the Chebyshev scheme depends strongly on the so called dominance ratio of the system under consideration, i.e. the ratio of

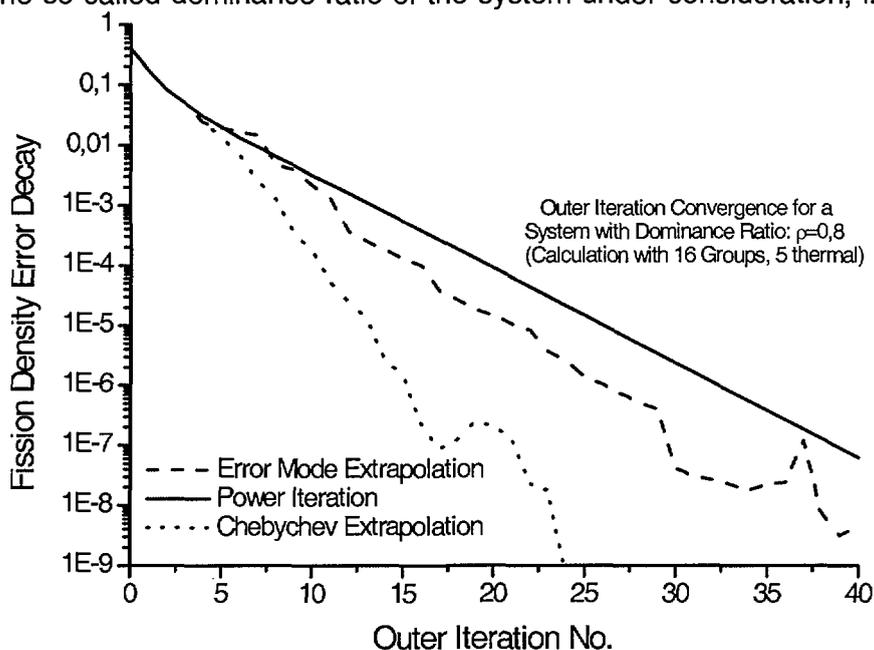


Figure **Erreur! Argument de commutateur inconnu.**: Comparison of extrapolation mechanisms for a system with a dominance ratio of 0.8

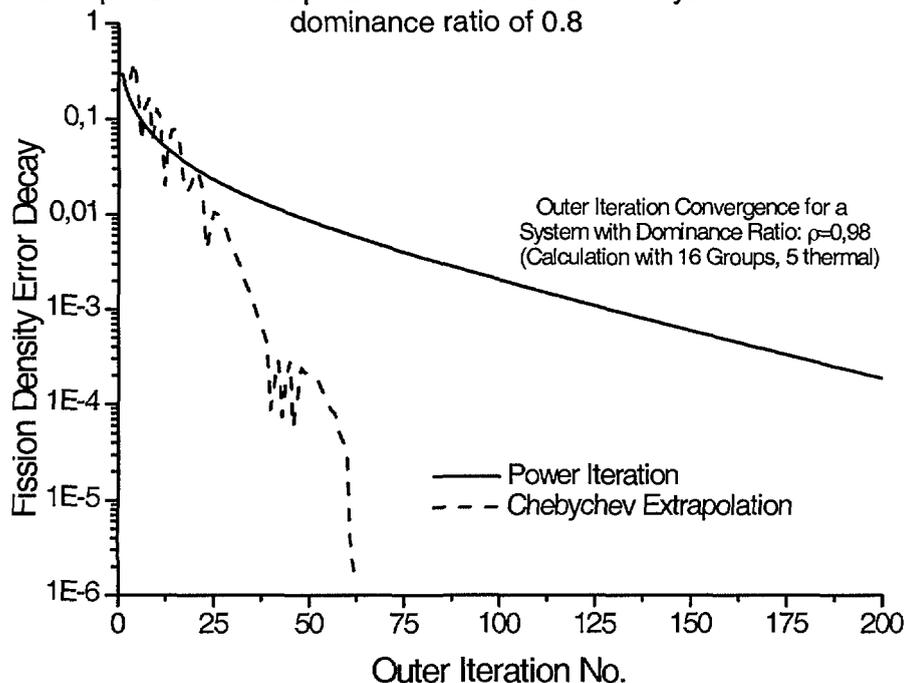


Figure **Erreur! Argument de commutateur inconnu.**: Comparison of extrapolation mechanisms for a system with a dominance ratio of 0.98

the two largest eigenvalues, but it is normally at least as effective as the error mode extrapolation.

Two examples are shown in **Erreur! Argument de commutateur inconnu.** and **Erreur! Argument de commutateur inconnu.** for systems with a dominance ratio of 0.8 and 0.98 respectively. In **Erreur! Argument de commutateur inconnu.** we compare Chebyshev and error mode extrapolation with the unaccelerated scheme, i.e. the well-known power iteration. Due to the rather small dominance ratio, the error decays quite quickly and, for the power iteration scheme, attains an asymptotic rate. The acceleration achieved by extrapolation mechanisms is modest and amounts roughly to a factor of 2-3, with the Chebyshev method slightly favourable over the simpler error mode extrapolation. In contrast, for a system with large dominance ratio (**Erreur! Argument de commutateur inconnu.**), Chebyshev extrapolation is extremely effective, as can be seen from the massive error reduction, compared to the asymptotic error decay rate for the unaccelerated power iteration.

Despite the use of effective acceleration schemes for inner and outer iterations, solving for a single timestep is still expensive. However, one can benefit from the fact, that the solution of timestep n is in general a quite good approximation (unless time steps are too large) to the fluxes to be calculated at timestep $n+1$, and can be used as an excellent starting guess for the iterative scheme. This guess may still be improved by doing a time-like extrapolation. By constructing “reactor periods” from timestep $n-1$ and n , which are resolved in space as well as in angle and energy:

$$\omega^{(n)}(\vec{r}, \vec{\Omega}, E, t_n) = \frac{1}{\Delta t_n} \ln \left(\frac{\psi^{(n)}(\vec{r}, \vec{\Omega}, E, t_n)}{\psi^{(n-1)}(\vec{r}, \vec{\Omega}, E, t_{n-1})} \right) \quad (6)$$

one can construct estimates for the angular fluxes $\tilde{\psi}^{(n+1)}$:

$$\tilde{\psi}^{(n+1)} = \psi^{(n)} \cdot \exp(\omega^{(n)} \Delta t_{n+1}) \quad (7)$$

Though this scheme is rather simple, it proves to be very efficient. This is demonstrated in **Erreur! Argument de commutateur inconnu.**, where we show a comparison for the convergence rate in a typical research reactor problem. On the left hand side, we have used the unextrapolated fluxes as a starting guess for solving the next timestep. Although the

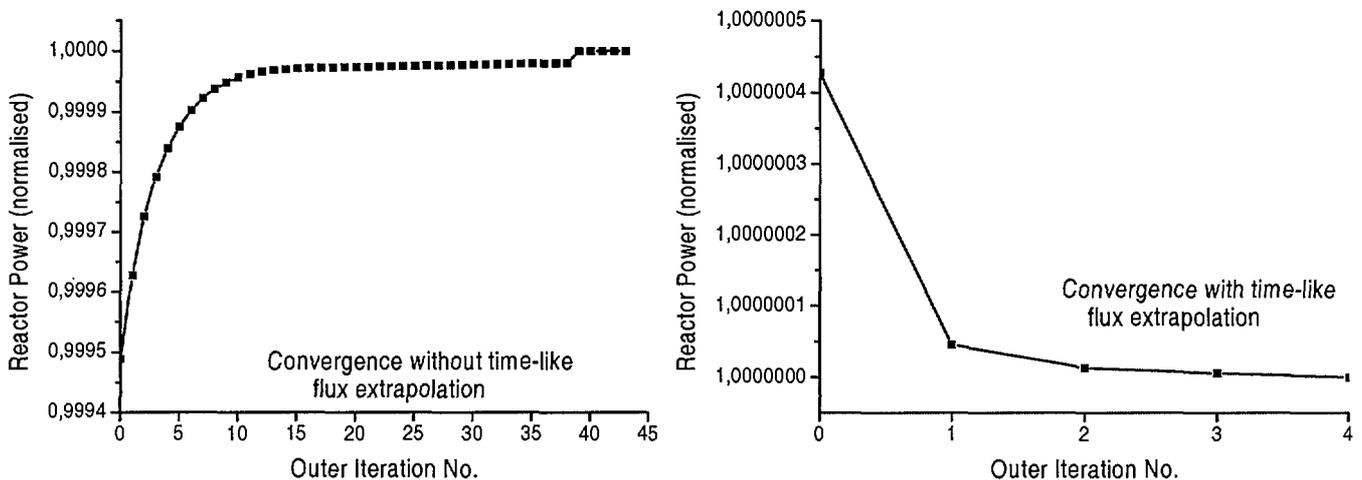


Figure **Erreur! Argument de commutateur inconnu.**: Comparison of convergence rate with and without temporal flux extrapolation

power values corresponding to timestep n and $n+1$ differ only by approximately $5 \cdot 10^{-4}$, roughly 40 iterations are necessary to find the new flux and fission density values, which are converged to better than 10^{-7} . The asymptotic behaviour of the iteration history is typical for fixed-source problems in multiplying media (the time-dependent problem is effectively a series of such problems). If one uses the time-extrapolated fluxes instead, convergence is dramatically improved: the power value from the first iteration is already quite close to the true value (in the example, it is even slightly overestimated by the extrapolated fluxes), and it takes only a few more iterations to obtain very accurate fluxes as well as fission and power densities.

The extensive use of the above mentioned Chebyshev acceleration, upscatter cycle and time-like extrapolation methods within the transient code system reduces the computing time needed for a single time step by more than an order of magnitude compared to the steady-state problem, although the numerical effort is formally the same. This makes the use of the transport code quite attractive, even when compared to quasistatic approaches. One should also keep in mind, that the effort using a fully implicit scheme may well be worth it due to the higher level of accuracy we achieve.

The phase-space resolved reactor periods additionally serve for adapting the maximum time step size. By comparing the temporal flux change to a simple exponential, a rough estimate of the truncation error, which is due to the implicit time discretisation, can be obtained. This time step is usually further refined to ensure a strictly conservative step size. Another problem is due to the iterative solution method: the longer the time step, the worse the convergence of outer iterations will be. This may pose much tighter restrictions on the maximum time step size than the truncation error.

4. COUPLING OF TRANSPORT CODE AND ATHLET

The transport code has finally been coupled to the well-known thermal hydraulic system code ATHLET [3], which has been developed at GRS and extensively applied to a wide variety of systems in the past twenty years. The coupling of ATHLET to a multidimensional neutron kinetics has been successfully carried out for several codes like QUABOX/CUBBOX, BIPR-8

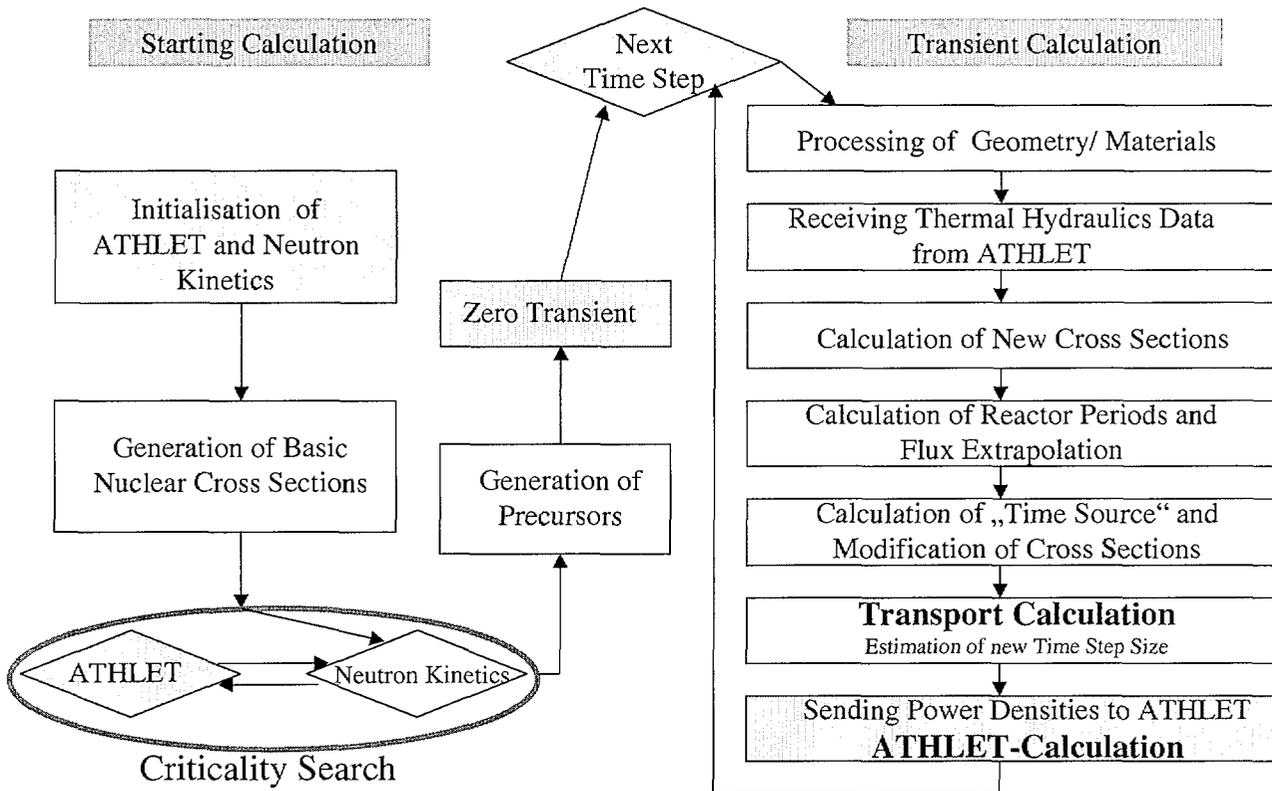


Figure **Erreur! Argument de commutateur inconnu.**: Flow chart of the coupled code system

and DYN3D and is described elsewhere [4]. We will not go into detail here, but only show a flow chart (**Erreur! Argument de commutateur inconnu.**) of the coupled model.

Any transient analyses starts with a criticality search. At the very beginning the pre-calculated nuclear cross section files are transferred to memory. With an approximate user-supplied power profile ATHLET can compute a first estimate of local temperature and density distributions in the reactor core. From these thermal hydraulic parameters, local multigroup cross sections are generated by interpolation from basic cross section sets. Using these cross sections, a first transport calculation is performed, resulting in a k_{eff} , which will most likely not be equal to unity. After adjusting e.g. control rod positions, a new iteration between ATHLET and neutron kinetics can be started. This loop is carried out until both thermal hydraulics system parameters and neutron fluxes, power densities as well as the effective multiplication

factor are sufficiently converged. In this starting phase it is possible to achieve a k_{eff} , which is converged as close as 10^{-8} to unity, corresponding to reactor periods of several years.

At this stage, an arbitrary transient can be initiated, e.g. by moving control rods, failure of coolant pumps etc. We usually apply a so called "zero transient" for a certain time to assure that under steady-state conditions the system remains in global balance before the "real" transient is started. In the code system we have provided for an eventual change of material composition or geometry during a transient, which can be done at the beginning of any time step. By this means we can e.g. achieve a quasi-continuous movement of control rods or other geometrical structures. After that, the thermal hydraulics data are received from the ATHLET calculation and new cross sections are generated. We emphasise, that this interpolation is done for every single time step and may be a rather time consuming task, depending on how many different cross section sets one wants to apply to represent feedback effects. The remaining steps in **Erreur! Argument de commutateur inconnu.** have

thoroughly been discussed in the previous section. Finally, the nuclear power density is returned to ATHLET in order to compute the new system state, and a new time step can be started.

5. THE RESEARCH REACTOR FRM-II

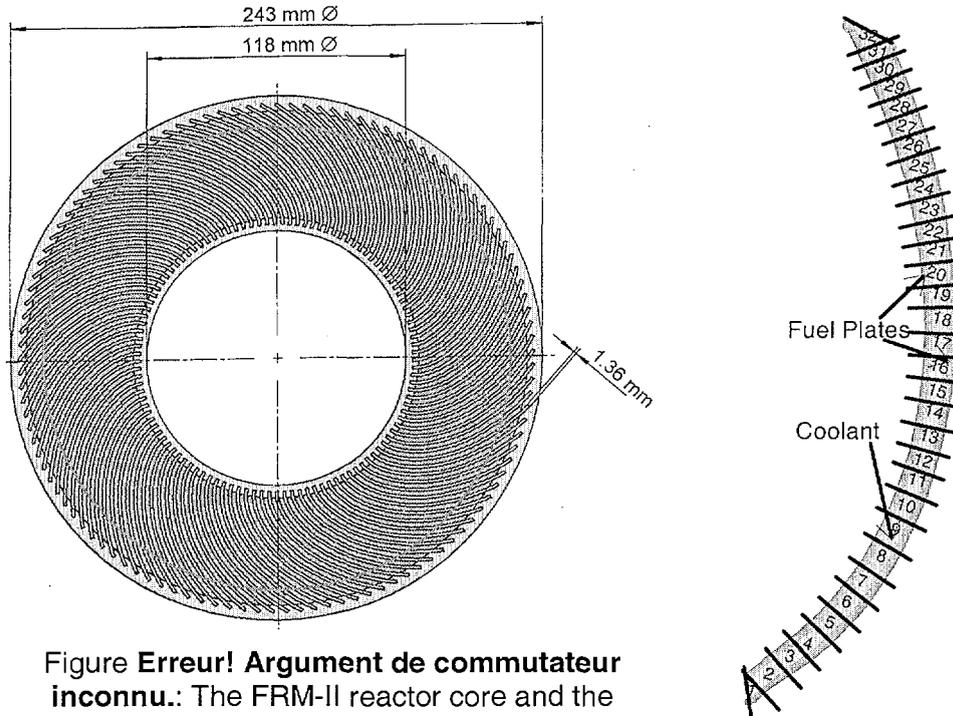


Figure **Erreur! Argument de commutateur inconnu.**: The FRM-II reactor core and the ATHLET model of one coolant channel

To test the capabilities of the coupled neutron transport code, we decided to analyse the research reactor FRM-II in Garching near Munich [5], which will start operation very soon. The system, which is quite similar to the HFR in Grenoble, is characterised by a small, compact reactor core in a large heavy water environment. As opposed to the HFR, the FRM-II is cooled by forced light water flow (mass flow rate ~ 300 kg/s). The cylindrical fuel assembly contains 113 evolvment shaped fuel plates of 1.36 mm thickness, separated by water gaps of 2.2 mm width. The moderator tank containing the reactor core is situated in a large light water pool.

The reactor has a few features, which make it interesting both from the thermal hydraulics and the neutron kinetics point of view. One main advantage is the fact, that the FRM-II can readily be represented by a 2D-geometry, when one neglects the experimental facilities inside the moderator tank. On the other hand, the reactor core itself is quite inhomogeneous and contains, apart from the fuel, several water channels and a central control rod, which can be moved up- and downwards inside the core. This rather sensitive control device is used for reactor start-up, power control and to compensate for burn-up. Due to the complicated geometry it may be advisable to perform coupled analyses using transport theory instead of the usual point kinetics or diffusion approach.

The thermal hydraulics of the FRM-II has been considered for the core region only. In **Erreur! Argument de commutateur inconnu.** we show a radial cut through the fuel assembly. Due to the rotational symmetry of the neutron kinetics model it is sufficient to model only one of the 113 coolant channels, as it is shown on the right hand side of **Erreur! Argument de commutateur inconnu.** Along the fuel plates, the coolant channel was subdivided into 32 parallel ATHLET subchannels. Each of these subchannels is nodalised

into 28 nodes in axial flow direction. Changes in mass flow, inlet temperatures or pressures are simulated by appropriate boundary conditions at the core inlet. While the description of single-phase flow is very well established, the modelling of two-phase flow under research reactor conditions (i.e. low system pressure, high fluent velocity, narrow, strongly heated coolant channels) is not very well validated in the ATHLET code, which is primarily applied to large nuclear power plants. We will thus restrict ourselves to transients, where the fluent flow remains single-phased, though the coupled code system certainly also works, when steam generation occurs.

One typical transient often considered in research reactor studies is the unintended withdrawal of a control rod. The maximum reactivity insertion of the FRM-II central control rod drive is

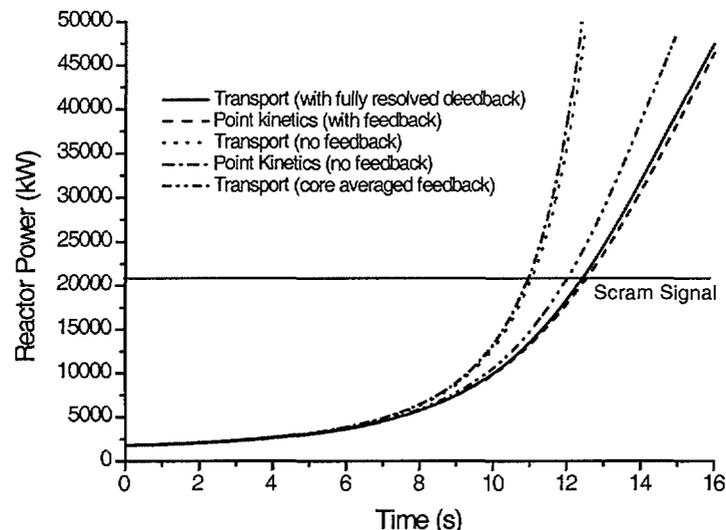


Figure **Erreur! Argument de commutateur inconnu.**: Comparison of transport results and point kinetics solutions for an unintended control rod withdrawal with and without feedback

$4 \cdot 10^{-4} \text{ s}^{-1}$, i.e. an accidental rod movement of 15-18 seconds duration corresponds to roughly one 1β (=1\$) of reactivity. It is instructive to compare the predictions of transport theory, diffusion and point kinetics. Starting from a reactor power of 10% nominal power (corresponding to ~2 MW), we did several calculations with and without feedback. Some results are depicted in **Erreur! Argument de commutateur inconnu.**.

The dotted and dashed-dotted curves show the power excursion without feedback effects from thermal hydraulics, revealing small deviations only. All scram signals, of which the last is usually activated at 114% overpower, has been ignored in this calculation. The good agreement between both approaches is not too surprising, since the main assumption of point kinetics, a constant flux shape function in time, is a good approximation for this type of transient. One should, however, not forget, that the evaluation of the point kinetics parameters already requires a couple of steady-state transport calculations, while none of those parameters is needed in our transient transport code. This also holds for calculations with feedback, drawn in **Erreur! Argument de commutateur inconnu.** as solid and dashed lines. Coolant and fuel temperature/density coefficients for the point kinetics solution were determined from *fully coupled* steady-state calculations. Due to the careful evaluation of reactivity coefficients the agreement between both approaches is again very good. In contrast, if one uses global, core averaged temperatures and densities for the determination of reactivity coefficients or cross sections instead, one obtains misleading results (cf. the dashed-double dotted line in **Erreur! Argument de commutateur inconnu.**). It is obviously necessary to represent feedback effects by spatially resolved cross section sets, i.e. for different thermal hydraulic ATHLET-nodes one should also evaluate individual nuclear data sets with the aid of the interpolation routine mentioned above.

It is also interesting to compare the diffusion predictions to transport theory.

For this purpose, we employed the diffusion module of the transient code system. The results are shown in **Erreur! Argument de commutateur inconnu..**

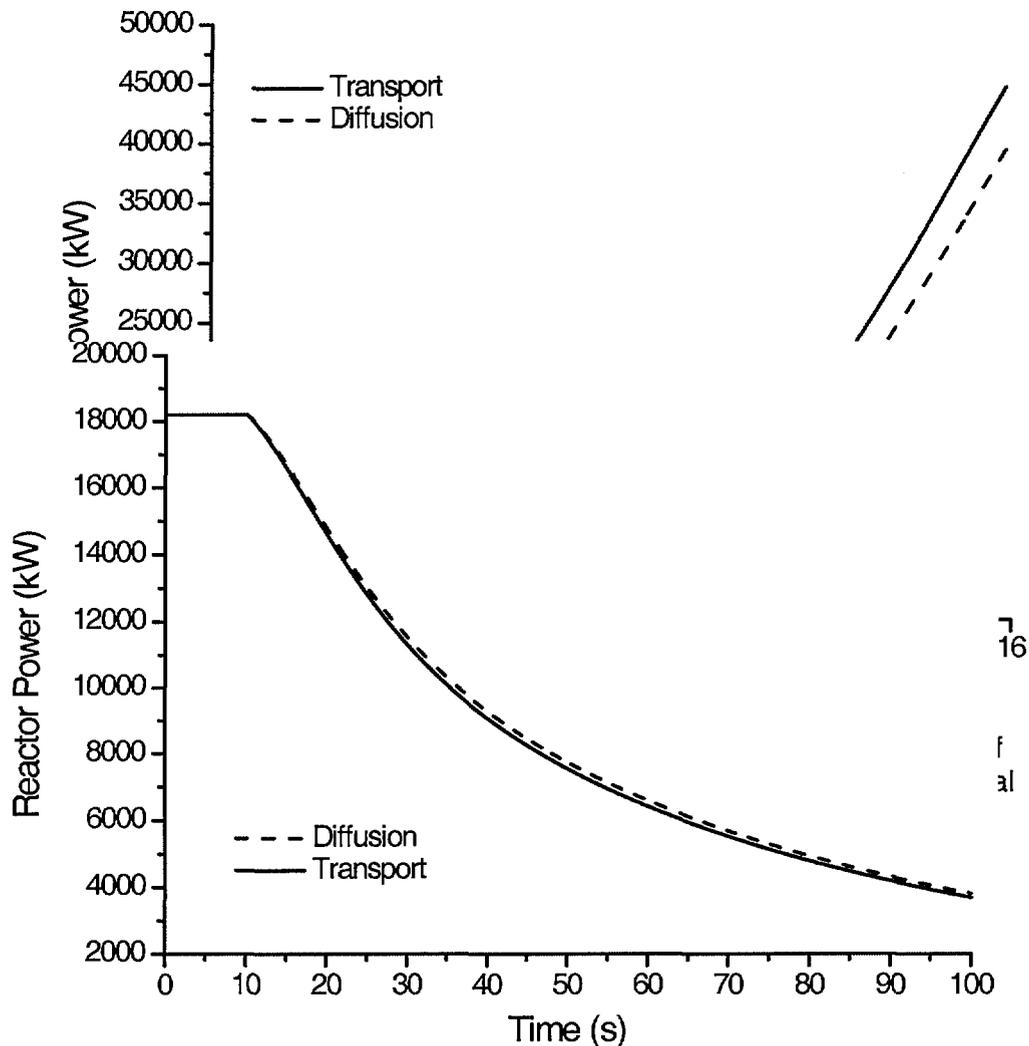


Figure **Erreur! Argument de commutateur inconnu.:** Power during loss of offsite power; shown are both the transport and diffusion solution

On first sight, the significant deviations are surprising. However, these stem exclusively from the fact, that diffusion theory cannot determine the control rod worth correctly, but gives a slightly lower rod worth than transport theory. If one adjusts the control rod velocity in the diffusion calculation

such that the net reactivity insertion for transport and diffusion theory is identical, the prediction of both approaches is in very good agreement (not shown in **Erreur! Argument de commutateur inconnu.**).

The second accident to be discussed is the loss of offsite power. This causes the primary coolant pumps to run down. Due to the big fly wheels on the main pumps the mass flow reduces to 10% of nominal flow within approximately 100 seconds. Of primary interest in this transient is to ensure, that the heat flux in the core remains at all times below some critical value. This may be characterised by the DNB ratio or the onset of flow instabilities.

Erreur! Argument de commutateur inconnu. shows the power during this transient. Here we postulated, similar to the control rod withdrawal, a failure of all scram signals. Obviously, the reduction of mass flow gives rise to a higher heat-up of the coolant and hence to a reduction of water density in the core. This in turn causes a rather strong negative feedback effect and a corresponding decrease in power.

To judge the thermal hydraulic behaviour of the core, we furthermore show as an example the maximum fuel temperatures (**Erreur! Argument de commutateur inconnu.**) and the minimum DNB ratio (**Erreur! Argument de commutateur inconnu.**) for this transient, as they occur in different ATHLET subchannels.

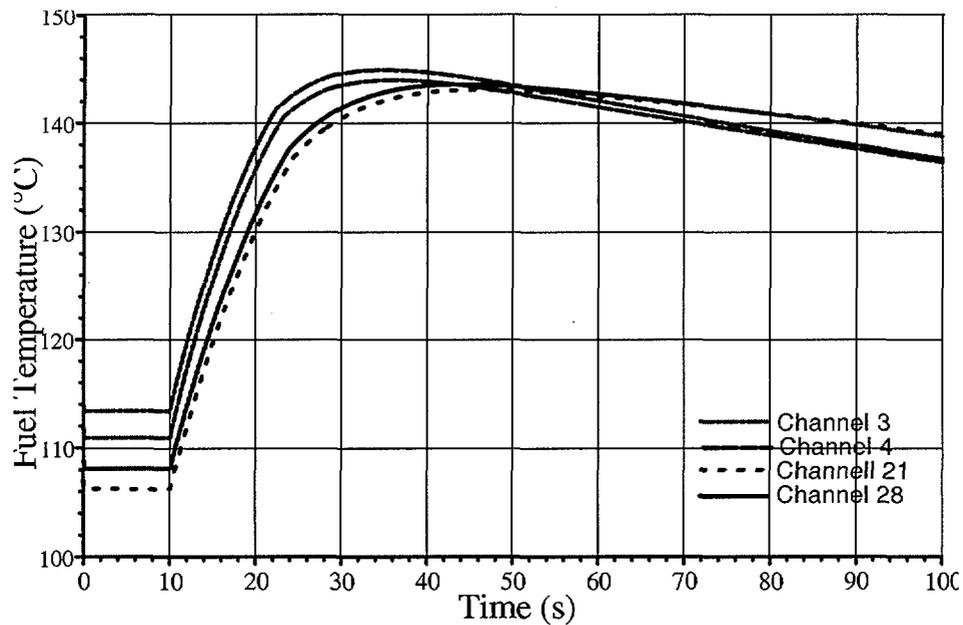


Figure **Erreur! Argument de commutateur inconnu.**: Fuel temperature during loss of offsite power

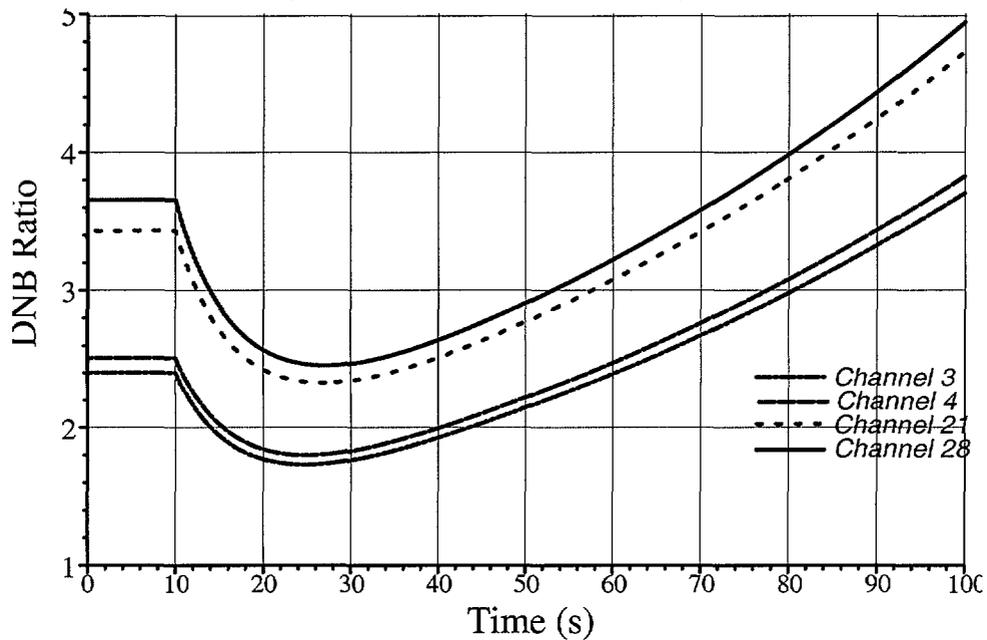


Figure **Erreur! Argument de commutateur inconnu.**: DNB ratio during loss of offsite power

The fuel temperature rises by about 30 K during the transient and assumes a flat maximum roughly 20 seconds after the failure of coolant pumps. At the same time, the DNB ratio attains its minimum value of ~ 1.8 , which is still quite far away from the critical value of unity. Finally, the severity of a transient may best be judged by the OFI (onset of flow instability) ratio, which is the most sensitive safety parameter for research reactors of the HFR-type and is depicted in **Erreur! Argument de commutateur inconnu.**

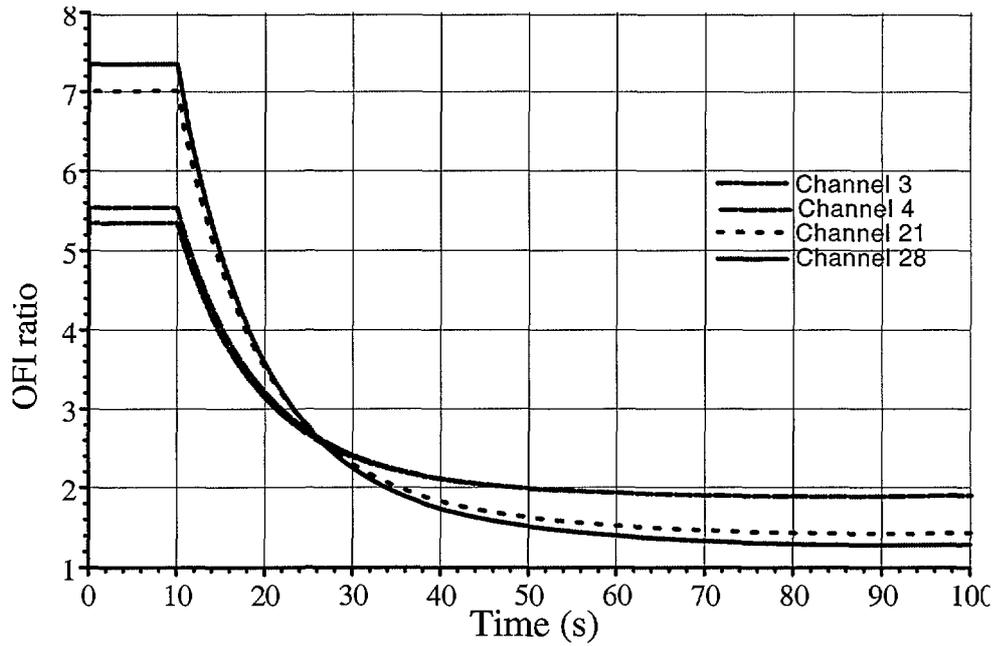


Figure **Erreur! Argument de commutateur inconnu.**: Distance to onset of flow instabilities during the loss of offsite power accident

However, even this parameter remains at all times above one and thus signals clearly, that the reactor can withstand a loss of offsite power even without reactor scram.

6. CONCLUDING REMARKS

In this paper a new coupled code system was presented and applied to a realistic nuclear system, the research reactor FRM-II. The capabilities of the neutron transport code DORT were extended to transient analyses via the implementation of a fully implicit, unconditionally stable time discretisation, which guarantees a very high accuracy in time-dependent calculations. Furthermore, DORT was coupled to the thermal hydraulics system code ATHLET.

Besides from the application to research reactors with compact core, the code system might as well be used to treat other kinds of nuclear systems, such as ADS (provided a proper source term can be constructed) or innovative reactor designs (like HTR or fast systems). An extension to three dimensions, using DORT's 3D counterpart TORT is already in progress.

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