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The Code DYN3D/M2 for the Calculation of Reactivity Initiated Transients in Light Water Reactors with Hexagonal Fuel Elements

- Code Manual and Input Data Description -

Herausgeber:
U. Grundmann und U. Rohde

Contents

1. Introduction

The code DYN3D/M2 is used for investigations of reactivity transients in cores of thermal power reactors with hexagonal fuel elements. The 3-dimensional neutron kinetics model HEXDYN3D of the code is based on a nodal expansion method for solving the two--group neutron diffusion equation. The thermo-hydraulic part FLOCAL consists of a two-phase flow model describing coolant behaviour and a fuel rod model. The fuel elements are simulated by separate coolant channels. Additional, some hot channels with power peaking factors belonging to chosen fuel elements can be considered. Several safety parameters as temperatures, DNBR and fuel enthalpy are evaluated. Macroscopic cross sections depending from the thermo-hydraulic parameters and boron concentration are input data of the code. The stationary state and transient behaviour can be analyzed.

The model and the basic equations of the code are described in Ref. /1/. The codes and its precursor codes HEXNOD23, HEXDYN3D, FLOCAL and DYN3D/M1 are validated by comparison with benchmarks, other codes and experiments /1,2,3,4,5/. Some analyses of reactivity accidents in the reactor VVER-440 were considered by the help of DYN3D/M2 (for example /6/,/7/).

2. Features of the Code

Analyzing a static state, there exists some possibilities to make the reactor critical:

- Division of multiplication cross sections by K_{eff}

- Variation of boron acid concentration

- Variation of reactor power

If a transient calculation should be carried out, the following perturbations can be treated:

· Movements of single control rods or control rod banks

· Variation of core coolant inlet temperature

· Variation of boron acid concentration

· Changes of core pressure drop or total mass flow rates

· Changes of pressure

If a control rod is partially inserted in a node during the control rod motion, the weight of the control rod is overestimated by using only geometrical weighting of the corresponding cross sections. A neutron flux - volume weigthing can be applied with help of axial distributions above and below of the interface between both materials, if the interface corresponds with a axial node boundary in the static case. The polynomial coefficients of these axial distributions can be transferred to a file after a stationary calculation with the described control rod positions. It is assumed, that the axial distributions depend weakly on the different positions of the lower end of control rod during the movement through the core. For example, a position of control rod in the middle of the core can be chosen for the calculation of the coefficients.

The macroscopic cross sections and the coefficients of their parametrization described in Ref. /1/ can be included in the dataset of neutron kinetics. Using the code PREPAR-EC /8/ for generating the cross section sets on the basis of the MAGRU - library, the data are written on a separate file, which is read by the code. If other sources of neutron physical constants should be used, some changes in the subroutines INPFLO,INCOEF and FEEDBA are necessary. Perhaps it will be more convenient to replace some of these subroutines.

3. Actual Modifications of the Thermo-Hydraulic Part of the Code

The physical model for the code DYN3D/M2 (neutron kinetics and thermal-hydraulics) is described in the report /1/. In this section, actual modifications and improvements of the thermo-hydraulic part FLOCAL of the code, developed in the last time, are outlined.

The actual version of FLOCAL comprises following developments:

- improvements of numerical algorithms for solving the basic equations,
- a new, fast running version of water and steam properties representation,
- an improved and extended model for heat transfer and fuel rod behaviour in the high temperature region, including the estimation of several fuel rod failure parameters,
- the implementation of several mixing models for the lower plenum, allowing the estimation of coolant temperature and boron acid distributions at the core inlet from the loop parameters.

The improvement of numerical algorithms concerns, first of all, the modification of the numerical scheme, allowing the solution of the balance equations under the conditions of partial flow reversal in the core. Partial flow reversal can occur during very fast power transients leading to intensive coolant boiling with a very fast change of coolant density. In this case, an expulsion of coolant from the core in both upward and downward directions can take place, so that the coolant flow velocity changes the sign along the channel. Difficulties arise, first of all, with the hyperbolic energy balance equation, solved by an implicit method of characteristics. In the case of flow reversal, an explicit MOC-scheme is applied. The Courant condition, neccessary for the stability of the scheme

$$r = |w| \frac{\Delta t}{\Delta z} \leq 1$$

(r-Courant factor, w coolant velocity, Δt-time step, Δz-axial mesh size) is fulfilled by the application of a local spatial mesh refinement only in the region and the time interval where flow reversal occurs. This method guarantees numerical stability, the account for specific boundary conditions for flow reversal and requires a minimum of calculational amount.

The fast running version for the determination of water and steam properties is based on the representation of KOKOREV and BOIKO from Ref. /9/. This representation is valid in the pressure range from 5.0 to 18.85 MPa and the temperature interval from 0 to 700 °C. In comparison with the fluid properties representation used in FLOCAL before, the consistency between subcooled or superheated region and the saturation line is improved. The total computation time of the module FLOCAL is reduced appr. by 20 %.

The recent heat transfer and fuel rod behaviour models used in FLOCAL are described in detail in the paper /10/. In that paper, some results of the validation of the model on RIA experiments from literature are given also.

The heat transfer regime map is extended to consider practically stagnant fluid conditions (taking into account natural convection and pool boiling heat transfer correlations). For the estimation of post-crisis heat transfer the GROENEVELD-DELORME model /11/ is implemented. It allows to take into account thermodynamic non--equilibrium in the post-crisis region also (in the previous FLOCAL version non-equilibrium was limited to subcooled boiling in the pre-crisis region).

In the report /1/, the development of a lower plenum mixing model is mentioned. This model is now implemented in the recent FLOCAL version. A mixing model developed by Draeger /12/ for the down-comer and the lower plenum of WWER-440-type reactors is used.

This model is based on a superposition of normalized, dimensionless reference distributions for a perturbation in one of the 6 loops of the WWER-440. The model describes exclusively hydraulic effects, and so it can be applied to the estimation of both coolant temperature and boron acid concentration inlet distributions. Optionally, experimentally defined or calculated by a simplified analytical model reference distributions can be used. Both options are available for the versions W-230 and W-213 of the reactor WWER-440, showing significant differences in the construction of the lower plenum. The version W-213 has an elliptical perforated plate, decreasing the turbulent mixing of the coolant flow. The reference distributions are available on special input data files (see Input Description for FLOCAL). The superposition of the reference distributions is carried out taking into account the transformation to the coordinate system used in DYN3D with consideration of the primary circuit loop positions.

The application of the mixing model developed by Draeger is limited to:

- WWER-440 type reactors (versions W-230 and W-213),
- working regimes with all 6 loops in operation and approximately equal coolant mass flow rates in all loops,
- under steady-state conditions.

That's why simple mixing models with ideal homogenous mixing in the lower plenum are implemented also, actually:

- ideal mixing with constant specific heat of the coolant (temperature based ideal mixing),
- ideal mixing with taking into account the dependence of c_p from pressure and temperature (based on enthalpy balance).

The ideal mixing models can, of course, be used for an arbitrary number of operating loops with arbitrary mass flow rates and for WWER-1000 type reactors also.

In the recent FLOCAL version, the mixing model is used for instationary problems in a quasistatic manner. Delay times for fluid transport and mixing are not taken into account.

Regardless of this limitations and simplifications, by the help of the mixing model one can carry out methodical investigations, in what cases and under what conditions the frequently used assumption of ideal mixing can give conservative results with respect to nuclear safety.

The simplified crossflow model for the core, described in the report /1/, is not yet implemented in the released FLOCAL version, that's why a further development and validation seems to be necessary.

4. Code Structure

In the main program DYN3D/M2 the three working arrays IARR,RARR and FTH are declared, two of type REAL and one of type INTEGER. The needed length of arrays depends on the size of the problem and can be changed in a simple way.

The working array IARR having the length MIAR is of the type INTEGER. Data describing the form of the considered part of the core and distributions of the different materials are stored there. The length MIAR is determined in the case of feedback by

$$\mathrm{MIAR} = 2*\mathrm{NIMAX}*\mathrm{NJMAX}+10*\mathrm{NJMAX}+\mathrm{NIMAX}+4*\mathrm{NZ}*\mathrm{NCAS}+2*\mathrm{NCAS}+2*\mathrm{NOBOU}+ \mathrm{NZ}*\mathrm{NOBOU}+2*\mathrm{NOSYMS}+2*\mathrm{NOTP}+3*\{\sum_{I=1}^{\mathrm{NOTP}} \mathrm{NOPN}(I)\} + 4$$

The REAL-values of neutron kinetics are arranged on the array RARR, the length of which MRAR is calculated for problems including feedback by

$$\mathrm{MRAR} = 61*\mathrm{NZ}*\mathrm{NCAS}+12*\mathrm{NZ}*\mathrm{NNC}+7*\mathrm{NCAS}+3*\mathrm{NOBT}+7*\mathrm{NZ}+64*\mathrm{NOMAS}+2*\mathrm{NODN}+ 2*(\mathrm{NODN}+1)*\mathrm{NOBET}+\mathrm{NODN}*\mathrm{NZ}*\mathrm{NCAS}+2*\mathrm{NOVET}+\mathrm{NOTP}+34*\mathrm{NOW}+ 8*\max_{I} \mathrm{NOPN}(I) \qquad (I=1,\ldots,\mathrm{NOTP})$$

The variables of these relations are described in the input data list of file LUNR, except NNC, which is given by

$$NNC = NCAS+NJMAX+\sum_{J=1}^{NJMAX} \{ \theta(ILEFT(J)-ILEFT(J-1)+1) + \theta(IRIGHT(J-1)-IRIGHT(J)-1)\}+ IRIGHT(NJMAX-1)-ILEFT(NJMAX-1)+2$$

with

$$\begin{array}{lcl} \theta(I) \equiv I & & I > 0 \\ & \text{for} & \\ \theta(I) \equiv 0 & & I \leq 0 \end{array}$$

The working array FTH of type REAL contains the arrays V,W and F, used by the part FLOCAL of DYN3D/M2. During a thermohydraulic time step the old and new values are stored on the arrays W and V respectively. The values on the array V are transferred to the array W at the begin of a new time step. Thermohydraulic results stored on the array F can be transferred to an output file. The length L of the array FTH is determined by

L = (NC+1)*((2*NR+47)*NH+28)

with

NC - number of fuel elements increased by the number of hot channels

NH - number of the axial planes of core

NR - number of radial segments of the fuel

During the run of the code, a temporary file LUNH with capacity S is needed. The capacity S must be greater than

$$S \geq 4*\{(6+4*(NZ+1)+NODN+1)*NZ*NCAS+12*NNC*NZ+100\} \text{ Byte}$$

Fig. 1 shows the scheme of the main code. Besides some of subroutines used for the input of data, the subroutines STATIT and RUNTRA for control of stationary and transient calculation are called.

Crude charts of STATIT and RUNTRA can be seen in FIG. 2 and 3 respectively. The subroutines and their position relative to the main program (level 0) are characterized shortly:

		level
BOIL	- Static calculation of thermo-hydraulic parameters of one mesh point for given enthalpy	3
BESI	- Calculation of modified Bessel functions I_0 and I_1 (without exponent, if asymptotic form is used)	5
BESJ	- Calculation of Bessel functions J_0 and J_1	5
BESSEL	- Calculation of all needed values of Bessel functions using recurrence formula	4
BSETYP	- Input of data for one type of fuel rods	3
CALINC	- Calculation of neutron fluxes for given sources (inner iteration). Entries CALINI, CALTIM and CALTII	3
CARTIN	- Input of core map and the distribution of different sets of macroscopic cross sections	2
CINOUT	- Input and output of group cross section sets	2
COMPST	- Input of strings of the used data sets	2,3

		level
CONTIN	- Input of control variables for the iteration of stationary neutron flux calculation	2
CORREC	- Preparation of input parameters for calculation of matrix elements (transient case)	2
CRIFLU	- Estimation of critical heat flux and the dry-out point	4
CRITCA	- Distributions of precursors of delayed neutrons at beginning of transient calculation	2
DATRAN	- Transfer of data needed for restart of transient calculation to a file	1,2
FEEDBA	- Calculation of macroscopic cross section using the parametrized form. Entries FEEDBR, FEEDBT and FEEDBS	2
FINEST	- Control of feedback iteration to achieve a given value of keff	2
FLOSTA	- Calculation of the thermal hydraulics of the stationary state	2
FLOTIM	- Thermo-hydraulic calculations for one time step	2
FPOT1	- Calculation of the thermo-physical properties of water and steam on the saturation line	≥3
FPOT2	- Thermo-physical properties of water and steam in the subcooled and superheated region are determined	≥3
GAPCON	- Estimation of heat transfer coefficient for the gas gap of the fuel rod (some functions describing the thermophysical and thermochemical properties of fuel and clad are called).	4

		level
GEOIN	- Input of geometrical parameters of the core	2
HETEMP	- Heat transfer conditions are calculated during the transient calculation using cladding surface temperature	3
HETRAN	- The heat transfer conditions are calculated for the steady-state case with known heat flux density	3
HYPEQU	- Complex of subroutines for the solution of a hyperbolic differential equation using a 4-point method of characteristics (MOC) or Lagrange MOC and Lax-Wendroff scheme	3,4
IBOUND	- Input of boundary conditions for the neutron flux calculation	2
INCOEF	- Input of the polynomial coefficients for the macroscopic cross sections	3
INDATA	- Input of thermo-hydraulic data	2
INIBOU	- Preparation of some variables being helpful for the consideration of boundary conditions	2
INISTA	- Determination of initial values for neutron flux calculation	2
INPATH	- The weights of the fuel elements for the considered sector of core are determined	2
INPBET	- Input of data for delayed neutrons	2
INPDIS	- Input of the identifiers of the sets of macroscopic cross sections, delayed neutrons or neutron velocities	3

		level
INPERT	- Input data describing the perturbation of control rod position, thermohydraulics or boron concentration	2
INPSTA	- Input of data for neutron flux calculation (static part)	1
INPFLO	- If feedback is considered, input of some data of the power state of the core	1
INPTRA	- Controlling the input of data needed for the transient calculation	1
INPVE	- Input of neutron velocities for both energy groups	2
IPOL	- Interpolation of transient thermo-hydraulic boundary conditions from a input table	3
ITERST	- Control of outer iterations of the stationary flux calculation	2
KOEFF	- Calculation of the coefficients for the MIRONOV - scheme, used for the solution of the continuity and momentum equations	3
MATCON	- Set of subroutines for the determination of fuel and and cladding thermo-mechanical properties and metal-water reaction rate	≥4
MATDIS	- Simulation of control rod motion in the reactor with feedback	2
MATHEX	- Calculation of the matrix elements of the nodal expansion method in the stationary case. Entry MATTRA for the transient calculation	3,4

		level
MATMOV	- Simulation of the control rod motion in the case without feedback	3
MATRIX	- Calls MATHEX in the static case	3
MATTIM	- Calls entry MATTRA of MATHEX in the transient case	3
NB30, NB60, NB120, NB181, NB182, NB360	- Preparation of data, used by the code for the different symmetry sectors	3
OBOUND	- Output of the used boundary conditions	2
OMEGAN	- Calculating the exponent of exponential transformation at the begin of a neutron kinetic time step (recalculation by subroutine OMEGA during the iteration process)	2
OUTADD	- Output of partial currents at the end of the stationary calculation	2
OUTCTG	- Printing the map of the sector and the identifiers of cross section set for each node	2
OUTPO1	- Output of flux distribution, power peaking factors and power distribution	2
POLYCO	- Output of flux weighting coefficients used for mixing group parameters in the case of partially insert control rods in a node	2

		level
POWCAL	- Calculation of static Xe - equilibrium distribution and power density using neutron fluxes. Entries POWCAT, POWCAR	2
PRECUR	- Calculation of precursor densities and estimation of the next time step	2
RANDBT	- Axial boundary conditions are taken into account during iteration process. Entry RANDAX	3
RASEC	- The same function as RANDBT for adial boundary conditions. Entry RASECI	3
RESULT	- Output of results	2
RODCAL	- Solution of the transient heat conduction equation for the fuel rod	3
RODSTA	- Solution of the static heat conduction equation for fuel rod	3
RUNTRA	- Control of transient calculations (Fig. 3)	1
STATIC	- Control of static calculations (Fig. 2)	1
STODAT	- Calculation of physical properties of water and steam on the saturation line with given pressure (calling FPOT1 for several parameters)	3
TEBEIN	- Calculation of inlet temperature distribution using mixing model of Dräger (calling TKBKIN or TKBKST)	4
TEHILF	- SET of auxiliary subroutines SUPOS6, RETRAN, TRANTS, DYNFMT, TMPHEX and BORHEX called in TEBEIN	5

		level
THERMO -	Transient calculation of thermohydraulic parameters of a mesh point at given enthalpy	3
TKBKIN -	Calculation of inlet temperature distribution of coolant temperature and boron acid concentration in the stationary case (called by FLOSTA)	3
TKBKST -	The same function as TKBKIN for the transient case (Called by FLOTIM)	3
TSCHEB -	Chebyshev acceleration for outer iterations of neutron flux calculation. Entry EXTPOL	2,3
WASTEA -	Containing several subroutines needed in FPOT1 and FPOT2	4

5. Description of Input Data

The input data are transferred by the code from the source files LUNR (neutron kinetic data) and LUNTHE (thermohydraulic data). The neutron physical constants can be included in the file LUNR or contained on a separate file LUNR1, which is generated by the code PREPAR-EC using the MAGRU-library of WWER data /8/. The input data on the file with number 5 are used for control of code run and determine the used files.

The data for continuation of the calculation are saved at given steps of problem-time on a file. After a stop of the transient calculation at a given CPU-time consumed by the run or a abnormal end of job, the calculation can be continued at one of timepoints, for which the data of calculation were saved. New values of data controlling the transient calculation are given on file 5 as input.

Data file 5

Record	Format	Cols.	Item	Comment
1	4I5, 2X, F9.1, 4X, F6.1	1-5	ICON	ICON < 0 Static calculation only ICON = 0 Static and transient calculation ICON = 1 Transient calculation beginning at a stationary state ICON = 2 Continuation of the transient calculation after the end of the previous calculation (only at IBM computers) ICON = 3 Continuation of the transient calculation at any time point, at which the needed data were stored
		6-10	LUNB	File-No. for saving the data of static calculation, if ICON=0. File No., from which the data needed for transient calculation or restart are transferred, if ICON>0.
		11-15	LUNE	The restart-data after time steps (problem time) and at the end of computer time are saved on the file with No. LUNE.
		16-25	LUNH	File-No. of a temporary file
		23-31	TOTIM	Computer time for stopping the transient calculation
		36-41	DTREST	Time steps (problem time), after which the data needed for restart are transferred to the File LUNE.

Record	Format	Cols.	Item	Comment
2	I5	1-5	ISTART	In the case ICON ≥ 2, the calculation will be continued by using the ISTART data set of RESTART file. Input in the case ICON ≥ 2 only.
3	4I5	1-5	LUNR	File-No. of the data set for neutron calculation (static and transient). Input only, if ICON < 2.
		6-10	LUNR1	If neutron physical group constants are generated by the code PREPAR-EC, LUNR1 is the No. of this file. If all data are included in the file LUNR, LUNR1 must equal LUNR. Input only, if ICON < 2.
		11-15	LUNTHE	File No. of the thermo-hydraulic data. Input only in the case ICON ≤ 0
		16-20	LUNPOL	File-No. for polynomial coeff. (used for axial weighting of group constants). Input only, if ICON < 2.
4	I5	1-5	LUNRES	Results are transferred to file with No. LUNRES all thermo-hydraulic time steps with output of detailed or reduced results on the file with No. 6 (for example printer). Input only for calculations with feedback. The data on the file LUNRES can be used for graphic drawing by help of small codes developed by the user.

If the transient calculation will be continued (ICON > 1), the control parameters (Records 55 - 64 of file LUNR) are read here. The data can obtain other values.

If the transient calculation starts from the results of the previous static calculation, the records of file LUNR are read only after record 52.

If ICON ≥ 2 the file LUNR (and also LUNR1) and LUNTHE are not necessary.

Input File LUNR

Record	Format	Cols.	Item	Comment
1	18A4	1-72	STR	Text record: Any characters for problem identification
2	18A4	1-12	SPRO	Text record: DATA HEXDYN
3	18A4	1-20	SCON	Text: CONTROL OF CALCULATION
4	14I5	1-5	ITIM	Without consequences
		5-10	IOINP	IOINP = 1: Output of input data without coefficients of parametrization of macroscopic cross sections IOINP > 1: Output of all input data IOINP = 0: Input data are not printed
		11-15	IH1	IH1 > 0: If the user has insight into details of code, some arrays will be printed to search errors of the input data IH1 = 0: No auxiliary output
		16-20	IH2	IH2 > 0: The table of outer iterations is printed in order to illustrate the convergence IH2 = 0: Output of the last iteration
		21-25	IH3	IH3 > 0: The addresses of partial currents at the outer boundary of the sector are printed IH3 = 0: No output of addresses
		26-30	IH4	At the end of the iterations several distributions are printed IH4 > 0: Output of neutron fluxes (normalized to a max. value of 9999.9) IH4 > 1, IH4 $\neq$ 4: The incoming partial currents are printed IH4 = 3 or IH4 > 0: Outgoing partial currents are printed

Record	Format	Cols.	Item	Comment
		31-35	IH5	IH5 > 0: The power peaking factors all nodes and and fuel elements are printed
5	5A4	1-20	SDI	Text: DIMENSIONS OF ARRAYS
6	14I5	1-5	ISYM	ISYM can assume the values 30, 60, 120, 181, 182 und 360 with reference to 30° -reflectional 60- und 120°-rotational /2/, 2 types 180°-reflectional symmetry (vertical and horizontal reflection respectively) and whole core /2/.
		6-10	NJMAX	Max. number of horizontal rows of the sector
		11-15	NIMAX	Max. number of columns of sector with reference to 60° co-ordinate system $NIMAX = \max_J IRIGHT(J) - \min_J ILEFT(J) + 1$ (J=JMIN,...,JMAX)
		16-20	NCAS	Number of hexagonal assemblies of the sector
		21-25	NZ	Number of slices in z-direction
		26-30	NOBOU	Number of outer faces of the hexagonal assemblies in radial direction
		31-35	NOSYMS	Number of faces of the hexagonal assemblies at the symmetry boundaries of the sector
		36-40	NOMAS	Number of different cross section sets
		41-45	NOBT	Number of different boundary conditions at the outer boundaries
7	3A4	1-12	SR	Text: MATERIAL MAP
8	14I5	1-5	JMIN	J-co-ordinate of the lowest row of the sector
		5-10	JMAX	J-co-ordinate of uppermost row

Record	Format	Cols.	Item	Comment
9	14I5	1-70	ILEFT(J)	I-co-ordinates of the left hex-agons (J=JMIN,...,JMAX) (J=JMIN,....,JMAX)
10		1-70	IRIGHT(J)	I-co-ordinates of the right hexagons (J=JMIN,...,JMAX)
The co-ordinates of the central assembly are (I,J)=(0,0)!				
11	1I5	1-5	INDC	INDC = -1: The identifiers of a given slice are read INDC = 0: The identifiers of the given slice are the same as for the previous slice INDC=NC, 1 ≤ NC ≤ NCAS NC identifiers of the given slice are different from those of the previous slice
12	14I5	1-70	IMAT(I)	IF INDC = -1, then IMAT(I) in the order I=ILEFT(J),.....,IRIGHT(J). A new record of type 12 for each J (J=JMIN,.....,JMAX).
12	14I5	1-5 6-10 11-15	I J KMAT	If INDC > 0, dann NC=INDC re-cords of type 12 with the hori-zontal co-ordinate I, the ver-tical co-ordinate J and the identifier of the cross section set KMAT are read
The records of type 11 followed by the records of type 12 with reference to INDC are repeated NZ-times with begin at the lower end of reactor.				
13	6A4	1-22	SV	Text: GEOMETRICAL PARAMETERS
14	6X, F10.6	7-16	SW	distance between opposite faces of the hexagons in cm
15	6X, 6F10.6	7-66	AN(IZ)	thickness of slices in cm for IZ=1,...,NZ
16	5A4	1-20	SIT	Text: CONTROL OF ITERATION
17	8X, 6F10.8	9-68	EPSBES	Trunction error of Bessel func-tions (recommended value 0.00002)
		19-28	EPSK	Truncation error of eigenvalue k_{eff} (recommended value 0.000001)

Record	Format	Cols.	Item	Comment
		29-38	EPSF	Truncation error of fission source (recommended value 0.000002)
		39-48	EPSMAT	without consequences
		49-58	EPSDS	without consequences
		59-68	EPS12	Limit for application of Ljusternik-acceleration (recommended value 0.025)
18	8X, F10.8	9-18	EKEFF	Initial value of k_{eff}
19	3I5	1-5	ITOUMA	Maximal number of outer iterations
		6-10	ITINMA	Number of inner iterations
		11-15	ITSCH	Order of Chebyshev-polynom used for acceleration (recommended values $3 \leq$ ITSCH ≤ 7)
20	I5	1-5	IOPT	Order of the expansion of the nodal neutron fluxes in z-direction (IOPT = 2 or 4)
21	5A4	1-18	S	Text: BOUNDARY RELATIONS
22	7X, 3F10.7	8-37	ALF11 ALF21 ALF22	Albedo coefficients α_{11}, α_{21}, α_{22} NOBT records of type 22
23	I5	1-5	IND	If NOBT > 0, for each slice beginning at the lower end of reactor the identifiers of boundary conditions for each outer boundary of the hexagonal plane are entered, i. e. IND = -1 Input of the identifiers IND = 0 The same identifiers as the previous slice are used
24	14I5	1-70	IBR(I)	If NOBT > 1 and IND = -1, the identifiers for all outer boundaries of hexagons are entered. The order of boundaries can be seen in Fig. 3,4,5,6 of /2/. ($1 \leq$ IBR(I) $\leq$ NOBT, $1 \leq$ I $\leq$ NOBOU)

Record	Format	Cols.	Item	Comment
25	14I5	1-70	IREFU(I)	If NOBT > 1, the identifiers of boundary conditions at the lower end of reactor are given in the order of assemblies (Fig. 3 of /2/. ($1 \le$ IREFU(I) $\le$ NOBT, I=1,...,NCAS)
26	14I5	1-70	IREFO(I)	Similar for the upper end of reactor
27	7A4	1-26	SR	Text: MACROSCOPIC CROSS SECTIONS
28	18A4	1-72	SF	If LUNR=LUNR1, input of the format SF of one set of the macroscopic cross sections
29	SF		DF(I) SIR(I) FNF(I) SFF(I) DT(I) SIA(I) FNT(I) SFT(I) TF(I)	If LUNR1=LUNR, input of the macroscopic cross sections in the order D_1 (cm), Σ_r (cm^{-1}), $\nu\Sigma_{f1}$ (cm^{-1}), Σ_{f1} (rel. units), D_2 (cm), Σ_a, $\nu\Sigma_{f2}$ (cm^{-1}), Σ_{f2} (rel. units), Σ_s (cm^{-1}) These cross sections are transferred for all NOMAS sets.
30	14I5	1-70	IDENT(I)	If LUNR$\neq$LUNR1, the NOMAS sets are contained in the file LUNR1 and the array IDENT(I) are the number of the set on the file LUNR1. (I=1,...,NOMAS)
31	3A4	1-11	SPOW	Text: TOTAL POWER
32	3X, F10.3	4-13	TOTPOW	Total thermal power of the reator in MW
If feedback isn't considered, the input is continued with record 48.				
33	10A4	1-38	SESS	Text: DATA FOR FEEDBACK
34	14I5	1-5	IBOR	IBOR=0 calculation with C_B=0 IBOR=1 boron acid concentration constant IBOR=2 boron acid concentration varies with time

Record	Format	Cols.	Item	Comment
		6-10	ITPOIS	ITPOIS = 0 no poison Xe and Sm ITPOIS > 0 Xe and Sm equilibrium distribution of the static state are considered.
		11-15	KRIPAR	KRIPAR=1: The critical state will be achieved dividing the multiplication cross sections by k_{eff}. KRIPAR=2: The critical state or a given value of $k_{eff,0}$ (different from 1.0) will be achieved by variation of boron acid concentrationration. KRIPAR=3: The critical state or a given value of $k_{eff,0}$ will be achieved by variation of reactor power.
		16-20	ITKRIM	Max. number of iterations flux distribution - temperature distribution in the case with feedback
34a	I5	1 -5	NHYCHA	Number of coolant channels (NHYCHA ≤ NCAS)
34b	7X, 3F10.7	8-17	EPSKRI	Truncation error of k_{eff} for the flux - temperature iterations.
		18-27	EPSTF	Truncation error of fuel temperatures for the flux - temperature iteration
		28-37	EPSRH	Truncation error of coolant density for the flux-temperature iteration
35	6A4	1-21	SFCO	Text: FEEDBACK COEFFICIENTS
The following records 36-42 are entered only, if LUNR1 = LUNR				
36	18A4	1-72	STR	If LUNR1=LUNR, input of the format STR of one set of parametrization coefficients
37	STR	1-72	RTMO(I)	Coefficients describing the dependence of cross sections from the coolant density for the group constants of the 1st set (same order as the group consants itself (I=1,...,9)

Record	Format	Cols.	Item	Comment
38	STR	1-72	RTM1(I)	linear coefficients of the dependence on the coolant density (same order as RTMO).
39	STR	1-72	RTM2(I)	quadratic coefficients of the dependence on the coolant density (same order as RTMO)
40	STR	1-72	RTB(I)	coeffiients of the dependence on the fuel temperature (same order as RTMO)
41	STR	1-72	RCB1(I)	If IBOR > 0, input of the linear coefficients of dependence on the boron acid cocentration (same order as RTMO)
42	STR	1-72	RCB2(I)	If IBOR > 0 input ot the quadratic coefficients of the dependence on the boron acid concentration (same order as RTMO)

After entering the recordes 37-42 for the 1^{st} set of group constants the input is repeated for the others of the NOMAS group sets.

Record	Format	Cols.	Item	Comment
43	5X, 6F10.5	6-15	TMO	Reference temperature of the moderator feedback in °C
		6-25	DENSO	Reference density of the moderator feedback in kg/m^3
		6-35	CBO	If IBOR > 0 iput of the reference value of boron acid concentration in g/kg H_2O

If LUNR1 $\neq$ LUNR, then the feedback coefficients are transferred from the file LUNR1.
Instead of records 36-43, the following record(s) 36 is read.

Record	Format	Cols.	Item	Comment
36	14I5	1-70	IDENT(I)	The order of sets of feedback parameters on the file LUNR1 is read (generally it will be the same array as in record 30, I=1,...,NOMAS).

Record	Format	Cols.	Item	Comment
The following records 44-45 are entered only in the case ITPOIS≠0				
44	5A4	1-20	STR	Text: XENON CROSS SECTIONS
45	1P 6E12.6	1-72	SIGXE(I)	Cross sections of XE (in barn) for I=1,...,NOMAS
The following record 46 are read only, if KRIPAR > 1				
46	7X, 2F10.7	8-17	BW	Input of the initial value of the derivation dk_{eff}/dc_B for KRIPAR = 2 or dk_{eff}/dP for KRIPAR = 3
		18-27	EKEFFO	$k_{eff,0}$: Destination of $_{keff}$
47	2I5	1-10	NZCB, NZCE	NZCB No. of the lowest slice of fuel and NZCE No. of the uppermost slice (different from 1 and NZ, if axial reflector zones are considered.
47a	14I5	1-72	IHYCHA(I) I=1,NCAS	Number of coolant channels belonging to fuel element I (order of input analogeous to IMAT(I), see Record 12) Input only if NHYCHA ≠ NCAS
48	10A4	1-40	SEST	Text: END OF DATA FOR STATIONARY CACULATION
The following records 49-52 are necessary, if polynomial coefficients for flux weighting of macroscopic cross sections in the case of partial in a node inserted control rods should be calculated. Otherwise continuation of input with record 53.				
49	10A4	1-40	STR	Text: ADDITIONAL DATA
50	I5	1-5	IRODST	Number of axial boundaries of nodes with different material in the upper and lower node, the polynomial coefficients of which will be transferred to the file LUNPOL
51	3I5	1-5	KI	Position of the fuel element in x-direction (60° co-ordinates system)
		6-10	KJ	Position of the fuel element in y-direction (60° co-ordinates system)
		11-15	KZ	Axial position of the upper node
Record 51 is read IROST-times.				
52	6A4	1-30	SSB	Text: END OF ADDITIONAL DATA
53	8A4	1-30	SI	Text: DATA FOR TRANSIENT CALCULATION
54	5A4	1-17	SC	Text: TIME STEP CONTROL

Record	Format	Cols.	Item	Comment
55	6X, 4F10.6	7-16	TE	Transient calculation will be carried until t=TE (in s) of problem time
		17-26	DTNK	Initial time step of neutron kinetics (in s)
		27-36	DTNKMI	minimal time step of neutron kinetics (in s)
		37-46	DTNKMA	maximal time step of neutron kinetics (in s)
		47-56	TCKIN	The initial time step of neutron kinetics isn't changed until time TCKIN (in s)
56	7X, F10.7	8-17	EPSF	Truncation error of flux iteration (recommended value 1.5E-6)
57	3I5	1-5	ITOUMA	Maximal number of outer iterations
		6-10	ITOUMI	Maximal Number of inner iterations
		11-15	ITSCH	Order of Chebyshev acceleration (recommended values 3,4,5, 6,7)
58	6X, 3F10.6	7-16	EPDOMM	Criteria using the change of mean value Ω of exponential transformation for time step control (recommended value 0.025)
		17-26	EPDOMR	Criteria using the maximal differences of Ω - distribution for time step control (recommended value 0.025)
		27-36	EPOM	Criteria using the mean value Ω for time step control (recommended value 0.25)
59	2I5	1-5	ITOUHA	If the number of outer iterations ITOU > ITOUHA, the time step of neutron kinetics is halved (recommended value 120)
		6-10	ITOUDB	If the number of outer iterations ITOU < ITOUDB, doubling the step of neutron kinetics (recommended value 40)

Record	Format	Cols.	Item	Comment
60	6X, F10.6	7-16	DTP	Thermohydraulic time step for detailed output at the printer (in s).
	I4	17-20	NPKL	Reduced output for each NPKL of thermohydraulic time steps
61	6X, 4F10.6	7-16	DTTH	Initial time step of thermo-hydraulics with DTTH > DTNK
		17-26	DTTHMI	Minimal thermohydraulic time step (in s)
		27-36	DTTHMA	Maximal thermohydraulic time step (in s)
		37-46	TCTH	The thermohydraulic is'nt changed before the time TCTH is reached (in s).
62	6X, 3F10.6	7-16	EPSTF	Relative change of the fuel temperature is used for con-trol of time step (recommended value 0.025)
		17-26	EPSRH	Similar value for moderator density (recommended value 0.025)
		27-36	EPSQ	Similar value for heat flux density (recommended value 0.5)
63	I5	1-5	ITHMAX	Max. number of iteration neu-tron kinetics-thermodydraulics
64	6X, F10.6	7-16	EPSPOW	Truncation error of change of power for the iteration neu-tron kinetics-thermohydraulics (Input only, if ITHMAX>0)
65	5A4	1-20	SM	Text: CONTROL ROD MOTION
66	3I5	1-5	NOTP	Number of time points, at which the material of some nodes is replaced completely by other material during the control rod movement.
		5-10	NOW	Number of different sets of polynomial coefficients for flux weighting in the case of partial in nodes inserted con-trol rods

Record	Format	Cols.	Item	Comment
		11-15	IHET	IHET = 0 (necessary)
67	7X, F9.6,I4	1-16	TIMP(I)	Time points for material replacement (I=1,...,NOTP)
		7-20	NOPN(I)	Number of nodes, in which the material at the time TIMP(I) is replaced completely (I=1,...,NOTP)
68	5I5	1-5	I	Horizontal co-ordinate of one of these nodes in the hexagonal plane.
		6-10	J	Corresponding vertical componente (60° co-ordinates)
		1-15	K	Number of the corresponding hexagonal slice
		6-20	ITYP	Type of the new material. If the control rod moves down, ITYP obtains a negative sign.
		21-25	IWEI	If NOW > 0 the type of weighting is entered.
Record 68 must be given NOPN(I) times. Records 67 with records 68 belonging to them are given NOTP times				
69	2I5	1-5	MDAT	MDAT ≤ 0: no thermohydraulic perturbation MDAT > 0: Number of time points with given values of perturbed function
		6-10	MSL	MSL = 0: The same perturbation of all primary loops (mixing model will not used). MSL > 0: perturbation of loop MSL (mixing model is used)
The following record 70 is entered MDAT times, if MDAT > 0				
70	6X, 5F12.8	7-16	TP(I)	Time points of perturbation (in s)
		17-26	TEDAT(I)	Absolute change of coolant inlet temperature at time TP(I) from to the stationary value
		27-36	PDAT(I)	The relative values of pressure

Record	Format	Cols.	Item	Comment
		37-46	DDAT(I)	Relative pressure drop or mass flow rate
		47-56	BDAT(I)	Absolute change of boron acid concentration at time TP(I) from the stationary value
71	4A4	1-10	SD	Text: DELAYED NEUTRONS
72	2I5	1-5	NODN	Number of groups for precursors of delayed neutrons
		6-10	NOBET	Number of different sets of β_{eff} values
73	5X, 6F10.5	1-65	RLAM(I)	Decay constants in s^{-1} of precursors (I=1,2,.,NODN)
74	5X, 6F10.5	1-65	BETAF(I)	β_{eff} values of fast fissions (I=1,2,...,NODN) of the first set
75	5X, 6F10.5	1-65	BETAT(I)	β_{eff} values of thermal fissions (I=1,2,...,NODN) of the first set
Records 3 - 75 are entered for all NOBET sets				
76	14I5	1-70	INDBE(I)	IF NOBET > 1, the identifiers of sets are read for all nodes. The same order as the identifiers of cross section sets
77	7X, 1P E12.5	8-20	RLIFE	Mean neutron lifetime of core in s (used for approximate evaluation of reactivity)
78	8X, 6F10.8	9-68	BETAFF(I)	Effective values β_{eff} of core (used also for evaluation of reactivity)
79	5A4	1-20	SV	Text: NEUTRON VELOCITIES
80	I5	1-5	NOVET	Number of sets of different neutron velocities
81	7X, 2E10.3	8-17	VEF	Velocity of fast neutrons of the first set
		18-27	VET	Velocity of the thermal neutrons of the first set
Record 81 is entered for all NOVET sets.				

Record	Format	Cols.	Item	Comment
82	14I5	1-70	INDVE(I)	If NOVET > 1, the identifiers of sets are read for all nodes. The same order as the identifiers of cross section sets
83	A4	1-4	SE	Text: FINE

Input Data Description for the Module FLOCAL (Thermal-Hydraulics)

The input data set for the thermo-hydraulic module FLOCAL of the code DYN3D/M2 is read from a file with the number LUNTHE. If the mixing model by Draeger /12/ for the lower plenum is used, the reference distributions have to be available. A survey of the available reference distributions is given in the following table.

Reactor type	Type of the reference distribution	File number	File name
W 213	calculated	60	W213B6
W 213	experimental	61	W213E6
W 230	calculated	85	W230B6
W 230	experimental	66	W230E6

Most of the file LUNTHE is read in free format. That's why the input data records must be no-numbered. Repeat factors for data can be given, what is a significant increase of comfort for large core sectors with a lot of identical fuel elements. The total number of fuel elements NK (each fuel element is identical with one coolant channel) and the number of axial nodes NH is taken from DYN3D and are not read in FLOCAL, but this values must be known for the construction of the data set.

Input Data Set LUNTHE

Record	Format	Variable	Comment
1	3I5	NSTO	Controler for the solution of the coolant mass balance equation: NSTO $\leq$ 0 constant - NSTO > 0 space dependent - instationary mass flow rate
		NST1	Controler for selecting the significant DNBR-correlation NST1 $\leq$ 0 no post-crisis heat transfer, NST1 = 1 DNBR by correlation IAE-4 NST1 = 2 DNBR by correlation OKB-2 NST1 = 3 DNBR by correlation BIASI
		NSTD	Controler for the dynamical correction of DNBR calculation: NSTD $\leq$ 0 no dynamical correction NSTD > 0 dynamical correction for IAE4 and BIASI correlations
2	3I5	NST2	Key for the fuel rod model: NST2 > 0 constant heat transfer coeff. and thermal conductivity NST2 $\leq$ 0 determination of this values in the frame of the model
		NR	Number of radial zones in fuel for solution of heat conduction equation
		NTYP	Number of different types of fuel rods
3	2I5	NST3	Controler for selecting the hydraulic boundary conditions: NST3 < 0 given pressure drop, NST3 = 0 given average total mass flow rate through the core, NST3 > 0 given inlet mass flow rate for each coolant channel
		MISCH	Key for selecting the mixing model for the lower plenum: MISCH< 0 no mixing MISCH= 0 mixing model by Draeger MISCH= 1 ideal mixing with c_p=const. MISCH= 2 ideal mixing with enthalpy balance
4	2I5	NS	Number of special hot channels
		NST5	Key for the calculation of a fictive channel with core averaged power (only for steady state) NST5 $\leq$ 0 no NST5 > 0 yes, NST5 is the number of fuel rod type for this channel

In the records 5 - 11 the data for each fuel rod type ityp=1,NTYP are given (repeating NTYP times).

Record	Format	Variable	Comment
5	free	DBI	Diameter of inner fuel pellet hole (must be > 0), in m
		DBA	Outer diameter of fuel pellet, in m
			Fuel material parameters:
		FB	portion of heat release
		RLB	thermal conductivity, in kW/mK
		ROB	density, in kg/m^3
		CB	specific heat, in kJ/kg K
6	free	DHI	Inner diameter of cladding, in m
		DHA	Outer diameter of cladding, in m
			Cladding material parameters:
		FH	portion of heat release
		RLH	thermal conductivity, in kW/mK
		ROH	density, in kg/m^3
		CH	specific heat, in kJ/kg K
7	free	DHYD	Equivalent hydraulic diameter of the rod bundle, in m
		S	Fuel rod lattice pitch, in m
		AST	Free flow cross section per fuel element, in m^2
		RPIN	Number of heated rods per fuel element
8	free	QR(1,NR)	Heat release distribution over the radial fuel zones
9	free	ALSP	Heat transfer coefficient for the gas gap, in kW/m^2

The input of records 10 and 11 is desired only in the case NST2 $\leq$ 0 (gas gap data).

Record	Format	Variable	Comment
10	free	DO	Reference gap width, in cm
		TBO	Reference fuel temperature, in K
		THO	Reference cladding temperature, in K
		PGAS	Cold gas pressure, in MPa
		XHE	Helium mole fraction
11	free	RAUB	Surface roughness of the fuel
		RAUH	- of the cladding, in cm

It follows the input of the hydraulic data for the core.

Record	Format	Variable	Comment
12	free	HE HA P DP ZA	Height of the fuel element foot, in m Height of the fuel element header, in m Coolant pressure at the upper boundary of the core, in MPa Pressure drop over the core, in kPa Flow resistance coefficient at the core outlet
13	frei	ZD(1,NH)	Flow resistance coefficients at the spacer grid for each axial node

Record 14 is entered only in the case NST3 > 0 (given mass flow rate).

14	free	TFRO, if NST3=0 FR(1,NK), if NST3>0	Total mass flow rate through the core (related to the whole core, take into account symmetry) in kg/s Mass flow rate for each coolant channel (per one fuel rod) in kg/s
15	free	ZE(1,NK)	Inlet flow resistance coefficient for each fuel element
16	free	ITYP (1,NK)	Number of fuel rod type for each fuel element

In the records 17 - 21 the description of the special hot channels is given. The input is demanded only if NS > 0 .

17	free	IZO(1,NS)	Number of the connected core channel for each hot channel
18	free	ITYP (1,NS)	Number of the fuel rod type for each hot channel
19	free	RK(1,NS)	Power peak factor of the hot channel relative to the connected core channel
20	free	HME(1,NS)	Difference of the coolant inlet temperature relative to the connected channel, in K
21	free	ZE(1,NS)	Inlet resistance coefficients for the hot channels

In the records 22 and 23 truncation errors and relaxation parameters for iteration procedures are given (including recommendations for this values).

Record	Format	Variable	Comment	
22	5F12.5		Truncation error[1] of:	
		EPS1	- mass flow rates (for given pressure drop)	0.002
		EPS2	- pressure distribution (steady state)	0.002
		EPS3	- - instationary	0.005
		EPS4	maximum absolute truncation error of the void fraction	0.001
		EPS5	weight factor for the heat flux at old and new time step	0.5
23	5F12.5		Truncation errors of:	
		EPS6	- fuel temperatures	0.001
		EPS7	- cladding temperatures	0.001
			Relaxation parameters[2] for:	
		EPS8	- pressure drop iteration in the case of given total mass flow rate	1.0
		EPS9	- the gas gap heat transfer coefficient iteration	0.5
		EPS10	- the cladding temperature in the post-crisis region	0.5

1) An iteration process is stopped, if the relative deviation of a variable x between previous and actuall iteration steps is less than the truncation error ε :

$$\left| \frac{x^{(n+1)} - x^{(n)}}{x^{(n)}} \right| < \varepsilon$$

2) The relaxation parameter Θ determines the weight of old and new value of a variable in the start value for the next iteration step

$$\tilde{x}^{(n+1)} = \Theta x^{(n)} + (1 - \Theta)\, \tilde{x}^{(n)},$$

$\tilde{x}^{(n)}$ - start value for the iteration step n, $x^{(n)}$ - value determined in the iteration step n, $0 < \Theta \leq 1$. A value of Θ near 1, in general, leads to an accelaration of the iteration process, but can induce some stability problems.

The records 24 and 25 contain the inlet values of coolant temperature and boron acid concentration for each fuel element, if no lower plenum mixing model is used (input only in the case MISCH < 0).

Record	Format	Variable	Comment
24	free	HME(1,NK)	Inlet temperature of the coolant for each fuel element, in deg C
25	free	CBOR(1,NK)	Inlet value of the boron acid concentration for each fuel element, in g/kg

The input of the records 26 - 29 is desired only in the case MISCH > 0 (use of a mixing model for the lower plenum), actually:
record 26 only if MISCH = 0 (mixing model by Draeger),
record 27 only if MISCH > 0 (ideal mixing).

26	A4,1X, A4	TYP BVERT	Reactor type (string "W213" or "W230") Type of the reference distribution (string "BER " or "EXP ")
27	I5	NSL	Number of loops of the primary circuit (if MISCH=0 is set NSL=6)
28	3F12.5	RMS(I) TS(I) BS(I)	Portion of total mass flow in loop I (not nessesary normalized) Coolant temperature in the loop, deg C Boron acid concentration in the loop I, in g/kg

The record 28 must be given for all loops I=1,NSL.

29	I5	IDRU	Output controler for the mixing model, IDRU=0 no output via printer, IDRU=1 printer output of the calculated inlet distributions map, IDRU=2 additional output of the reference distribution, (significant only if MISCH=0)

The input of the control parameters for the thermo-hydraulic calculation, including the table for the description of time-dependent variations of the primary circuit parameters, is accomplished in the subroutines INPTRA and INPERT (see description of input data file LUN record 70.

6. Output and Analysis of the Results

The output of input data is accomplished dependent from the value of the key IOINP. After each calculation of the steady-state neutron flux distribution, the number of outer iterations, the eigenvalue KEFF, the deviation of the eigenvalue DKF and the fission source term distribution DSOU, the relation between second and first eigenvalue EVA12 and the maximum relative deviation of fission sources are printed. If feedback is taken into account, the number of thermo-hydraulic iterations ITKRIT and the maximum deviations of fuel temperature and coolant density between the last two iterations DTF and DRHO are printed. Additionally, the resulting eigenvalue for the sequence of iterations is given. In the case of achieving criticality by variation of the boron acid concentration, the actual value of this parameter is printed. If the critical state is attained by variation of the reactor power, the power value is printed. In this case, possibly, several thermo-hydraulic iterations are necessary before obtaining a new value for the critical power. This procedure is used that's why in the case of coolant boiling the power density distribution sensibly depends on the moderator density distribution, what can disturb the iteration process for the critical power value.

In accordance with the choice of the output control parameters IH4 and IH5 after finishing a steady-state calculation, neutron flux distribution, power peaking factors and distribution of absolute power density values are printed.

Besides the described above output, the results of steady-state and instationary calculations are printed in a unique form by the help of the subroutine RESULT. The choice of a reduced or detailed output option is possible (see Input Description of DYN3D, Sect.4). A reduced output is induced after NPKL time steps, a detailed output in given time intervalls.

The reduced output includes:

- global parameters of the core (thermal power, power supply to coolant, coolant mass flow rate, pressure, pressure drop, average coolant inlet temperature and outlet vapour mass fraction, average moderator density and temperature, average fuel temperature),

- averaged parameters (thermal power per fuel rod, power supply coolant per fuel rod, mass flow rate, fuel tempera ture, coolant temperature and density) and maxima of safety relevant parameters (vapour mass fraction, maximum fuel temperature, fuel enthalpy, cladding temperature oxide layer thickness, stress criterion RUPT) for each coolant channel (or fuel element).

Fuel enthalpy (in kJ/kg fuel), oxide layer thickness (in microns) and the stress criterion are used for the evaluation of fuel rod failure. The stress criterion is defined as

$$\text{RUPT} = \begin{cases} 0 & \text{if the stress } \sigma \text{ in the dadding} \\ & \text{is below the yield strength the } \sigma_y, \\ 1 & \text{if } \sigma \geq \sigma_y. \end{cases}$$

The detailed output comprises, moreover, the printing of axial distributions of following parameters for each fuel element (coolant channel):

- linear power rate (in kW/m),
- heat flux (in kW/m^2),
- fuel enthalpy (in ky/kg),
- fuel central temperature
- radial averaged fuel temperature in deg C,
- cladding surface temperature
- coolant temperature
- mass flow velocity (in kg/m^{2}s),
- heat transfer coefficient in gas gap (in kW/m^{2}K),

- DNB ratio determined by correlations IAE-4, OKB-2 and BIASI,
- oxide layer thickness (in microns),
- key defining the heat transfer regime
  (0 = natural convection, 1 = forced convection,
  2 = developed boiling, 3 = forced convective boiling,
  4 = transition boiling, 5,6 = film boiling,
  7 = convection to superheated steam),
- stress criterion.

The detailed output can be very paper consuming. That's why besides the printing of the results, all significant output data can be written on a sequential file number LUNRES. This file can be used for an off-line analysis of the results or the construction of plot files.

The output on the file LUNRES is accomplished for each call of the subroutine RESULT (independently from the actual choice of reduced or detailed output option) in a unique format. The output data file LUNRES consists of following unformatted data records for each time step:

1.	TAU	actual process time	(in s),
2.	Array VRES(10) containing:		
	QNU	total nuclear reactor power	(in kW)
	QTH	thermal power supplied to coolant	(in kW)
	TFR	total coolant mass flow rate	(in kg/s)
	HCE	average coolant inlet temperature	(in deg C),
	XCA	average vapour mass fraction at the core outlet,	
	P	coolant pressure	(in MPa),
	DP	pressure drop over the core	(in kPa),
	REAC	reactivity	(in $),
	QNA	nuclear reactor power,	
	VRES(10)	dummy.	

3. Array F

The array F contains successively the axial distributions of following parameters (the length of each segment is given in parentheses, NH = number of axial nodes):

- pressure (NH+1),
- vapour mass fraction (NH+1),
- void fraction (NH+1),
- coolant density (NH+1),
- coolant mass flow velocity (NH+1),
- linear power rate per rod (NH),
- heat flux (NH),
- key for the actual heat transfer regime (NH),
- DNBR by correlation IAE-4 (NH),
- DNBR by correlation OKB-2 (NH),
- DNBR by correlation BIASI (NH),
- cladding surface temperature (NH),
- radial averaged fuel temperature (NH),
- fuel centerline temperature (NH),
- heat transfer coefficient between cladding and coolant (NH),
- heat transfer coefficient for the gas gap in fuel rod (NH),
- oxide layer thickness (NH),
- stress criterion (NH).

This parameter composition is repeated for all fuel elements. At the beginning of the file LUNRES (before the steady-state results) several records are written, containing some necessary informations (length of arrays, constants), actually:

1.	record	length	35	type	integer
2.	"	"	360	"	"
3.	"	"	15	type	real
4.	"	"	750	"	"

The analysis of the file LUNRES can be accomplished by the help of a programm OUTGRA and a subroutine REGRAF. This routines construct from the file LUNRES a new sequential file NOUT, containing the most significant parameters for one given channel (fuel element) IK, what can be used for graphical representation of the results. The file NOUT contains for each time point TAU a record with the format 16(F10.3,2X) including the parameters:

TAU	actual process time	(in s),
QROD	nuclear power per fuel rod	(in kW),
QCOL	thermal power supplied to coolant	(in kW),
GAV	average mass flow velocity	(in kg/m^{2}s),
TAV	average fuel temperature	(in deg C),
TCOL	average coolant temperature	(in deg C),
DENS	average coolant density	(in kg/m^3),
XOUT	vapour mass fraction at the fuel element outlet	(in %),
DNB1, DNB2, DNB3	DNBR by correlations IAE-4, OKB-2 and BIASI,	
TMAX	maximum value of fuel centerline temperature	(in deg C),
HMAX	maximum value of fuel enthalpy	(in ky/kg),
TSMX	maximum value of cladding surface temperature	(in deg C),
PHMX	maximum value of rod fraction	(in %),
DMAX	maximum value of oxide layer thickness	(in microns).

The programm OUTGRA is an off-line post processing procedure. For a call of the programm OUTGRA following parameters are read in free format:

LUNRES	number of the file with detailed results,
NOUT	number of the plot file,
IK	number of the fuel element of interest.

The subroutine REGRAF used in OUTGRA calls the function HCB, included in the set of functions MATCON (see Section 4).

Additionally to the output accomplished by the subroutine RESULT, for each neutron kinetics step following parameters are printed:

TNNK	actual time for neutron kinetics,
DTNK	new value for the neutron kinetics time step,
ITOU	number of outer iterations,
DSOU	average deviation of fission source distributions between 2 outer iterations,
DFMAX	maximum deviation of fission sources between 2 outer iterations,
EW	eigenvalue of the homogeneous equation.

After finishing a thermo-hydraulic time step the average exponent OM for the neutron flux time behaviour of the actual neutron kinetic time step DTNK and the approximately estimated reactivity value REACT are printed.

During instationary calculation, additionally, for each thermo-hydraulic time step, several control parameters are printed. That are

TNTH	end time of the actual thermo-hydraulic time step,
DTTH	actual size of the thermo-hydraulic time step,
ITFEED	iteration index for the thermo-hydraulic time step,
POWER	actual nuclear power,
DP	relative change of the nuclear power between 2 iterations,
DHF	maximum relative change of the heat flux,
DFT	maximum relative change of the fuel temperature,
DMD	maximum relative change of the moderator density.

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Fig. 1: Flow chart of main program DYN3D/M2

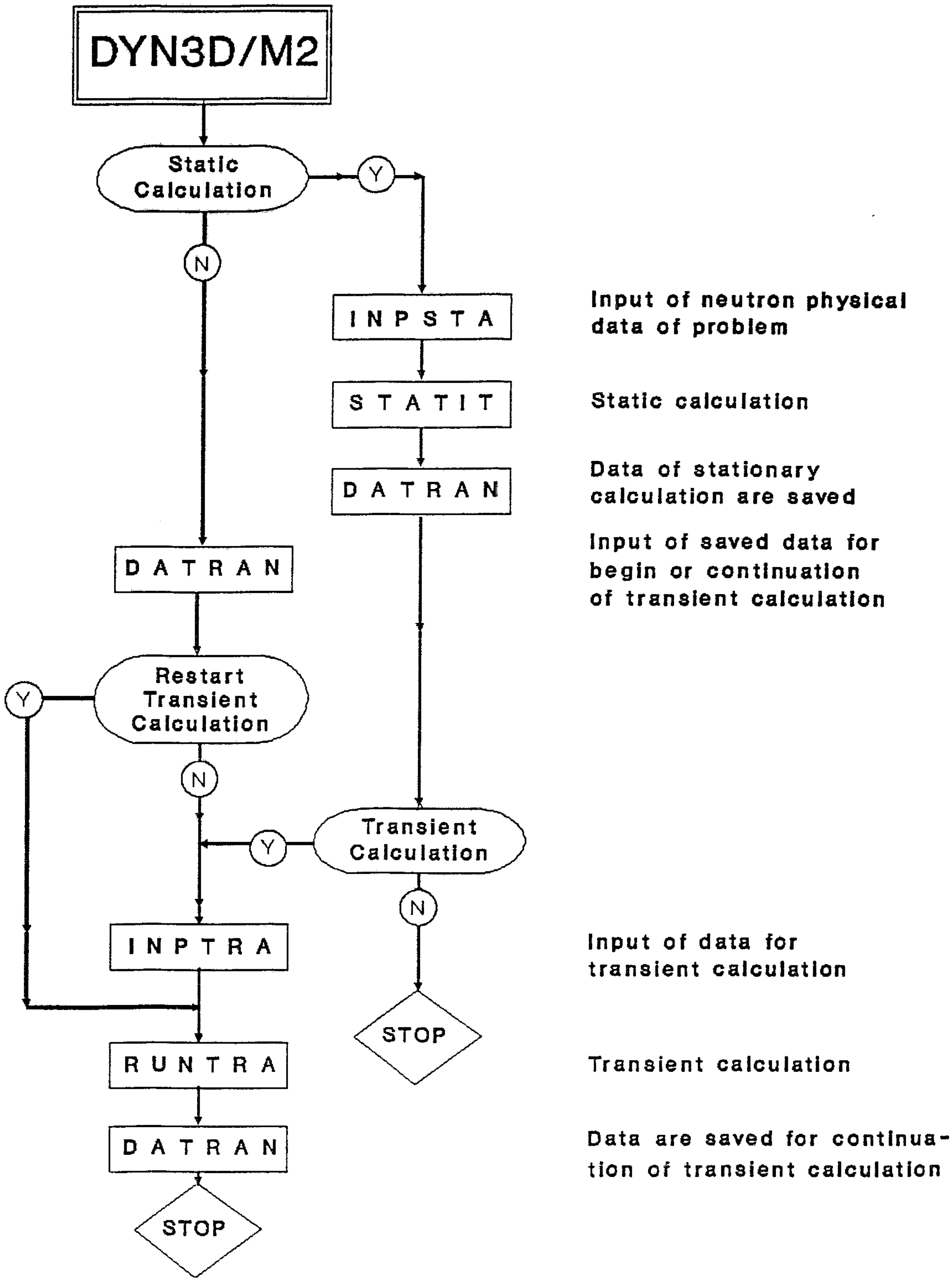
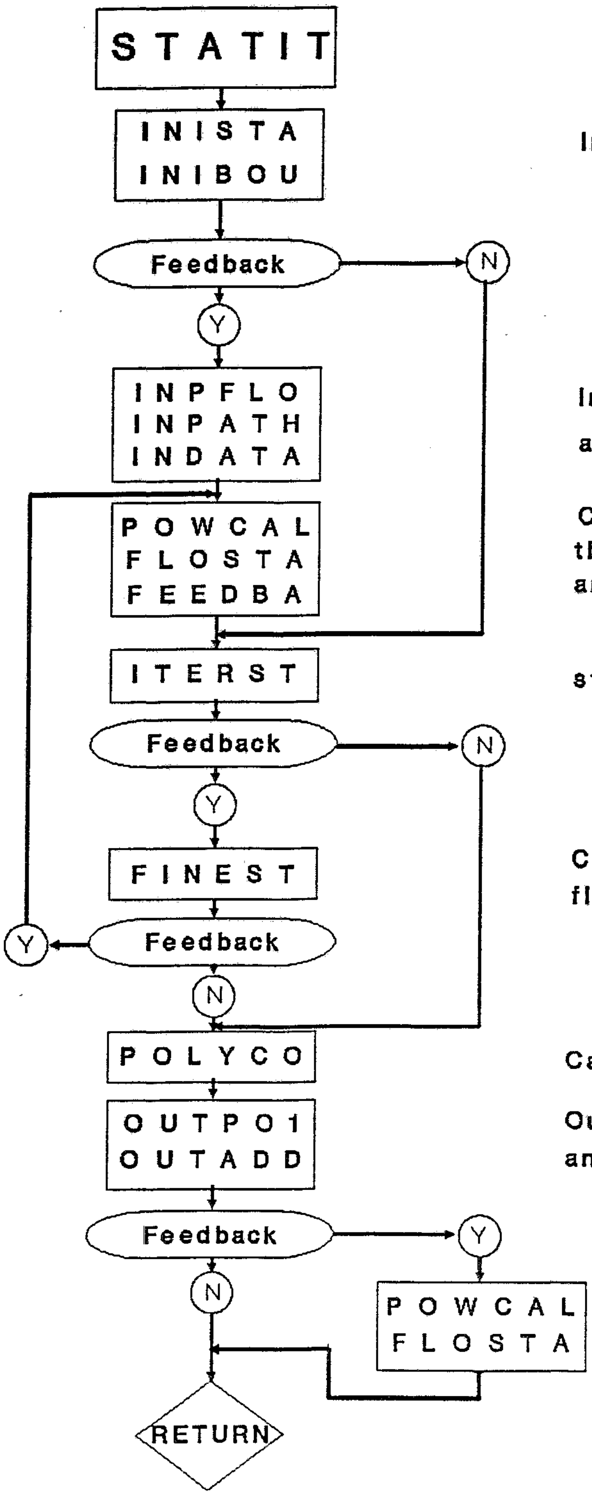


Fig. 2: Flow chart of subroutine STATIT



Initial values and symmetry

Input of feedback coefficients
and thermohydraulic data

Calculation of power density,
thermohydraulics, fuel rod
and feedback

static neutron flux calculation

Control of iteration
flux calculation - thermohydraulics

Calc. of polynomial coeffic. (if wanted)

Output of flux distribution
and mean power densities

Calculation of power densities
Thermohydraulics and output

Fig. 3: Flow chart of subroutine RUNTRA