Faculty
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VALIDATION OF THE POINT KINETIC NEUTRONIC MODEL OF THE PBMR

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EXECUTIVE SUMMARY

This study introduces a new method for the validation of the point kinetic neutronic model of the PBMR. In this study the diffusion equation solution, as implemented in the TINTE PBMR 268 MW reactor model, replaces the point kinetic model, as implemented in the Flownex V502 PBMR plant model. An indirect coupling method is devised and implemented in an external program called *Flownex-Tinte-Interface (FTI)* to facilitate the data exchange between these two codes.

The validation study of FTI indicates that the indirect coupling method introduces small errors in data transfer between the two codes and therefore FTI is not suitable for very fast thermal hydraulic and detailed reactor simulations. However, it is accurate enough for the point kinetic validation study.

The comparison between transient simulation results shows that the point kinetic parameters as implemented in V502 do not model the PBMR 268 MW correctly. Changes to some of the point kinetic parameters produced results that are more acceptable. The results also reveal that Flownex disregards any neutronic calculations after an explicit power change.

Further studies on the newest PMBR 400 MW reactor should determine if the point kinetic parameters used are valid under all transient conditions. Thought must also be given into a low-level integration of TINTE and Flownex. This could solve the problem of the induced errors by the coupling method, but would increase computational time dramatically.

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- My Creator to whom I'll forever be in debt.

Declaration

I, the undersigned, hereby declare that the work contained in this project is my own original work.

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Deon Marais

Date: 3 May 2007

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LIST OF ABBREVIATIONS

AGR	Advanced Gas cooled Reactor
AVR	Arbeitsgemeinschaft Versuchsreaktor (working group test reactor)
CBCS	Core Barrel Conditioning System
CFD	Computational Fluid Dynamics
CPU	Central Processing Unit (computer)
CRCC	Control Rod Cooling Channel
ECS	Emergency Cooling System
FDS	Flownex Diagramming System
FTI	Flownex-TINTE Interface
GUI	Graphical User Interface
HTGR	High Temperature Gas cooled Reactor
HTR	High Temperature Reactor
I/O	Input and Output
INEL	Idaho National Engineering Laboratory
Panther	PWR and AGR Neutron and Thermal Hydraulic Evaluation Route
PBMR	Pebble Bed Modular Reactor
PCU	Power Conversion Unit
PWR	Pressurized Water Reactor
RCS	Reactivity Control System
ROMO	Linear Rod Motion Model
RPV	Reactor Pressure Vessel
RSS	Reactivity Shutdown System
SAS	Small Absorber Spheres
SPECTRA	Sophisticated Plant Evaluation Code for Thermal-hydraulic Response Assessment
TCRW	Total Control Rod Withdrawal
TINTE	TIme dependant Neutronics and TEmperatures
USNRC	United States Nuclear Regulatory Commission
V&V	Validation and verification
VSOP	Very Superior Old Programs

LIST OF SYMBOLS

$\dot{V}_{in(tnt)}$ -	volume flow at reactor inlet calculated using the inlet mass flow according to TINTE
$\dot{V}_{in(fnx)}$	volume flow at reactor inlet calculated using the inlet mass flow according to Flownex
$\dot{m}$	mass flow through core
$\dot{m}_{in(fnx)}$	inlet mass flow calculated by Flownex
Δt_{Ni}	i^{th} nuclear time step in TINTE
Δt_T	temperature time step in TINTE
Δp	pressure drop over pebble bed core
C	ratio of K_{fnx} to k_{pipe}
C_i	normalised density or concentration of precursor atom group i [atoms/cm^3]
d/H	normalised insertion depth of the controls rods
K_{fnx}	characteristic loss factor of PCU
K_{tnt}	characteristic loss factor of reactor
k_{pipe}	pipe loss factor
P_{he}	thermal power transferred to coolant
P_n	normalized reactor power
P_t	total reactor thermal power
p_{in}	reactor inlet pressure
$p_{in(fnx)}$	inlet pressure of reactor calculated by Flownex
$p_{in(tnt)}$	inlet pressure of reactor calculated by TINTE
p_{out}	reactor outlet pressure
$p_{out(fnx)}$	outlet pressure calculated by Flownex
$p_{out(tnt)}$	outlet pressure of reactor according to TINTE
Q_{ex}	external sources [neutrons/(cm^2.s)]
$Q_{fission}$	fission power of the reactor
T_{in}	reactor inlet coolant temperature
$T_{in(fnx)}$	inlet temperature calculated by Flownex
T_{out}	reactor outlet coolant temperature
T_{surf}	temperature distribution of solid structures in core

v	neutron speed
β	total delayed neutron fraction
Λ	generation time or average neutron lifetime [s]
λ_i	decay constant for delayed neutron group i [s^{-1}]
ρ	dynamic reactivity
ρ_{ex}	reactivity due to external effects, i.e. the control rods
ρ_f	reactivity due to fuel
ρ_m	reactivity due to moderator
ρ_{max}	maximum external reactivity obtained with the control rods withdrawn
ρ_{min}	minimum external reactivity obtained with the control rods fully inserted
ρ_X	reactivity due to Xenon
σ_a^{Xe}	Xenon-135 absorption cross section
ϕ_{ave}	Average neutron flux [neutrons/(cm^2.s)]

1 INTRODUCTION

1.1 Introduction

According to the Energy Information Administration (2006:3), the world's energy consumption will increase by 71 percent from 2003 to 2030. Currently, most of the energy that is consumed is produced by the burning of fossil fuels like oil, coal and natural gas. This burning process releases the greenhouse gas CO_2 into the environment, which is a major contributor to global warming. Fossil fuels are also a limited commodity, and with its depletion, its price will increase dramatically over the years to come making energy more and more expensive if no alternative fuel source is used.

A new cost effective, reliable and safe method of energy production must therefore be developed. Many solutions have been proposed like wind and solar power, but it has been noted by Bradley (1996) that these renewable energy sources all have some major issues limiting their use. Nuclear power generation is a very attractive alternative but is by no means a new technology. A recent study by the International Atomic Energy Agency (2006:64) indicates that there are 443 power-generating nuclear reactors in operation throughout the world with 418 belonging to the water-reactor family. Although water-reactors have relatively high performance factors, they still suffer from problems such as safety and proliferation possibilities. One solution, which takes care of both these problems, is the high temperature gas cooled reactor (HTGR). The pebble bed modular reactor (PBMR) which is currently being developed by the South African company PBMR (Pty.) Ltd. is such a reactor (Slabber, 2004:1).

Like any nuclear reactor, the PBMR plant consists of thermal-hydraulic and neutronic systems. The thermal-hydraulic system can be subdivided into two models, one for the flow through the nuclear core, and one for the flow through the rest of the power conversion unit (PCU). The neutronic system of the core directly influences the thermal-hydraulic system of the core as almost all of the energy released by nuclear fission is deposited locally (Stacey, 2001:12). The thermal-hydraulic system of the PCU is directly influenced by the thermal-hydraulic system of the core. This can be described in the sense that what flows out of the one, flows into the other. The neutronic system therefore indirectly influences the thermal-

hydraulic system of the PCU. Figure 1.1: gives a schematic representation of the interaction between these three systems.

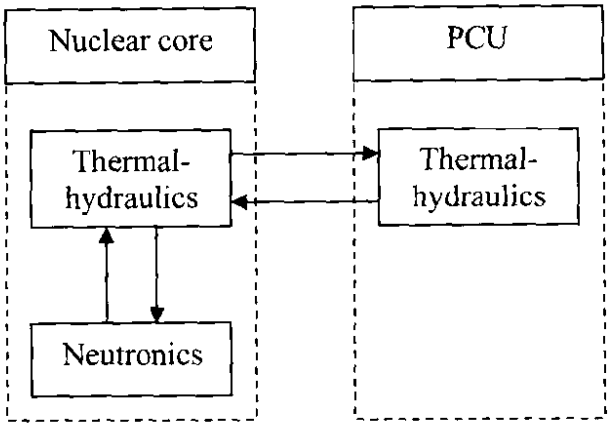


Figure 1.1: *Interaction between core and CPU*

Before a nuclear plant can be commissioned, it must be modelled very precisely and as accurately as possible. A complete simulation of the balance of plant is therefore desired.

There are many thermal-hydraulic simulation codes commercially available and some of them can solve neutronic models as well. Flownex is such a code (Anon, 2005a:2). It uses a very simple zero-dimensional point kinetic model to simulate the neutronic behaviour of the core. Other codes, such as TINTE (TIme dependant Neutronics and TEmperatures), use a very sophisticated two-dimensional neutron diffusion equation to model the neutronic behaviour of the core (Gerwin *et al.* 1989).

The objective of this project is to validate the point kinetic model of the PBMR as used in Flownex by comparing it to the neutron diffusion equation solver, TINTE. This comparison will be done by analysing transient responses obtained while using the two different core models in a complete plant simulation.

1.2 Background

The PBMR plant is a generation IV helium cooled, graphite moderated, advanced gas cooled reactor (AGR) fuelled with uranium dioxide (Koster *et al.* 2003:231). The whole system consists of numerous pipes, pumps, compressors, heat exchangers, valves, turbines and of cause the nuclear core. The PBMR plant layout is seen in Figure 1.2:.

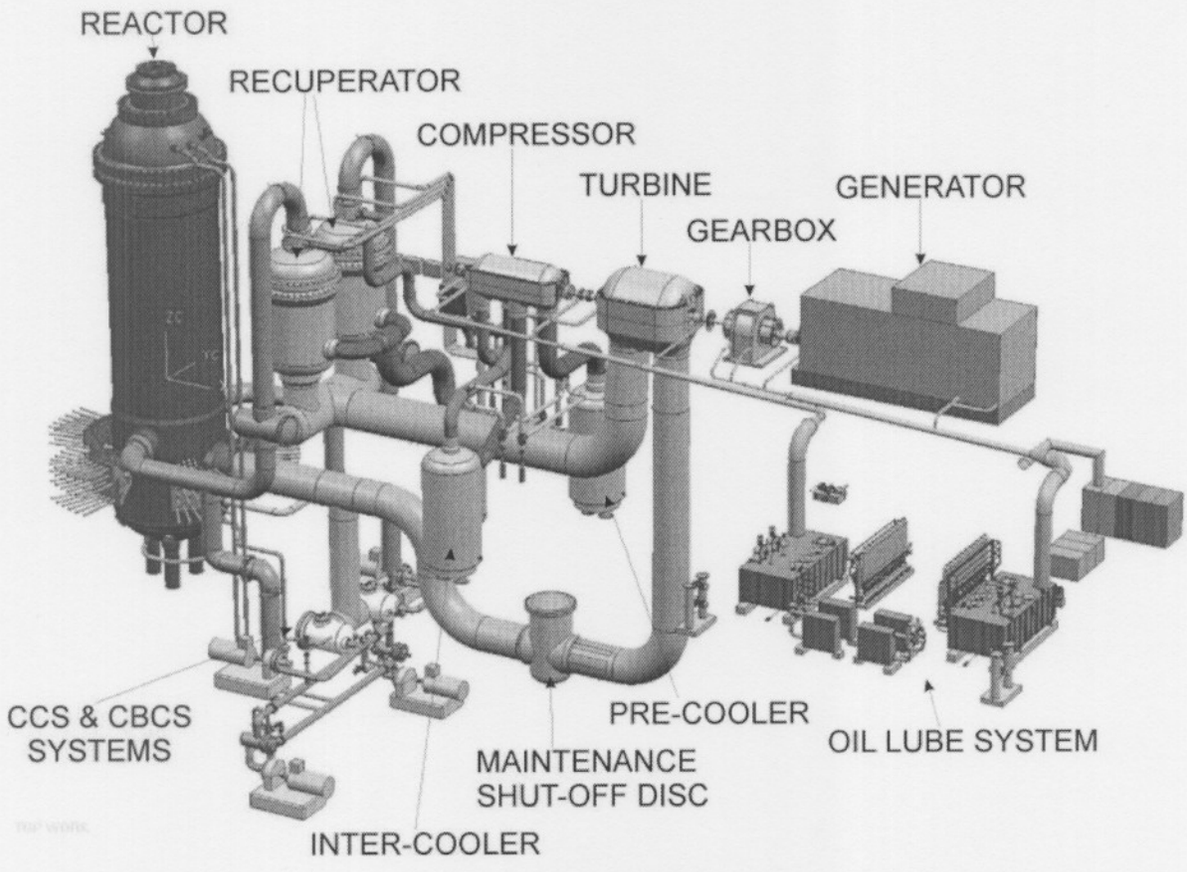


Figure 1.2: *PBMR Diagram*

PBMR (Pty.) Ltd. have decided to use Flownex as its primary development tool for the thermal-hydraulic system of the PCU because of fast computational times, robustness and versatility. Flownex was developed by the Potchefstroom based, South African company M-tech Industrial. Flownex is a systems CFD code that solves networks by using the conservation of mass, momentum and energy equations. A point kinetic neutronic model of the nuclear core has now also been added to Flownex.

Another important software package used in the development of HTGR's is the German designed TINTE. The TINTE code system deals with the nuclear and the thermal transient behaviour of the nuclear core of an HTGR taking into consideration the mutual feedback effects in two-dimensional cylindrical (r-z) geometry. The neutronic component of the core is simulated by solving the neutron diffusion equation. A drawback of TINTE is that only one external thermal-hydraulic component can be connected to the core. This raises the issue that a complete simulation of the balance of plant is not possible (Gerwin & Scherer, 2004:40).

1.3 Problem Statement

The neutronic model as implemented in Flownex was not designed to facilitate detailed reactor design, but rather to do fast, integrated simulations of the reactor and PCU. The detailed reactor design is done with TINTE and therefore only separate effect simulations are currently possible. It would therefore be beneficial to determine to what extent the point kinetic equations can model the reactor by analysing any deviations from the neutron diffusion equation in a fully integrated plant simulation.

1.4 Objective

The objective of this project is to verify the point kinetic solver as employed in Flownex. This will be done by comparing transient analysis in full-integrated plant simulations using both the point kinetic and neutron diffusion models. The TINTE reactor model simulates both the neutronic and thermal-hydraulic aspects of the core. The complete reactor model in Flownex will therefore be replaced by the TINTE model. This replacement of the core will be done by creating a high-level interface between Flownex and TINTE. Once the interface has been developed, comparisons between the point kinetics model of Flownex and the diffusion equation solver of TINTE can be made. Based on the results of this comparison, the viability of low-level integrating of Flownex with TINTE can be considered.

1.5 Lay-out of the study

Chapter 1 gives an overview of the problem that will be addressed by this project. A number of previously coupled HTGR codes are explored in the next chapter. Flownex and its interface are described in Chapter 3. Summaries of the point kinetic model and pebble bed element as implemented in Flownex are also given. Chapter 4 introduces the reactor development code TINTE and pays attention to some basic input parameters. The PBMR 268MW TINTE model is also described.

In Chapter 5 two methods of coupling are explored, and due to some technical issues, it is decided to implement an indirect coupling approach. A number of changes are made to the original TINTE code and Flownex-TINTE-Interface (FTI) is developed. FTI manages data exchange between Flownex and TINTE.

The validation studies in Chapter 6 focus on the error which in introduced by the indirect coupling method and the influence of the time step. It is shown that the introduced error is within reasonable limits and time step independence can be achieved by choosing appropriate time step lengths.

Three simulation studies are performed in Chapter 7. These are steady state, load follow, and total control rod withdrawal (TCRW) simulation. It becomes evident that reactor simulation with a point kinetic model is very sensitive to the point kinetic parameters and these must be chosen with care. After some adjustments of the neutronic parameters, the results obtained with the Flownex point kinetic model are very close the results obtained with the coupled code FTI.

In Chapter 8 some conclusions about the study and recommendations for future work are made.

2 LITERATURE SURVEY

2.1 Introduction

Commercially there are a number of software packages available for thermal-hydraulic system analyses, which include the computational fluid dynamics (CFD), and systems CFD approach. There are also a number of HTR neutronic solvers, which are used in HTR design. Some of these codes have already been successfully coupled. A brief description of a few commonly used code systems follows.

2.2 Related software

2.2.1 WKIND and RZKIND

WKIND and RZKIND were developed by Siemens Interatom (Kindt & Hauque, 1992). WKIND solves the one group neutron diffusion equation in the axial direction. This is done with regard to prepared cross sections of the fuel, moderator, reflector, control rods, small absorber spheres (SAS) and xenon. These cross sections are dependant on the neutron energy spectrum. The thermal-hydraulic system in the core is modelled with regard to the average axial fuel, moderator, gas and reflector temperature distribution. It contains a very detailed heat transport model from the fuel particle to the moderator, which is important for fast transients.

RZKIND uses a different solver for the neutronic system and solves in two dimensions (r-z). The thermal-hydraulic solver of RZKIND does not include a detailed fuel temperature model yet, but the original 1D WKIND or the 2D THERMIX/KONVEK thermal-hydraulic module can be used in RZKIND.

WKIND and RZKIND can be used for a number of quasi-stationary and transient simulations. These include the following:

- Slow transients due to load changes, start up, and shut down.
- Analysis of slow xenon transients after load changes.
- Slow transients after restart from a hot stand-by.

- Slow transients due to recriticality after core heat-up accidents.
- Fast transients due to changes of control rod position, SAS position or loss of absorbing substances.
- Fast transients due to changes of coolant mass flow.
- Fast transients due to changes of coolant inlet temperature.
- Fast transients due to ingress of moderating substances (e.g. water).
- Fast transients due to reactivity increase because of compression of the pebble bed.

Walter et al. (2004:6) successfully coupled the 1D WKIND code with Flownex by means of an independent program. This program exchanges data on a time-step basis. A pipe was used to replace the reactor model of Flownex. After every time-step the pressure drop, outlet temperature and power transfer to the pipe, was updated. The inlet temperature and mass-flow of the pipe then served as the boundary conditions for the WKIND model. Time step synchronization and control interaction problems were experienced, but overall the study showed good results.

2.2.2 SPECTRA

SPECTRA (Sophisticated Plant Evaluation Code for Thermal-hydraulic Response Assessment) was developed by NRG Netherlands in 1994. Numerous V&V tests demonstrate SPECTRA as a robust and reliable tool for thermal-hydraulic design and analyses of nuclear and conventional power plants. The modelling approach of SPECTRA is based on the control volume concept where physically bounded space is connected by junctions. The SPECTRA includes a point kinetics model, which was verified by the 3D neutronic code OCTOPUS/PANTHERMIX. SPECTRA was used in the V&V of the Flownex PBMR models and the results showed adequate consistency (De Geus & Stempniewicz, 2006:2).

2.2.3 PANTHERMIX

PANTHERMIX consists of a combination of three different codes namely PANTHER, THERMIX and DIRECT (Oppe *et al., 1998).*

PANTHER is a 3D neutron diffusion equation solver, which can calculate steady state or time dependant power distribution in the reactor core. PANTHER solves with regard to few energy groups and delayed precursor groups. The thermal hydraulic model in PANTHER is

not capable of modelling heat transfer in a pebble-bed configuration. Therefore the code THERMIX and DIREKT is used to replace the built-in model.

The THERMIX code was developed in order to describe the heat transport by conduction within a pebble-bed HTR. It solves the heat conduction equation for the solid materials in two dimensions (r and z direction). Although its uniqueness lies in the treatment of the pebble-bed, the reflector regions, cavities and piping can also be included in the model. For each mesh point an appropriate material composition can be defined, if necessary with a certain degree of porosity when part of the mesh volume has to be occupied by a fluid. The solid part of the mesh volume is homogenized with respect to conductivity, heat capacity, and heat transfer. Thus, only one local temperature characterizes the solid temperature of a mesh volume. This approach is only valid if the heat production in a fuel pebble is low, resulting in low temperature gradients. This is the case for incidents with a scrammed reactor and not for operational transients in general. For operational transients, the heterogeneous solid structure model in THERMIX can be used.

The DIREKT code was developed in order to solve the time-dependent equations for convection and to establish the gas temperature distribution for the reactor. As such, it can be seen as the complementary calculation of THERMIX. It describes the fluid part of a mesh volume. The heat convection calculation allows cross element heat transfer. This is to describe the circulation and eddying of the gas in the pebble-bed in transient cases with a halted mass flow rate. The first step to obtain the gas temperature distribution is to combine the equations of continuity and motion. As solution, it yields the pressure and mass flow rate distribution over the reactor at fluid temperatures of the previous iteration. The second step is to solve the energy equation. The input is the new pressure and mass flow rate distribution and the solid temperature distribution from THERMIX. This results in the new gas temperature distribution. The iteration over the two steps leads to convergence in the solutions (Verkerk, 2000:21).

2.2.4 RELAP5/mod3.2

Relap5 was developed at the Idaho National Engineering Laboratory (INEL) for the United States Nuclear Regulatory Commission (USNRC) as a light water transient analysis code. Specific applications of the code included simulations of transients in LWR systems such as

loss of coolant, anticipated transients without scram, and operational transients such as loss of feed water, loss of offsite power, station blackout and turbine trip. According to Borges *et al.* (2000:5), the RELAP5/Mod3.2 is based on a non-homogeneous, non-equilibrium set of six partial differential balance equations for the steam and the liquid phases. A non-condensable component in the steam phase and a non-volatile component (boron) in the liquid phase can be treated by the code. A fast, partially implicit numeric scheme is used to solve the equations inside control volumes connected by junctions. Heat flow paths are also modelled in a one-dimensional sense, using a staggered mesh to calculate temperatures and heat flux vectors. Several specific models are included in the code to simulate special components like pumps, valves, steam separators, etc.

Verkerk (2000:24) claims it is possible to use RELAP5 with only helium and no steam, and in that case, the working fluid only exists in one phase and behaves like an ideal gas. This is quite a simplification, as many empirical theological models are no longer necessary in the emergency cooling system (ECS) calculations. On the other hand, from a well-known code, validated and tested with experiments, one enters an area in which virtually no testing and benchmarking were done. However, reasoning that all mass and energy balances are still valid, and that the correct properties of helium are present in the code, it seems that there is no fundamental objection to using the code with helium as working fluid. Simple analytical problems such as pressurised helium flowing into or from a tank were tested and are correctly calculated. Problems that are more serious are to be expected with the two main dynamic components, the turbine and compressor. There is a basic gas turbine model - not validated - which has been used in developmental stages and then only as a single stage turbine. The gas compressor altogether lacks as a component, which is understandable: water-cooled systems use a pump to make up for pressure losses and the water is pressurised in the liquid phase.

RELAP5 also contains a point kinetic neutronic model. This point kinetics code uses core-average fluid conditions, weighting factors, and feedback coefficients to determine a total reactivity to drive the kinetics calculation of the total core power. Once the total core power is determined, it is then distributed among the fuel heat structures in a fixed power profile. Fletcher and Schultz (1995:126) state that for many simulation problems this model may be an adequate approximation of the physical processes, but if it is determined that point kinetics is inadequate, then it may be possible, through an iterative process between RELAP5 and a more functional kinetics code, to converge upon the true solution.

2.2.5 TALINK

Verkerk (2000:26) developed the TALINK code to supply RELAP5 with the more accurate neutronic solver, Panther. TALINK controls the data transfer required for the execution of these two coupled transient analysis codes. The calculations take place in separate operating system processes. The TALINK code is regarded as being the governing component in this transfer structure with the other codes as clients. In order to provide flexible data coupling, the data values transferred by TALINK are stored in its internal database. The user can specify operations to be performed on the data before transfer. TALINK writes the requested data in a temporary file, which is read by the client code. In addition to the data transfer files, another set of files is created and checked to identify when each data transfer operation can begin. In turn, Panther can pass some of the data to THERMIX-DIREKT. In general, Relap5 will offer the new inlet conditions for the core, based on the core outlet conditions Panther supplied to it at the start of the time interval. The RELAP5 program will then temporarily be halted by TALINK until Panther and THERMIX-DIREKT have processed the data and come up with new core outlet conditions. At that moment, the data transfer between RELAP5 and Panther takes place, and a new time step starts. TALINK lets the codes communicate with time intervals of typically several seconds, but RELAP5 internally has to divide such a time interval in transient time steps in the order of milliseconds in order to obtain a stable calculation of the transient behaviour. Figure 2.1: gives a graphical representation of the data exchange between RELAP5, TALINK, Panther and THERMIX-DIRECT.

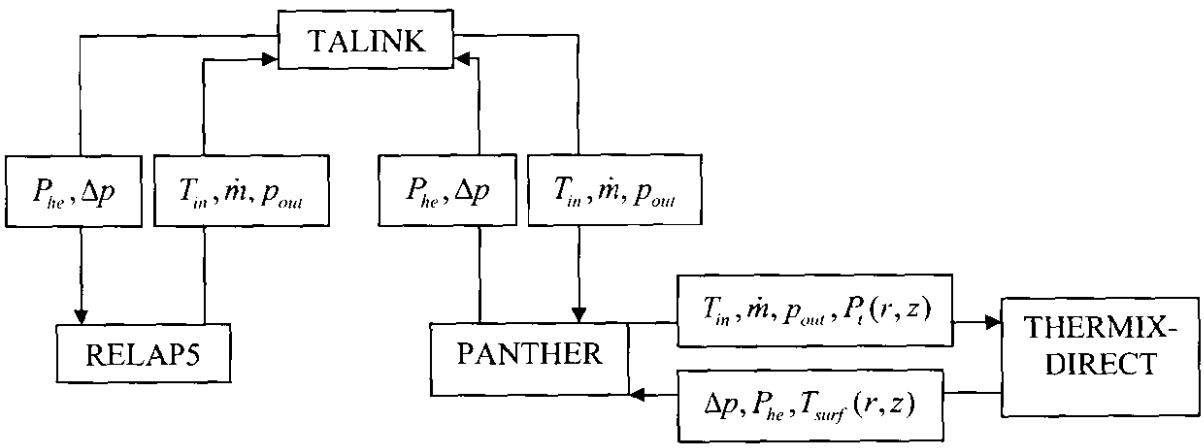


Figure 2.1: *TALINK Data Exchange*

The core neutronics are modelled in 3D by Panther, the core heat transfer in 2D by THERMIX-DIREKT, and the PCU in 1D by RELAP5. The parameter P_t is the total reactor thermal power, P_{he} the thermal power transferred to the helium, $\dot{m}$ the mass flow rate, Δp the pressure drop over the pebble-bed core, T_{in} the core gas inlet temperature, $T_{surf}(r,z)$ the temperature distribution of the solid structures in the core and p_{out} the core outlet pressure of the helium.

2.3 Conclusion

A number of neutronic and thermal hydraulic codes have already been coupled which would suggest that there is a need for such programs. Such coupled programs combine the detailed neutronic and thermal-hydraulic behaviour of the nuclear core with the thermal-hydraulic behaviour of the PCU to give a complete balance of plant. PBMR (Pty.) Ltd. has selected Flownex and TINTE as part of their main development and analysis tools, so it would be advantageous to have a coupled version of these two codes. This coupled version can then be used to validate the point kinetic module employed in Flownex.

3 DESCRIPTION OF FLOWNEX

3.1 Introduction

Flownex (Anon, 2005a:2) is a systems CFD code which was developed in South Africa. It uses the principles of conservation of mass, momentum and energy to solve thermal hydraulic networks in one dimension. The transient versions of Flownex can perform steady state and transient network solutions. It employs a state-of-the-art implicit pressure correction algorithm that results in fast and accurate analysis. By using this implicit algorithm, the time step is not as restrictive as with an explicit algorithm. Flownex can perform fast and slow transients and computational time is relatively short because of the one dimensional network approach. Flownex can perform detail analysis on a variety of complex systems such as conventional and nuclear power plants, ventilation systems, gas, water and compressed air distribution networks.

The components (also known as elements) that are available in the Flownex database for network construction are pipes, resistive ducts, conductive heat transfer elements, compressors, turbines, fans, pumps, rotating pipes, labyrinth seals, heat exchangers, restrictors, valves, controllers, gearboxes, shafts, pebble bed and advanced pebble bed reactors. Thermal-fluid networks are represented in by a combination of nodes and elements. Figure 3.1: shows a schematic representation of an unstructured thermal-fluid network consisting of nodes and several different types of elements. In the Flownex Graphical User Interface (GUI), nodes are indicated with a square box symbol while elements are indicated with a circle.

Networks are created by placing and connecting elements and nodes in any unstructured fashion. Flownex caters for any number of elements and nodes per network, limited only by the available computer memory. It is therefore possible to create very complex thermal-fluid networks using Flownex. Nodes act to connect elements and to represent boundaries for a network.

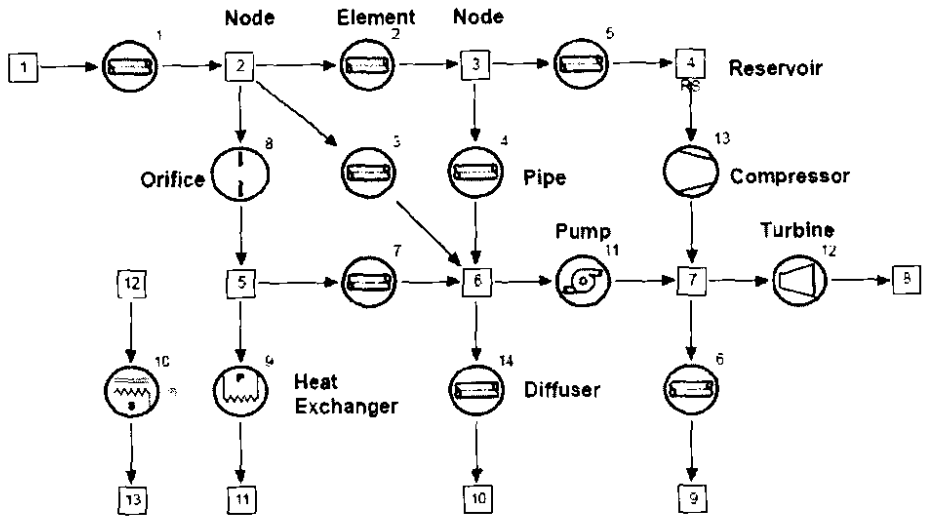


Figure 3.1: *Flownex network representation*

3.2 Flownex reactor model

Flownex employs a comprehensive 2D porous CFD reactor model for the simulation of the thermal-flow behaviour of the reactor core and core structures (Du Toit *et al.* 2003:3). The model is based on the fundamental equations for the conservation of mass, momentum and energy for the compressible fluid flowing through a fixed bed, as well as the equations for the conservation of energy for the pebbles and core structures. Through a rigorous analysis, the equations are reduced and recast in a form that is suitable for incorporation in a network code. This formulation of the equations results in a collection of one-dimensional elements (models) that can be used to construct a comprehensive multi-dimensional model of the reactor. The elements account for the pressure drop through the reactor, the convective heat transport by the gas, the convection heat transfer between the gas and the solids, the radiative, contact and convection heat transfer between the pebbles and the heat conduction in the pebbles. The numerical formulation of the equations is based on a staggered grid approach and is solved with the implicit pressure correction method.

Two reactor models are available in Flownex namely Pebble Bed Reactor model and Advanced Pebble Bed Reactor model. These reactor models consist of three main units as shown schematically in Figure 3.2: (Rousseau & Greyvenstein 2003:25).

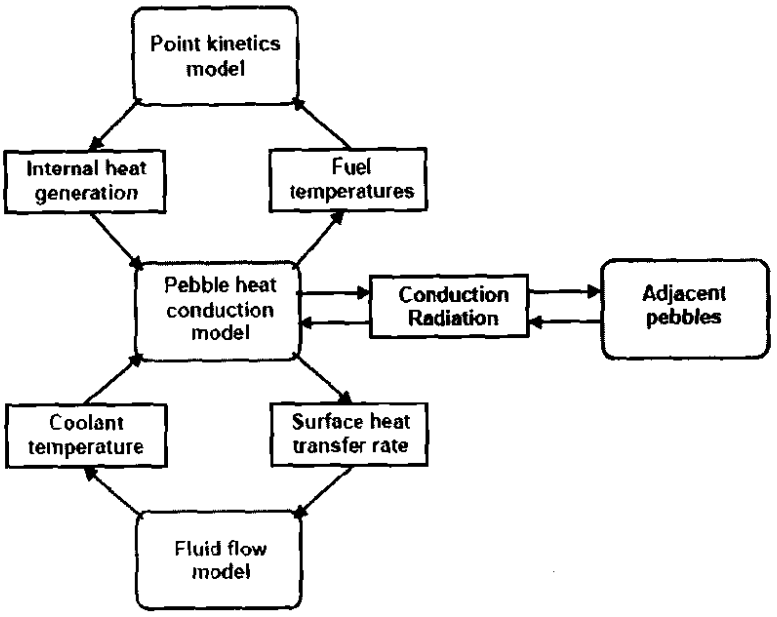


Figure 3.2: *Interaction in Flownex between the three models.*

- *The transient point kinetic neutronics and decay heat generation model.* It requires as input the temperatures within the fuel spheres and provides as output the total internal heat generation within all the fuel spheres contained in the reactor core.
- *The detailed transient internal heat conduction for each representative sphere in each core section.* It requires as input the heat generation density within the fuel as well as the temperature of the gas surrounding the spheres. It provides as output the temperature distribution within the spheres as well as the heat transfer through convection between the surfaces of the spheres and the surrounding coolant.
- *The transient fluid flow model that determines the temperature and pressure variations in the gas contained in each core section.* It requires as input the surface heat transfer rate and provides as output the coolant temperatures and pressures.

The governing equations for the fluid, solid and neutronic models are fully documented by Du Toit *et al.* (2003:7) and in the Flownex user manual (Anon, 2005b:334).

3.2.2 Point Kinetic Model

The neutronic solver of Flownex is based on the well-known point kinetic equations, which describes the neutron density of the whole reactor assuming a constant spatial shape (Stacey, 2001:142). As the neutron density is directly proportional to the reactor power, the point kinetics equation can be written in terms of normalized power P_n:

$$\frac{dP_n}{dt} = \frac{\rho - \beta}{\Lambda} + \frac{1}{\Lambda}\left(\sum_{i=1}^{6} \lambda_i C_i + Q_{ex}\right)$$

where

P_n = normalized reactor power,

ρ = dynamic reactivity,

β = total delayed neutron fraction,

Λ = generation time or average neutron lifetime [s],

λ_i = decay constant for delayed neutron group i [s^{-1}],

C_i = normalised density or concentration of precursor atom group i [atoms/cm^3] and

Q_{ex} = external sources [neutrons/(cm^2.s)].

The change in neutron precursor concentrations in time is given by:

$$\frac{dC_i}{dt} = \beta_i P_n - \lambda_i C_i$$ where i = 1…6 and β_i is the neutron fraction for group *i*.

Special care is taken to account for Xenon poisoning as it contributes greatly to the absorption of neutrons and therefore to power density. Total reactivity is obtained by the addition of the different reactivities: $\rho = \rho_f + \rho_m + \rho_X + \rho_{ex} - Q_{ex}$ with

Q_{ex} = the external source necessary to start up the reactor [neutrons/(cm^2.s)],

ρ_f = reactivity due to fuel,

ρ_m = reactivity due to moderator,

ρ_X = reactivity due to Xenon and

ρ_{ex} = reactivity due to external effects, i.e. the control rods.

The power distribution profile is fitted using a power distribution curve, which is obtained from VSOP calculations. The power distribution profile can then be used to calculate temperatures in the fuel pebble and gas. Control rods and SAS are modelled to the so-called fitting to an s-curve. According to the Flownex user manual (Anon, 2005b:342), the relation of the reactivity to the position of the control rods are then given by the equation:

$$\rho_{ex} = -0.5\frac{\rho_{max} - \rho_{min}}{\tan^{-1} C}\tan^{-1}\left(C\left(2\frac{d}{H} - 1\right)\right) + 0.5\left(\rho_{max} + \rho_{min}\right)$$

where

ρ_{max} = maximum external reactivity obtained with the control rods withdrawn,

ρ_{min} = minimum external reactivity obtained with the control rods fully inserted,

d/H = normalised insertion depth of the controls rods and

C = curvature factor depending on the design of the reactor and the control rods.

3.2.3 Pebble Bed Reactor Element

The Pebble Bed Reactor element is a simplified model of the pebble bed nuclear reactor and is subject to a number of assumptions. The thermal hydraulics of the reactor is modelled by a 2D axi-symmetrical model, therefore it is assumed that variations in the mass, momentum and energy in the tangential direction are negligible compared to the variations in the axial and radial direction. The fluid velocity in the tangential direction is also zero. The outlet temperature of the active core region is mixed with gas from the dynamic inner region to obtain a fully mixed exit temperature. Shear stresses are negligible compared to the flow resistance terms, so the convective, diffusive and dilatational terms in the momentum equations for the pebble bed may be neglected. In the case of compressible flows, the convection and static pressure gradient terms in the equation for the conservation of momentum are rewritten into stagnation temperature and stagnation pressure gradient terms. This manipulation is done assuming an ideal gas with constant specific heat. The error induced in the case of real gasses is assumed negligible.

A constant porosity is specified for the whole reactor, which does not vary in the radial direction. It is assumed that the outside reflector is adiabatic. The gas and pebble conduction, contact conduction and radiation in the pebble bed are modelled with the Zehner-Schlunder effective conductivity correlation and the Kugeler-Schulten convection correlation is applied uniformly to the pebble bed (Anon, 2005b:346).

The pebbles in the packed bed can be considered as heat exchangers each with a constant surface temperature. All pebble spheres modelled by a representative pebble have exactly the same temperature distribution and internal heat generation density. For the pebbles it is assumed that the temperature only vary in the radial direction.

The global reactor neutronic behaviour is modelled dynamically as a single point having certain weighted average properties that may be assumed a constant over time. The neutron spectrum does not change during a transient. The flux tilt in the radial co-ordinate must also

be negligible as is the case when the control rods are symmetrically placed. The single power value calculated by the point kinetics model is assigned to the bed according to an axial power profile. The power profile does not vary in the radial direction. All the RCS Bank 1 control rods are always fully inserted before the insertion of any RCS Bank 2 control rods are initiated. All the control rods of both RCS Bank 1 and Bank 2 are always fully inserted before the RSS is activated. The Bank 1 control rods can only be inserted up to a specified depth.

The validity of the Reactor model is subject to the following constraints:

- The simplification of modelling the neutronic behaviour of the reactor using the point kinetics equation is valid when the reactor is sufficiently small to be well coupled with the space and time variables essentially separable. This means that the spatial neutron flux shape changes negligibly during a transient. Stated more simply it means that the normalised neutron flux shape factor has a weak dependence on time even though the actual amplitude may have a strong dependence on time. The flux tilt in the radial co-ordinate must also be negligible as is the case when the control rods are symmetrically placed.
- None of the reactor solid structures are modelled, e.g. outside reflector, riser channels, core barrel, etc.
- The inner pebble column is not fixed; it consists of non-nuclear graphite pebbles moving with the bed.
- The Mach number is less or equal to one over any of the entire flow path lengths.
- The flow conduit cross-sectional area is completely filled with fluid over the entire element length.
- No work is done on the fluid in the reactor other than flow work and gravitation.
- The decay heat approximation by means of three exponentially decaying functions is only valid for approximately three days after shutdown.

The input parameters to the Pebble Bed Reactor Element can be summarized as follows:

- Core dimensions, fuel dimensions, porosity, steady state heat transfer rate
- Graphite conductivity and specific heat coefficients
- RCS curvature factor and offsets
- RSS curvature factor and offsets
- Axial power distribution coefficients

- Iodine and Xenon decay constants
- The product of the average absorption cross section and equilibrium neutron flux
- External neutron source flux
- Average neutron lifetime
- Decay constants and average fractions of the six delayed neutron groups
- Decay constants and fractions of the decay heat groups
- Coefficients for the conductivity between pebbles
- Fuel and moderator baseline temperatures
- Minimum and maximum insertion depths and reactivity for the RCS and RSS
- Fuel, moderator and Xenon reactivity feedback coefficients

3.2.4 Advanced Pebble Bed Reactor Element

The phenomena that can be simulated in the Advanced Pebble Bed Reactor model but cannot be simulated with the previous Pebble Bed Reactor model include the following:

- The presence of a central reflector column that implies that the core itself does not extend outward from the centre but has an inner and outer diameter.
- The addition and extraction of gas via purpose provided channels and/or leak flow paths along the inner or outer perimeters of the core.
- The simulation of heat transfer and fluid flow through porous and solid core structures surrounding the core.
- The simulation of fluid flow and heat transfer, including radiation and natural convection, in purpose provided cavities between core structures with a two-dimensional rather than one-dimensional nature.
- The ability to specify normalised radial power distribution profiles within the different axial layers in the core.
- The ability to take into account heat generation that may occur in any of the core structures.

3.3 Flownex interface

In Flownex, a network is created by means of the Flownex Diagramming System (FDS). The FDS provides the functionality of a drawing application like placing and linking components, grid functionalities, aligning, spacing etc. The FDS assists users by automatically drawing

nodes and elements in proper sequence. Moreover, a rule system exists to help enforce "diagrammatical correctness". This is a basic level of checking to ensure that the user creates a fluid-flow network that may be solved by the Flownex Solver. Furthermore, FDS provides the tools that accelerate the creation of "diagrammatically correct" networks and therefore enhances the quality of the user's experience while utilizing Flownex. This includes modes of operation that automatically select the correct type of component to be placed, while additionally placing the correct type of link between them, with a minimum of user interaction. The FDS does not prohibit the user from placing components in a fashion commonly found in other diagramming applications.

Flownex uses a memory mapped file (Anon, 2005a:206) for data transfer with external programs and Microsoft® Windows events for synchronization. This gives the developer the ability to interface directly with Flownex without the need to alter the Flownex source code. For the user to be able to communicate between Flownex and an external program during simulations, the layout of the memory map file is important. The structure of the memory map file is shown in Figure 3.3:. The memory map structure (and the code that utilizes it) must be compiled with eight-byte alignment, if not the mapping will be incompatible with the Flownex internal representation.

```
enum controlType {Running = 0, Stopped = 1};
struct MemoryFileStruct
{
        enum controlType m_Control;
        double m_dT; //*< simulation time
        int m_iNumberOfInputs; //*< to the Flownex simulator
        int m_iNumberOfOutputs; //*< from the Flownex simulator
        double m_iaInputs[1000]; //*< data = inputs[0] ... inputs[NumberOfInputs-1]
        double m_iaOutputs[1000]; //*< data = outputs[0] ... outputs[NumberOfOutputs-1]
        int m_iUpdateStatesWhenFinished; /**< = 0 ... states will not be updated
                                            = 1 ... states will be updated*/
        double m_dSimulinkClock;
        int m_iEventNumber; //Simulink event number (0 = no event)
        int m_iExternal;
        double m_daInputValues[1000];
        int m_iaInputElements[1000];
        int m_iaInputVariables[1000];
        char m_caInputTypes[1000];
        double m_daOutputValues[1000];
        int m_iaOutputElements[1000];
        int m_iaOutputVariables[1000];
        char m_caOutputTypes[1000];
};
```

Figure 3.3: *Flownex Memory Map File Structure*

The different input and output variables are specified in the *External Control Specification Dialog* of Flownex. The plant input variables are the parameters that will be controlled from

the external controller. Conversely, the plant output variables are parameters passed to the external controller for further processing.

3.4 Time Steps

As already stated, Flownex uses an implicit algorithm when solving a network (Greyvenstein, 2002). Although this is more complex to solve, this method does not suffer from instability when the simulation time is long as is the case with explicit methods. In the event of a transient simulation, time-step independence must be assured. A too large time step will result in some variations, which occur in a smaller time then the specified time step, not to be observed. Thus, the time step should be decreased until an insignificant change in the results is realized. This will then lead to a solution that is time step independent.

3.5 PBMR plant model

The network that was used for all the simulations is the 268MW_{th} V502 PBMR plant. It is a three shaft, gas turbine, inter-cooled, recuperated plant based on the Breyton-cycle. The flow path is as follows. From the reactor core, helium is expanded first through a high-pressure turbine and then through a low-pressure turbine. It then drives the power turbine which is connected to the generator and hence to the electric grid. The helium then flows through the recuperator and the start-up blower system. The gas then flows through two pre-cooled compressors, through the recuperator and back to the core. Figure 3.4: shows a schematic diagram of the process flow. The detailed thermal-hydraulic specifications can be found in the V502 Datapack compiled by Correia (2000:8).

The pebble bed core has an outer diameter of 3.5m, inner diameter of the graphite pebble core of 1.75m, a height of 8.5m and a pebble bed void fraction of 0.39. Control rod cooling channels (CRCC) are also present. The reactor is modelled by the *Pebble Bed Reactor Element* as described in Par 3.2.3. Rousseau (2000:9) set out the detailed specifications and parameters of the point kinetic neutronics model which are used in the Flownex V502 simulation.

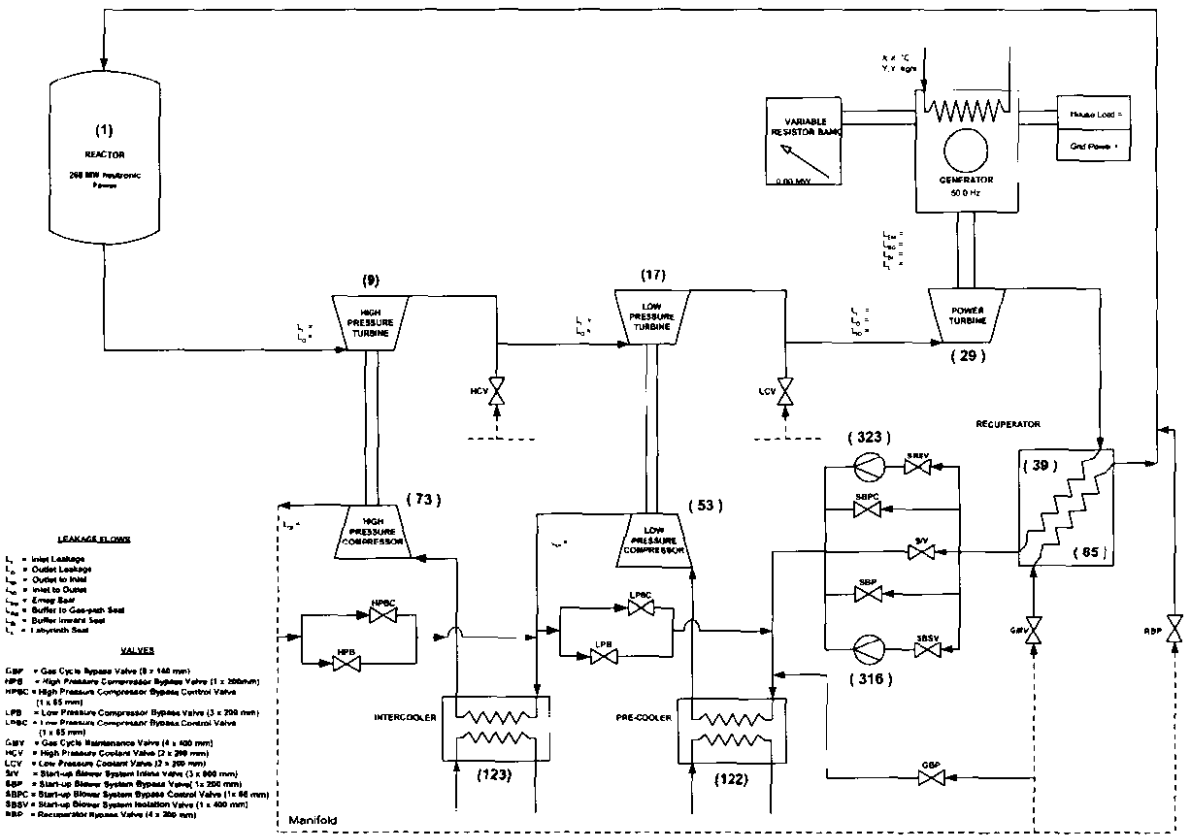


Figure 3.4: *V502 Process Flow*

4 DESCRIPTION OF THE TINTE CODE

4.1 Introduction

Gerwin (1987:1) states that the modular TINTE (Time-dependent Neutronics and Temperatures) code deals with the nuclear and the thermal transient behaviour of an HTR core taking into consideration the mutual feedback effects in two-dimensional (r-z) geometry.

TINTE solves the following sub problems:

- Time-dependent neutron flux calculation
- Time-dependent heat source distribution (local and non-local fractions)
- Time-dependent heat transport from the fuel to the fuel element surface
- Time-dependent global temperature distribution
- Gas-flow even under natural circulation conditions for both a given total mass flow and a given pressure difference
- Convection and its feedback to the circulation

The TINTE code was developed because of the dynamic experiments that had been conducted at AVR, which required an improved spatial representation on the point kinetics approach typically used. Corresponding to the initial tests, which only lasted a few minutes, TINTE was conceptualised as a short time dynamic code. The computation speed achieved made it possible to use the programme for longer transients. The good convergence properties also permit the use of TINTE for calculation of natural convection and complicated flow problems.

The modular design of the code (see Figure 4.1:) makes it possible to select individual calculation paths. For example, a targeted temperature change in the nuclear section can be used to determine local temperature coefficients. When the nuclear section is switched off, TINTE can be used as a stable thermo-fluid dynamic code. TINTE was also designed for use in the HTR with helium as the coolant. However, the code was adapted for use with other coolants (Gerwin & Scherer, 2004:38).

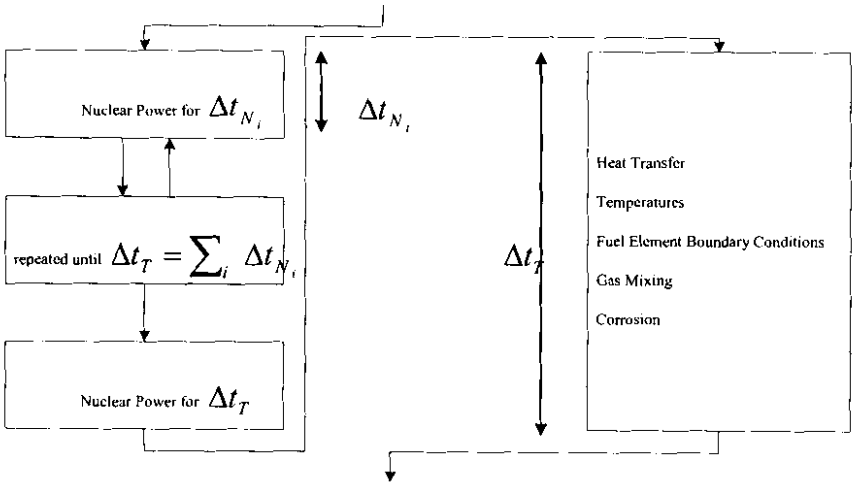


Figure 4.1: *Modular structure of TINTE*

In the dynamic approach to a nuclear reactor, nuclear events are influenced by changes in the core composition and the temperatures. The former factors include the following:

- An increase in burnup, which is associated with a decrease in reactivity.
- Changes in the composition of the coolant and possible corrosion of the structural material (e.g. water incursion and associated temperature-dependent graphite corrosion). Such effects are provided for in TINTE, but have not yet been implemented in the present version.
- The change in concentration of short-lived strong absorbers, particularly ^{135}Xe.
- Material movements in the core or in its proximity, particularly absorber rod movements.

Time-dependence is taken into account by means of discretisation into time intervals of varying length. The half-life of the nuclide partner with the shortest half-life generally limits the step width that can be used. For the dynamic calculation of neutron flux, time steps as small as 10^{-3} seconds would be necessary for the delayed neutron calculations. In most cases, the curves representing the output and the flux changes are so smooth, that time steps varying from a few seconds to minutes, appear to be perfectly adequate.

For the treatment of spatial dependence, a (wide-meshed grid) differential method was selected. The use of this method necessitates a common grid network for transfer of the

results of all the calculation steps. If not, continuous recalculation (necessary when different networks are used) will lead to unacceptable loss of information.

Before time-dependent problems can be solved with a dynamic programme, an initial stationary condition must be found. A temperature transient can start in any given condition. However, this is not the case with a nuclear transient, because the initial condition must be critical. That is to say in relation to the Eigen-value k, k = 1, must apply. Only in this condition can stationary reactor operation occur. To eliminate small deviations, which are determined by computer technology, one can adjust the neutron poison concentration. In TINTE the number of neutrons per fission was adjusted, i.e. the k Eigen-value was determined, and $\nu\Sigma_f$ was replaced by $\nu\Sigma_f / k$. This method is not only characterised by simplicity, but it also has the advantage that energy or spatial flux distortions can be avoided. Both the determination of the Eigen-value and the time-dependent nuclear calculation, are based on the same theory and use the same subprograms. Consequently the linear behaviour over longer time steps becomes stable, for instance at fixed temperatures, time steps of up to fifteen minutes are feasible.

4.2 TINTE reactor model

4.2.1 Neutron Diffusion Equation

To calculate nuclear heat sources one requires information about time-dependent neutron flux changes. This information can be obtained by solving the neutron transport equation or by satisfactory approximation with the diffusion equation. These calculations have to be carried out many times in a dynamic code. Hence one must attempt to achieve very rapid calculation speeds. In other words, only the diffusion approximation can be considered. For the same reasons, variable numbers of groups are not used when calculating the energy-dependence of the neutrons.

TINTE uses the leakage iteration method to solve the diffusion equation. During the realisation of the leakage iteration method it became evident that a very accurate determination of the leakage was required to ensure the method's stability. During location discretisation, when the fluxes are calculated at the end of the intervals or the interval

corners, this accuracy was not attainable. The discretisation where fluxes are defined as mean interval values, achieves the required accuracy.

Temperature-dependent cross-sections are calculated with respect to changes in the xenon concentration. To make provision for the pronounced dependence of the few-group cross sections on the flux curve, they were specified as leakage-dependent.

The time dependent two-group neutron diffusion equations solved by TINTE are as follows:
In the fast energy region:

$$\frac{1}{v_1}\frac{\partial \varphi_1}{\partial t} = \nabla D_1 \nabla \varphi_1 - (\Sigma_{a1} + \Sigma_{s1})\varphi_1 + \Sigma_{s2}\varphi_2 + (1-\beta)P + \sum_l \lambda_l C_l + Q$$

and in the thermal energy region:

$$\frac{1}{v_2}\frac{\partial \varphi_2}{\partial t} = \nabla D_2 \nabla \varphi_2 - (\Sigma_{a2} + \Sigma_{s2})\varphi_2 + \Sigma_{s1}\varphi_1$$

where

$P = \sum_{g=1}^{2} \nu\Sigma_{fg}\varphi_g$ = the neutron production rate,

$(1-\beta)$ = the portion of neutrons that is promptly obtained,

$\beta = \sum_i \beta_i$ = delayed neutron fraction of all i precursors with decay constants λ_i,

$1/v_g$ = the mean reciprocal neutron velocity in group g,

D_g = the diffusion constant of group g,

Σ_{ag} = the absorption cross-section of group g,

Σ_{sg} = the scattering cross-section from group g and

$\nu\Sigma_{fg}$ = the production cross-section of group g.

The change in precursor concentrations over time is described by:

$$\frac{dC_i}{dt} = \beta_i P - \lambda_i C_i$$

These ordinary differential equations, which contain constant coefficients for the delayed neutrons, can be directly integrated to give:

$$C_i = e^{-\lambda_i(t-t_0)}\left\{C_i(t_0) + \beta_i \int_{t_0}^{t} dt' P(t') e^{\lambda_i(t'-t_0)}\right\}$$

The nuclear (fission) power calculation starts by calculating the temperature distribution inside the fuel elements by using extrapolated boundary conditions for the heat transfer to the gas. From this fuel element temperature data, the moderator and fuel temperatures are derived. These are used to calculate the nuclear cross-sections. During the neutron flux iteration the ^{135}Xe concentrations are adjusted. In transient cases an iteration between the power distribution and the fuel element temperatures takes place. Figure 4.2: illustrates how the nuclear power and temperature distribution are calculated (Gerwin & Scherer, 2004:21).

Gerwin (1987:7) gives a more detailed description of the neutronic and thermal hydraulic equations used in TINTE.

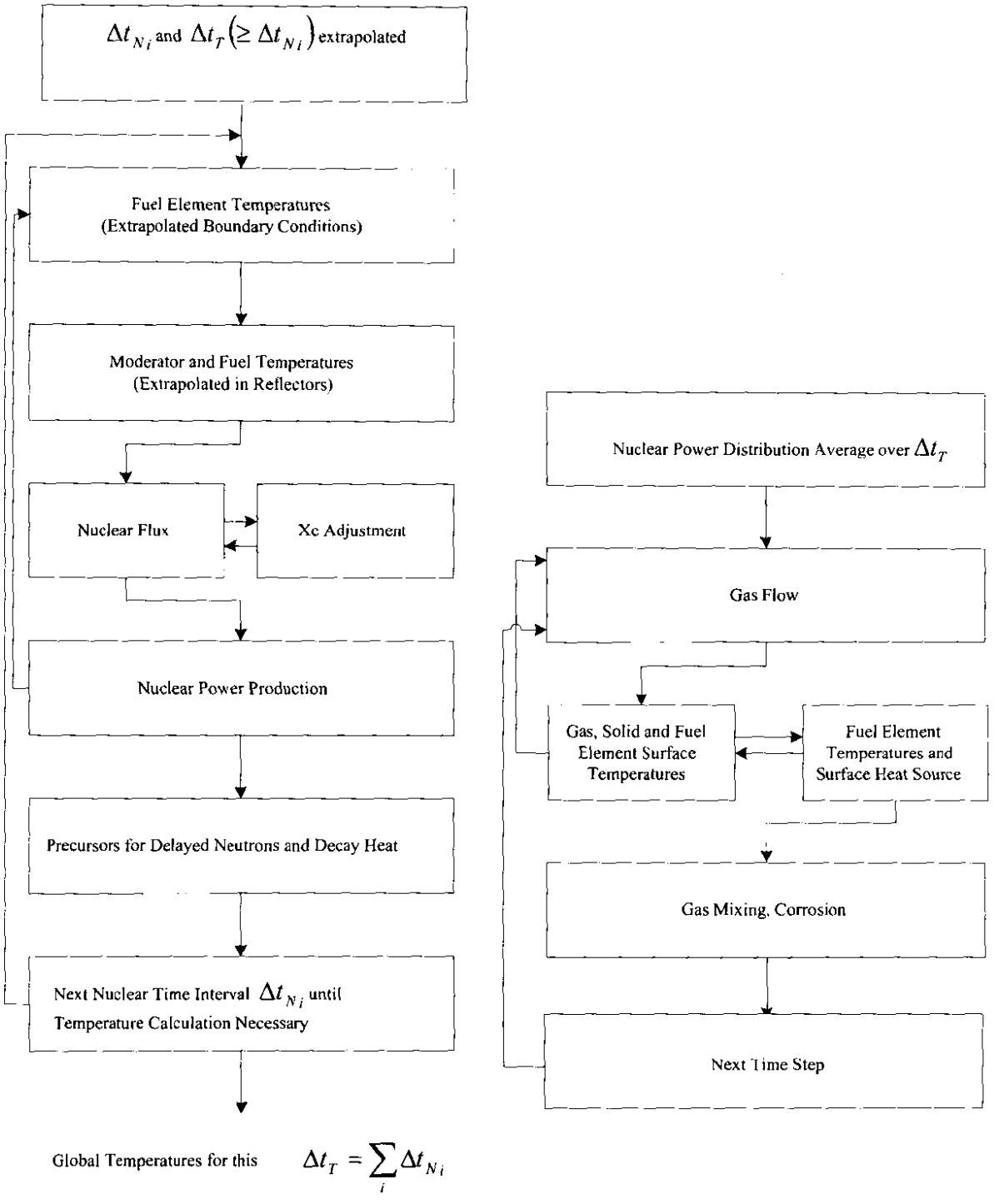
4.3 TINTE interface

The TINTE input consists of seven blocks. These blocks collect the input parameters for several fields of interest. The input data of blocks 1 to 5 have to be stored in a file with the extension '.tn3'. They must follow the same sequence as described by Gerwin and Scherer (2004:32). The blocks that are not used may be omitted. If the blocks stage out of sequence in the input stream, they are neglected by the TINTE code. The cross section data in block 6 have to be stored in a file with the extension '.tn4'. Block 7 data may either be fed in from the console in an interactive way by the user, or stored in a file with the extension '.tn1'.

4.3.1 General Control Parameters.

Block 1 specifies the main control parameters and is as follows:

- Specification of which modules are called and how they are interpreted
- The reactor power in MW and fixed external neutron source
- Maximum temperature and power change per time step
- Convergence parameters
- Gas property correlations



(a) *(b)*

Figure 4.2: *Calculation of (a) power and (b) temperature distribution in TINTE*

4.3.2 Geometry and Spatial Mesh Definitions.

Block 2 specifies the calculations for the solid temperatures that are performed in the total defined mesh grid. The gas temperature and the fluid flow are calculated only in those meshes that are declared as being flow meshes. For memory saving reasons, this may usually be done in a smaller mesh grid that has to be a subset of the total mesh grid. This holds true for both the nuclear calculations and the heterogeneous temperature calculations.

The definition of the mesh grid includes the mesh boundaries in the axial and radial direction, an optional division in finer meshes and information of the type of calculations to be performed in the mesh. The defined mesh grid is a "material" mesh grid. It should be constructed in such a way that a well-defined material assignment is possible for both the thermal-fluid and the nuclear calculations. Furthermore, temperatures, flows and power densities should not be homogenised. In defining the boundaries of this mesh grid, it is not necessary to account for a sufficiently small discretisation with respect to the finite difference solution of the differential equations. Smaller mesh subdivisions can be introduced by the user, in which the basic leakage iteration process is used only for the 1D calculations.

4.3.3 Material Assignment to the Mesh grid for Thermal-fluid Calculation.

After the input of the previous data block the total number of meshes is known. In Block 3, materials have to be assigned to these (coarse) meshes. Only numbers of the different components/materials are assigned here and the detailed description is done in the next block.

4.3.4 Material Description for Thermal-fluid Materials.

Block 4 defines the different materials that are available. These include solid material without gas flow, pebble bed with and without gas flow, the boundary layer between pebble bed and reflector, flow tubes, cavities and burst discs. The parameters for all these components including general pebble bed parameters such as fuel sphere diameter are specified in this block.

4.3.5 Material Assignment to the Mesh Grid for Nuclear Calculations.

Block 5 declares numbers to the nuclear materials for the part of the grid where nuclear calculations have to be done. The material meshes are also grouped into leakage iteration meshes in this block.

4.3.6 Nuclear Cross Section Data Base.

Block 6 consists of three parts: In the first part, the nuclear cross-sections and their polynomial expansions are specified. In the second part, information is read for the treatment of large cavities or holes (non-isotropic diffusion regions). The third part contains information on the treatment of the decay heat, and optionally on multi-fuel element properties. The two-group nuclear cross section data can be generated via spectrum codes like TISPEC. TINTE can also read results from the burn-up code system VSOP(99).

4.3.7 Control Commands for the Programme Operation.

All TINTE transient calculations (i.e. calculations where a variable, or more than one variable, changes in value over time) must be preceded by a complete steady-state calculation, or at least use the steady state restart file as a starting point for the transient. The easiest way of controlling the transient calculation is by specifying the transient control parameters in an input file (block 7) with the mandatory extension of *.tnl*. The input can also be given by the user in real time during the transient. Since typical transients takes 2-4 hours to complete, this is not always a practical solution. A wide variety of transient control options is available to the user.

In general, nuclear and thermal transient control commands are specified in the following manner:

- The time at which the ramp starts.
- End time of the ramp. The variable is changed linearly from the start time to the end time. When a ramp on the same variable starts before the first is finished, a polygonal dependence for the variable will be established.
- The final value of the parameter when the ramp ends.
- Identification of the material number on which the ramp is imposed, or specification of global ramps.
- The type of ramp or control command.

Global nuclear ramps include changes of effective multiplication constant k_{eff}, the equilibrium fission power, or the desired fission power.

Solid material ramps include temperature ramp, heat source ramp, ZETA ramp in tubular component and pressure jumps for a burst disc or safety valve.

Gas flow ramps include gas inlet temperature in gas source regions, pressure ramp, mass flow ramp and a ramp to change the volumetric flow source relative to the steady state value as well as a ramp to change the relative power removed by convection.

The user enters the times at which detailed two-dimensional output should be given. TINTE also displays one-dimensional data such as inlet and outlet temperature, pressure drop and mass flow after each temperature time step.

4.4 Time Steps

As was already shown in Figure 4.1:, the time steps are divided into nuclear time steps (Δt_N) and temperature times steps (Δt_T). At the start of a new time interval all the relevant parameters, which have been calculated in the previous time interval or with a stationary calculation, are available. In addition, the previous changes in certain variables over time are also known - either from a variable's prior maximum change or from the changes that have occurred in each one of the grids.

The maximum changes together with the slopes of the curves depicting the specified values which change over time, are used to determine the new step widths. Experience confirmed that it is necessary to limit the time step for nuclear calculations to a maximum of 60 seconds, or else the feedback to the start of the interval, which is primarily caused by long-lived delayed neutrons, is too small and causes instabilities. If the temperature changes slowly, the step width for the next temperature calculation can be longer (up to about 5 minutes).

These step sizes are not fixed at the start of the interval. If the flux changes in an interval are too pronounced, the calculation is interrupted and restarted for a shorter interval. If the nuclear calculations produce a pronounced heat production deviation, the thermal time interval is ended. After the step widths have been determined, the specified variables for the nuclear calculation (superimposed cross-sections) are interpolated to the end of the nuclear step from the specified timetable.

4.5 TINTE Core model

The following geometrical simplifications were made to the 268 MW core design in essentially two-dimensions (r-z). The pebble bed's upper surface was flattened (i.e. no fuel heaps or valleys were modelled). The dynamic central column and mixing zone widths were defined to be constant over the total axial core height. Control rods in the side reflector were modelled as a cylindrical skirt (also referred to as a grey curtain), with a given B-10 concentration.

Thermal-hydraulic simplifications include the specification of stagnant helium between the core barrel and RPV, and stagnant air between the RPV and heat sink (outer boundary). The coolant flow was restricted to flow upwards from the inlet below the core within a porous ring in the reflector, and downwards through the pebble bed to the outlet plenum. No reflector cooling or leakage paths were defined, i.e. all flow entering the reactor inlet plenum flows through the core. General simplifications included the assumption that all heat sources from fission should be deposited locally (i.e. in the fuel) and that no other heat sources exist outside the core (for example neutron absorption in the control rods). Simplifications were also made in the material thermal properties in as far as constant values or specific correlations were employed.

The core layout is shown in Figure 4.3:. The following table shows the core geometrical specifications at room temperature. Reitsma *et al.* (2005:9) provide more details.

Table 4.1: *TINTE 268MW Core Specification*

#	Description	Unit	Value
1	Equivalent core outer radius	m	1.75
2	Cylindrical height of the core (Flattened core surface at the top and flat bottom reflector).	m	8.5
3	Total core volume	m^3	81.779
4	Dynamic central column radius filled with graphite only spheres	m	0.786
6	Mixing zone outer radius with 50:50 mixture of fuel and graphite	m	1.109
7	Fraction of graphite spheres in the core	%	~30

Table 4.1: *TINTE 268MW Core Specification (continues)*

#	Description	Unit	Value
8	Effective height of the upper void cavity (levelled core surface to bottom of top reflector).	m	0.25
9	Effective annular thickness of the inner reflector block (graphite).	m	0.75
10	Equivalent annular thickness of the outer reflector block (carbon).	m	0.25
11	Inner radius of the core barrel	m	2.87
12	The wall thickness of the core barrel.	m	0.05
13	The inner radius of the RPV	m	3.0
14	The wall thickness of the RPV.	m	0.17
15	Radius of cooling system / 20° C temperature isothermal boundary	m	4.17
16	Radii of five material meshes in core (5 radial meshes in core)	m	0.786 1.109 1.357 1.566 1.750
17	Thickness of core radial meshes	m	0.786 0.323 0.248 0.209 0.184
18	Axial material mesh: 8.5 m / 17 meshes	m	0.5
19	Outlet plenum inner diameter	m	0
20	Outlet plenum outer diameter	m	1.75
21	Outlet plenum height	m	0.5
22	Inlet plenum inner diameter	m	2.15
23	Inlet plenum outer diameter	m	2.35
24	Inlet plenum height	m	0.5
25	He up-flow skirt / porous region inner radius	m	2.15
26	He up-flow skirt /porous region outer radius	m	2.35
27	Distance from bottom of core to top of the inlet plenum	m	1.0
28	Centre line axial distance between inlet and outlet plenum	m	1.0
29	Top inlet plenum inner diameter (mixture zone mesh)	m	0.0

Table 4.1: *TINTE 268MW Core Specification (continues)*

#	Description	Unit	Value
30	Top inlet plenum outer diameter	m	2.35
31	Top inlet plenum height	m	0.5
32	Distance from bottom of top reflector to bottom of top inlet plenum	m	1.0
33	Porous region (vertical flow) in top reflector inner diameter	m	0.0
	Porous region (vertical flow) in top reflector outer diameter (i.e. downward flow in the total top reflector volume directly above the pebble-bed)	m	1.75
34	Total height of top reflector (including top plenum)	m	1.5
35	Carbon block height at top of top reflector (above top inlet plenum)	m	0.25
36	Total height of bottom reflector (Distance from top of bottom plate to bottom of core).	m	2.75
37	Top steel plate thickness	m	0.3
38	Bottom steel plate thickness	m	0.3

	0	78.6	110.9	135.7	156.6	175	181	194	215	235	250	275	287	292	300	317	417	418
-231		78.6	32.3	24.8	20.9	18.4	6	13	21	20	15	25	12	5	8	17	100	1
-230	1	20	20	20	20	20	20	20	20	20	20	20	20	20	20	20	20	18
-200	30	13	13	13	13	13	13	13	13	13	13	13	13	12	15	16	17	18
-175	25	7	7	7	7	7	7	7	7	7	7	7	11	12	15	16	17	18
-125	50	9	9	9	9	9	9	9	9	9	3	7	11	12	15	16	17	18
-75	50	2	2	2	2	2	3	3	3	6	3	7	11	12	15	16	17	18
-25	50	2	2	2	2	2	3	3	3	6	3	7	11	12	15	16	17	18
0	25	1	1	1	1	1	3	5	3	6	3	7	11	12	15	16	17	18
50	50	a	b	c	d	e	3	5	3	6	3	7	11	12	15	16	17	18
100	50	a	b	c	d	e	3	5	3	6	3	7	11	12	15	16	17	18
150	50	a	b	c	d	e	3	5	3	6	3	7	11	12	15	16	17	18
200	50	a	b	c	d	e	3	5	3	6	3	7	11	12	15	16	17	18
250	50	a	b	c	d	e	3	5	3	6	3	7	11	12	15	16	17	18
300	50	a	b	c	d	e	3	5	3	6	3	7	11	12	15	16	17	18
350	50	a	b	c	d	e	3	5	3	6	3	7	11	12	15	16	17	18
400	50	a	b	c	d	e	3	5	3	6	3	7	11	12	15	16	17	18
450	50	a	b	c	d	e	3	5	3	6	3	7	11	12	15	16	17	18
500	50	a	b	c	d	e	3	5	3	6	3	7	11	12	15	16	17	18
550	50	a	b	c	d	e	3	5	3	6	3	7	11	12	15	16	17	18
600	50	a	b	c	d	e	3	5	3	6	3	7	11	12	15	16	17	18
650	50	a	b	c	d	e	3	5	3	6	3	7	11	12	15	16	17	18
700	50	a	b	c	d	e	3	5	3	6	3	7	11	12	15	16	17	18
750	50	a	b	c	d	e	3	5	3	6	3	7	11	12	15	16	17	18
800	50	a	b	c	d	e	3	5	3	6	3	7	11	12	15	16	17	18
850	50	a	b	c	d	e	3	5	3	6	3	7	11	12	15	16	17	18
900	50	4	4	4	4	4	3	3	3	6	3	7	11	12	15	16	17	18
950	50	4	4	4	4	4	3	3	3	6	3	7	11	12	15	16	17	18
1000	50	4	4	4	4	4	3	3	3	8	3	7	11	12	15	16	17	18
1050	50	4	4	4	4	4	3	3	3	3	3	7	11	12	15	16	17	18
1100	50	10	10	10	10	10	3	3	3	3	3	7	11	12	15	16	17	18
1125	25	7	7	7	7	7	7	7	7	7	7	7	11	12	15	16	17	18
1155	30	14	14	14	14	14	14	14	14	14	14	14	14	12	15	16	17	18
1156	1	19	19	19	19	19	19	19	19	19	19	19	19	19	19	19	19	18

Neutronics

a	Dynamic central column (graphite sheres only)
b	Mixture region with 50:50 ratio of fuel and graphite spheres
c	3rd flow channel with only fuel spheres
d	4th flow channel with only fuel spheres
e	Outer flow channel with only fuel spheres
1	Void area above the pebble bed
2	Top reflector graphite
3	Side reflector graphite
4	Bottom reflector graphite
5	Control rod grey skirt / graphite
6	Side reflector graphite (with helium channels)
7	Carbon side reflector / insulation

Thermal Hydraulics

a	Pebble bed
b	Pebble bed
c	Pebble bed
d	Pebble bed
e	Pebble bed
1	void area above core with vertical helium flow
2	Top reflector porous region with downwards helium flow
3	Side reflector (no helium flow)
4	Bottom reflector porous region with downwards helium flow
5	Control rod region
6	Porous side reflector graphite (with upward flow in helium channels)
7	Carbon top / side / bottom reflector / insulation
8	Bottom inlet plenum
9	Top Inlet plenum
10	Bottom Outlet plenum
11	Stagnant helium at pressure between side reflector and barrel
12	Barrel
13	Top plate
14	Bottom plate
15	Stagnant helium at pressure between barrel and RPV
16	RPV
17	Air outside RPV
18	Heat sink
19	Adiabatic boundary condition
20	Adiabatic boundary condition

Figure 4.3: *TINTE Core Layout and Identification*

5 INTERFACE DESIGN

5.1 Introduction

Data transfer between Flownex and TINTE will be accomplished by creating an external program aptly called Flownex-TINTE-Interface (FTI). FTI must receive data from one code, do manipulations if necessary and then pass it to the other code. FTI must also synchronize Flownex and TINTE.

5.2 Preliminary Design

5.2.1 Introduction

In the preliminary design phase, two different methods of coupling are investigated, namely the direct method and the indirect method. It will be shown that while both are conceptually feasible, the direct method suffers from problems introduced by the TINTE solver.

5.2.2 Direct Method

This method constitutes the passing of temperatures, mass flow and pressures between TINTE and Flownex to apply as boundary conditions for the next time step. The methodology is as follows:

1. Perform independent steady state calculations for both TINTE and Flownex using approximations as boundary conditions.
2. Pass the reactor inlet mass flow, inlet temperature and outlet pressure from Flownex to TINTE to be used as new boundary conditions.
3. Perform a time step with the new TINTE boundary conditions and pass the reactor inlet pressure, outlet temperature and outlet mass flow to Flownex to be used as new boundary conditions.
4. Perform a Flownex time step.
5. Repeat the procedure from step 2 until the desired time is reached.

This procedure is shown schematically in Figure 5.1:.

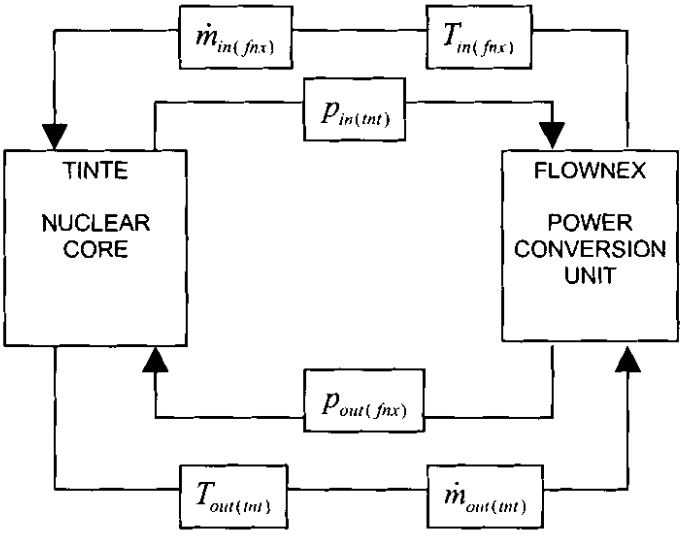


Figure 5.1: *Direct coupling method*

Although this method is conceptually very simple, it has negative consequences. After the steady state calculations for TINTE and Flownex, different mass flows and pressure drops will be observed for the PCU and core model. It is tempting to just supply the same mass flow to both models, but this will still result in different pressure drops across the core and PCU. The only way to rectify this is to use an iterative method (such as Newton-Raphson) in order to attain the same mass flow and pressure drop. This is shown visually in Figure 5.2:. The problem with this is that TINTE cannot redo a temperature time-step, and therefore a new steady state would have to be calculated after each time step. Even if computational time were not an issue, this would suppress all nuclear transients.

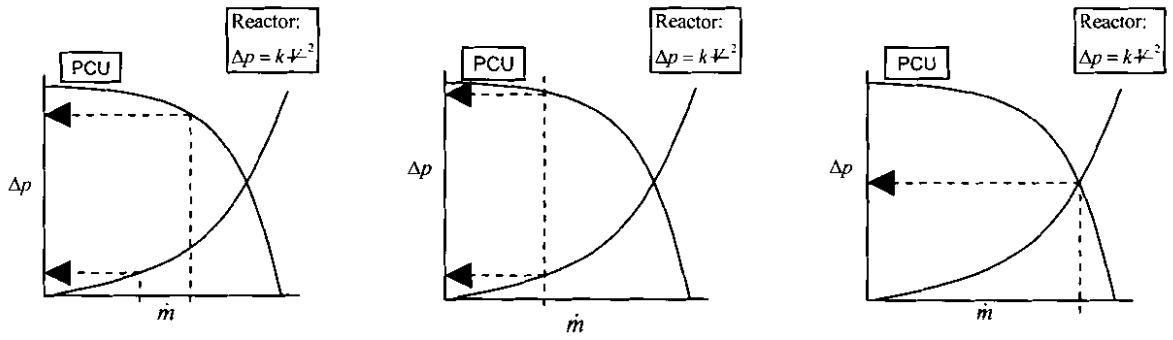


Figure 5.2: *Problem with direct coupling method: (a) Different Steady States, (b) Different delta P for the same mass flow, (c) Iteration.*

5.2.3 *Indirect Method*

With the indirect coupling method, the reactor model in Flownex is replaced by a pipe with (variable) losses, which has about the same fluid volume and cross sectional flow area as the reactor. A characteristic loss factor K is determined for the TINTE reactor model as well as the Flownex PCU network. These values are then related to the friction factor k of the pipe to complete the match. Stepwise the methodology is as follows.

1. *Model the reactor core as a pipe with losses:*

$$\Delta p = k_{pipe} \frac{1}{2} \rho V_{pipe}^2$$

where

k_{pipe} = pipe loss factor,
ρ = average density of the gas and
V_{pipe} = gas velocity.

2. *Perform steady state calculations for both TINTE and Flownex using approximations as boundary conditions.* In TINTE the boundary conditions that must be specified are the inlet temperature T_{in}, the inlet mass flow $\dot{m}_{in}$, and the outlet pressure p_{out}. In Flownex it is the pipe loss factor k_{pipe} and the power transferred to the pipe P_{he}.

3. *Exchange data from Flownex to TINTE.* Pass the reactor inlet mass flow $\dot{m}_{in(fnx)}$, inlet temperature $T_{in(fnx)}$ and outlet pressure $p_{out(fnx)}$ from Flownex to TINTE.

4. *Perform a TINTE time step.* The maximum time step is calculated internally by TINTE and is based on the maximum allowable temperature change per time step. At the end of the time step new values for $\dot{m}_{in(tnt)}$, $p_{in(tnt)}$ and P_{he} are available.

5. *Calculate loss factor K_{tnt} for TINTE and K_{fnx} for Flownex.* The overall loss coefficient values can be defined as follows:

$$K_{tnt} = \frac{p_{in(tnt)} - p_{out(tnt)}}{V_{in(tnt)}^2} \qquad \textit{and} \qquad K_{fnx} = \frac{p_{in(fnx)} - p_{out(fnx)}}{V_{in(fnx)}^2}$$

where

K_{tnt} = Characteristic loss factor of reactor,

$p_{in(tnt)}$ = Inlet pressure of reactor calculated by TINTE,

$p_{out(tnt)}$ = Outlet pressure of reactor according to TINTE,

$\dot{V}_{in(tnt)}$ = Volume flow at reactor inlet calculated using the inlet mass flow according to TINTE,

K_{fnx} = Characteristic loss factor of PCU,

$p_{in(fnx)}$ = Inlet pressure calculated by Flownex,

$p_{out(fnx)}$ = Outlet pressure calculated by Flownex and

$\dot{V}_{in(fnx)}$ = Volume flow at inlet calculated using the inlet mass flow according to Flownex.

6. *Determine the ratio of the Flownex loss factor K_{fnx} to the pipe loss factor k_{pipe}:*

$$C = \frac{K_{fnx}}{k_{pipe}}$$

7. *Determine the new loss factor for the pipe:*

$$k'_{pipe} = \frac{K_{tnt}}{C}$$

8. *Update the pipe properties.* The new pipe loss factor k'_{pipe} and the heat that is transferred to the helium $P_{he,}$ which was calculated by TINTE, are used as the new boundary conditions for the next Flownex time step.

9. *Perform a Flownex time step.* At the end of this time step, new reactor inlet mass flow $\dot{m}_{in(fnx)}$, inlet temperature $T_{in(fnx)}$ and outlet pressure $p_{out(fnx)}$ are obtained.

10. *Repeat the procedure from step 3 for a number of time steps.*

This procedure is shown schematically in Figure 5.3:.

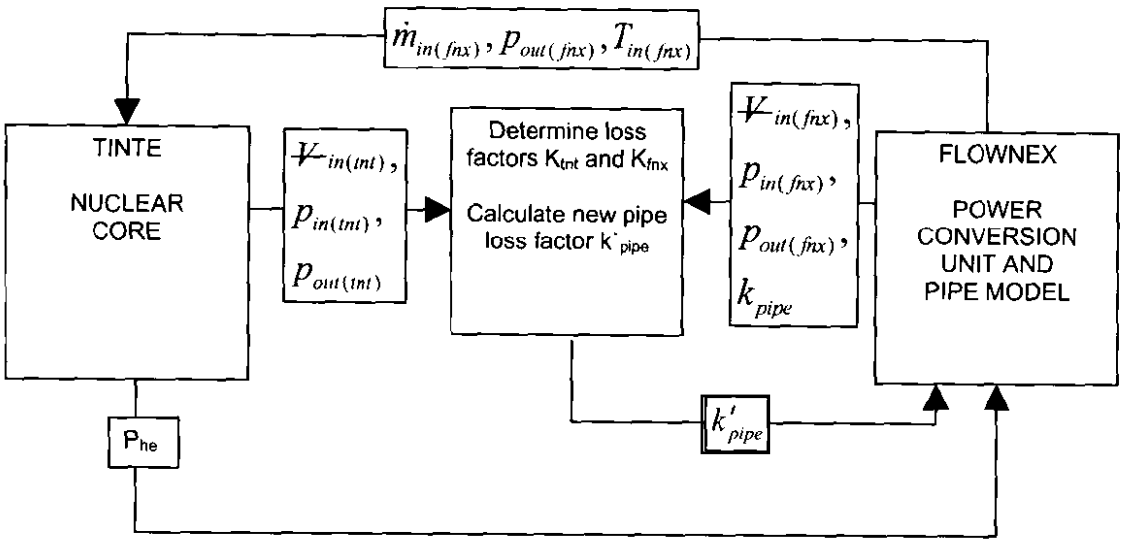


Figure 5.3: *Indirect Coupling Method*

5.2.4 Conclusion

By using the indirect coupling method, a combined steady state for both the core and the PCU can be obtained immediately after a few time steps. No iteration is needed and therefore this method can be implemented to couple Flownex and TINTE.

5.3 Detailed Design

5.3.1 Introduction

FTI was designed to implement the indirect coupling method described above to provide a high-level integration of TINTE and Flownex.

5.3.2 Programming language

Most of the TINTE source is coded in FORTRAN-77, but some of the later subroutines and extensions have been added in FORTRAN-90 format. Alterations to the TINTE code have thus been done in FORTRAN-90.

Flownex is coded in C++ and uses the Windows Application Program Interface (Windows API) for external control. Therefore, no code changes were needed in Flownex.

The Flownex-TINTE-Interface program was coded using Microsoft® Visual C++ .NET 2003.

5.3.3 *Program flow*

The program flow of FTI can be broken down into two main parts. The first is the creation of the memory structures together with the start-up of Flownex and TINTE. This can be seen in Figure 5.4:. The second part handles the data transfer needed to couple TINTE and Flownex. The process flow of this second part of the program is pictured in Figure 5.5:.

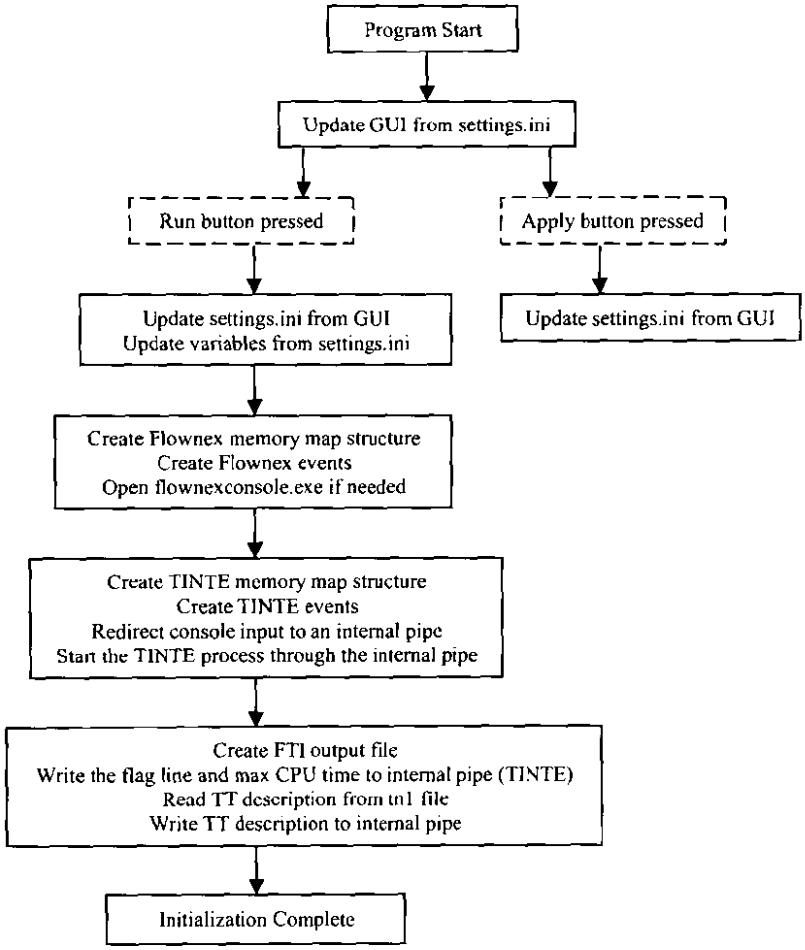


Figure 5.4: *FTI initialization*

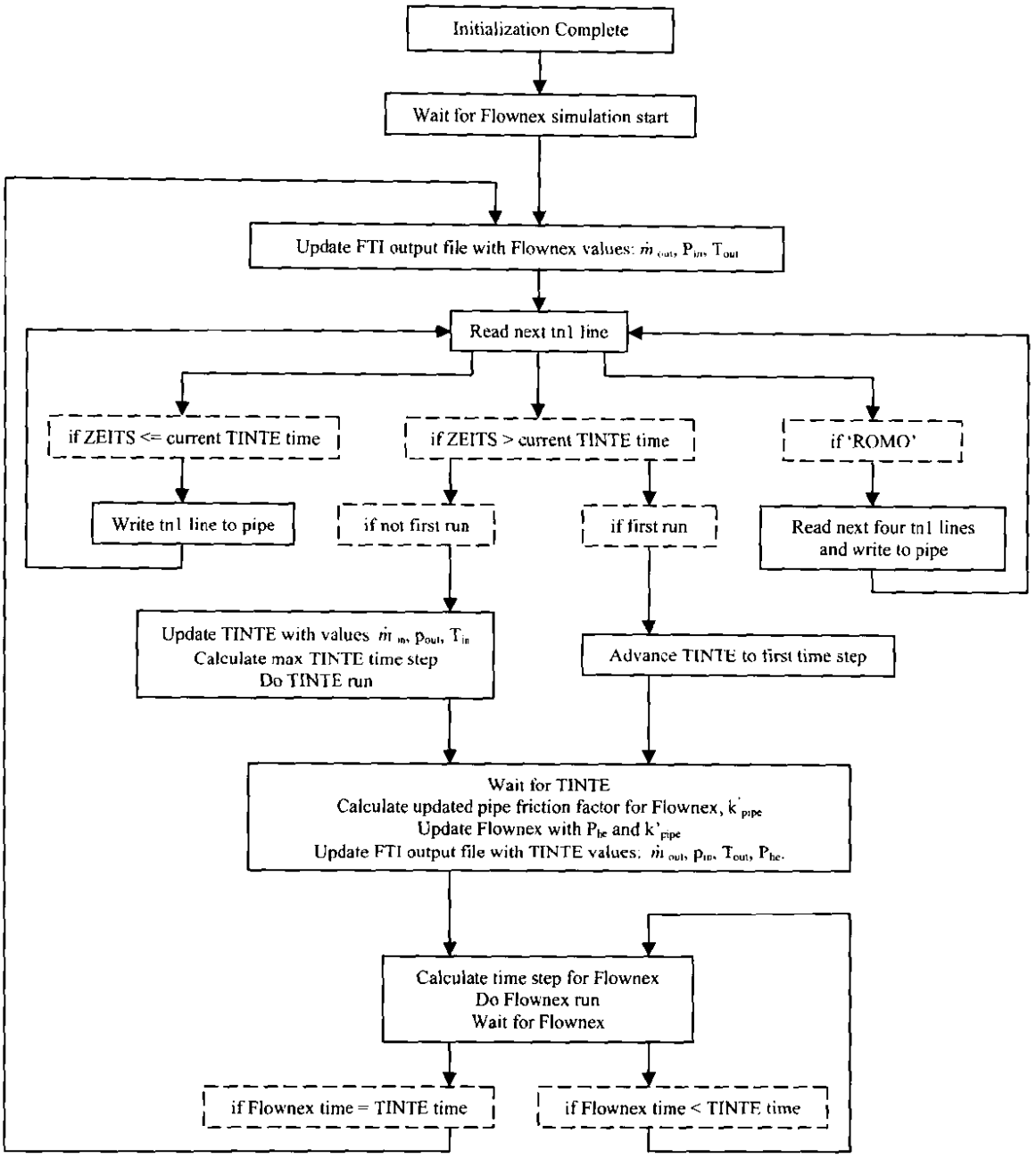


Figure 5.5: *FTI Process flow*

Data is transferred after each TINTE time step and Flownex then adapts to this time step. This time scheme can be seen in Figure 5.6:. After a steady state is reached by both programs, a time step is done in TINTE. At the end of this time step, the output data from TINTE and the current Flownex data is processed and passed as the input data for the next Flownex time step. Flownex then continues the simulation until the Flownex time is the same as the TINTE time. The output data of Flownex is then used to update TINTE.

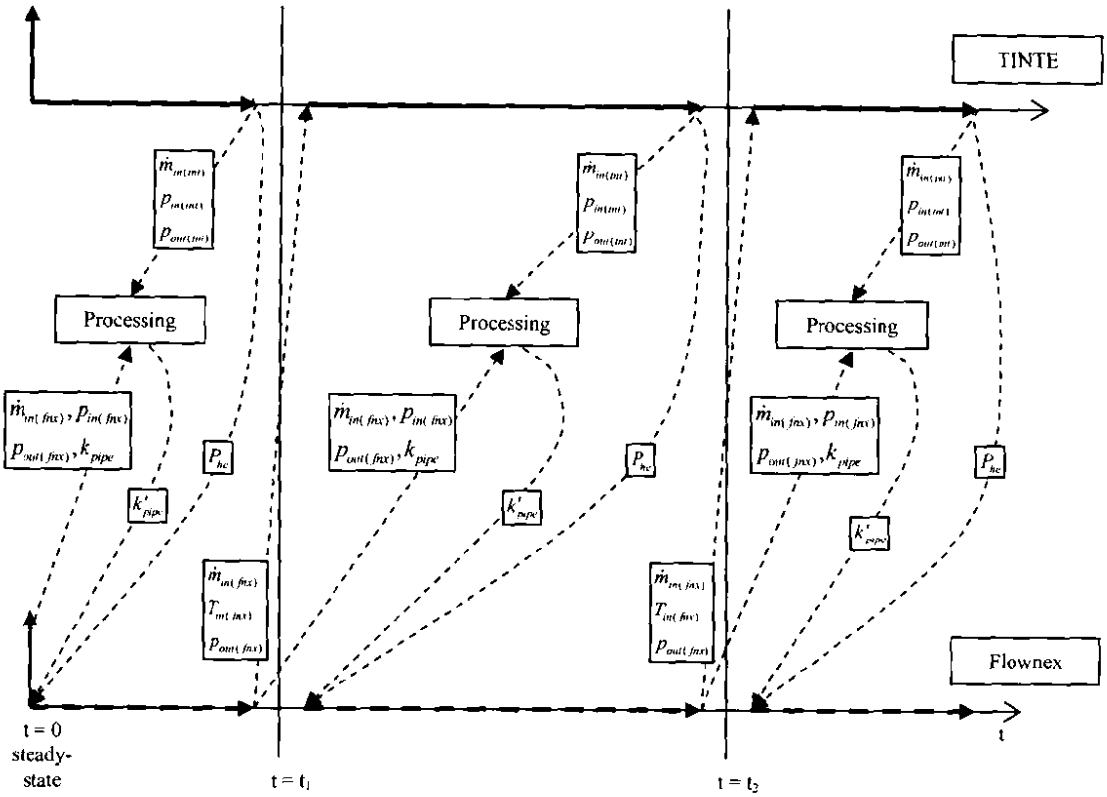


Figure 5.6: *Time-wise data exchange between Flownex and TINTE*

5.3.4 TINTE Code Changes

The transient input commands of TINTE is done either via the console or via an ASCII text file and a new input is only read after the TINTE solver has reached the destination time specified by the user. TINTE displays a number of parameters such as the inlet and outlet temperatures, pressure drops and mass flows to the screen after each temperature time step. The time step is internally calculated and therefore, it would be unwise to limit it as to accommodate FTI. Hence, it was inevitable to make a number of changes to the TINTE source code.

1. Time Step Changes

Changes to the code were introduced to manipulate the time specified by the user to coincide with the end of a temperature time step. For example, the user might specify a simulation of 300 seconds. In the old code, this would have resulted in TINTE performing a number of nuclear and temperature time steps until 300 seconds were reached. The user could then enter a new transient, or another destination time. In the changed code, the user will specify a simulation of 300 seconds, but TINTE will now only perform the necessary nuclear time

steps to perform a single temperature time step, which might result in say, five seconds. After this time step, control is given back to the user and a new transient or a destination time must be specified. This was done by setting the variables D(KF+3) = D(KF) as well as ZEITS = D(KF) in node 580 of the *tinte.f* file.

2. Memory Map

The output needed from TINTE is displayed to the screen, as well as written to a binary file after each temperature time step. This file must then be converted in order to access the data, causing an unnecessary delay. To overcome this problem, a memory map structure was implemented into TINTE. After each temperature time step, the required variables were updated in the memory map. This method enables one to access variables directly in the computers memory and thus eliminating the need for data conversion and transfer via files on the computer's hard drive.

Care must be taken when using the memory map across different programming languages as the starting index of arrays and vectors differ. The first element of an array in C++ is indexed as 0, whereas the first element of an array in Fortran is indexed as 1. The memory map is defined in the new module TIXIO and has the following structure:

```
TYPE MemoryFileStruct
        SEQUENCE
        REAL(8)::Running              //Used as flag - for future use
        REAL(8)::T                    //Current TINTE time
        REAL(8)::dT                   //Temperature time step length
        REAL(8)::Inputs(1000)         //Input vector - redundancy for future use
        REAL(8)::Outputs(1000)        //Output vector - redundancy for future use
END TYPE MemoryFileStruct
```

Figure 5.7: *TINTE Memory Map File Structure*

3. Windows Event Signaling

Windows event signalling was also implemented into the TINTE code. After each temperature time step, a signal is given to the Windows environment, which can be monitored by external programs. This event signalling results in straightforward synchronization between the different codes as an external program can now wait until a signal is received, before attempting to access data. The event is created in the global memory space and has the name "Tinte External Start".

4. *Input Output Changes*

Two new control commands were added to the TINTE input command list. The command "-12" will read input values from the memory map into the variables ZEITS, ZEITZ, WERTZ, IORT and IART. ZEITS is assigned the value of the first element of the input array, ZEITZ to the second and so forth. The control command "-13" will only read a value into the variable ZEITS and is used for starting the next temperature time step.

The memory map output-array definitions are shown in Table 5.1:.

Table 5.1: *TINTE Memory map output-array*

Element number	Element Description
1	Reactor outlet pressure
2	Reactor inlet pressure
3	Reactor outlet temperature
4	Reactor inlet temperature
5	Reactor outlet mass flow
6	Reactor inlet mass flow
7	CBCS outlet pressure
8	CBCS inlet pressure
9	CBCS outlet temperature
10	CBCS inlet temperature
11	CBCS outlet mass flow
12	CBCS inlet mass flow
13	Reactor nominal power
14	Power transferred to gas

5.3.5 FTI Design

Interaction with FTI is done by means of a GUI where all the relevant files and settings for the current simulation are chosen. This includes the following:

1. *Global Settings*

- *Path of FTI settings file.* This file contains information on the GUI variables. The file will replace the contents of the current settings.ini file in the program directory. Any

subsequent changes to the GUI variables will thus only be in the settings.ini file leaving the original chosen file intact.

- *Path of the FTI output file.* This file contains information about the data that is passed between TINTE and Flownex.

2. *Flownex Setup*

- *Path of 'flownexconsole.exe'.* Flownexconsole is the command line interface of Flownex.
- *Path of Flownex Project (*.fnz).* This is the Flownex file that contains the network data relevant to the current simulation.
- *External Control Set Name.* This is the name of the external control set which is specified in Flownex. The alignment of the variables of the external control set must be in exactly the order shown in Table 5.2:.

Table 5.2: *Flownex external control set alignment*

Plant Input Number	**Element / Node**	**Variable**
0	Reactor simulation pipe	Transferred Power
1	Reactor simulation pipe	Sum forward losses
Plant Output Number	**Element / Node**	**Variable**
0	Reactor inlet downstream node	Pressure
1	Reactor inlet downstream node	Temperature
2	Reactor inlet	Total mass flow
3	Reactor inlet downstream node	Mean density
4	Reactor outlet downstream node	Pressure
5	Reactor outlet downstream node	Temperature
6	Reactor outlet	Total mass flow
7	Reactor outlet downstream node	Mean density
8	Reactor simulation pipe	Mean density
9	Reactor simulation pipe	Maximum Velocity
10	Reactor simulation pipe	Sum forward losses

- *Transient Control File (*.ftc).* This file has not yet been implemented. It is however envisaged to enable the user to start a Flownex transient event, which has previously been exported and is not saved in the current simulation network.
- *Run with Flownexconsole checkbox.* If it is enabled, FTI will open the selected network with a *flownexconsole* to do the simulation.
- *Max time step(s).* This is the maximum time step length to be used in the Flownex simulation.

3. *TINTE Setup*

- *Path of TINTE.exe.* This specifies the path of the TINTE executable.
- *Path of *.tn1 file.* This file contains the transient control commands of the TINTE simulation.
- *Path of *.tn3 file.* As already mentioned, this file contains data blocks 1 – 5 which describes the nuclear core.
- *Path of *.tn4 file.* This file contains the cross section data calculated by VSOP.
- *Path of *.rtn file.* This is the restart file. A restart file can be used in the following manner: A steady state is calculated for the core and stored in the restart file. When a new simulation is to be started, the restart file can be loaded and a new steady state calculation is not necessary. A restart file can also be created at any time during a transient. This point can then be reloaded to perform other transients without the need to simulate the first part again.
- *Path of *.p2d file.* This file contains two-dimensional binary information of calculated variables for graphical representation.
- *Path of output file.* This file contains two-dimensional information of the core at specific times as requested by the user in the tn1 file.
- *Max CPU time (min).* If no limit on CPU time is to be set, this must be left blank or set to zero.
- *File flag line.* These are control parameters used by TINTE. Gerwin and Scherer (2004:34) present a complete and detailed description of these parameters. It is important to note that the console input of TINTE was intercepted by redirecting it to an internal pipe. Therefore, when specifying the flag line, the input has to be set to the console (usually device #5) and not the *tn1* file itself. The *tn1* file is therefore virtually read in through the console.

- *Max time step(s).* This value might not be the actual maximum time step taken by TINTE as it is also specified in the project files. This value is added to the current TINTE time to create destination times.
- *Core inlet node.* As specified in the reactor setup files.
- *Core outlet node.* As specified in the reactor setup files.
- *CBCS inlet node.* Core barrel conditioning system inlet node as specified in the reactor setup files.
- *CBCS outlet node.* Core barrel conditioning system outlet node as specified in the reactor setup files.

It is necessary to note that CBCS simulation is currently not implemented. The reason is that the Flownex plant model that was obtained for academic purposes (V502), does not include the CBCS network. However, only a small alteration to the code will be necessary to include the CBCS network as well.

5.3.6 Conclusion

The modifications to the TINTE code make direct memory access of the variables possible and the Windows events makes synchronization with external programs easy. The FTI GUI provides an easy way to locate the relevant files and enter parameters needed for the simulation. These settings are saved in a file and are reloaded when FTI is started up again. Care must be taken to keep the right order when setting up the external control file of Flownex.

5.4 Summary

Two methods of coupling were explored, but due to the internal workings of TINTE, it is not viable to couple Flownex directly. An indirect method was proposed which attempts to match characteristic values of the TINTE core with the Flownex PCU. These characteristic values are updated after every time step, which eliminates the need for an iterative procedure when analysing transients. Some changes were made to the TINTE I/O modules, which allows for better data transfer and synchronization with external programs.

6 VALIDATION STUDY

6.1 Introduction

The validation study focuses on the error introduced by the indirect coupling method. The study further concentrates on the influence of the time step as the coupling procedure incorporates a time based data exchange.

In the studies that follow, reference is made to the inlet and outlet temperatures, pressures, and mass flow. Figure 6.1: gives an indication to the positions where these parameters are calculated. As can be seen, the reactor model of both Flownex and TINTE include a control rod cooling channel (CRCC) and therefore the entire mass flow entering the core does not flow through the pebble bed itself.

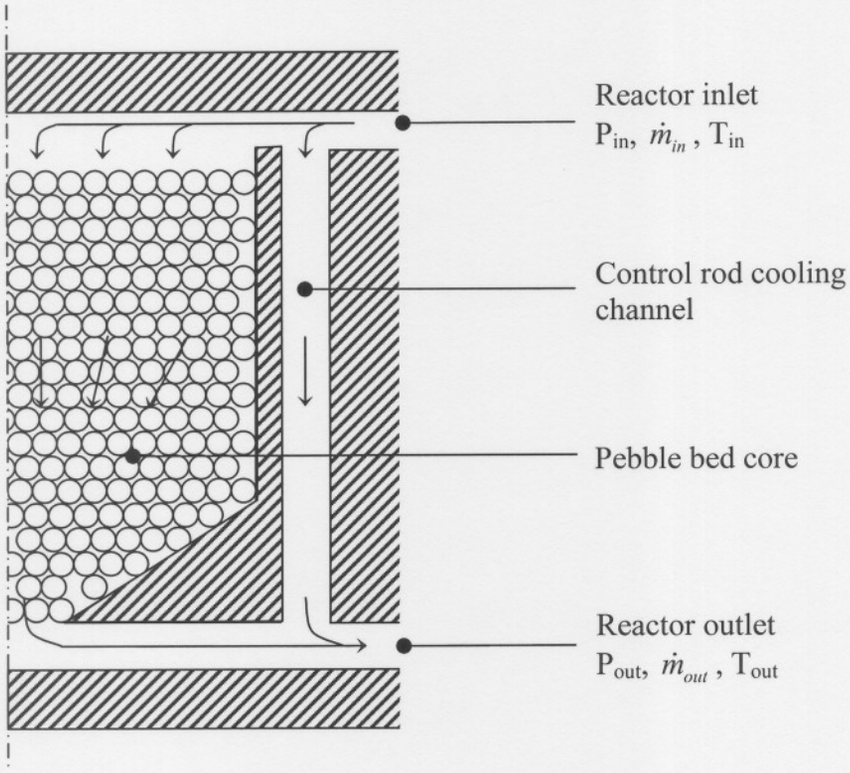


Figure 6.1: *Core parameter calculation positions*

It must be noted that in the results of the validation study, Flownex refers to the Flownex part of FTI and not to the stand-alone V502 Flownex model.

6.2 Margin of error – Steady State

6.2.1 Introduction

Both TINTE and Flownex use iterative schemes to determine steady state values before any transient analysis can be done. As already explained in Chapter 5.2.2, the TINTE and Flownex models reach different steady states when simulated independently. The indirect coupling method, which is employed in FTI, attempts convergence of these different solutions by matching characteristic values after each time step. Theoretically, the two solutions should be the same if infinite time has passed. In practise, it is not possible to simulate for infinite time. However, if the values stay constant over some time or show little change from one time step to another, it can be assumed that a steady state was reached. The converged steady state should therefore be the same for the Flownex and TINTE parts of FTI and any deviations could therefore be an indication of the error introduced by the indirect coupling method.

6.2.2 Implementation

A combined steady state solution was reached as will be described in Chapter 7.2. The solution was then allowed to continue a time-wise data exchange for 30 hours of simulated time, which took around 2 hours of CPU time. A maximum TINTE time step of 50 seconds and a maximum Flownex time step of 2.5 seconds were used.

6.2.3 Results

Table 6.1: shows the minimum, maximum and average values obtained during the entire simulation. Δ_{max} is the absolute difference between the minimum and maximum values obtained. $Q_{fission}$ is the fission power of the reactor and is shown as a percentage of the plant nominal power. In this case, the nominal power is 268MW. P_{he} is the heat that is transferred from the fuel to the coolant. This is also the power that is coupled to the Flownex pipe emulating the reactor.

Table 6.1: *FTI steady state results – original models*

TINTE										
	T_{in} [°C]	T_{out} [°C]	P_{out} [bar]	ΔP [bar]	$\dot{m}$ [kg/s]	$T_{fuel\text{-}max}$ [°C]	$T_{fuel\text{-}ave}$ [°C]	Xe	P_{he} [kW]	$Q_{fission}$ [%]
Min	491.77	880.13	68.87	0.7304	132.13	1090.60	839.12	2.12	266793	99.67
Max	491.97	880.90	68.87	0.7309	132.17	1091.70	839.78	2.13	267222	100.0
Ave	491.86	880.54	68.87	0.7308	132.17	1091.16	839.57	2.12	267001	99.83
Δ_{max}	0.20	0.77	0.00	0.0004	0.0400	1.10	0.66	0.01	429	0.33

Flownex							
	T_{in} [°C]	T_{out} [°C]	P_{out} [bar]	ΔP [bar]	$\dot{m}$ [kg/s]	P_{he} [kW]	k – losses
Min	491.65	880.55	68.86	0.7476	132.13	266793	2730.22
Max	491.85	881.37	68.86	0.7483	132.17	267222	2732.02
Ave	491.75	880.95	68.86	0.7482	132.17	267001	2731.95
Δ_{max}	0.20	0.82	0.09	0.0007	0.0407	429	1.79

The results obtained were processed to give an indication of the average and maximum error that is introduced by the indirect coupling method. This is shown in Table 6.2:.

Table 6.2: *Absolute difference between TINTE and Flownex when using FTI*

	T_{in} [°C]	T_{out} [°C]	P_{out} [bar]	ΔP [bar]	$\dot{m}$ [kg/s]	Q [kW]
Max	0.32	1.24	0.0029	0.0179	0.0447	0.0
Ave	0.11	0.41	0.0004	0.0174	0.0003	0.0

6.2.4 Discussion

The above tables show that the indirect coupling method does not give a perfect reproduction of the temperatures, pressures and mass flow. However, a relatively inconsequential error exists. As expected, there is no error introduced by the transferred power.

6.3 Time step influence

6.3.1 Introduction

From Figure 5.6: it is clear that TINTE is the driving force and Flownex is the slave. This means that data is passed to Flownex only after TINTE has completed a full temperature time step. This time step length is irregular and can range from milliseconds up to a few minutes. This is calculated internally by TINTE. The length is based on the maximum allowable temperature change per time step, as well as a user-defined maximum. To synchronize Flownex with TINTE, FTI adapts the Flownex time steps to ultimately match the TINTE time step. The TINTE time step is usually much longer than the maximum Flownex time step. It is thus necessary to perform a number of Flownex time steps to catch up with TINTE.

The values obtained at the end of a TINTE time step act as the boundary conditions for the beginning of the Flownex time steps, which will range over the same time span. This brings forth the problem that Flownex always present data with a one time step delay. In slow transients, this is not as great a problem as in fast transients.

When using FTI, two different transients can be defined, namely Flownex transients and TINTE transients. When a transient is specified in TINTE, the simulation will run up to the defined starting point and then continue. The time step lengths usually shorten dramatically to accommodate the change in temperature and pressure. Therefore, when a TINTE transient is initiated, more data will automatically be exchanged from the beginning to the end of the transient.

The state of a Flownex transient is different. FTI is unaware when a Flownex transient will start and as already mentioned Flownex follows TINTE in time. This introduces the problem that transient data will only be exchanged with TINTE after two temperature time steps. As a temperature time step can be in the order of minutes, this is unacceptable.

6.3.2 Implementation

One way of rectifying this problem is to decrease the maximum time step duration taken by TINTE. This will however increase the CPU time dramatically. The solution is to specify the

starting time of the Flownex transient in the TINTE *tn1* file as well as to define a number of small fixed time steps thereafter. TINTE will now be forced to exchange data with Flownex during the start of the transient. As soon as TINTE receives the transient data from Flownex, it will automatically shorten its time step length and the simulation can continue as normal.

This procedure will be demonstrated by a mass flow reduction through the core. The gas-cycle bypass valves are opened to reduce the mass flow to 51.6 kg/s in 5 seconds. A number of simulations were done using different maximum time steps in TINTE. The maximum Flownex time step was set to one second in all the simulations. The results of some of these simulations are shown in Figure 6.2: to Figure 6.4:.

6.3.3 Results

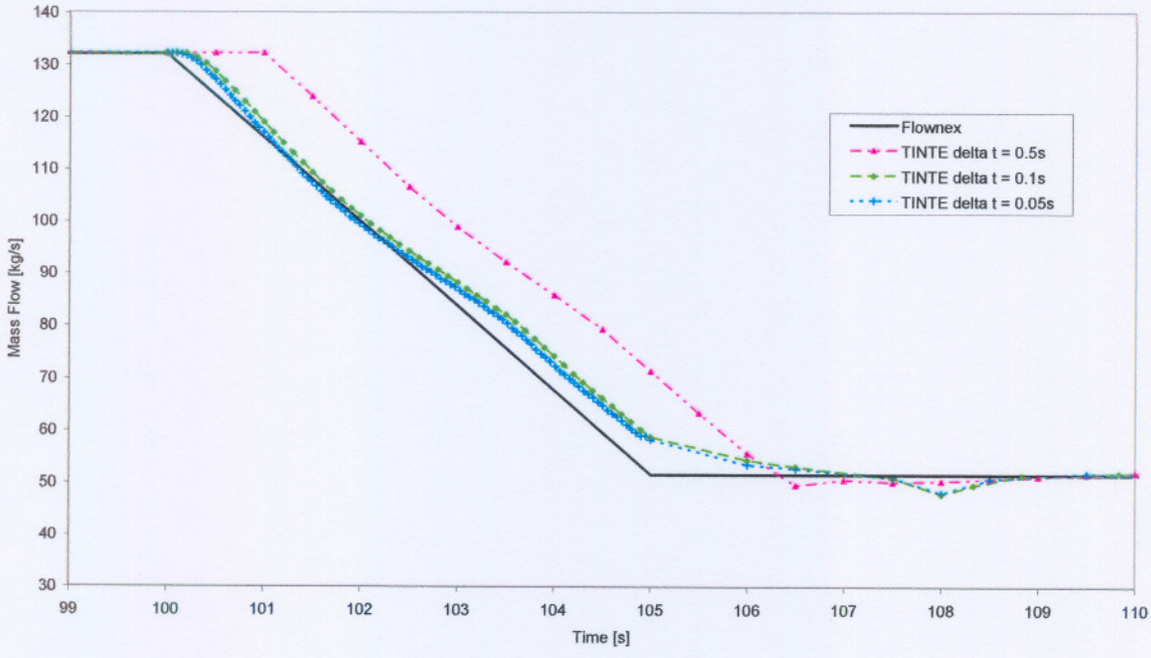


Figure 6.2: *Time step influence - Mass flow*

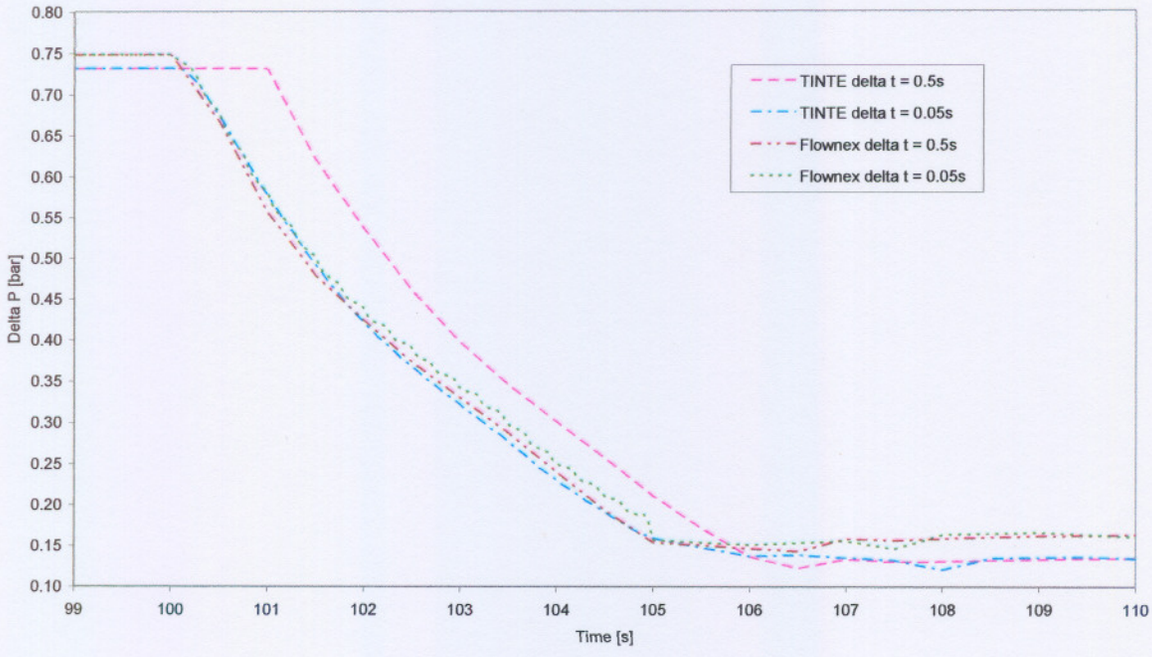


Figure 6.3: *Time step influence - Pressure drop*

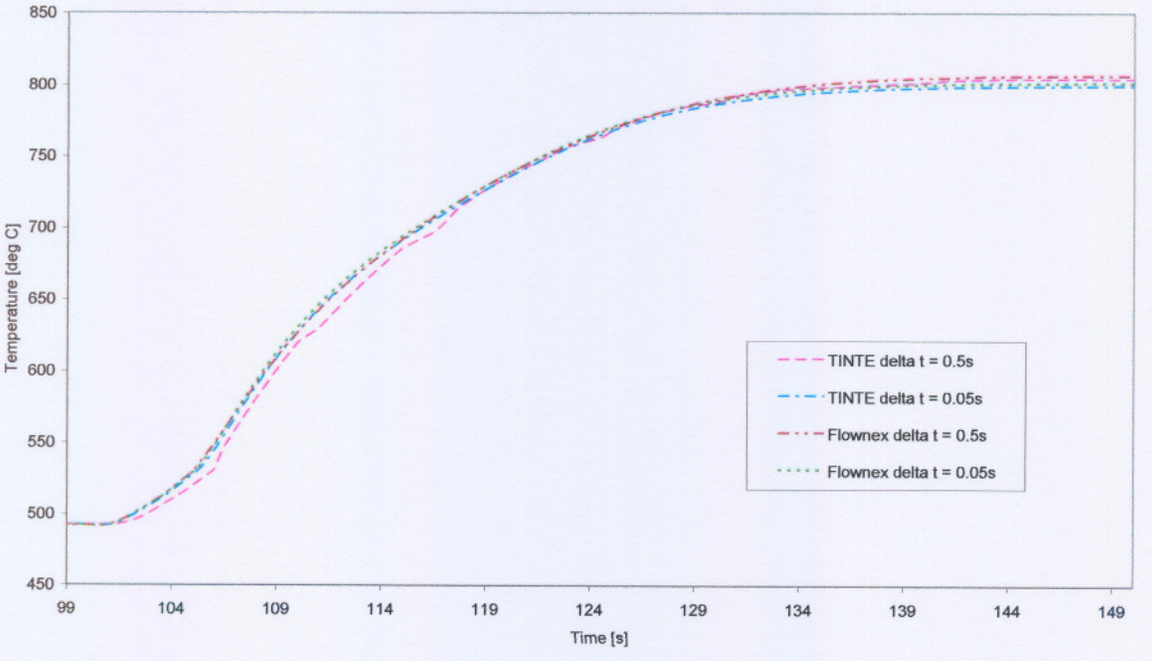


Figure 6.4: *Time step influence – Inlet temperature*

6.3.4 Discussion

The results confirm that FTI is very dependent on the time step length. This is especially true when Flownex initiates a thermal hydraulic transient. Reducing the maximum TINTE time step reduces the error caused by the coupling method, but this also increases the CPU time. The greatest errors are introduced at the start of transients but diminish as a new steady state is reached.

6.4 Summary

The validation study shows that the coupling method introduces errors when reproducing mass flow, temperature and pressure. These errors are relatively small in steady state analysis, but become more pronounced when dealing with fast thermal hydraulic transients. Decreasing the maximum time step length in TINTE reduces this problem, but increases the computational time dramatically.

7 SIMULATION STUDY AND RESULTS

7.1 Introduction

During the lifetime of a nuclear reactor, many transient situations can occur, such as start-up, power transitions, shutdown and much dreaded accidents. The purpose of the simulations is to analyse the steady state and transient conditions using the PBMR 268MW V502 plant model and comparing it to the results of the V502 model when using the alternative TINTE PBMR 268MW reactor model. Based on these results, it will be possible to determine if the point kinetic model as incorporated in Flownex simulates the neutronic aspects of the reactor accurately enough.

7.2 Test case 1 - Steady state

7.2.1 Introduction

As with many engineering simulations, a time independent solution must be reached before transient or accident conditions can be modelled. This is done by choosing appropriate boundary conditions and initial values, which are then used in an iterative scheme to determine the final steady state values.

7.2.2 Implementation

The steady state of the coupled plant was acquired through a number of manual iterations. First, an independent TINTE steady state was calculated using estimates as boundary conditions. This steady state solution was used to create a TINTE restart file. This restart file was then used as a starting point for the TINTE part of FTI. Initial conditions were also estimated for Flownex and a steady state calculation was performed. The coupled simulation was started and ended after some time. The values available at the end of the simulation were then used as new initial conditions in the TINTE *tn3* file as well as for the new Flownex initial conditions. New steady state calculations in both TINTE and Flownex were performed again and the process was repeated until convergence was reached.

7.2.3 Results

Table 7.1: shows the results obtained by determining the plant steady state. The 'CPU time' is the actual real time that the simulation was running while the 'Sim time' is the simulated time. These times were not optimized and the final steady state can more readily be reached by implementing more iterations while shortening CPU time.

Table 7.1: *Iterative steady state calculation*

			TINTE				Flownex	
Iter.	Sim time [min]	CPU time [min]	T_{in} [°C]	T_{out} [°C]	P_{out} [bar]	$\dot{m}$ [kg/s]	Q [kW]	k – losses
1			500.00	900.00	68.00	130.00	280000	500
2	1.6	3	493.92	895.68	68.85	131.97	275230	2770
3	1.6	3	492.56	883.91	68.85	132.15	268427	2731
4	16.6	5	491.97	881.97	68.86	132.16	267519	2731
5	416.6	35	491.77	881.06	68.86	132.17	267050	2732
6	500.0	34	491.73	880.87	68.86	132.17	266946	2731
7	1800.0	139	491.77	881.08	68.86	132.17	267064	2732

A steady state calculation was then performed with the stand-alone Flownex model. The stand-alone Flownex model refers to the $268MW_{th}$ V502 PBMR plant, which includes the point-kinetic neutronic model. The result of this simulation is shown in Table 7.2:. This table also shows an averaged FTI steady state result for comparison purposes. These averages were calculated from the averages of the TINTE and Flownex parts of FTI.

Table 7.2: *(a) Stand-alone Flownex and (b) averaged FTI steady state results*

	T_{in} [°C]	T_{out} [°C]	P_{out} [bar]	ΔP [bar]	$\dot{m}$ [kg/s]	$T_{fuel\text{-}max}$ [°C]	$T_{fuel\text{-}ave}$ [°C]	Xe	P_{he} [kW]	$Q_{fission}$ [%]
a	501.28	898.48	68.22	1.247	130.00	1059.44	858.01	1	268000	100.0
b	491.81	880.75	68.87	0.740	132.17	1109.16	839.57	1	267001	99.83

A large difference in pressure drop was observed and further investigations revealed that the stand-alone Flownex model accounts for secondary losses in other reactor components such as the inlet plenum holes, outlet slots and the CRCC whereas the TINTE model does not. The secondary losses in the Flownex model were therefore changed to zero. Different mass flows in the CRCC were also observed and the TINTE model was adjusted to give the same mass flow as the Flownex model. Table 7.3: shows the results obtained with the modified models.

Table 7.3: *(a) Stand-alone Flownex and (b) averaged FTI steady state results – modified models*

	T_{in} [°C]	T_{out} [°C]	P_{out} [bar]	ΔP [bar]	$\dot{m}$ [kg/s]	$T_{fuel\text{-}max}$ [°C]	$T_{fuel\text{-}ave}$ [°C]	Xe	P_{he} [kW]	$Q_{fission}$ [%]
a	492.90	884.14	68.67	0.944	131.97	1047.57	839.27	1	268000	100.0
b	494.64	885.41	68.66	0.930	131.48	1037.21	820.69	1	266973	99.81

7.2.4 Discussion

The difference in P_{he} comes from the fact that TINTE does not transfer all the thermal power generated by the reactor to the gas. Radiation and conduction by the solid materials also plays a role and therefore a typical steady-state value for P_{he} is 99.55% of the reactor thermal power.

In the steady state run, TINTE adjusts the fission power to keep the reactor critical, therefore $Q_{fission}$ ranges between 99.67% and 100%. This is caused by the minor changes in temperature and mass flow introduced by the coupling method.

As can be seen in Table 7.2:, there exists a large difference in pressure drop between the Flownex and TINTE (FTI) model. The reason for this is that the thermal-hydraulic specifications of the two acquired reactor models are not exactly the same. In TINTE, the control rod cooling channels are modelled as an annular skirt around the reactor because of 2D symmetry. The mass flow in this combined cooling channel is 16% of the inlet mass flow, which amounts about 21kg/s. In Flownex, the cooling channel is modelled as a 1D pipe with 3.2kg/s mass flow. The mass flow of helium through the packed bed is therefore much larger than in the TINTE model, which naturally results in a larger pressure drop.

The mass flow through the CRCC of the TINTE model was adjusted to 3.2kg/s. This corrected the pressure drop through the bed itself, but different pressure drops for the whole core were still observed. This was rectified by setting the secondary losses of the core components of the stand-alone Flownex model to zero.

The main parameters that were observed are very close to one another, except for the average and maximum fuel temperatures. These temperatures are very dependent on the mesh spacing in both models, but in the Pebble Bed Element of Flownex, only the axial mesh increments can be increased.

The xenon concentrations are normalized to the individual steady state values and are therefore equal to one.

7.3 Test case 2 - Load Follow

7.3.1 Introduction

One of the main advantages in the PBMR design is its load follow capability. Load follow is achieved by changing the helium inventory. This in turn adjusts the fission power of the reactor. A decrease in fission power results in an increase in ^{135}Xe concentration. This is a result from the decay of ^{135}I and a decrease in xenon transmutation. A new equilibrium concentration will result some time after the transition. ^{135}Xe has a very large absorption cross section for thermal neutrons, and with the increase, it would be impossible to return to the original fission power before the xenon concentration has decayed below a critical value. The core has some excess reactivity that is obtained by circulating the fuel faster. The excess reactivity in the core can override this xenon poisoning effect and an upward power transition will be possible at any requested time. The control rods are only moved to overcome the reactivity effects of changes in temperature and xenon concentration.

7.3.2 Implementation

The load follow simulation was implemented by only changing the reactor fission power in both models. No explicit change in mass flow was introduced. At one hour, the fission power was reduced to 40%, which is 107.2 MW. The power was then returned to 268 MW seven

hours into the simulation. The transitions were performed linearly lasting five minutes each. The simulation was continued up to 72 hours as to account for the xenon effects.

In TINTE this was accomplished by defining a global nuclear parameter ramp in the *tn1* file. The reactor fission power was changed to 107.2 MW over a time span of five minutes and then returned to 268 MW, which lasted another five minutes. In Flownex the total power was adjusted to the desired values.

A maximum TINTE time step of 50 seconds and a maximum Flownex time step of 2.5 seconds were used in the FTI simulation. In the pure Flownex simulation a maximum time step of two seconds were used in the non-transition time spans and a maximum time step of one second during the transitions.

7.3.3 Results

CPU time for FTI was just under four hours and the stand-alone Flownex simulation lasted one hour.

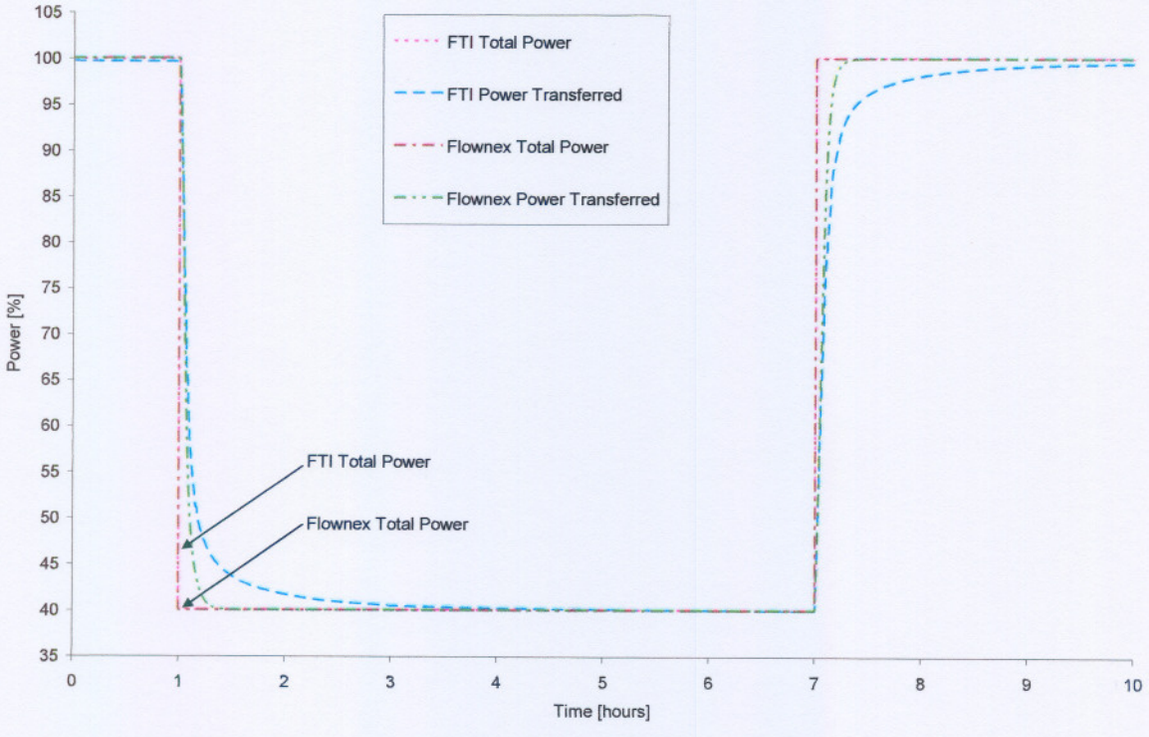


Figure 7.1: *Load Follow – Reactor Power*

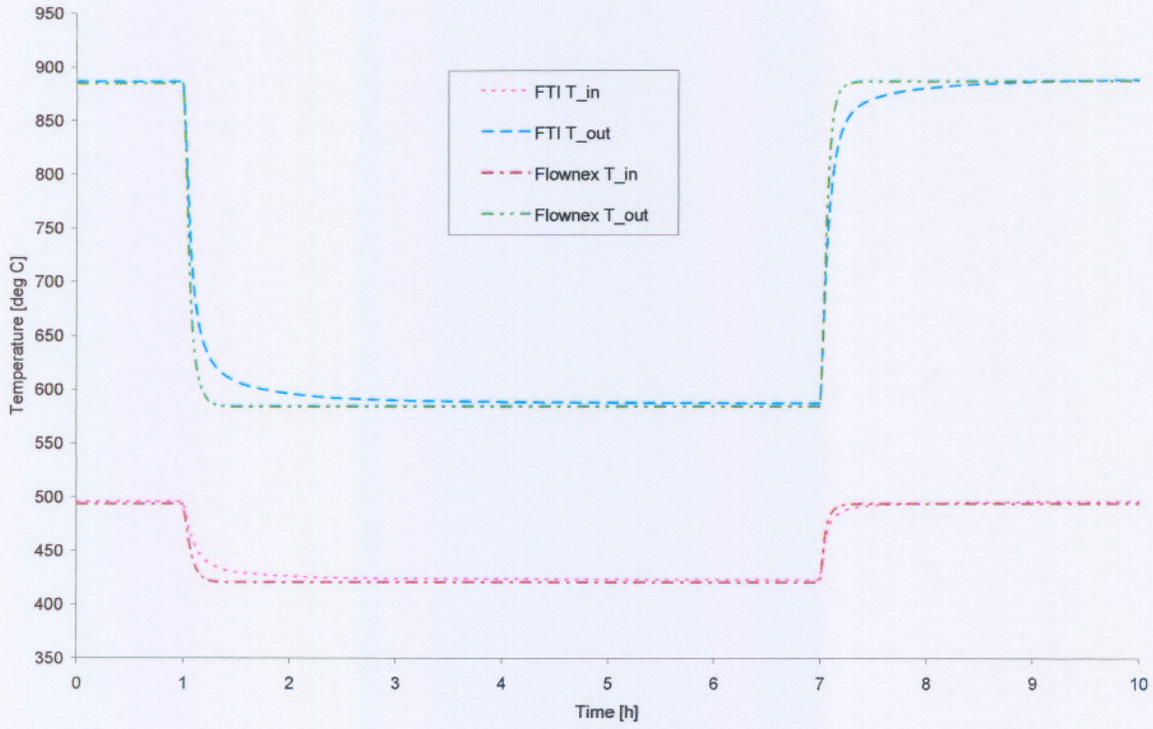


Figure 7.2: *Load Follow – Helium temperatures*

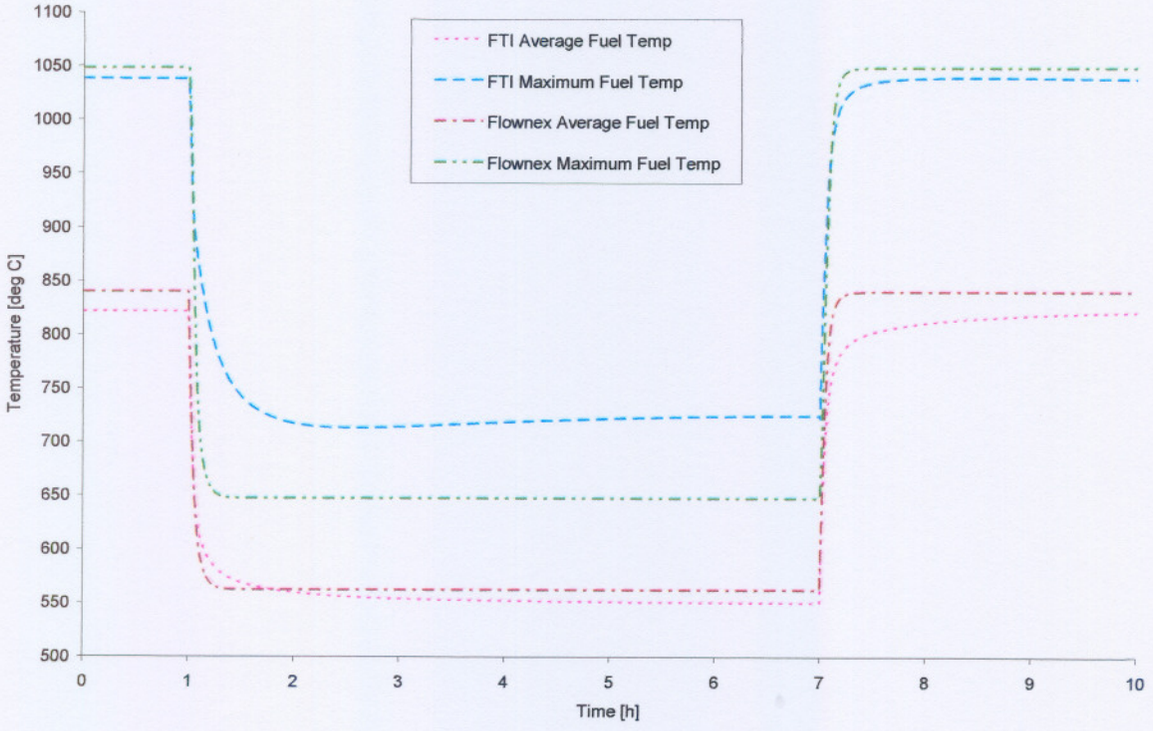


Figure 7.3: *Load Follow – Fuel temperatures*

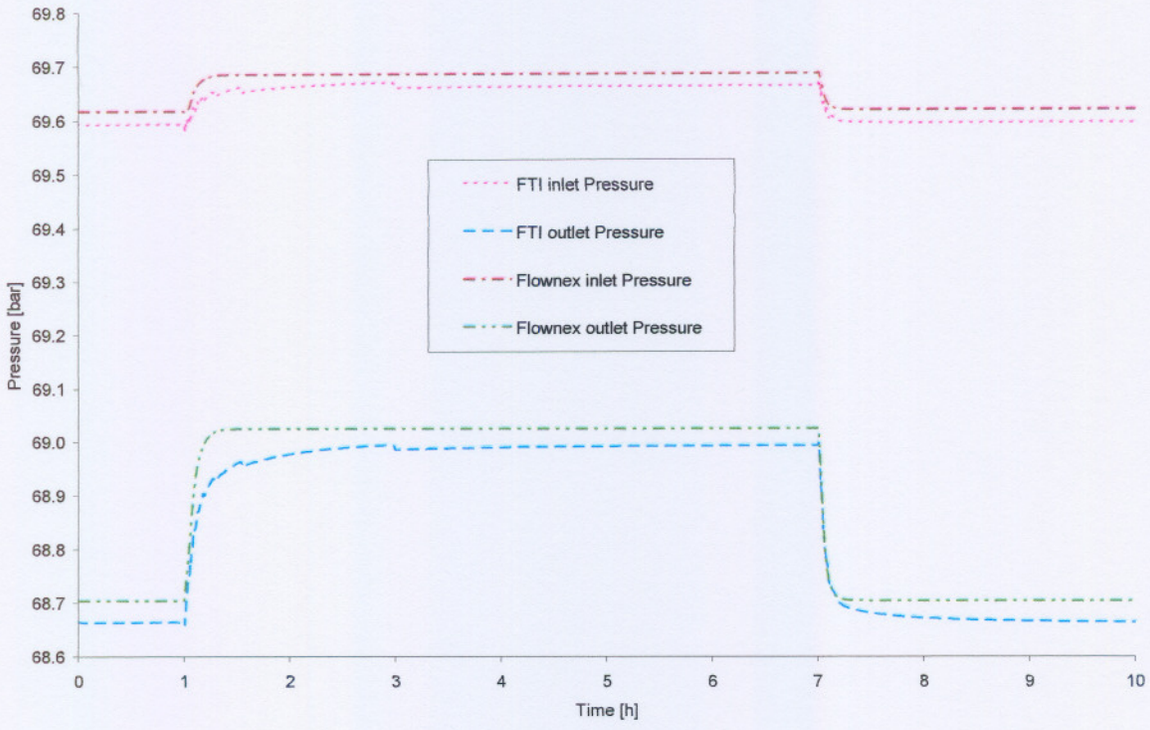


Figure 7.4: *Load Follow - Pressures*

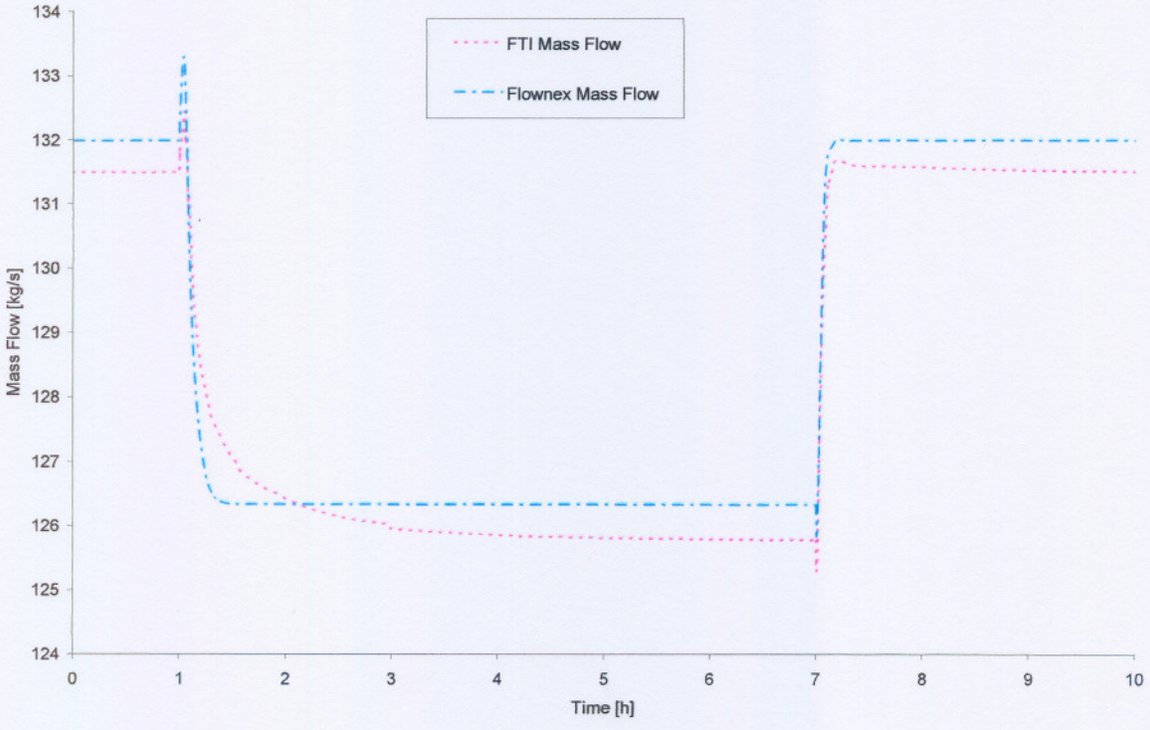


Figure 7.5: *Load Follow – Mass flow*

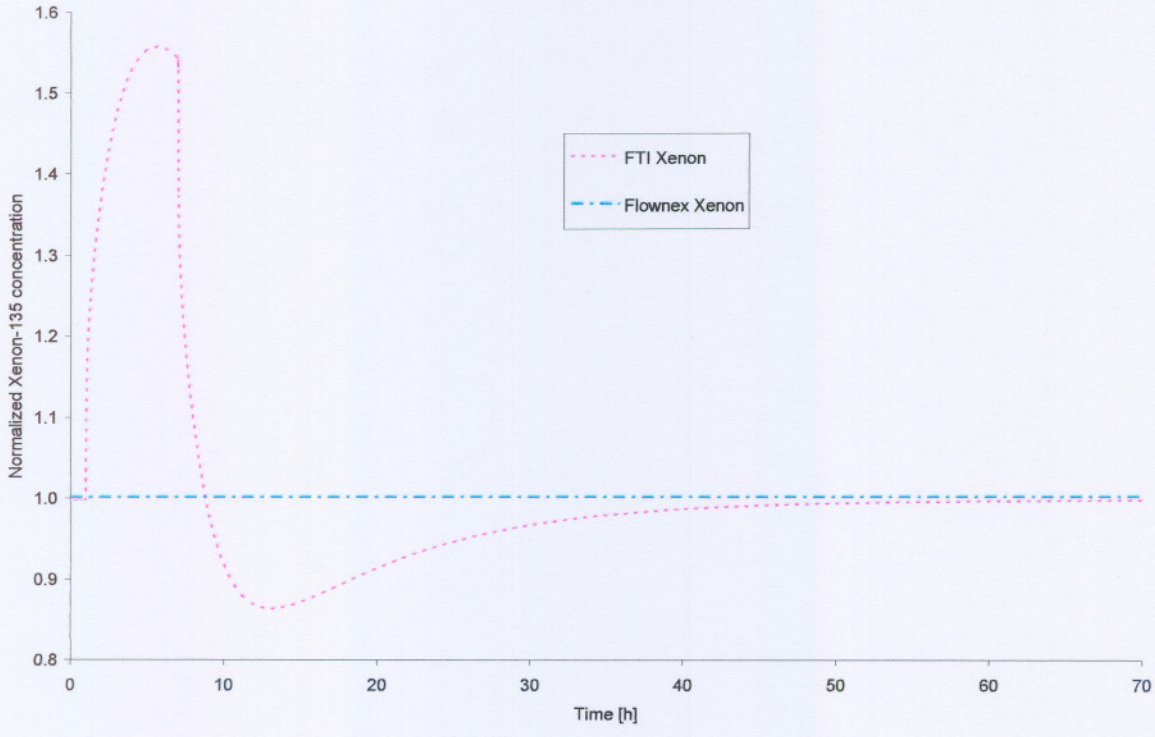


Figure 7.6: *Load Follow – ^{135}Xe concentrations*

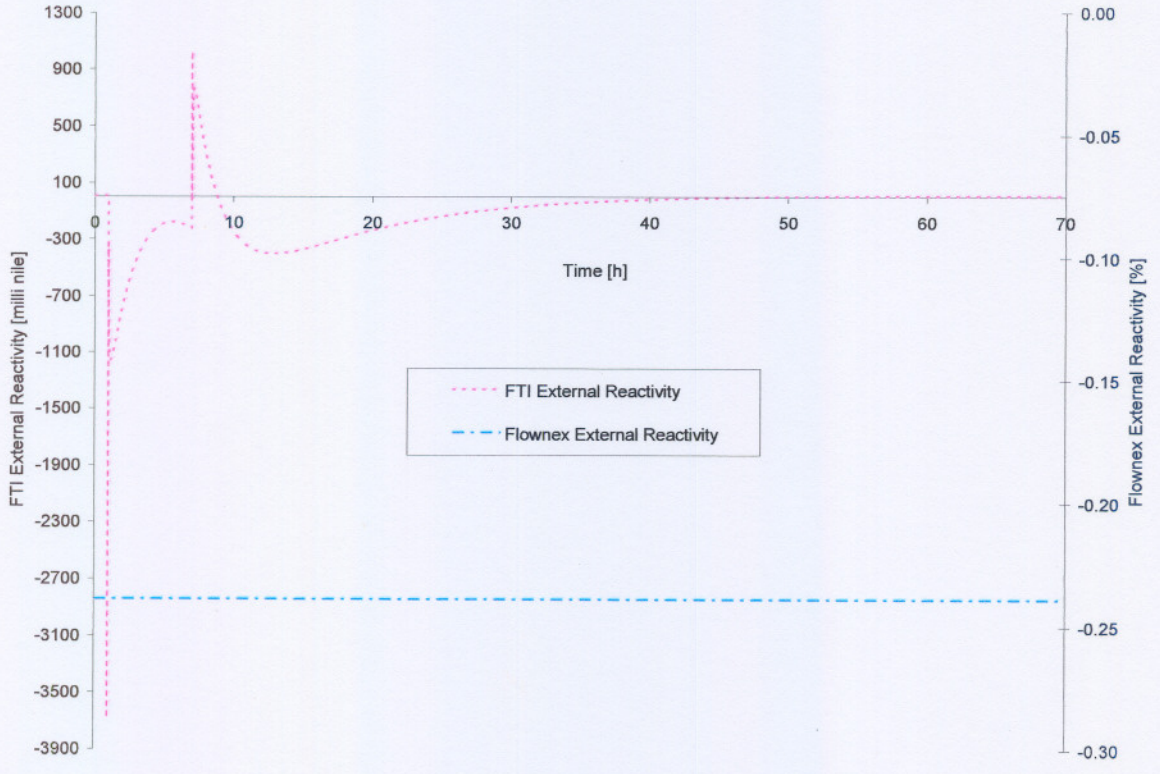


Figure 7.7: *Load Follow – External reactivity*

7.3.4 Discussion

Again, as was seen from the steady state calculations, the temperatures and pressures of V502 and TINTE PBMR 268 are very close together. Figure 7.1: shows the reactor power and the heat transferred to the coolant. From this figure, it is evident that the changes in heat transfer occur faster in the Flownex simulation than the FTI simulation. This is due to a larger heat capacity of the TINTE model. In TINTE, the side reflector is also modelled and this adds to the thermal capacity of the reactor. Temperature changes will therefore be slower than in the Flownex Pebble Bed Reactor Element where the heat capacity is only accounted for by the fuel and moderator spheres. This can be seen in Figure 7.2: where the maximum difference in outlet temperature is 47°C. The same steady state is reached after both power transitions.

Figure 7.3: indicates that the average fuel temperatures of both models are very similar, but the maximum fuel temperature of FTI is 70°C higher than the Flownex simulation after the first power transient. This can once again be accounted to the radial mesh dependence of the Pebble Bed Reactor Element of Flownex.

The pressure and mass flow response as seen in Figure 7.4: and Figure 7.5: follow the same trends and are both very close together. A maximum difference in outlet pressure of 0.06 bar is found during the first power transient. The maximum difference in mass flow is 1.18 kg/s, but diminishes to about 0.59kg/s in the steady state condition after the first power transient.

Figure 7.6: indicates that Flownex did not adjust the xenon concentrations and Figure 7.7: shows that the external reactivity needed for a power transition stayed constant. No nuclear computations are therefore performed when a power transient is implemented by adjustment of the 'Total Power' variable. Attention should be given not to adjust any nuclear parameters after a Flownex power transition was initiated in this manner.

7.4 Test case 3 – Slow Total Control Rod Withdrawal

7.4.1 Introduction

In this postulated accident, all the reactor control rods are slowly withdrawn from the core. This will have the effect of adding positive reactivity to the core, which will result in an increase in fission power and therefore a rise in temperatures. A sudden withdrawal of the rods will generally result in a faster increase of fuel temperature as well as a higher fuel temperature, but the initial power excursion will be shorter due to the fast temperature increase. After a long time the final conditions for both fast and slow control rod withdrawal will be the same.

7.4.2 Implementation

The control rods were withdrawn with a constant velocity over a time span of 150 seconds. In Flownex, this was achieved by reducing the parameter 'Control Fraction' from its steady state value to zero over the indicated time. The so-called ROMO module was used in the TINTE reactor model to achieve rod movement. The transients were simulated over 100,000 seconds (27 hours) to allow the time dependant fission power to stabilize. A maximum time step of 1s was used up to 600s in both simulations. Thereafter the maximum TINTE time step was set to 50s and the maximum Flownex time step to 2.5s.

The modified stand-alone Flownex model was simulated and compared to the FTI results. Large differences were observed in the neutronic behaviour of the two models. The maximum total power of Flownex was 419% compared to 150% for FTI. Further investigation showed that the point kinetic parameters in V502 did not coincide with the TINTE model. New control rod reactivities were calculated from the TINTE model and updated in V502. The ^{135}Xe and ^{135}I decay constants were changed from $3.053*10^{-5}$ s^{-1} and $4.202*10^{-5}$ s^{-1} to $2.09*10^{-5}$ s^{-1} and $2.87*10^{-5}$ s^{-1} respectively. These values are given by Lamarsh and Baratta (2001:378). The product of the average neutron flux (ϕ_{ave}) and absorption cross section (σ_a^{Xe}) of ^{135}Xe were recalculated as well. The ϕ_{ave} was derived from the equilibrium TINTE model as $7.717*10^{13}$ neutrons/(cm^2.s) and σ_a^{Xe} was taken as $2.65*10^6$ barns. It must be remembered however, that ^{135}Xe is a non-1/v absorber and σ_a^{Xe} is therefore very dependant on temperature (Lamarsh & Baratta, 2001:75).

7.4.3 *Results*

The following graphs were plotted against a logarithmic time scale to show the transient during the rod withdrawal as well as the long-term effects. The FTI values were taken from the TINTE *ptr* result file. Some important results are the following. Figure 7.8: shows that different maximum power peaks are still observed in the two models: 150% for FTI and 137% for Flownex. The ^{135}Xe concentration graphs are shown in Figure 7.9:. These results are normalized to steady state concentrations where a value of one would indicate the steady state concentration at the rated reactor power.

The FTI simulation lasted just under three hours and Flownex finished the simulation in fifty minutes.

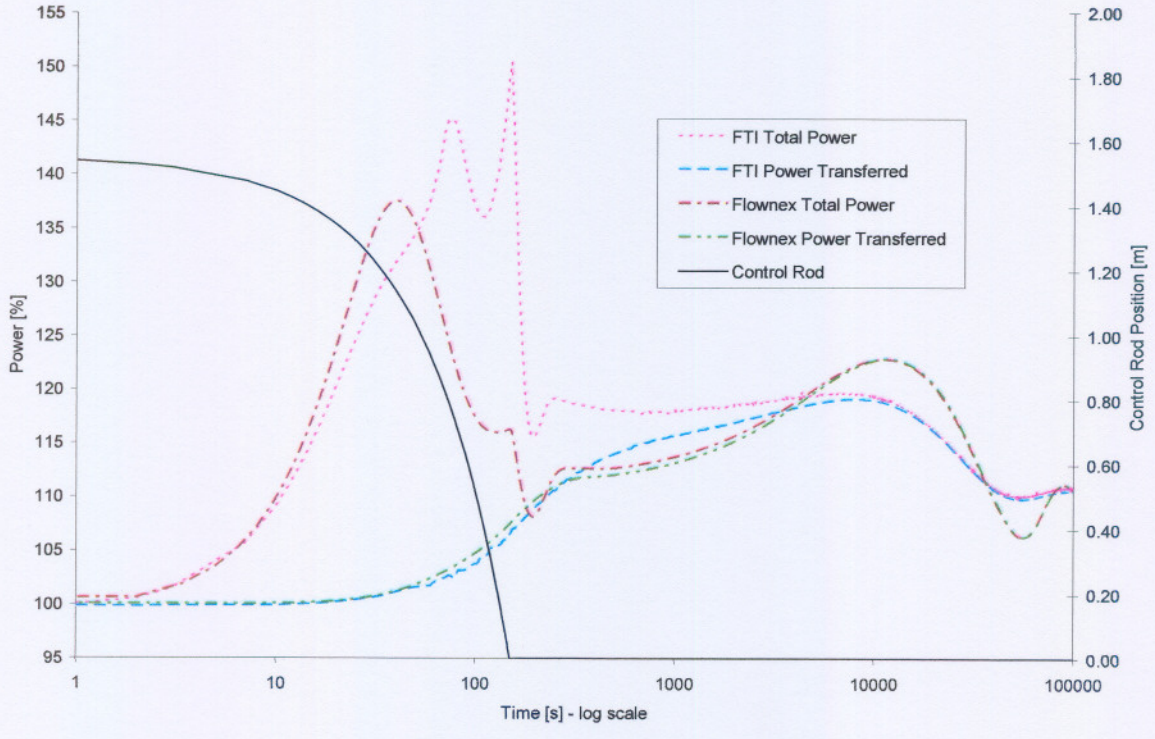


Figure 7.8: *Slow TCRW – Reactor power*

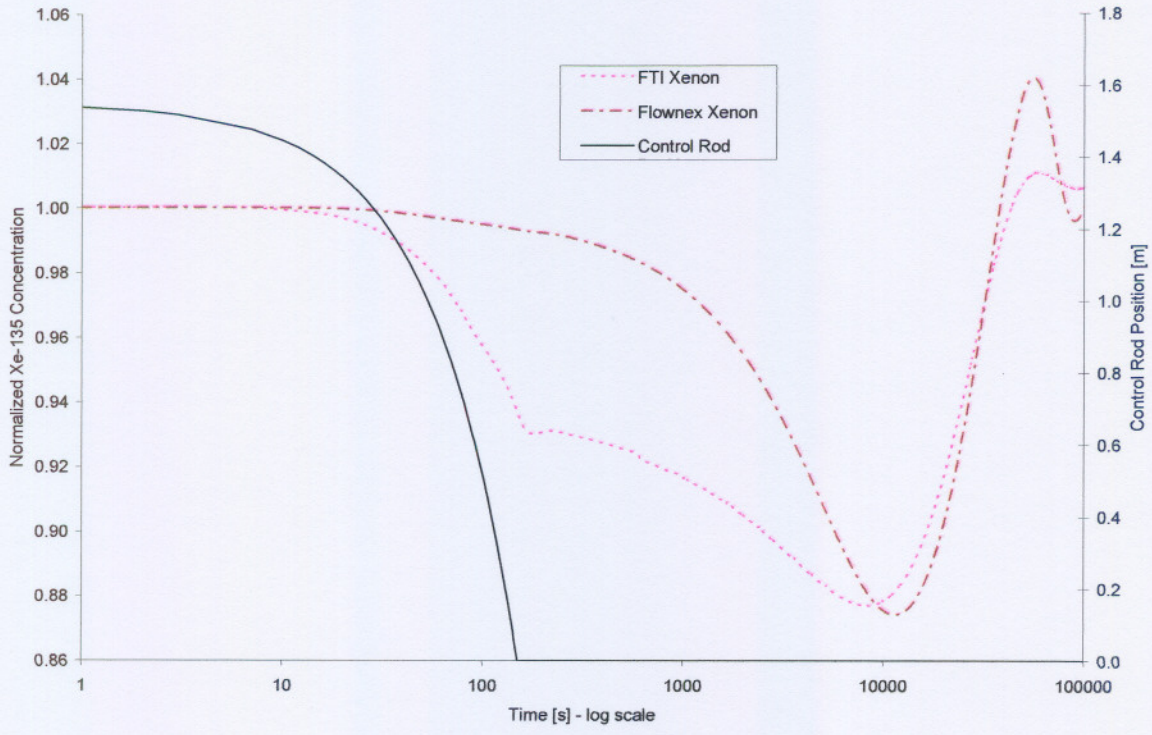


Figure 7.9: *Slow TCRW – ^{135}Xe Concentrations*

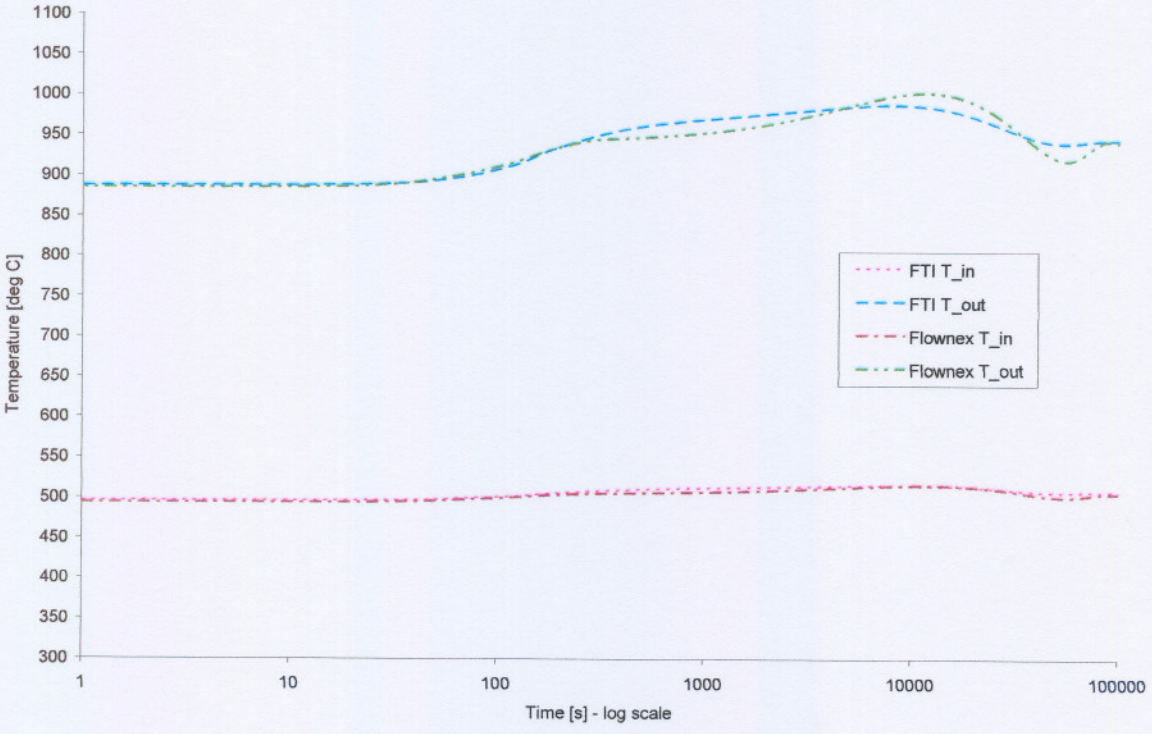


Figure 7.10: *Slow TCRW – Helium Temperatures*

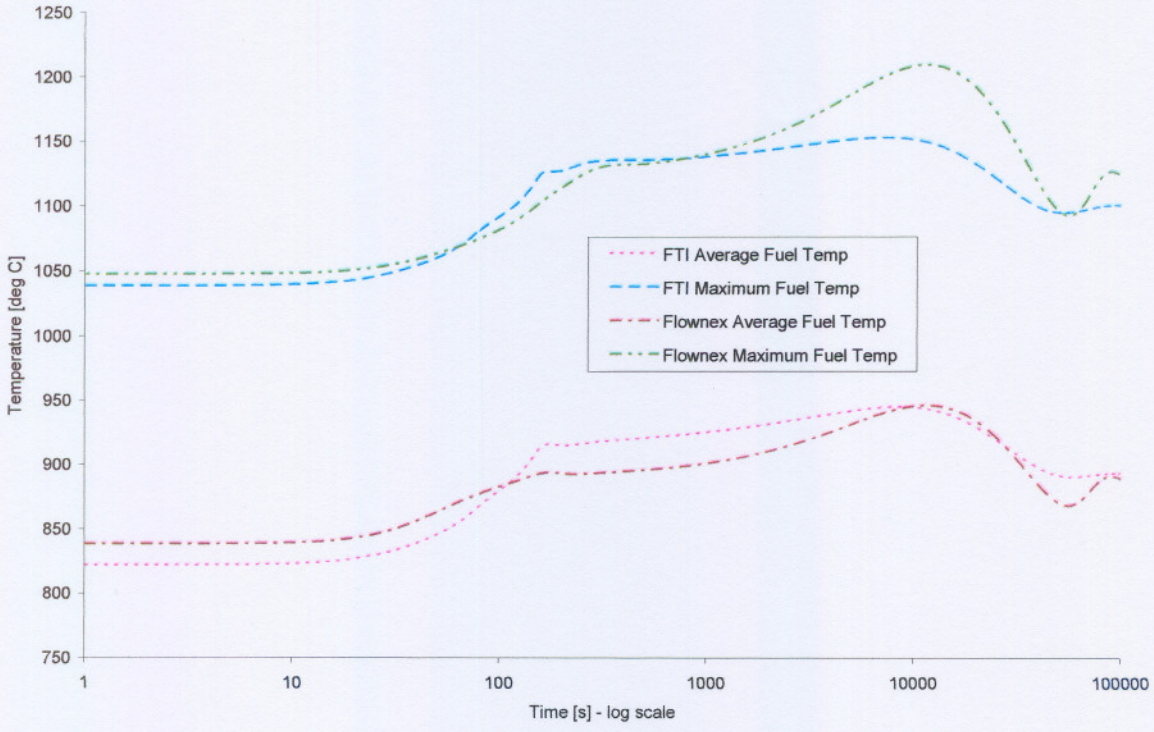


Figure 7.11: *Slow TCRW – Fuel Temperatures*

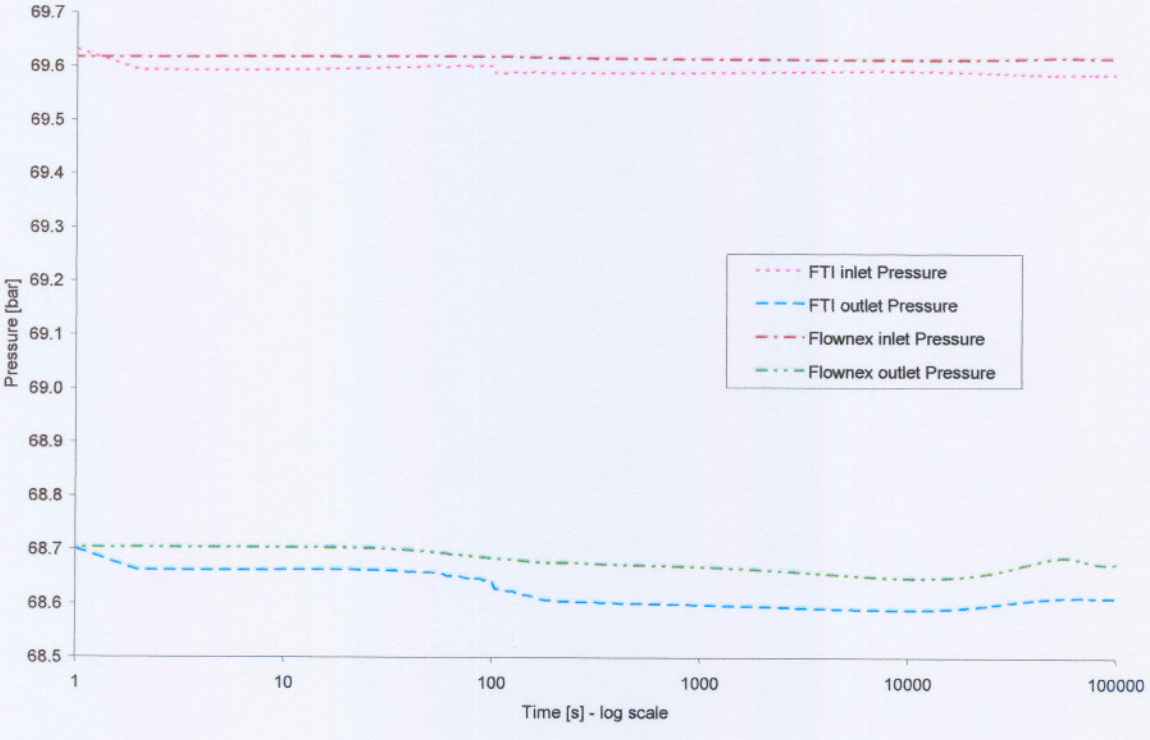


Figure 7.12: *Slow TCRW – Pressures*

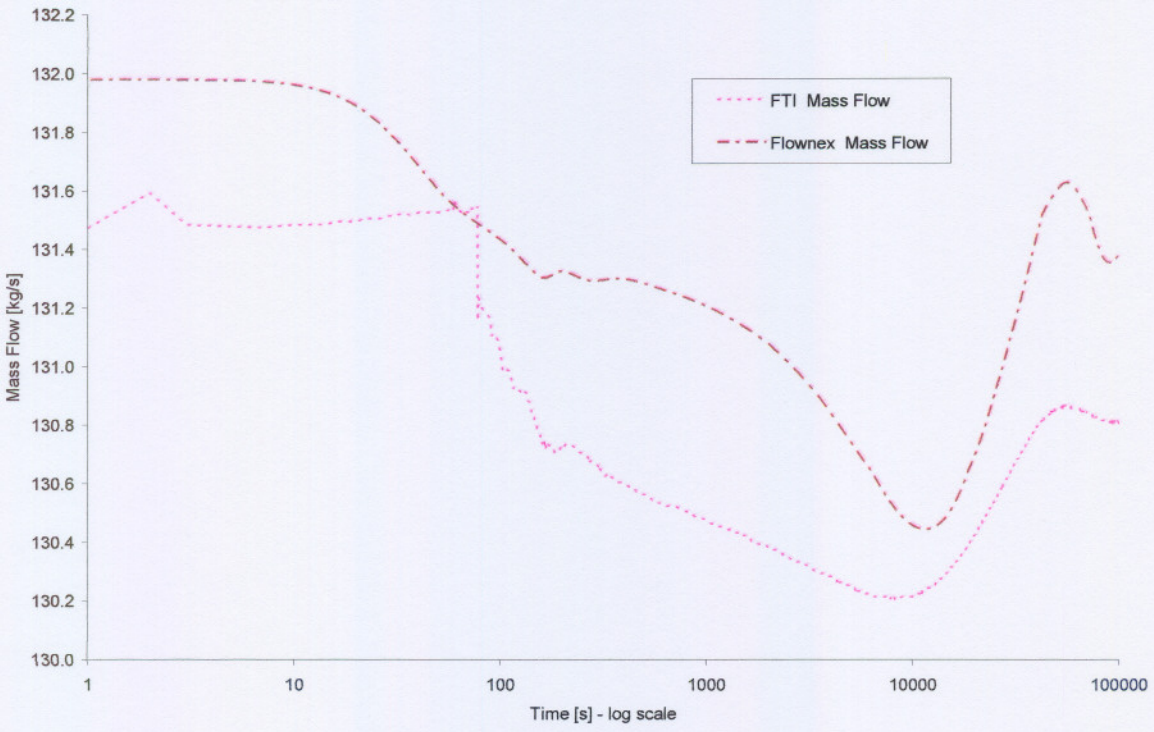


Figure 7.13: *Slow TCRW – Mass Flow*

7.4.4 Discussion

It was found that the point kinetic model is very sensitive to the product of σ_a^{Xe} and ϕ_{ave}. These values must therefore be chosen very carefully.

As the control rod is removed, the addition of external reactivity increases the fission power, which causes an increase in fuel temperature. At 80s (in the FTI simulation), the negative temperature feedback causes a decrease in fission power. This continues until the increase in fuel temperature stops at about 10000s. At 111s, the positive reactivity insertion, which is added by the withdrawal of the control rods, exceeds the negative temperature feedback and this result in the second increase in fission power. Once the rods have stopped, the negative feedback is stronger and therefore the power rapidly decreases. It stays above 100% because reactivity was inserted and that has to be compensated by negative temperature feedback, i.e. by higher temperatures. To obtain and maintain the higher temperatures more power is necessary. By xenon burning at higher power more reactivity is induced, leading to slowly increasing temperatures later and for many hours.

The absence of a second increase of power in the Flownex simulation indicates that the fuel, moderator and Xenon feedback parameters of V502 are not equivalent to those that would be obtained through the TINTE PBMR268 model. This is also evident in the decrease of ^{135}Xe concentration in the different models.

The maximum difference in reactor coolant temperatures is 20°C, as can be seen in Figure 7.10:. This is because of a large difference in maximum fuel temperature at 13200s of 60°C. This can once again be accounted to the radial mesh dependence of the Flownex Pebble Bed Reactor Element.

The pressure response of the two models as shown in Figure 7.12: is very similar and the maximum difference of the reactor outlet pressure is 0.07 bar.

Figure 7.13: shows that there is a sharp decrease in mass flow at 78s in the FTI model. This is due to the coupling method and can be corrected by choosing a smaller time step. Care must therefore be taken when very accurate simulations are necessary. The maximum difference in mass flow is 0.74 kg/s, which is still very accurate.

As the control rods are removed or inserted into the core, the power profile is changes from the steady state calculation. It would therefore be beneficial to update the power profile of Flownex with control rod movement, but in the current version, this is not possible.

7.5 Summary

After some changes were made to the Flownex V502 model, the steady state results showed that the thermal hydraulic specifications of the two reactor models were very much alike. It was found that the CRCC are modelled differently in the two codes. TINTE approximates the channels as an annular skirt around the core whilst Flownex models it as a 1D pipe. This difference induces diverse mass flows through the pebble bed and results in a variation in pressure drop. The mass flow through the annular skirt in TINTE was adjusted to coincide with the mass flow found in the V502 model.

The larger heat capacity of the TINTE model became evident in the load follow transient. This is mainly due to the simulation of the side reflector in TINTE, where it is assumed

adiabatic in the Pebble Bed Reactor Element. The load follow transient indicated that Flownex does not perform any neutronic calculations when a power transient is initiated by changing the 'Total Power' parameter. The ^{135}Xe concentration is not updated and the external reactivity needed for the power transient remains unchanged. Any subsequent neutronic transients would therefore be inaccurate.

The simulations further showed that the point kinetic parameters as used in V502 do not model the behaviour of the TINTE 268 MW precisely. The ^{135}Xe and ^{135}I decay constants were changed as well as the external reactivity of the control rods. The product of σ_a^{Xe} and ϕ_{ave} were calculated from the steady state TINTE model and updated in V502 as well.

Using the TINTE code to model the neutronic and thermal hydraulic behaviour of the core increases the computational time of Flownex dramatically, especially when neutronic transients, such as control rod movements, are simulated.

8 CONCLUSION AND RECOMMENDATION FOR FURTHER WORK

In Chapter 2 different neutronic and thermal hydraulic software packages were investigated. It was seen that a number of these codes have already been coupled to create more complete nuclear reactor design and analysis tools. Chapter 3 and 4 describes two commonly used packages used by the PBMR company in the design of the new PBMR helium cooled, graphite moderated nuclear reactor. Details of the Flownex V502 PBMR 268 MW plant as well as the TINTE PBMR 268 MW reactor model were given.

It was decided to couple Flownex and TINTE to provide accurate boundary conditions for the TINTE reactor model during transient simulations. Comparison between the two core models in transient conditions can then be used for validation purposes concerning the Flownex point kinetic model.

8.1 Coupling method and code design

Two coupling methods were explored in Chapter 5. It was shown that, although conceptually simple, a direct coupling algorithm would be unfavourable. An indirect coupling method has been proposed where the algorithm matches the characteristic values of the PCU and nuclear core after every time step. For this method to work, the reactor in Flownex has to be replaced by another thermal flow element. A pipe with roughly the same dimensions as the reactor was chosen. Properties such as the friction factor and heat transferred to the pipe are then updated after each time step.

The indirect coupling method was implemented in a program called FTI, which manages data exchange between Flownex and TINTE. FTI also synchronizes the two independent programs. Some code changes had to be made to the TINTE I/O modules to facilitate data exchange and enable synchronization. FTI is operated through a user-friendly interface where the relevant Flownex and TINTE project files are chosen.

The validation studies in of Chapter 6 showed that the indirect coupling method introduces errors in the temperature, pressure and mass flow data exchange. These errors are insignificant and typical deviation is in the order of $\pm 0.5\%$ of full range values during the steady state. These errors increase during fast thermal hydraulic transients. Decreasing the maximum time step of TINTE reduces this problem, but increases computational time.

8.2 Point kinetic validation

FTI was used to couple the TINTE reactor model to the Flownex PCU. The simulation studies in Chapter 7 showed a number of interesting results. The steady state simulation indicated that the thermal hydraulic specifications of the TINTE PBMR 268 MW reactor model and the Flownex V502 PBMR reactor model differ. This is mainly because the TINTE model simulates the CRCC as a skirt around the core, while Flownex simulates it as a 1D pipe. The larger bypass flow in TINTE results in a slighter pressure drop over the reactor. Changes to V502 and the TINTE model corrected these steady state issues.

The simulations showed large differences in the neutronic behaviour of the different codes. It was concluded that the point kinetic parameters as used in V502 does not model the TINTE PBMR 268 very accurately. Some of point kinetic parameters were updated and results that are more realistic were obtained. The point kinetic model of Flownex also does not implement reactivity feedback or ^{135}Xe concentration calculations when explicit power transitions are simulated.

8.3 Recommendations and future work

The CBCS coupling in FTI must be completed to present results that are more accurate during total plant simulation. The implementation of the *Flownex transient control file* must also be completed.

It is suggested to redo all the simulations using the current PBMR-400 Flownex and TINTE models. The PBMR-400 is simulated with the *Advanced Pebble Bed Reactor Element* in Flownex and although the thermal hydraulic model is more sophisticated than the *Pebble Bed Reactor Element,* it uses the same neutronic model. Furthermore, thought must be given to include reactivity and ^{135}Xe feedback when performing explicit transient power simulations

in Flownex. It would also be beneficial to incorporate power profile changes during control rod movement.

The errors introduced by the indirect coupling method limits the use of FTI. It is not recommended to use FTI in very detailed reactor analysis problems as inaccuracies may occur. It would therefore be beneficial to perform a low-level integration of TINTE into Flownex. This would eliminate the need for a coupling method and all the governing equations can be solved simultaneously.

The interfacing or integration of a fuel cycle code into Flownex such as the 3D diffusion equation solver VSOP could also benefit total plant simulation and validation of point kinetic parameters.

8.4 Conclusion

The results of this study conclude that the indirect coupling method can provide rough boundary conditions during transients when interfacing TINTE and Flownex. These are adequate to perform validation studies on the point kinetic behaviour during transient conditions. However, further investigation should be done into the validation of all the point kinetic parameters, especially during cold shutdown and non-equilibrium fuel situations such as new core loadings.

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