DYN3D - THREE-DIMENSIONAL CORE MODEL FOR STEADY-STATE AND TRANSIENT ANALYSIS OF THERMAL REACTORS

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ABSTRACT

1. **Program Name and Title:** DYN3D

2. **Computer for Which Program is Designed and Other Machine Versions Available:** SUN-ULTRA1 (original computer), Pentium PC

3. **Problem Solved:** DYN3D is a three-dimensional computer code for calculating transients in light-water reactor cores with quadratic (version DYN3D/R) or hexagonal (version DYN3D/H) fuel assembly geometry. Starting from the critical state (k_{eff} - value, critical boron concentration or critical power) the code allows to simulate the neutronic and thermal-hydraulic core response to reactivity changes caused by control rod movements and/or changes of the coolant core inlet conditions. Burn-up calculations can be performed. The depletion state can used as starting point for the transient. The steady state and transient Xe and Sm concentrations can be analyzed. The decay heat is taken into account.

4. **Method of Solution:** The neutron kinetic model is based on the solution of the three-dimensional two-group neutron diffusion equation by nodal expansion methods. Different methods are used for quadratic and hexagonal fuel assembly geometry. It is assumed that the macroscopic cross sections are spatially constant in a node being the part of a fuel assembly. In the case of Cartesian geometry, the three-dimensional diffusion equation of each node is transformed into one-dimensional equations in each direction x,y,z by transversal integrations. The equations are coupled by the transversal leakage term. The 1-dimensional equations are solved with the help of flux expansions in polynomials up to 2nd order and exponential functions being the solutions of the homogeneous equation. The fission source in the fast group and the scattering source in the thermal group as well as the leakage terms are approximated by the polynomials. In the case of hexagonal fuel assemblies, the stationary diffusion equation in the node is solved by factorizing the space dependency of neutron fluxes in the radial plane and the

axial direction. A 2-dimensional diffusion equation in the radial plane and a 1-dimensional equation in axial direction are obtained. The two equations are coupled by the transversal bucklings. In the hexagonal plane the fluxes are expanded by using Bessel functions being the solutions of the Helmholtz equation. The 1-dimensional equation in axial direction is solved by a polynomial expansion up to the fourth order. The outgoing partial currents in axial direction are given by the averaged fluxes, incoming partial currents and higher order coefficients. In the new version of DYN3D this technique is replaced by the HEXNEM method, applying transverse integrations over the z-direction or the hexagonal plane. The one-dimensional diffusion equation over the axial direction z and the 2-dimensional equation in the hexagonal plane are solved by similar expansions as in the Cartesian geometry. Considering the 2-dimensional problems in the hexagonal planes, not only the side averaged values of flux and current but also the corner point values can be used for the approximate solution of the diffusion equation which leads to a higher accuracy in the case of larger fuel assemblies loaded in the Russian type reactor VVER-1000. In the steady state, an eigenvalue iteration is performed by using an inner and outer iteration strategy. The outer iteration (fission source iteration) is accelerated by Chebychev extrapolation.

The steady-state iteration technique is applied for the calculation of the initial critical state, the depletion calculations and the Xe and Sm dynamics.

A two time step scheme consisting of thermal-hydraulic and neutron kinetic time steps is used for the transient integration. One or several neutron kinetic steps are used within a thermal-hydraulic step. Concerning the time integration over the neutronic time step an implicite difference scheme with exponential transformation is used. The exponents in each node are calculated from the previous time step or during the iteration process. The precursor equations are analytically solved, assuming the fission rate behaves exponentially over the time step.

A thermal-hydraulic model of the reactor core and a fuel rod model are implemented in the module FLOCAL being a part of DYN3D. The reactor core is modelled by parallel cooling channels which can describe one or more fuel elements. The parallel channels are coupled hydraulically by the condition of equal pressure drop over all core channels. Additionally, so-called hot channels can be considered for the investigation of hot spots and uncertainties in power density, coolant temperature or mass flow rate. Thermohydraulic boundary conditions for the core like coolant inlet temperature, pressure, coolant mass flow rate or pressure drop must be given as input for DYN3D. The module FLOCAL comprises a one- or two-phase coolant flow model on the basis of four differential balance equations for mass, energy and momentum of the two-phase mixture and the mass balance for the vapour phase allowing the description of thermodynamic non-equilibrium between the phases, a heat transfer regime map from one-phase liquid up to post-critical heat transfer regimes and superheated steam, a fuel rod model for the calculation of fuel and cladding temperatures and the determination of some parameters for fuel rod failure estimation.

The two-phase flow model is closed by constitutive laws for heat mass and momentum transfer, e.g. vapour generation at the heated walls, condensation in the subcooled liquid, phase slip ratio, pressure drop at single flow resistance's and due to friction along the flow channels as well as heat transfer correlations. Different packages of water and steam thermophysical properties presentation can be used.

A model for description of the mixing of coolant from different primary loops in the downcomer and lower plenum of VVER-440 type reactors is implemeted. It is based on the special feature of VVER-440 type reactors, that the coolant flow in the downcomer is nearly parallel without large re-circulation vortexes as they are known from Western type PWRs. Thus the flow can be well described in the potential flow approximation, where the Navier-Stokes equations can be solved analytically for the 2D flow in the downcomer. The velocity gradient in the radial direction was neglected. In the lower control rod chamber a parallel flow with constant velocity was assumed. With this approximation of the velocity field the diffusion equation for the temperature is solved. The solution is presented as a closed analytical expression based on series of orthogonal eigen-functions. The turbulence was taken into account by constant scalar turbulent Peclet numbers defined individually for the downcomer and the lower control rod chamber. The turbulent Peclet numbers describe the intensity of turbulent diffusion and were adopted to experimental data. For the validation of the model, measured values from an air operated 1:5 scaled VVER-440 model were used. Temperature measurements were taken at the end of the downcomer and at the inlet of the reactor core. Further, the model was validated against measured operational data from NPP with VVER-440 and CFD calculations.

Iterations between neutron kinetics and thermal hydraulics are carried out in the steady state as well as for each thermal-hydraulic time step. Based on the changes of the main physical parameters of the transient process the time step size is controlled during the calculation.

The decay heat model integrated in the code is based on 4 fissionable isotopes.

5. **Restrictions on the complexity of the Problem:** The number of the prompt neutron energy groups is restricted to two groups. The other parameters as the number of fuel elements, the number of axial layers, the number of cross section sets etc. can be enlarged in correspondence to the problem in two block data program units. There are only restrictions by the memory of the computers.

6. **Typical Running Time:** The running time is strongly influenced by the analysed problem. The CPU - time varies between minutes to hours on a SUN ULTRA 1. It depends on the time of transient, the size of the problem, the changes of the parameters during the transient, the given iteration limits, and time step restrictions.

7. **<u>Related and Auxiliary Programs</u>**: The program OG3.3 can be used for preparing the wanted output (time tables of single parameters or radial and axial distributions at given time points) from the binary output file of DYN3D.

The graphical postprocessor RS is available on SUN workstations for graphical imagines of the most important parameters during the transient or at the end of calculation.

The core model DYN3D is coupled with the plant model ATHLET, developed by the Gesellschaft fuer Anlagen- und Reaktorsicherheit (GRS) to analyse more complex transients where the coolant flow conditions are influenced by the core behaviour and vice versa. Two different ways of coupling DYN3D/ATHLET were realized. The first one uses only the neutron kinetic part of DYN3D and integrates it into the heat transfer and heat conduction model of ATHLET. This is a very close coupling, the data have to be exchanged between all core nodes of the single models (internal coupling). For this reason a great number of data have to be transferred.

In the second way of coupling the whole core is cut out of the ATHLET plant model (external coupling). The core is completely modelled by DYN3D. The thermal-hydraulics is split into two parts: the thermal-hydraulic model of DYN3D describes the thermal hydraulics and the fuel rods of the core, the ATHLET code models the rest of the coolant system. As a consequence of this local cut it is easy to define the interfaces. They are located at the bottom and at the top of the core. The pressures, mass flow rates, enthalpies and concentrations of boron acid at these interfaces have to be transferred. So the external coupling needs only a few parameters to be exchanged between the codes. It is effectively supported by the general control simulation module (GCSM) of the ATHLET code. For this reason almost no changes of the single programs are necessary and the two codes can be developed independently.

Depending on the application each of the two versions of coupling has its advantages and disadvantages:

Internal coupling:

- solution of the thermohydraulic equation system in the ATHLET code
- description of reverse flow is possible
- mixture levels in the core can be described
- longer CPU times by using a larger number of coolant channels in the core

External coupling:

- whole core simulation with a large number of coolant channels possible
- integration of mixing models
- more detailed fuel rod model of DYN3D available
- no reverse flow in the core
- no mixture level in the core

8. **Status:** The original computer is a SUN ULTRA1 workstation, but the code is running also at other machines by minor changes. Benchmarks and experiments were calculated for the verification and validation of the code. The hexagonal version of DYN3D is used in some countries with operating VVER-reactors for safety assessments. The problems calculated for verification and validation are listed below.

Steady State Problems:

- 3D IAEA Benchmark (Cartesian)
- 2D IAEA benchmarks modified for hexagonal geometry
- 2D and 3D Seidel benchmark for VVER-440 (hexagonal)
- 3D Schulz benchmark for VVER-1000 (hexagonal)
- AER (Atomic Energy Research) benchmark problem concerning control rod worth of VVER-440 reactor (hexagonal)
- Comparison of calcualted reactor parameters (critical boron concentration, reactivity coefficients, control rod efficiency) with measured data for different VVER-reactors

Transient Problems

- Kinetic experiments at the zero power reactor LR-0 (hexagonal)
- 1st – 3rd kinetic benchmarks of AER (Atomic Energy Research) of ejection of asymmetrical control rods in a VVER-440 (hexagonal)
- 4th kinetic benchmark of AER (Atomic Energy Research) of a boron dilution transient in a VVER-440 (hexagonal)
- NEACRP benchmarks on control rod ejections in PWR (Cartesian)
- NEA-NSC benchmarks on uncontrolled withdrawal of control rods at hot zero power in PWR (Cartesian)
- OECD-main steam line break benchmark – exercise 2 (Cartesian)
- OECD-main steam line break benchmark – exercise 3 analysed with DYN3D/ATHLET (Cartesian)
- 5th kinetic benchmarks of AER on main steam header break in a VVER-440 with DYN3D/ATHLET (hexagonal)
- Comparison of DYN3D/ATHLET results with measured data for operational transients in nuclear power plants with VVER type reactors.

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11. **Hardware Requirements:** The codes runs at Unix workstations and PC. A large problem with 10 000 nodes requires about 30 MByte memory.

12. **Programming Language(s):** FORTRAN-77 with elements of FORTRAN-IV

13. **Operating System:** Solaris, Windows NT, Windows95/98

14. **Other Programming or Operating Information or Restrictions:**

15. **Name and Affiliation of Author or Contributor:**

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16. **Material Available:**

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17. **Category:** F, C
Keywords: LWR reactors, fuel rods, heat transfer, hexagonal-z, Cartesian geometry, neutron diffusion equation, flux, power distribution, reactivity, reactor cores, reactor kinetics, reactor safety, thermodynamics, three-dimensional, steady state conditions, transients, two-group theory, two-phase flow

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