

PROJECT ADMINISTRATION DATA SHEET

ORIGINAL REVISION NO. _____

Project No. E-26-B04 (Follow-on to E-26-B03) GTRI/~~EES~~ DATE 12/15/83

Project Director: J. M. Kallfelz School/~~EES~~ Nuclear Engr.

Sponsor: Union Carbide Corp., Nuclear Div., Oak Ridge, TN

Type Agreement: S.A. #8, Proj. Auth.#X01 under B.A.#7802 (DOE Prime)

Award Period: From 10/1/83 To 6/30/84 (Performance) 7/30/84 (Reports)

Sponsor Amount:	This Change	Total to Date
Estimated: \$ <u>32,828</u>		\$ <u>32,828</u>
Funded: \$ <u>32,828</u>		\$ <u>32,828</u>

Cost Sharing Amount: \$ None Cost Sharing No: N/A

Title: Analytic Reactor Physics and Design Investigations

ADMINISTRATIVE DATA

OCA Contact William F. Brown x4820

1) Sponsor Technical Contact:

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Defense Priority Rating: N/A Military Security Classification: N/A

(or) Company/Industrial Proprietary: _____

RESTRICTIONS

See Attached _____ Supplemental Information Sheet for Additional Requirements.

Travel: Foreign travel must have prior approval - Contact OCA in each case. Domestic travel requires sponsor approval where total will exceed greater of \$500 or 125% of approved proposal budget category.

Equipment: Title vests with None proposed or anticipated.

COMMENTS:

Supplemental Agmt.#8 adds \$32,828 for additional 9 months of effort.

Total amount of subcontract (including previous project numbers) is \$279,352.

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SPONSORED PROJECT TERMINATION/CLOSEOUT SHEET

Date 8/31/84

Project No. E-26-B04

School ~~XXX~~ NE/NEHP

Includes Subproject No.(s) _____

Project Director(s) J. M. Kallfelz

GTRI / ~~XXX~~

Sponsor Martin - Marietta (formerly Union Carbide) Oak Ridge Facility

Title "Analytic Reactor Physics and Design Investigations"

Effective Completion Date: _____ (Performance) _____ (Reports)

Grant/Contract Closeout Actions Remaining:

- None
- Final Invoice or Final Fiscal Report
- Closing Documents
- Final Report of Inventions
- Govt. Property Inventory & Related Certificate
- Classified Material Certificate
- Other _____

Continues Project No. E-26-B03

Continued by Project No. E-25-627

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Georgia Institute of Technology
A UNIT OF THE UNIVERSITY SYSTEM OF GEORGIA
SCHOOL OF NUCLEAR ENGINEERING AND HEALTH PHYSICS
ATLANTA, GEORGIA 30332
6 December 1983

(404) 894-3720

MEMORANDUM

TO: D.G. Cacuci, C.R. Weisbin and B.A. Worley (ORNL)
FROM: J.M. Kallfelz
SUBJ: Progress Report for ORNL Subcontract 7802
Months of October and November, 1983

Accomplishments During Report Period

The subcontract extension was not mailed by Union Carbide until 20 November 1983. Thus there was no activity on this project in October, and only 10% of my time was spent on this project in November.

Literature concerning the DSNP language (1-7) for simulating LMFBR systems was reviewed, and initial steps were taken to obtain this code system at Georgia Tech. For our investigations related to development of an LMFBR monitor, the DSNP code system could prove useful for studies of overall LMFBR system behavior. I plan to attend a workshop on the DSNP system to be held at Argonne (8,9).

Plans for Work for Next Month

Studies of the application of the DSNP system in LMFBR monitor development will be continued. If the code is obtained soon enough (already ordered) at Georgia Tech, work on initiation of the code on our computer system will start.

REFERENCES

1. D. Saphier, "A Special Purpose Simulation Language for Nuclear Power Plants," in Simulation of Systems '79, L. Dekker et al., Eds., North-Holland Publishing Co., p. 1055, 1980.
2. D. Saphier and J.T. Madell, "The DSNP Simulation Language and Its Application to LMFBR Transient Analysis," Nucl. Tech., 56, p. 493, (March, 1982).
3. D. Saphier, "The Simulation Language of DSNP: Dynamic Simulator for Nuclear Power Plants," ANL-CT-77-20, Rev. 02, Argonne National Laboratory, 1978.
4. D. Saphier, "The Level-One Modules Library for DSNP: Dynamic Simulator for Nuclear Power Plants," ANL-CT-77-22, Rev. 02, Argonne National Laboratory, 1978.
5. D. Saphier, "The Precompiler for DSNP: Dynamic Simulator for Nuclear Power Plants," ANL-CT-77-22, Rev. 02, Argonne National Laboratory, 1978.
6. J.T. Madell and D. Saphier, "A Comparative Transient Analysis of Metal and Oxide Fueled LMFBR," ANL-CT-80-1, September, 1979.
7. D. Saphier, Soreq. Nucl. Res. Cen., letter to J. Kallfelz, dated 30 June 1983.
8. M. Harris, ANL, letter concerning DSNP Workshop at Argonne, dated October 20, 1983.
9. J. Madell, Fauske and Associates, letter to J. Kallfelz, dated November 10, 1983.

E-26-804



Georgia Institute of Technology
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SCHOOL OF NUCLEAR ENGINEERING AND HEALTH PHYSICS
ATLANTA, GEORGIA 30332

(404) 894-3720

January 18, 1984

MEMORANDUM

TO: D. G. Cacuci, C. R. Weisbin and B. A. Worley (ORNL)
FROM: J. M. Kallfelz *JMK*
SUBJECT: Progress Report for ORNL Subcontract 7802
Month of December, 1983

Accomplishments During Report Period

- o Studies of the application of the DSNP code system²⁻⁴ in LMFBR monitor development was continued. This code was received at Georgia Tech during December⁵, and work on initiation of the code on our computer was begun.
- o On December 19 I visited ORNL, and discussed the plans for our studies concerning a Power Distribution and Reactivity Monitor (PDRM)⁶ for an LMFBR. I met with D. Cacuci, B. Worley and F. di Fillipo, and the attached page summarizies our agreement on the distribution of responsibilities for this effort.

Plans for Work for Next Month

- o Work on the initiation of the DSNP system on the Georgia Tech computer will be continued. If successful operation is achieved, we will start performing runs related to LMFBR operations.

JK/vw
Attachment

cc: Dr. F. di Fillipo

Responsibilities for PDRM Studies

Following is a summary of the agreements made at ORNL on December 19, concerning our studies related to a Power Distribution and Reactivity Monitor (PDRM)⁶ for an LMFBR.

1. Capabilities of PDRM Desired by Operations Personnel.

J. Kallfelz will attempt to better define these capabilities through discussions with staff members at vendors, national laboratories, and FFTF.

2. In-core Calculations.

F. di Fillipo and B. Worley will investigate the methods for on-line calculations of core parameters (see sec. 2.4 of Ref. 6), including

- (a) Nodal/modal codes
- (b) Influence function methods, and
- (c) Eigenfunction expansion methods

This topic includes the use of measured values for adjustment of parameters used in on-line calculations. (see sec. 2.5.3. of Ref. 6).

F. di Fillipo will have primary responsibility for this topic, and B. Worley will provide assistance in investigating method (a).

3. Two-dimensional Adjoint Flux Calculations.

These calculations include transport and diffusion theory calculations, to determine the sensitivity of ex-vessel detectors to core power distributions (see sec. 2.5.1. of ref. 6).

B. Worley will be responsible for these investigations, to be initiated in the spring.

4. Application of the DSNP Code System.

Application of the DSNP code²⁻⁴ for LMFBR systems analyses related to PDRM development will be studied.

J. Kallfelz is responsible for this topic.

REFERENCES

1. J. M. Kallfelz, "Progress Report for ORNL Subcontract 7802, Months of October and November, 1983," memorandum to D. G. Cacuci, C. R. Weisbin and B. A. Worley (ORNL) dated December 6, 1983.
2. D. Saphier, "A Special Purpose Simulation Language for Nuclear Power Plants," in Simulation of Systems '79, L. Dekker et al., Eds., North-Holland Publishing Co., p. 1055, 1980.
3. D. Saphier and J. T. Madell, "The DSNP Simulation Language and Its Application to LMFBR Transient Analysis," Nucl. Tech., 56, p. 493, (March, 1982).
4. See references 3-7 of ref. 1.
5. John T. Madell, Fauske & Associates, Inc., letter of transmittal to J. M. Kallfelz dated December 9, 1983.
6. J. M. Kallfelz and L. A. Belblidia, "Studies for a Power Distribution and Reactivity Monitor (PDRM) and Peak Core Power Densities," GITNEHP-83/1, School of Nuclear Engineering and Health Physics, Georgia Institute of Technology (Annual Progress Report for ORNL Subcontract 7802-X01), October 1983.

E-26-B04

Georgia Institute of Technology

A UNIT OF THE UNIVERSITY SYSTEM OF GEORGIA

SCHOOL OF MECHANICAL ENGINEERING

ATLANTA, GEORGIA 30332

March 16, 1984

Reply to:
Nuclear Engineering and
Health Physics Program
Cherry Emerson Building

MEMORANDUM

TO: D. G. Cacuci and C. R. Weisbin

FROM: J. M. Kalfelz, *JMK* L. A. Belblidia and L. S. Choi

SUBJECT: Progress Report for ORNL Subcontract 7802
Months of January and February, 1984

Accomplishments During Report Period

- o L. A. Belblidia attended a workshop at Argonne National Laboratory on the simulation language code system DSNP (Dynamic Simulator for Nuclear Power Plants)^{1,2} During the Hands-on session he ran various DSNP cases, and had an opportunity to resolve many of the questions concerning the initiation of this code on the Ga. Tech IBM 4341. Details are given in Attachment A.
- o The initiation of the DSNP program at Georgia Tech has progressed so that sample Problem #1 (described in Appendices B and C) has been successfully run. Steps taken to reach this stage are described in attachment A.
- o The physical significance and logical structure of DSNP test case 1 has been thoroughly analyzed as described in Attachments 2 and 3.

- o Steps have been taken³ to obtain the reactor transient code RELAP 3B⁴ a program which J. Kallfelz utilized in reactor operational transient studies⁵ performed in Germany.

Plans for Work for Next Month

- o Initialization of DSNP on the Ga. Tech computer system will continue, and further test cases will be run and analyzed.
- o If the RELAP3B code package is received in time, work will start to make this code operational on the Ga Tech CYBER 170/855 system.

REFERENCES

1. D. Saphier, "A Special Purpose Simulation Language for Nuclear Power Plants," in Simulation of Systems '79, L. Dekker et al., Eds., North-Holland Publishing Co., p. 1055, 1980.
2. D. Saphier and J. T. Madell, "The DSNP Simulation Language and Its Application to LMFBR Transient Analysis," Nucl. Tech., 56, p. 493, March, 1982.
3. J. M. Kallfelz, letter to F. Oda, USNRC, dated Feb. 23, 1984, Subject: Request for RELAP 3B Program
4. BNL-NUREG - 22011, "User's Manual for RELAP3B-MOD 110, A Reactor System Transient Code", Department of Applied Science, Brookhaven National Laboratory.
5. J. M. Kallfelz et al, "ATWS Studies for a Tight-Lattice LWR", in proceedings of ANS Topical Meeting on Anticipated and Abnormal Plant Transients in Light Water Reactors, Jackson, Wyoming, 26-29 Sept. 1983.
6. DSNP Manual, ANL-CT-77-20, Argonne Nat. Lab.

ATTACHMENT A - ORNL PROGRESS REPORT - 15 MARCH 84

REPORT ON DSNP-RELATED ACTIVITIES (Lotfi Belblidia/3-13-84)

On February 21 and 22, 1984, I attended a DSNP Workshop held at Argonne National Laboratory. The workshop was directed primarily at university professors who would be interested in using DSNP in their research and for teaching. It combined presentations by John Madell, Eric Dean and Howard Larson with hands-on experience at computer terminals. Over thirty people attended this workshop.

The presentations involved a description of DSNP features, discussions on how to set up and run DSNP programs, and descriptions of some component modules and of DSNP MACRO capabilities. Sample problems, such as solving an ordinary differential equation with DSNP, simulation of SATURN V rocket launch, and a reactivity insertion problem in a LMFBR with no hydraulics involved, were used to illustrate the talks.

During the hands-on sessions, I ran a simple heat exchanger problem and three of the sample cases given in DSNP's users manual. I have complete outputs for Sample Problem #1 which simulates the kinetics and thermodynamics of the primary loop of a LMFBR, Sample Problem #2 which simulates a startup accident in a 10 Mw thermal reactor, and Sample Problem #6 which is an LMFBR problem that includes neutronics, thermodynamics and hydrodynamics. I will use these outputs for comparison with the results I obtain here.

It appears now that the problems I have been having in my attempts to get DSNP to run on our IBM 4341 are specific to this system and to the compiler used. For example, the DATA statements in the DSNP precompiler and library routines appear sometimes before the explicit type statements. This causes an error on our machine. Changing the order of these statements took care of this problem. Another problem which I am still trying to resolve has to do

with the way the precompiler writes the BLOCKDATA routine of the generated FORTRAN program. There are three ways in DSNP to set a variable a value: (1) it is set to zero originally, (2) it appears in the default input data set, and (3) it is provided by the user, and all three contribute DATA statements to the BLOCKDATA routine. Thus, a variable may appear as many as three times in this routine. This goes against standard FORTRAN practice; a variable cannot be reinitiated by a DATA statement. The question is how to change the precompiler such that the generated BLOCKDATA routine includes each variable only once. For now, I have bypassed this problem by deleting the redundant DATA statements from the generated FORTRAN program and have managed to successfully run Sample Problem #1.

The second task I am working on is to obtain plots of the numerical results. DSNP has a potential option which enables the user to output the results in graphic form. This option is not yet available because it is system dependent and each user must develop its own capability. The other complication is due to the fact that the IBM 4341 at GIT does not support any graphics package at this time. Therefore, I plan to use the CALCOMP routines of the CYBER. The procedure would write the results to a file which could then be transferred from IBM to CYBER. Once on the CYBER, there are numerous routines to generate plots.

Explanation of the Sample Problem #1

02/20/84

1. Declarations

In Sup Choi

o. PLIST

will cause data set I1 to be copied to data set I2 at the end of the pre compilation step.

usage:

- PLIST I1 TO I2

o. DDFILE

insert a data set having DSNP data statements into the simulation program

- DDFILE (data set number)

A DSNP data set must be in DSNP data format.

o. There are two kinds of DSNP modules.

One is those that require prior definition, and the other is those that don't require prior definition.

The latter should be identified.

In this problem #1, these are PIPE01 and IHXMS1.

o. PIPELC

Although this pipe segment connecting the lower plenum to the reactor core is not actually shown in the flowchart (p.8), it is needed in order to provide a delay equivalent to the holdup time in the lower plenum.

TE#1: upper plenum inlet temp in °C (LPLEN)
 TE#2: core coolant exit temp in °C (CORTP)
 TE#3: lower plenum exit temp (LPLEN)
 TE#4: upper plenum exit temp. ()
 TE#5: core coolant inlet temp.
 TE#6: IHX primary sodium inlet temp. (IHXM#) Ser. TC#PI of TE#HI secondary coolant
 TE#7: " " exit " (IHXM#) TE#PX TE#HX
 TE#8: lower plenum inlet temp.

PIPE*i* (FIN, FEX, VPLD, EF, ET, G/RO, KK, MP);

where

- FIN: a pipe inlet parameter usually temperature (Variable)
- FEX: a pipe exit value at any chosen time (Variable)
- VPLD: the volume of the pipe (Numeric value)
- EF: the minimum change in the pipe inlet variable initiating a storage operation (Variable or Numeric Value)
- ET: the minimum increase in time since last storage, initiating a new storage operation (Variable or Numeric Value)
- G: the flow rate through the pipe (Variable)
- RO: the fluid density at the pipe inlet (Variable)
- KK: the degree of interpolation used to evaluate the pipe exit value, FEX, if requested at times falling between two tabulated values (Numeric value)
- MP: the number of storage locations to be reserved for the *i* pipe (Numeric value)
- determine the pure time delay between the inlet variable and its outlet value.

ZFLCOR (CORTP) Flow through the core, in kg/s
 FPAI to be defined
 ZROPXA# (IHXM#) H-X. primary side fluid exit density, in kg/m³
 ZROCX (CORTP) Core exit coolant density, in kg/m³

IHXMa1 (IHX, NXW, ZPDI, ZMATP, ZMATH);

where

- a: a single character name given to the heat exchanger (Alphanumeric)
- a is A in Problem #1
- IHX: the number of axial nodes (Numeric value)
- NXW: the number of channels (Numeric value)
- ZPDI: the pitch to diameter ratio (Numeric value)
- ZMATP: the primary side fluid (Alphanumeric)
- ZMATH: the secondary fluid (Alphanumeric)

2. Initial Conditions

- o. The function of this segment is to compute the initial conditions for the dynamic simulation.
- o. The first statement of this segment must be `BEGIN AT TIME`, where `TIME` stands for the actual time at which the dynamic simulation is to start.
- o. The statements should be listed in a logical order according to the flow of the physical quantities.
- o. Seeking Convergence, `CONVERG`.

There are two convergence statements :

- `CONVERGA (Y, ER, LBL, ITMX, P or N) ;`
- `CONVERGR (Y, ER, LBL, ITMX, P or N) ;`

The first statement will seek absolute convergence of the variable `Y` being evaluated while the second statement will seek relative convergence of the variable `Y`.

`ER` : the relative or absolute allowed error between two subsequent iterations.

If the variable `Y` did not converge, control is transferred to the program location having a label `LBL`.

If convergence is not achieved after `ITMX` iterations, the simulation is terminated.

`P` → the printing of each iteration step
`N` → no printing

- CNTRL1 Simulates the Rx. Control Sys.
This module calculates the reactivity generated by the reactor control system and external perturbations.
Only three external modes of reactivity are available to initiate a transient. (a step, a ramp, or an arbitrary function)
- FDBEK1 Simulates the Core Temp. Feedback
This module is a simplified model for calculating the reactivity feedback effects ~~upon~~ given the core average temperatures.
This module is usually used with the lumped parameter core module.
- GAMAR1 Simulates the Core Radioactive Decay Heat
simple lumped parameter model to calculate the power contributed by the radioactive decay of the fission products accumulated in the reactor fuel. steady state contribution WGP0
- NEUTR1 Simulates the Neutron Kinetics
In this module, the neutron kinetics equations with six groups of delayed neutrons are solved.
- TPWR1 Simulates the Core Power Distribution.
calculates the power distribution in the reactor. However for a lumped parameter model, it only converts the normalized reactor power into kW
- CORTP1 Simulates the Core Thermodynamics
represents a simple lumped parameter approximation to the core and blanket TD. The fuel, cladding and coolant are represented by a single node.
Should be used only when detailed temp. distns in the core are not necessary.
- UPLN1 Simulates the Upper Plenum Thermodynamics
metal node + coolant node.
- PIPEII Refer to next page (page 2)
- IHXMA1 Simulates a Counterflow Heat Exchanger Refer to next page.
- PIPEII Refer to next page (page 2)

3. First Dynamic Loop.

SIMULATE LOOP01 STIFF1 [DEBUG] [ALL] [TDV=2]
 ↑ ↑
 identification integration
 number method

identification number

LOOP01 : the first dynamic LOOP

LOOP02 : the second dynamic LOOP
and so forth

integration method (8 kinds)

STIFF 1 : STIFF1 integration method

TRAPZ 1 : TRAPZ1 integration method
etc.

[] stands for "optional"

DEBUG

indicates that the LOOP will be debugged.

ALL

can be inserted only after DEBUG and
indicates that all of the modules in this loop
will be debugged.

TDV = 2

defines the initial time step of LOOP01
to be half the master time step, $DELTM/2$.

Any number of up to four digits can
replace the "2".

The default value is $TDV = 1$.

All of the statements in this segment should be listed in a logical order.

The same sequence as in the initial conditions segment is used.

The "Simulation Loop" should not be confused with a physical loop, although they may be identical. A simulation loop includes a set of elements or differential equations that can be integrated as separate entities in the simulation process and which are usually strongly coupled.

In this sample problem, two SIMULATE segments are used.

This was chosen because the core neutronic and thermodynamic coupling is very "stiff" on the other hand while the thermodynamic processes in the intermediate heat exchanger are slow and decoupled from the core by the two pipe segments UI and IL.

4. Terminal Section

TERMINATE AT 20.0

↑
physical termination time

5. Data Statements and Printing Instructions

- o. Data statements are used to input data into the simulation through the COMMON blocks. They are used to assign values to the DSNP common block variables and cannot be used to input other data.

Examples:

- DATA (PROGR) = A/5.00/, D(5)/3.0-4/, X/1.0, 3.0, 6.0, 7*0.00/;
- DATA (PROGR) = Z(5,6)/3.4/;

After the "DATA(name) =" statement, the data follows in a free format as VARIABLE/DATA/.

Vectors can be assigned values by members, in which case the appropriate index of the vector member is written out explicitly, or a vector can be assigned all its values at once, in which case the numbers are written subsequently separated by commas as in the above example where ten values were assigned to vector X.

All data entries must be separated by commas as shown above.

o. The DSNP printing instructions provide a tabular output in which each column contains the output variables at a certain predetermined time. The printing intervals are determined by the BY-TO statement.

Example: P X, Y, Z(3-5), I

Output:

TIMEM	0.5000 D-02	0.1000 D-01	0.2000 D-01	0.4000 D-01
DELTM	0.5000 D-02	0.5000 D-02	0.5000 D-02	0.5000 D-02
JTIM	0.1000 D+01	0.2000 D+01	0.4000 D+01	0.8000 D+01
X
Y
Z(3)
Z(4)
Z(5)
I

TIMEM : the master time

DELTM : the master time step size

JTIM : the number of master time steps.

o. Defining the printing intervals with BY-TO

BY indicates the size of the printing interval and TO indicates the time span up to which this printing interval is to be used.

<u>NAME</u>	<u>COMBLK</u>	<u>DESCRIPTION</u>
EXRECI	CNTRL	External Reactivity inserted to initiate a transient, in absolute units
TIEXR(I)	CNTRL	can't be found.
IEXR	"	Indicates which type of external reactivity inserted 1-step, 2-ramp, 3-function
DTMAX	TIMER	Maximum time step allowed, in s
DTMIN	"	Minimum "
DELT	"	Current integration time step, in s
LETM	"	1 → use fixed time step, 2 → use variable time step
TIMEDS	"	At this time, printing of requested variables starts at each time step.
TIMEDF	"	At this time, printing of time step ends, in s.
PWD	TPOWR	Initial PWJ
PWJ	"	Thermal Power generated by the reactor, in W
WLA	CORTP	Total heat transfer between the fuel cladding and the core coolant, in W/c
WCA	"	" between the fuel and the cladding, in W/c
CC		Primary coolant specific heat, in J/kg-c
CF		Core fuel specific heat, in J/kg-c
CL		Fuel cladding specific heat
VC		volume of coolant in the core, in m ³
QMF		Total weight of fuel in the core, in Kg
QML		Total weight of cladding in the core, in Kg
IGAM	GAMAR	Number of radioactive FP decay groups assumed in the model.
GAM(I)		Disintegration constant of the Ith FP
BGM(I)		Fraction of the FP produced by the I-th group
GA		Fraction of the decay heat liberated in the coolant
WGPO		Initial WGP, in W
WGP		Power produced by the decay of FP, in W
RKFB	FDBEK	Total feedback reactivity
RKDC		Coefficient of the Doppler reactivity
RKEC		Coefficient of core expansion reactivity
RKSC		Coefficient of Sodium expansion reactivity
VNC (VNC)	LPLEN (LPLEN)	Volume of coolant in the Lower plenum, in m ³ (upper plenum)
WMA (WMA)		Total heat transfer from metal to coolant in the lower plenum, in W/c (UP)
CM (CN)		Specific heat of the metal in the lower plenum, in J/kg-c (UP)
QLPF (QLPF)		Weight of metal in the lower plenum, in Kg (UP)
ARAP	IHXMA	Total IHX primary cross sectional flow area (shell side), in m ²
XARH (XARP)		IHX tube inner (external) radius, in m
RAW		IHX tube metal density, in Kg/m ³
BAW		Thermal conductivity of the IHX tube, in W/m-c
CAW		Thermal conductivity of the IHX tubemetal, in J/kg-c Specific heat
FHAI → FAHI		Secondary IHX flow, in Kg/s
XAWL		IHX tube length, in m
APSHA (EAPSHR)		Maximum absolute (relative) error allowed in the IHX temperature calculation
TIAMX (ITOMX)		Maximum number of inner (external) iterations in the IHX initial temperature calculation.

NAME	COMBLK	DESCRIPTION
DELTL(I)	TIMER	The time step DELT of the Ith integration loop, in S
TIMEL(I)	"	The recent time in the Ith integration loop, in S
LSTEP(I)	DERIV	Number of subimesteps in the Ith loop to integrate from TIME to TIME + DELTM
P NJ	NEUTR	Normalized neutron power at the Jth time step
P WJ	TPOWER	Thermal power generated by the reactor, in W
RKS	FDBEK	Sodium density reactivity feedback
RKD	"	Doppler reactivity feedback
RKE	"	Core expansion reactivity feedback
RKFB	"	Total feedback reactivity
REAK	NEUTR	Total core excess reactivity
SUMC	NEUTR	Sum of $CLJ(I) * ELAMDA(I)$ for $I=1, \dots, 6$
CLJ(I)	"	Delayed precursor concentration at the Jth time step, $I=1, \dots, 6$
ELAMDA(I)	"	Delayed precursor decay constants, $I=1, \dots, 6$
WGN	GAMAR	Total normalized radiation power produced by the FP
WGT	"	Power produced by the decay of FP, in W
TEMPACTENFA)	LPLEN(LPLEN)	Average metal temperature in the lower (upper) plenum, in C
TELA	CORTP	Core cladding average temp., in C
TEFA	"	Core fuel average temp., in C
TEAW	IHXMA	IHX tube wall temp. distribution, in °C

questionnaire

1. The first line of p. 11 is

P ZDELTL(1), ZTIMEL(1), ILSTEP(1), ZDELTL(2),
ZTIMEL(2), ILSTEP(2)

Why aren't they shown in the output?

2. What's the meaning of the scratch file,
default data set and default value?

3. On the 12th line of page 11,

TIMEDS = 50 sec and TIMEDE = 5.3 sec.

Is this reasonable?

4. On the 16th line of page 11

IGAM/41.

I think it should be IGAM/51.

1.1.3. A Sample Solution.

The big advantage of DSNP over any other simulation language is the simplicity by which the above simulation flowchart is directly transformed into a simulation program. This transformation is done here and the resulting DSNP program is shown below. Each of the block names shown in the above diagram appear in the simulation program below. The modules can be identified once in the simulation static part and once in the simulation dynamic part. The simulation program has two dynamic "LOOPS". One uses the STIFF integration method, while the other uses the TRAPEZoidal algorithm. In the static part, two iteration loops can be observed for converging the static calculation. The inner loop iterates until the core exit temperature TECK converges and the external loop iterates until the heat exchanger primary exit temperature TEAPX converges.

A DSNP PROGRAM TO SIMULATE THE LMFBR PRIMARY LOOP.

```
//DSNP1 JOB (F####,07,,4) , 'NAME', PRTY=N, NOTIFY, REGION=250K
ACCOUNTING INFORMATION
//*MAIN ORG=RM019
//STEP1 EXEC FGICIG
//FGI.SYSPRINT DD DUMMY
//SYSIN DD DSN=B####.COMP2.FORT, DISP=SHR
//GO.SYSIN DD *
C*****DSNP REV 02 *** SEPTEMBER 3, 1978 *****C
C
C DSNP SAMPLE PROBLEM #1: THE SIMULATION OF THE KINETICS AND PRIMARY C
C THERMODYNAMICS OF AN LMFBR. TRANSIENTS ARE INITATED BY A STEP CHANGEC
C IN REACTIVITY AT 1.0 SECONDS.
C
C THE FOLLOWING DATA SETS ARE USED WITH THIS SAMPLE PROBLEM: C
C 05 SYSIN INPUT DATA SET, PROGRAM AND DATA. C
C 06 SYSOUT OUTPUT DATA SET, USUALLY LINE PRINTER. C
C 08 &&ISCR5 SCRATCH FILE FOR INPUT DSNP PROGRAM. C
C 10 &&ISCR1 SCRATCH FILE FOR INTEGRATION AND DERIVATIVE. C
C STATEMENTS DURING PRECOMPILATION STEP AND DATA. C
C SCRATCH DURING THE SIMULATION. C
C 13 IDEFD2.DATA * DEFAULT INPUT SIMULATION DATA SET. C
C 20 DSNP2.FORT GENERATED FORTRAN SIMULATION PROGRAM. C
C 21 &&ISCR2 SCRATCH FILE FOR FIRST PASS COMPILATION. C
C 22 &&ISCR3 SCRATCH FILE FOR TEMPORARY I/O STATEMENTS. C
C 23 &&ISCR6 SCRATCH FILE FOR TEMPORARY MAIN AND DERIV ROUT. C
C 24 &&ISCR4 SCRATCH FILE FOR TEMPORARY COMMON BLOCKS. C
C 30 LIB12.DATA THE DSNP LIBRARY. C
C
C * USER CAN SUPPLY ANY NUMBER OF INPUT DATA SETS WITH DIFFERENT C
C NUMBERS (EXCLUDING THOSE ALREADY IN USE) USING THEM FOR INPUT C
C BY DDFILE(II). C
```

```

C
C ALL REAL DATA SHOULD BE GIVEN IN DOUBLE PRECISION'
C***** DECLARATIONS *****C
C
.   FLIST 13 TO 06;
.   PLIST 20 TO 06;
.   DDFILE(13);
.   CNCTCU(TENI,TECX);
.   DFPIPELC(TEMX,TECI,1.5D1,1.D-3,1.0,ZFLCOR/ZROPXA,2,50);
.   DFPIPEUI(TENX,TEAPI,1.D1,1.D-3,1.D0,FPAI/ZROCKX,2,26);
.   DFPIPEIL(TEAPX,TEMI,1.D1,1.D-3,1.D0,FPAI/ZROPXA,2,24);
.   IHXMA1(5,250,1.4,SO,SO);
C***** INITIAL CONDITIONS *****
C
BEGIN AT 0.0
.   1 CONTINUE
.   CNTRL1;
.   FDBEK1;
.   NEUTR1;
.   GAMAR1;
.   FPAI=ZFLCOR
.   2 CONTINUE
.   TPOWR1;
.   LPLEN1;
.   PIPELC;
.   CORIP1(SO);
.   CNCTCU;
.   UFLEN1;
.   PIPEUI;
.   IHXMA1;
.   PIPEIL;
.   CONVERGR(TECX,0.001,2,10,P);
.   CCNVERGA(TEAPX,0.1,1,10,P);
C***** FIRST DYNAMIC LOOP *****
C
SIMULATE LOOP01 STIFF1 IDV=2
.   CNTRL1;
.   FDBEK1;
.   NEUTR1;
.   GAMAR1;
.   TPOWR1;
.   LPLEN1;
.   PIPELC;
.   CORIP1(SO);
.   CNCTCU;
.   UFLEN1;
.   PIPEUI;
SIMULATE LOOP02 TRAPZ1 TDV=1
.   IHXMA1;
.   PIPEIL;
C***** TERMINAL SECTION *****
C
TERMINATE AT 20.0
C***** DATA STATEMENTS AND PRINTING INSTRUCTIONS ****
C

```

```

P ZDELTL (1) , ZTIMEL (1) , ILSTEP (1) , ZDELTL (2) , ZTIMEL (2) , ILSTEP (2) ,
P ENJ , PWJ , RKS , RKD , RKE , RKFB , REAK ,
P SUMC , CLJ (1-6) , WGN , WGP , WGT , TECI , TEMA , TENA , TEMFA , TENFA , TENI , TEMI ,
P TECX , IECA , TELA , TEFA ,
P TEAP (1-5) , TEAH (1-5) , TEAW (1-5) , TEAPI , TEAPX , TEAHI , TEAHX
P BY1.D-3TO1.D-2BY1.D-2TO1.D-1BY1.D-1TO1.D0BY5.D-1TO1.D1BY1.D0TO2.D1
P 'SAMPLE PROBLEM #1: SIMULATION OF THE KINETICS AND PRIMARY '
P 'THERMODYNAMICS OF AN LMFBR.'
. DATA (CNTRL) = EXREC (1) / 3.D-3 / , EXREC (2) / -3.D-3 / , TIEXR (1) / 1.0D0 / , TIEXR (2) /
. 5.D0 / , EXREC (3) / 0.D0 / , TIEXR (3) / 1.D2 / , IEXR / 1 / ;
. DATA (TIMER) = DTMAX / 5.0D-2 / , LETM / 2 / , DELT / 5.D-3 / ,
. DTMIN / 1.0D-7 / , TIMEDS / 5.D1 / , TIMEDE / 5.3D0 / ;
. DATA (TPOWR) = PW0 / 9.D+8 / ;
. DATA (CORTP) = UCA / 3.14D6 / , ULA / 4.689D6 / , CF / 1.047D3 / , CL / 6.28D2 / ,
. CC / 2.512D3 / , VC / 5.D0 / , QML / 1.D3 / , TECI / 3.5D2 / , QMF / 2.D4 / , ZFLCOR / 3.893D3 / ;
. DATA (GAMAR) = IGAM / 4 / , GAM / 1.D-5 , 3.D-3 , 1.D-2 , 4.D-2 , 0.D0 / , BGM / 2.7D-1 ,
. 2.1D-1 , 1.6D-1 , 3.6D-1 , 0.D0 / , GA / 5.0D-1 / , WGP0 / 7.5D7 / ;
. DATA (FDBEK) = RKFB / 0.D0 / , RKDC / -3.D-2 / , RKEC / -4.D-6 / , RKSC / 4.5D-6 / ;
. DATA (LPLEN) = TEMA / 3.5D2 / , TEMI / 3.5D2 / , VMC / 7.5D-1 / , UMA / 2.5D6 / , CM / 8.37D2 / ,
. CLPF / 1.5D3 / ;
. DATA (UPLEN) = TENA / 7.D2 / , TENI / 7.D2 / , VNC / 1.85D0 / , UNA / 3.77D6 / , CN / 8.37D2 / ,
. QUPF / 2.742D3 / ;
. DATA (IHXMA) = TEAPI / 449.7D0 / , TEAHI / 3.D2 / , ARAP / 4.D-1 / , XARH / 1.D-2 / ,
. XARP / 1.3D-2 / , TEAHX / 405.D0 / , TEAPX / 2.D2 / ,
. RAW / 7.8D3 / , CAW / 4.187D2 / , BAW / 8.374D2 / , FHAI / 3150.D0 / ,
. XAWL / 5.5D0 / , EAPSHA / 1.D-1 / , EAPSHR / 5.D-4 / , ITOAMX / 10 / , ITIAMX / 15 / ;
C ***** MACRO DEFINITIONS *****

```

C ***** MACRO DEFINITIONS *****

```

C
/*
//GC.FT08F001 DD DSN=##ISCR5, DISP=(NEW,DELETE), UNIT=SASCR,
// DCB=(LRECL=80, BLKSIZE=2960, RECFM=FB), SPACE=(CYL, 1)
//GO.FT10F001 DD DSN=##ISCR1, DISP=(NEW,DELETE), UNIT=SASCR,
// DCE=(LRECL=80, BLKSIZE=2960, RECFM=FB), SPACE=(TRK, (5, 5))
//GC.FT13F001 DD DSN=B#####.IDEFD2.DAT, DISP=(OLD,KEEP),
// DCB=(LRECL=80, BLKSIZE=2960, RECFM=FB)
//GC.FT20F001 DD DSN=B#####.DSNP2.FORT, DISP=(OLD,KEEP),
// DCB=(LRECL=80, BLKSIZE=2960, RECFM=FB)
//GO.FT21F001 DD DSN=##ISCR2, DISP=(NEW,DELETE), UNIT=SASCR,
// DCE=(LRECL=80, BLKSIZE=12800, RECFM=FB), SPACE=(CYL, 6)
//GC.FT22F001 DD DSN=##ISCR3, DISP=(NEW,DELETE), UNIT=SASCR,
// DCB=*.FT08F001, SPACE=(TRK, (5, 5))
//GC.FT23F001 DD DSN=##ISCR6, DISP=(NEW,DELETE), UNIT=SASCR,
// DCE=*.FT08F001, SPACE=(TRK, (5, 5))
//GO.FT24F001 DD DSN=##ISCR4, DISP=(NEW,DELETE), UNIT=SASCR,
// DCB=*.FT08F001, SPACE=(TRK, (5, 5))
//GC.FT30F001 DD DSN=B#####.LIB12.DAT, DISP=(OLD,KEEP),
// DCB=(LRECL=80, BLKSIZE=12800, RECFM=FB)
/*
//STEP2 EXEC FGICLG, CCND=(5,LT)
//FGI.SYSPRINT DD DUMMY
//SYSIN DD DSN=B#####.DSNP2.FORT, DISP=SHR
//GO.FT10F001 DD DSN=##ISCR1, DISP=(NEW,DELETE), UNIT=SASCR,
// DCE=(LRECL=80, BLKSIZE=2960, RECFM=FB), SPACE=(TRK, (5, 5))
//* OUTPUT DATA SETS FOR DUMP CAN BE INSERTED HERE IF NECESSARY.

```


A Sample Output from a DSNP Simulation.

SIMULATING THE THERMODYNAMICS OF AN LMFBR PRIMARY LOOP WITH DSNP.

DYNAMIC SIMULATOR FOR NUCLEAR POWER PLANTS, (DSNP); REV. 02, SEPTEMBER 19

SAMPLE PROBLEM #1: SIMULATION OF THE KINETICS AND PRIMARY THERMODYNAMICS

I N I T I A L C C N D I T I O N S.

TIMEM	0.0
DELTM	0.5000D-02
JTIM	0.0
PNJ	0.1000D+01
FWJ	0.9000D+09
RKS	0.0
RKD	0.0
RKE	0.0
FKFE	0.0
REAK	0.0
SUMC	0.1607D+05
WGN	0.1000D+01
WGF	0.7500D+08
WGT	0.7500D+08
TECI	0.3955D+03
TEMA	0.3955D+03
TENA	0.6468D+03
TEMFA	0.3955D+03
TENFA	0.6468D+03
TENI	0.6468D+03
TEMI	0.3956D+03
TECX	0.6468D+03
TECA	0.5212D+03
TELA	0.7211D+03
TEFA	0.1020D+04
TEAP (1)	0.6084D+03
TEAP (2)	0.5648D+03
TEAP (3)	0.5153D+03
TEAP (4)	0.4593D+03
TEAP (5)	0.3956D+03
TEAH (1)	0.5960D+03
TEAH (2)	0.5508D+03
TEAH (3)	0.4995D+03
TEAH (4)	0.4412D+03
TEAH (5)	0.3751D+03
TEAW (1)	0.5926D+03
TEAW (2)	0.5469D+03
TEAW (3)	0.4950D+03
TEAW (4)	0.4362D+03
TEAW (5)	0.3694D+03
TEAFI	0.6468D+03
TEAPX	0.3956D+03
TEAHI	0.3000D+03
TEAHX	0.5960D+03

SAMPLE PROBLEM #1: SIMULATION OF THE KINETICS AND PRIMARY THERMODYNAMICS

TIMEM	0.4000D+01	0.8000D+01	0.1200D+02	0.1600D+02	0.2000D+02
DEITM	0.1667D-01	0.1667D-01	0.1667D-01	0.1667D-01	0.1667D-01
JTIM	0.2440E+03	0.5200D+03	0.7850D+03	0.1048D+04	0.1302D+04
PNJ	0.1435D+01	0.8610D+00	0.8933D+00	0.9492D+00	0.9837D+00
PWJ	0.1292D+10	0.7749D+09	0.8040D+09	0.8543D+09	0.8853D+09
RKS	0.4444D-04	0.5523D-04	0.2953D-04	0.1979D-04	0.8323D-05
RKD	-0.1821D-02	-0.1393D-02	-0.5353D-03	-0.1059D-03	0.2763D-04
RKE	-0.2553D-03	-0.1939D-03	-0.7342D-04	-0.1442D-04	0.3755D-05
RKFB	-0.2032D-02	-0.1532D-02	-0.5792D-03	-0.1005D-03	0.3971D-04
REAK	0.9681D-03	-0.1532D-02	-0.5792D-03	-0.1005D-03	0.3971D-04
SUMC	0.2019D+05	0.1657D+05	0.1543D+05	0.1545D+05	0.1573D+05
WGN	0.1027D+01	0.1024D+01	0.1013D+01	0.1006D+01	0.1004D+01
WGP	0.7699D+08	0.7677D+08	0.7594D+08	0.7547D+08	0.7527D+08
WGT	0.7699D+08	0.7677D+08	0.7594D+08	0.7547D+08	0.7527D+08
TECI	0.3956D+03	0.3959D+03	0.3973D+03	0.3990D+03	0.3975D+03
TEMA	0.3958D+03	0.3966D+03	0.3992D+03	0.3978D+03	0.3967D+03
TENA	0.6561D+03	0.6594D+03	0.6529D+03	0.6497D+03	0.6478D+03
TEMFA	0.3958D+03	0.3963D+03	0.3991D+03	0.3980D+03	0.3968D+03
TENFA	0.6468D+03	0.6469D+03	0.6470D+03	0.6470D+03	0.6470D+03
TFNI	0.6665D+03	0.6710D+03	0.6581D+03	0.6520D+03	0.6485D+03
TEMI	0.3959D+03	0.3964D+03	0.3992D+03	0.3976D+03	0.3966D+03
TECX	0.6665D+03	0.6710D+03	0.6581D+03	0.6521D+03	0.6485D+03
TECA	0.5310D+03	0.5334D+03	0.5277D+03	0.5256D+03	0.5230D+03
TELA	0.7520D+03	0.7483D+03	0.7326D+03	0.7253D+03	0.7219D+03
TEFA	0.1083D+04	0.1068D+04	0.1038D+04	0.1023D+04	0.1019D+04
TEAP (1)	0.6096D+03	0.6211D+03	0.6170D+03	0.6125D+03	0.6103D+03
TEAP (2)	0.5658D+03	0.5754D+03	0.5726D+03	0.5686D+03	0.5667D+03
TEAP (3)	0.5162D+03	0.5238D+03	0.5219D+03	0.5186D+03	0.5171D+03
TEAP (4)	0.4599D+03	0.4652D+03	0.4641D+03	0.4618D+03	0.4606D+03
TEAP (5)	0.3960D+03	0.3989D+03	0.3984D+03	0.3970D+03	0.3963D+03
TEAH (1)	0.5971D+03	0.6085D+03	0.6047D+03	0.6002D+03	0.5983D+03
TEAH (2)	0.5518D+03	0.5612D+03	0.5585D+03	0.5547D+03	0.5529D+03
TEAH (3)	0.5003D+03	0.5077D+03	0.5059D+03	0.5028D+03	0.5013D+03
TEAH (4)	0.4419E+03	0.4470D+03	0.4459D+03	0.4437D+03	0.4426D+03
TEAH (5)	0.3755D+03	0.3781D+03	0.3777D+03	0.3765D+03	0.3759D+03
TEAW (1)	0.5938D+03	0.6048D+03	0.6010D+03	0.5967D+03	0.5945D+03
TEAW (2)	0.5479E+03	0.5570D+03	0.5544D+03	0.5506D+03	0.5488D+03
TEAW (3)	0.4958D+03	0.5029D+03	0.5012D+03	0.4982D+03	0.4967D+03
TEAW (4)	0.4367D+03	0.4416D+03	0.4406D+03	0.4384D+03	0.4374D+03
TEAW (5)	0.3697D+03	0.3720D+03	0.3716D+03	0.3705D+03	0.3700D+03
TEAPI	0.6483D+03	0.6613D+03	0.6558D+03	0.6510D+03	0.6486D+03
TEAPX	0.3960D+03	0.3989D+03	0.3984D+03	0.3970D+03	0.3963D+03
TEAHI	0.3000D+03	0.3000D+03	0.3000D+03	0.3000D+03	0.3000D+03
TEAHX	0.5971D+03	0.6085D+03	0.6047D+03	0.6002D+03	0.5983D+03



Georgia Institute of Technology

A UNIT OF THE UNIVERSITY SYSTEM OF GEORGIA
SCHOOL OF MECHANICAL ENGINEERING

June 22, 1984

Please reply to:

NUCLEAR ENGINEERING AND
HEALTH PHYSICS PROGRAM
CHERRY EMERSON BUILDING
GEORGIA INST. OF TECH.
ATLANTA, GEORGIA 30332 U.S.A.

MEMORANDUM

TO: D. G. Cacuci and C. R. Weisbin, ORNL

FROM: J. M. Kallfelz, L. A. Belblidia, L. S. Choi and
J. F. King

SUBJECT: Progress Report for ORNL Subcontract 7802
Months of March, April and May, 1984

Accomplishments During Report Period

- o The initiation of the simulation language code system DSNP (Dynamic Simulator for Nuclear Power Plants)¹⁻³ on the Georgia Tech IBM 4341 continued. Using version 3.3 of this code system, various sample problems³ supplied with the code system have been run and analyzed (see Attachment A). Three of these problems had been also run at ANL by L. Belblidia during the DSNP Workshop⁴. After changes to our version of DSNP to make it compatible with the ANL⁵ version 3.4, the results for the test case runs at the two sites were the same⁵.
- o The reactor transient code RELAP3B⁶ has been received from the USNRC⁷, and initiated on the Georgia Tech CYBER 170/835-855⁸. The test case supplied with the code has been successfully executed. Input for a "generic" (non-proprietary) model typical of a Westinghouse PWR⁹ has been obtained from BNL¹⁰.

Plans for Work for Next Month

- o A newer version of DSNP will be initiated on the Georgia Tech computer and work will be started on improving a PWR model for this code system.
- o Using the "generic PWR model" input^{9,10}, we will initiate work on reactor transient calculations with RELAP3B. It is planned to use these calculations as a benchmark for DSNP results.

References

1. D. Saphier, "A Special Purpose Simulation Language for Nuclear Power Plants", in Simulation of Systems '79, L. Dekker et al., Eds., North-Holland Publishing Co., p. 1055, 1980
2. D. Saphier and J. T. Madell, "The DSNP Simulation Language and Its Application to LMFBR Transient Analysis", Nucl. Tech., 56, p. 493, March 1982
3. D. Saphier, "The Simulation Language of DSNP", ANL-CT-77-20, Rev 02, Argonne National Laboratory, September 1978
4. J. M. Kallfelz, L. A. Belblidia and L. S. Choi, "Progress Report for ORNL Subcontract 7802, Months of January and February, 1984", memorandum to D. G. Cacuci and C. R. Weisbin (ORNL), School of Mechanical Engineering, Georgia Institute of Technology, March 16, 1984
5. In Sup Choi, "Progress Report; DSNP: Dynamic Simulator for Nuclear Power Plants", Georgia Tech internal memo to J. Kallfelz dated June 1, 1984
6. BNL-NUREG-22011, "User's Manual for RELAP3B-MOD 110, "A Reactor System Transient Code", Department of Applied Science, Brookhaven National Laboratory
7. L. Shotkin, USNRC, Washington, telegraphic message to W. Kato, BNL, dated May 3, 1984
8. J. King, "RELAP3B-MOD 110: Theory and Models Used; Implementation and Testing of Source Code on the Georgia Tech CYBER 170/835-855", Georgia Tech internal report to J. Kallfelz dated June 6, 1984
9. "ATWS for LWRs", NUREG-0460, U.S. Nuclear Regulatory Commission, April 1978
10. W. Shier, BNL, private communication to J. Kallfelz, May 1, 1984

(1) fundamental concepts of DSNP

DSNP is a special purpose block-oriented simulation language,⁽¹⁾⁻⁽⁴⁾ which was developed at ANL.

Although the execution times of the programs are less favorable with the high level language, this can be offset by the reduction in computer time usage during the program development phase.⁽⁵⁾ The advantage of higher level simulation language is obvious from Table 1.

Table 1. Comparison of size and time required for a particular program using different computer languages⁽³⁾

	Number of statements	Approximate programming time	Statements per DSNP statement
ASSEMBLER	22000	1 year	400
FORTRAN	1760	1 month	34
CSMP	630	2 weeks	12
DSNP	52	1 day	1

Because most programs developed so far or developing are for a particular system with a specific set of transients inducing perturbations, these programs are rather inflexible and applicable only to a particular system and to a limited set of situations.^(12, 16)

These are the incentives for developing a modular approach to the simulation of nuclear power plants.⁽³⁾

The DSNP language is in principal a modular approach to the problem of a power plant simulation.⁽³⁾ The philosophy to develop it was to facilitate the development of safety-oriented dynamic simulations for a large variety of nuclear power plants.⁽⁷⁾ The major advantage of DSNP is the simplicity with which a power plant can be simulated, the small number of statements necessary to write a simulation program, and the inherent flexibility that permits the simulation of a large variety of systems.⁽⁸⁾

(2) Progress

In the first quarter, we run four sample problems: Sample Problems #2, #3, #4, and #6. Sample Problem #1 had been already run before this work by Dr. Belblidia. All the five reports are attached with appropriate correction, some of which were already turned. They are:

Appendix 1: Explanation of the Sample Problem #1.

Appendix 2: " " #2

Appendix 3: " " #3

Appendix 4: " " #4 (see p. A-4)

Appendix 5: " " #6 (see p. A-5)

We have three "standard outputs" which Dr. Belblidia brought from the DSNP Workshop, Feb. 21-22, 1984 by Argonne National Laboratory (ANL). They are for Sample Problems #1, #2, and #6. The first one had been already verified. The others has been verified during this work (See Appendices 2 and 5). We can know that the DSNP program with out IBM system is O.K.

DSNF Sample Problem #4.

This sample problem simulates a hydraulic network. The DSNP DEFHPIP defining a hydraulic pipe segment, DFPUMP defining a pump, DFLOW defining a flow path, DFJUNC defining a hydraulic junction, and DFCAVT defining a cavity are used. The hydraulic network is composed of two separate branches or loops. Each loop starts at the cavity CAVT01 and consists of two pipe segments and a pump. The two flows FLOW01(WC1) and FLCW02(WC2) are then combined at a junction into a single flow FLOW03(WC) that reenters the cavity at the bottom. WC1 flows through pipe segments 1 and 2 and is driven by PUMP01. flow WC2 flows through pipe segments 3 and 4 and is driven by PUMF02. The combined flow WC flows through pipes 5 and 6 from the junction JUNC01 back to the cavity CAVT01. A transient is induced into the hydraulic loop^⓪ by increasing the temperature in flow path FLOW03 as a function of TIME² and^⓪ by initiating a ramp like valve closure at five seconds after the simulation starts. Note the COMON data blocks defined in this simulation for variables not defined by the system.

(From ref. 3, 22 June 84 ORNL Progress Report)

DSNP Sample Problem #6-

This sample problem can be considered a combination of sample problems #3 and #4. In this simulation, the primary loop thermodynamics of an LMFBR is combined with the hydrodynamic representation of a double loop sodium coolant system. The reactor initial power is arbitrarily set to 1000 MWt, and the system transients are initiated at one second after simulation start time by a step increase in core reactivity.

The major system components are the reactor with an upper and lower plenum and an intermediate heat exchanger with constant secondary inlet temperature. The heat exchanger is simulated for one loop only. In the second coolant loop, the temperatures are assumed to be constant and equal to their initial value.

The hydraulic loops or branches start at the cavity CAVT01, which is the hydraulic representation of the upper plenum. Each branch leaving the cavity includes two hydraulic pipe segments and a pump. FLOW01 passes through pipes 01 and 02 which also represents the hydraulic inertance of the heat exchanger and through PUMP01. FLOW02 passes through pipes 03 and 04 and PUMP02. Both flows are mixed at the inlet to the lower plenum in a hydraulic junction JUNCO1. Compare the mixing plenum definitions with sample problem #3, which has only one inlet flow to the lower plenum. FLOW03 is the flow leaving the hydraulic junction through the lower plenum to the core and into the upper plenum or the hydraulic cavity.

To compute the system initial conditions, three iteration "loops" are constructed. The innermost loop consists of the thermodynamic and hydrodynamic components and requires absolute convergence of the core exit temperature TEEX. The intermediate iteration "loop" includes the same components, however, relative convergence of the value of the flow through the core ZFLCOR is required. The outermost iteration "loop" includes all of the system components and requires absolute convergence of the IHX primary side coolant exit temperature TEAPX.

The transients in this simulation are initiated by a step change in the core reactivity at one second after the simulation starts.

(From ref. 3, 22 June 84 ORNL Progress Report)

REFERENCES FOR APPENDIX A

References

1. Saphier, D., "DSNP - A New Approach to Simulate Nuclear Power Plants", Argonne National Laboratory, CONF - 770728 - 2 (1977)
2. Saphier, D., "The Simulation Language of DSNP: Dynamic Simulator for Nuclear Power-Plants", Argonne National Laboratory, ANL-CT-77-20, Rev. 02 (1978), p.134
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4. Saphier, D. and Madell, J.T., "The DSNP Simulation Language and Its Application to Liquid-Metal Fast Breeder Reactor Transient Analyses", in Nuclear Technology, Vol. 56, Mar. 1982, p. 493
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E-26-B04

Georgia Institute of Technology
A UNIT OF THE UNIVERSITY SYSTEM OF GEORGIA
SCHOOL OF MECHANICAL ENGINEERING

August 24, 1984

Please reply to:
NUCLEAR ENGINEERING AND
HEALTH PHYSICS PROGRAM
CHERRY EMERSON BUILDING
GEORGIA INST. OF TECH.
ATLANTA, GEORGIA 30332 U.S.A.

MEMORANDUM

TO: D.G. Cacuci, ORNL

FROM: *JMK* J.M. Kallfelz, *J. Saphier* D. Saphier, L.A. Belblidia *DA*
J.N. Davidson, L.D. Koffman, and J.F. King

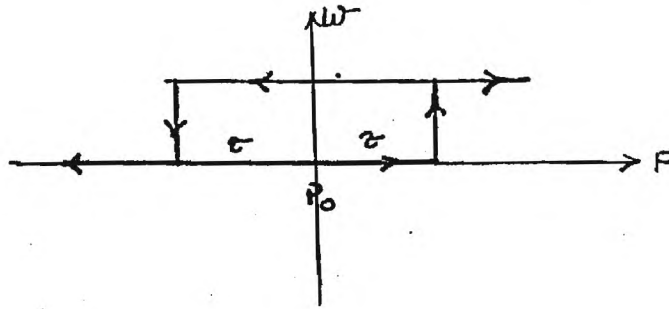
SUBJECT: Progress Report for ORNL Subcontract 7802
Months of June, July and August 1984

Accomplishments During Report Period and Plans for Future Week

1. DSNP¹⁻³ Revision 4.1 was installed at Georgia Tech on the IBM 4341 computer. Necessary modifications have been made to enable operation in the Ga. Tech. environment. These included:
 - Code modification for compatibility with the VS compiler (the only one available at Ga. Tech.)
 - Installation of a clock routine on the IBM 4341
 - Program control language modification to be compatible with the Ga Tech Environment.
2. Some of the DSNP Sample problems in the area of PWR, HTGR and LMFBR simulation were executed.
3. A DSNP nodalization diagram to simulate a generic Westinghouse PWR⁴⁻⁶ -data based mostly on the Trojan PWR - was developed and is shown in fig. 1. This nodalization scheme was made to be equivalent to an available RELAP-3B diagram shown in fig. 2.
4. The Thermal-hydraulic code RELAP 3B was installed at Ga. Tech. and successfully compiled. The initial part of the sample problem was executed up to 100 sec - the results are being analysed and the run will be continued. Due to excessive computer time required to run the RELAP code it will be only used as a backup and verification tool for DSNP.

5. A new pressurizer model for the DSNP was developed and included in the DSNP level 2 library. The model has the following features.
 - a. Two regions (water and steam) are treated separately, nonequilibrium conditions can exist in the pressurer, and level tracking is provided.
 - b. The pressurizer can be flooded (becomes solid) with the steam region disappearing.
 - c. The pressurizer can be emptied - water region disappears.
 - d. Steam flashing is permitted during appropriate conditions and a bubble rise model was included.
 - e. Condensation is permitted under the appropriate conditions and an exponential condensation model was included.
 - f. Spray and heater controls are provided.
 - g. Relief valve (PORV) and safety valve flow are determined by any desirable flow equation or correlation, and the valve area - water level interaction is precisely calculated from the given valve area and elevation.
6. The pressurizer response under extreme operating conditions was tested by combining the pressurizer surge line to a single large cavity having the volume of the primary coolant loop and a temperature transient was forced on the flow entering the volume. This temperature transient is equal to the primary loop temperature transient during a "Loss of off-site Power Transient". The pressure during the transient increases significantly as shown in fig. 3, and both the PORV and Safety valves are activated as seen in fig. 4. The Pressurizer becomes solid at 41s and the steam dome is reestablished at 460s into the transient. The temperature transient in the primary hot leg (driving the transient) is shown in fig. 5 (curve A) and the resulting primary loop pressure is shown in fig. 6. The pressurized surge line flow (A) and combined PORV and safety valve flow (B) are shown in fig. 7. The PORV operating point was set to 16.2MPa and the safety valve was set to 17.34 MPa.
7. With the above program (designated PWRI REV4P1) preliminary sensitivity studies were initiated. The PORV response characteristics and their effect on the primary loop was investigated. The PORV response is fast, it is assumed to be fully open when the pressure in the pressurizer is above 16.2 MPa, and it is assumed to be fully closed when the pressure is below 16.2 MPa. These characteristics cause the valve to oscillate rapidly and maintain a constant pressure of 16.2 MPa in the pressurizer until it becomes solid. Then the PORV capacity becomes insufficient to maintain a constant pressure and the pressure increases until the safety valves open.

The operating characteristics of the PORV were changed to provide some hysteresis in its operation, i.e. open at $P_o + \tau$ and close at $P_o - \tau$.



The values of τ chosen were 100 Pa, 10^4 Pa and 10^5 Pa. The number of oscillations was reduced significantly as the hysteresis was increased. This had a significant effect of reducing computer time with increasing hysteresis

$\tau = 10^2$ Pa	CPU=	5948 for 600s simulation
$\tau = 10^4$ Pa	CPU=	4979 for 600s simulation
$\tau = 10^5$ Pa	CPU=	1936 for 600s simulation

Then the valve opening was assumed to be a linear function of the pressure in the range of 16.2 MPa and 16.21 MPa. This caused the valve response to be stable, the pressure changed slowly between 16.2 and 16.21, and the flow became a time average of the previous cases. The execution time was however reduced by a factor of 10 and more (CPU time for 600s simulation was 485s.) The two types of responses are compared in figs. 8 and 9. All the other system state variables were not sensitive to the relief valve mode of operation. It is therefore recommended that in the future simulations all valve response characteristics be set for a ramp-like response to save computer time.

8. A two day DSNP workshop was conducted at Georgia Tech - with about 10 participants from Ga. Tech. and outside the campus. The workshop covered all aspects of DSNP programming and model development. The workshop enabled all members of the research team and some students to use DSNP. Attachment A is an outline of the workshop.
9. The lumped parameter steam generator (SG) model of DSNP was remodeled and improved by:
 - a. Adding a steam dome model to permit nonequilibrium condition.
 - b. Water level tracking was included.
 - c. Improved heat transfer correlations were included to account for both flow and water level variations.
 - d. The new model permits SG flooding and dryout phenomena in the U-tube SG to account for extreme operating conditions.

10. A quasistatic flow model was added to DSNP. This often reduces significantly the computation time by eliminating the integration of the momentum equation. The momentum equations have the smallest eigenvalue in a typical PWR thermohydraulic system and introduce a large degree of stiffness into the differential equations.
11. The Karman Nikuradze correlation used to calculate the friction factor, was replaced by an explicit formula by Techo, Tickner and James. This also gives significant time saving in solving hydraulic network problems.
12. A feature to permit on-line calculation was added to DSNP - the user can now observe the progress of his simulation and display online the transients of some of the system state variables.
13. The DSNP uses the Helmholtz on Gibbs free energy equations to produce water and steam material properties with great accuracy. However this is a very time-consuming process. A new DSNP module was developed which upon first entry will generate the tables of all the steam water properties and some of their derivatives, below and above the saturation line and also beyond the critical pressure. Preliminary results are very encouraging and computer calculational times are reduced to about 1/3 their values without the table method. The module is in the final testing stages, and if operation proves successful with a variety of programs, it will be included in the DSNP Level-2 library.
14. A full scale PWR simulation program in DSNP according to the nodalization diagram shown in fig. 1 has been developed. This program is now in an advanced stage of development and will be used to simulate operational transients and ATWS accidents. When the transient has been calculated and compared to RELAP calculation, sensitivity studies will be performed. All the transients sensitivities will be investigated with regards to modelling and theoretical assumptions made and the accuracy of the data used in the study.
15. On 10 August, J. Kallfelz, D. Saphier and J.N. Davidson visited the TVA Operations Training Center at Daisy, TN. We spoke to Dr. R.J. Johnson, director of the center, and with Mr. Fletcher Gibbs, an engineer responsible for maintaining and programming TVA nuclear and fossil power plant simulators. (Incidentally, both individuals are Georgia Tech graduates.) Dr. Johnson was quite interested in joint work on reactor operations simulation, and is willing to commit some of his manpower for cooperative work. This topic is of particular interest because of new NRC requirements concerning simulator licensing, and the need for a computer code system to validate the simulator results.

xc: Dr. John Brighton, Director ME

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5. W. Shier, BNL, private communication to J. Kallfelz, May 1, 1984
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7. Trojan Nuclear Plant. Final Safety Analysis Report, Vol. 4. Docket-50344-41, Portland General Electric Co., 5 Feb 1973.
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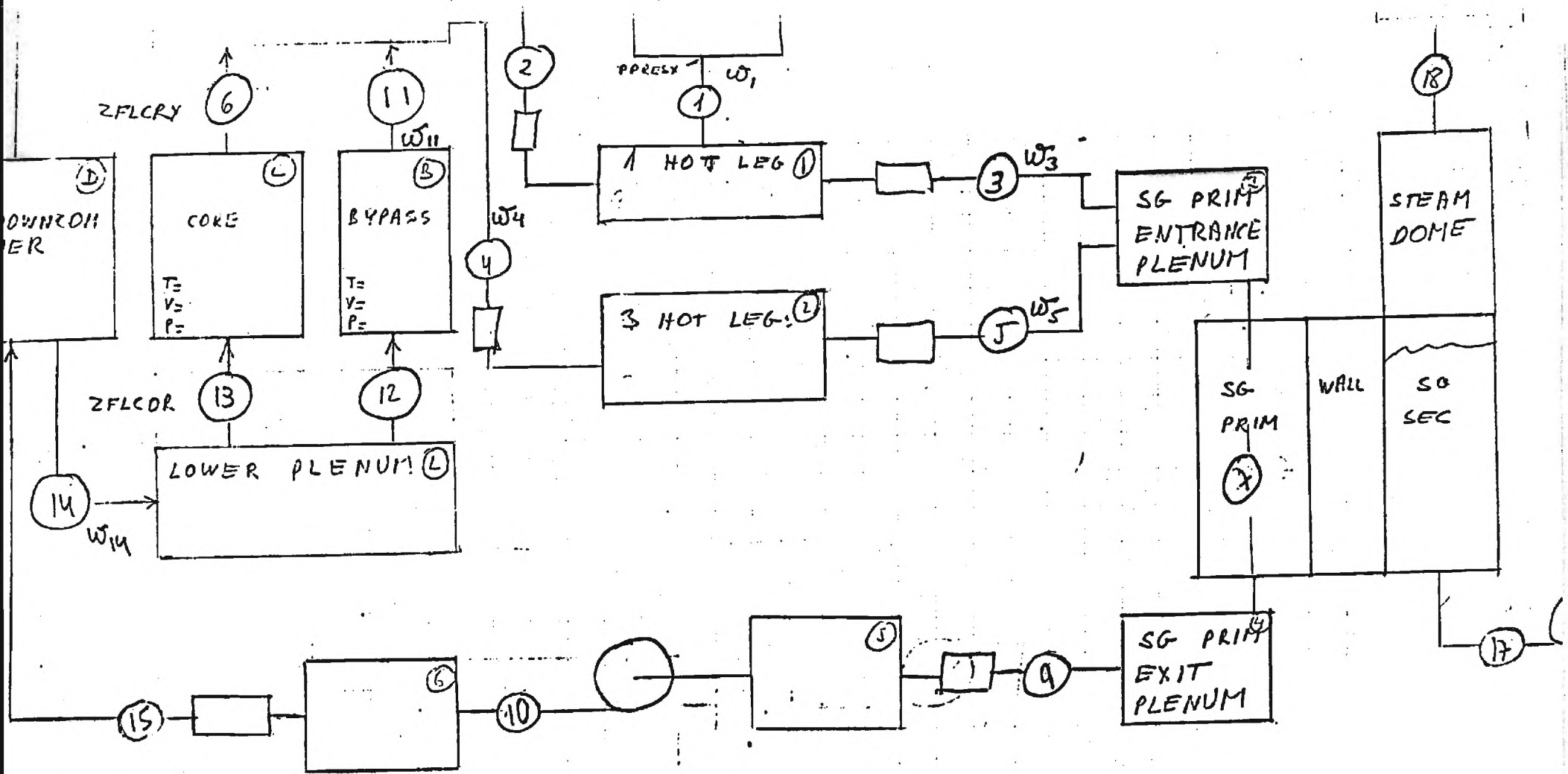


Fig. 1

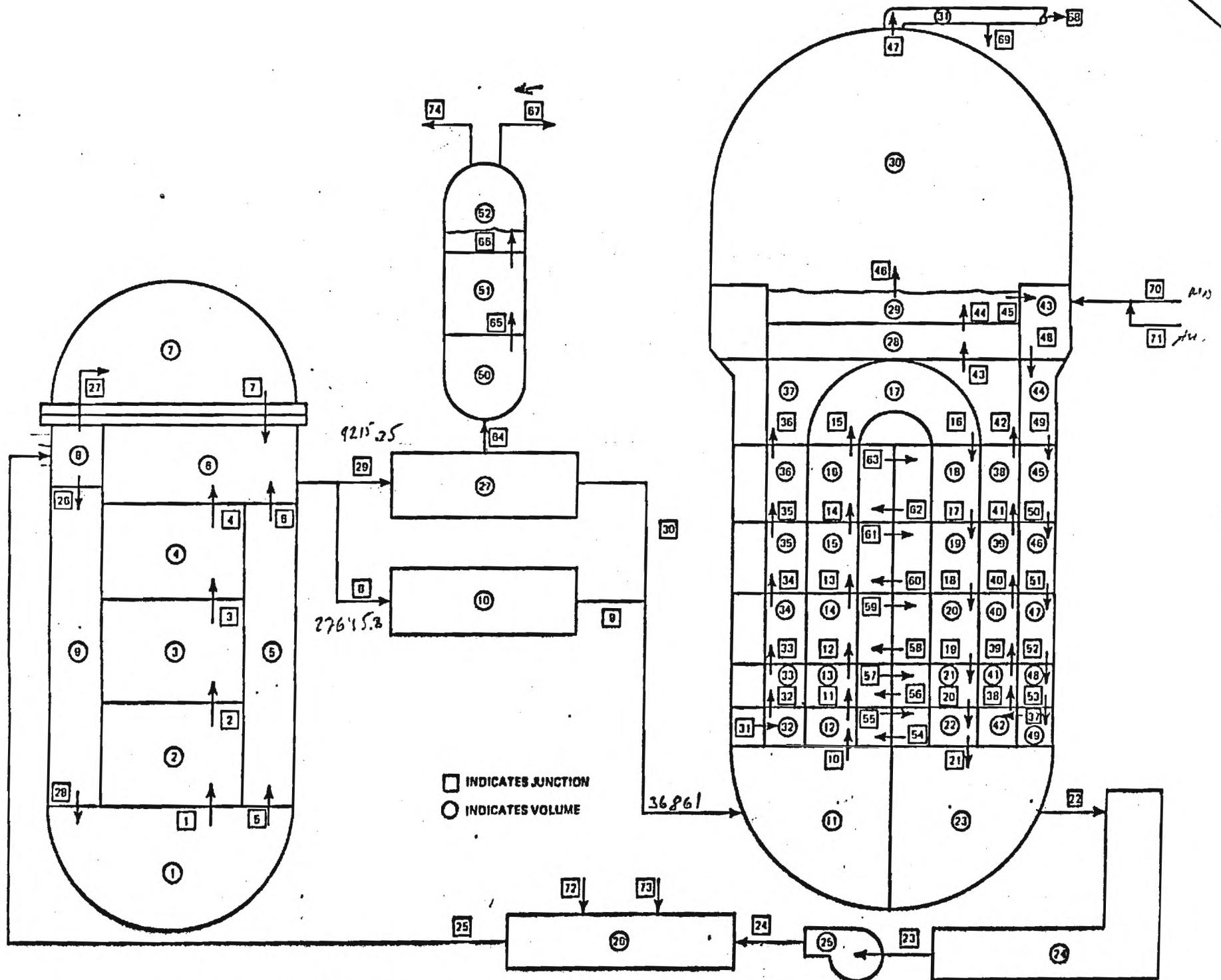
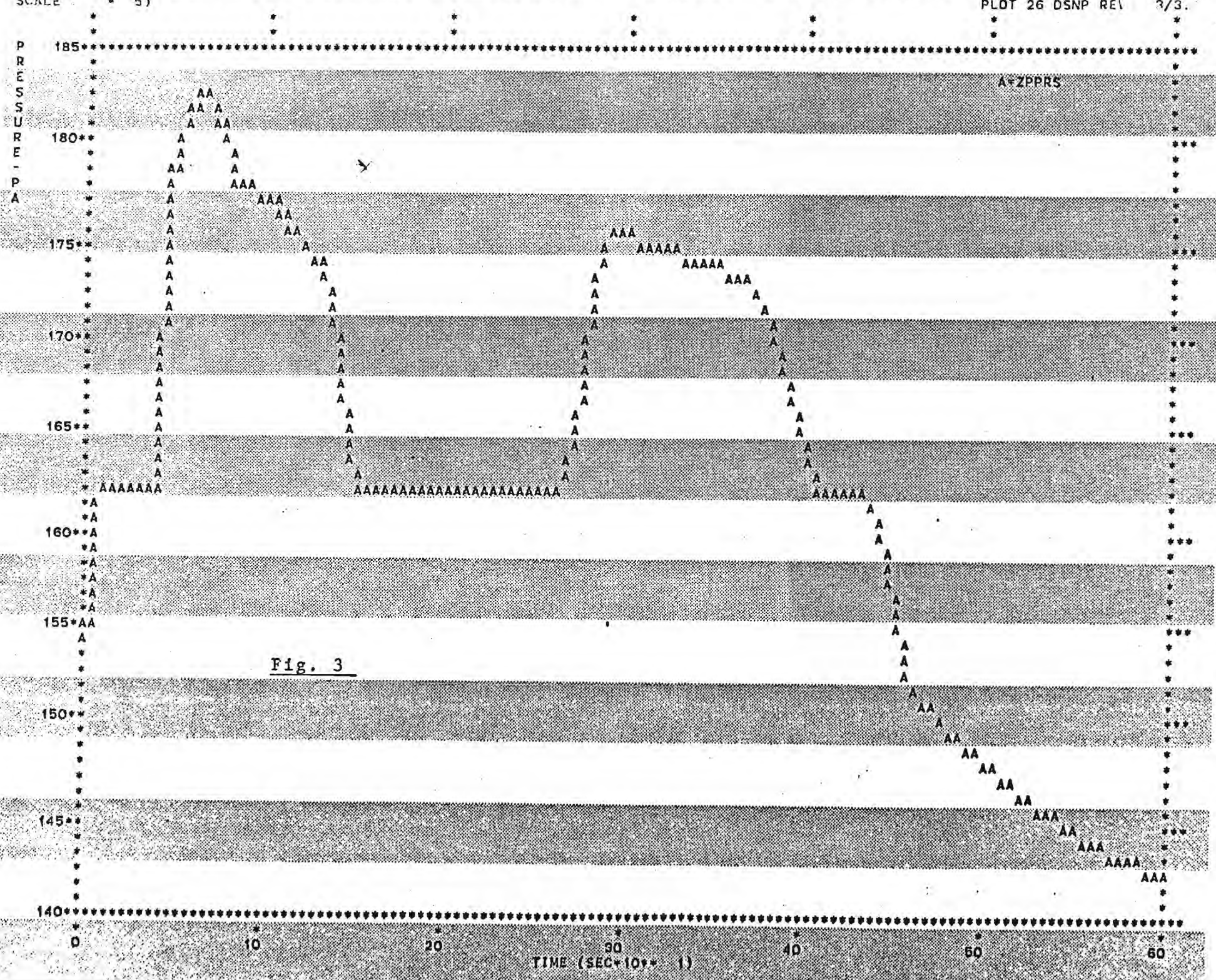


FIGURE 2. RELAP3B NODALIZATION MODEL FOR W PLANT (from ref, 4)



SCALE (* 1)

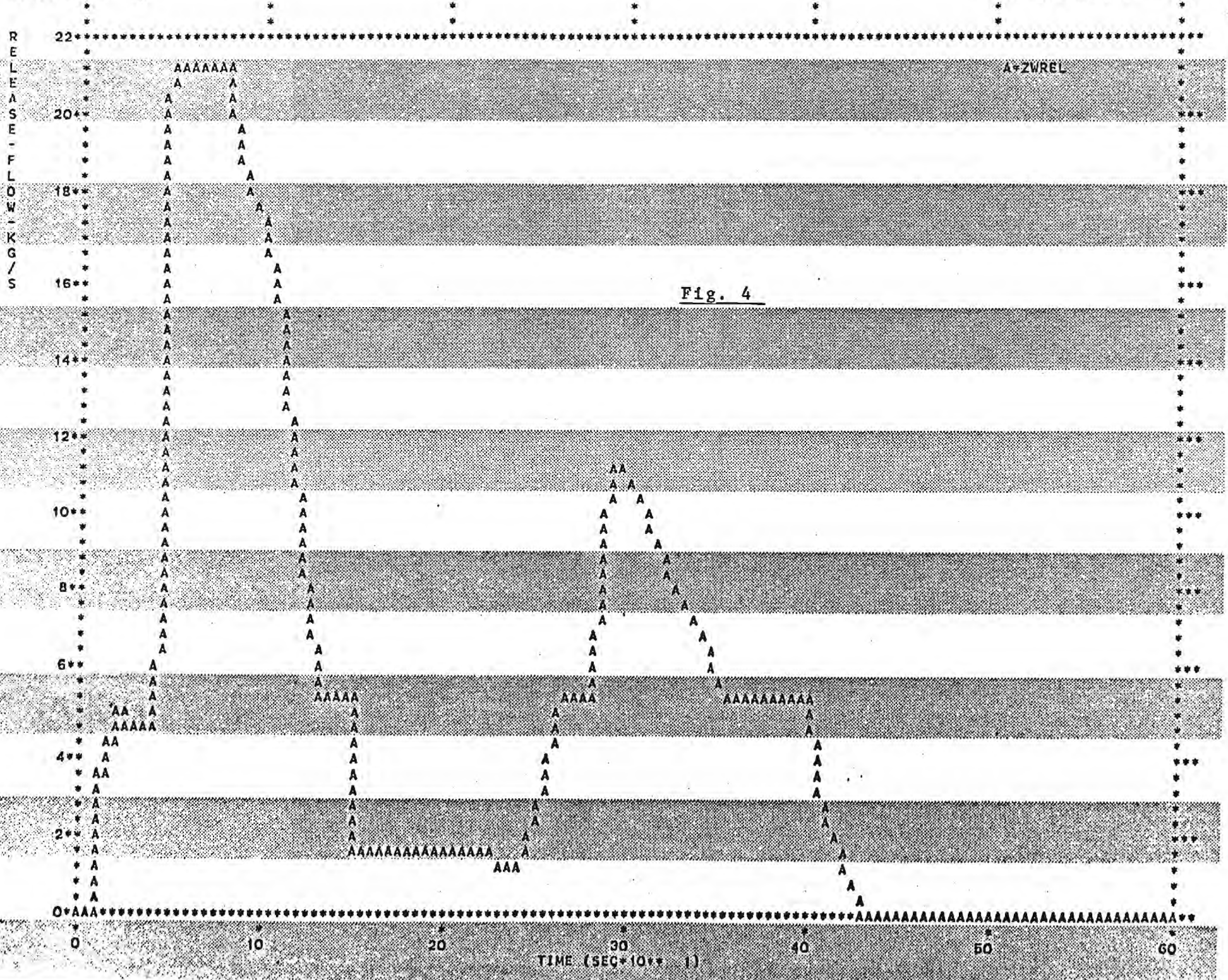


Fig. 4

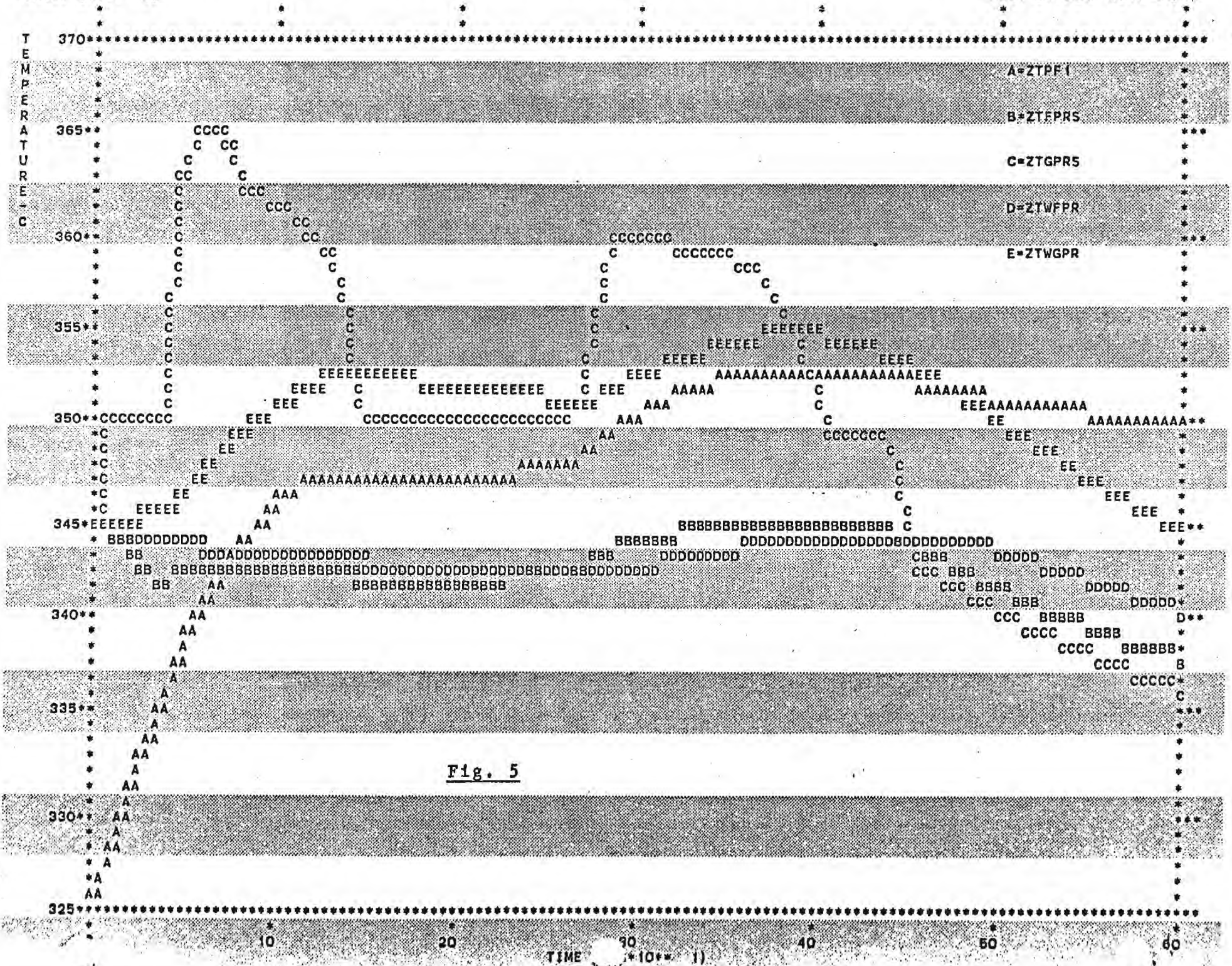


Fig. 5

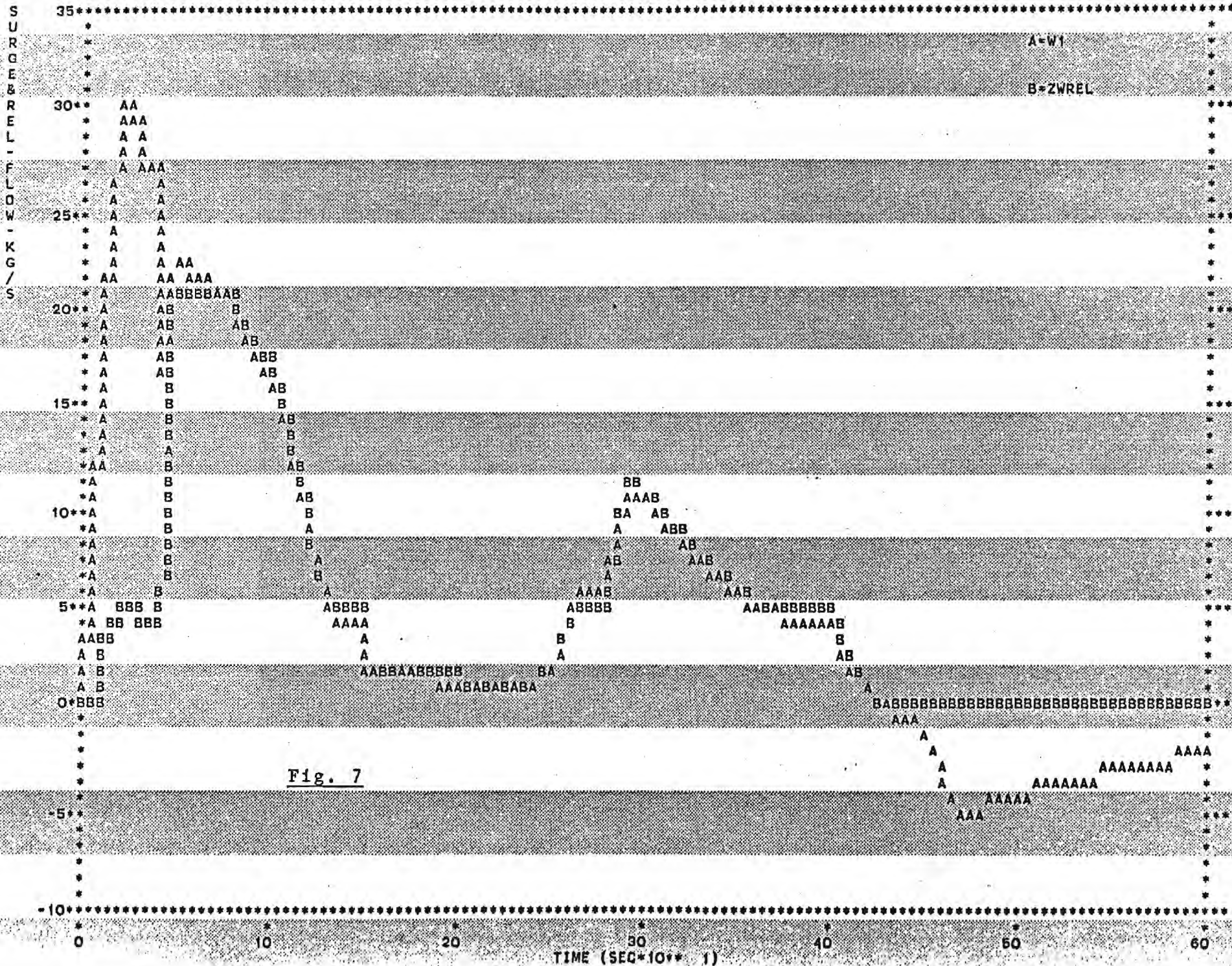


Fig. 7

SCALE (10** 0)

PLOT 3 DSNP REV. 03/3.

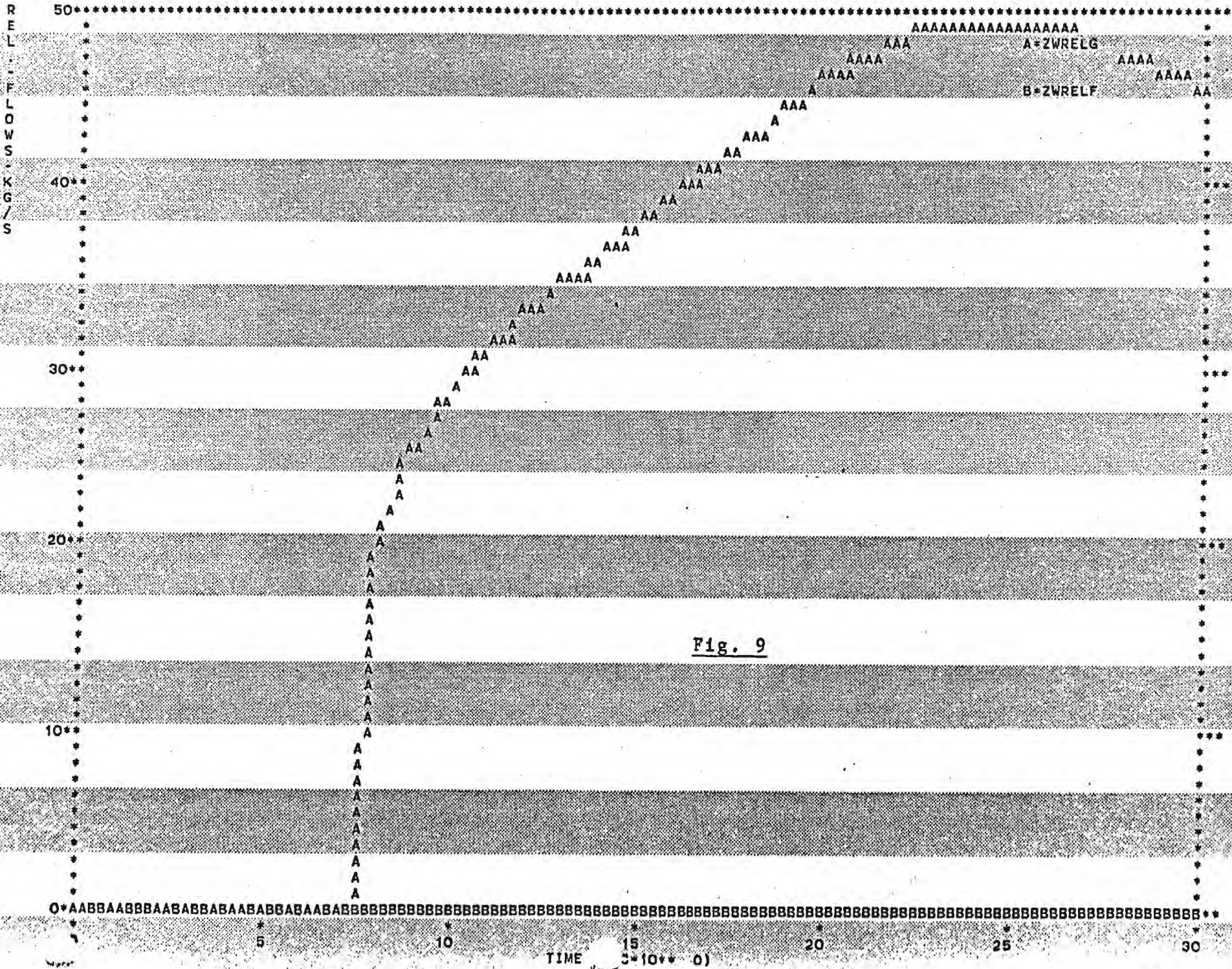


Fig. 9

MECHANICAL ENGINEERING/NUCLEAR ENGINEERING

A workshop on THE DSNP SIMULATION LANGUAGE will be held by

David Saphier
Visiting Professor

School of Mechanical Engineering, Nuclear Engineering Program

DATE: July 16 & 17
TIME: 9:00 a.m. to 4:00 p.m.
PLACE: Emerson Building, Room 206

GENERAL

The DSNP (Dynamic Simulator for Nuclear Power-Plants) is a special purpose block oriented simulation language by which a variety of nuclear power plants can be simulated. It is a modular modeling system by which safety oriented transient analysis of power plants can be performed. Its application to transient analysis of various power plants will be demonstrated. It is presently being applied to PWR ATWS (Anticipated Transients Without Scram) accident analysis. Although being a special purpose simulation language, it has also been used successfully in general purpose application in simulating dynamic systems and in solving large sets of differential equations. At the workshop a detailed description of DSNP will be presented and the participant will learn how to use the simulation language and how to develop modules for power plant components.

TENTATIVE PROGRAM

MONDAY - July 16

9:00 - 9:15	Introduction
9:15 - 9:35	Structure of a DSNP Program
9:35 - 10:45	General Description of the DSNP Simulation Language
10:45 - 11:00	COFFEE BREAK
11:00 - 11:20	Structure of the DSNP System

11:20 - 12:35 The DSNP Components Library
12:35 - 1:35 LUNCH
1:35 - 2:30 RELAP Modeling Methodology
2:30 - 3:00 INPUT/OUTPUT and Data Bases in DSNP
3:00 - 3:20 COFFEE BREAK
3:20 - 4:00 Integration Schemes in DSNP

TUESDAY - July 17

9:00 - 9:20 Use of Material Property Functions and the Material Property Library
9:20 - 10:00 The DSNP Hydraulic Network Solver
10:00 - 11:15 The DSNP Macro Processor
COFFEE BREAK
11:15 - 11:45 How to Develop a DSNP Module
11:45 - 12:30 Development of a Full Scale Simulation Program of a Power Plant - The Simulation Process Methodology
12:30 - 1:30 LUNCH
1:30 - 1:50 DSNP Debugging Features
1:50 - 2:10 A Simple Loop Example
2:10 - 2:30 COFFEE BREAK
2:30 - 4:00 Several Sample Problems will be Analyzed and Discussed.