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High-Temperature Gas-Cooled Reactor Safety Studies for the Division of Reactor Safety Research Quarterly Progress Report, April 1–June 30, 1977

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J. C. Cleveland
J. C. Conklin
J. P. Sanders

Prepared for the U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
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HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF REACTOR SAFETY RESEARCH QUARTERLY PROGRESS REPORT, APRIL 1—JUNE 30, 1977

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CONTENTS

Page

PRIOR HTGR SAFETY REPORTS v

FOREWORD. vii

ABSTRACT. 1

1. HTGR SYSTEMS AND SAFETY ANALYSIS 1

1.1 Development of the FSV Nuclear Steam Supply System Simulation Code (ORTAP-FSV) 1

1.2 Analysis of FSV Postulated Rod-Withdrawal Accidents Using the ORTAP-FSV code 4

1.3 University of Tennessee Subcontract. 13

1.3.1 Introduction 13

1.3.2 Evaluation of modeling assumptions for an HTGR steam generator 13

1.3.3 Detailed nonlinear model of the FSV steam generator 14

1.3.4 Nonlinear model for FSV 15

2. COMPUTER CODE IMPLEMENTATION 17

REFERENCES. 19

Prior HTGR Safety Reports

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Sept. 30, 1974	ORNL/TM-4798
Dec. 31, 1974	ORNL/TM-4805, Vol. IV
Mar. 31, 1975	ORNL/TM-4914, Vol. IV
June 30, 1975	ORNL/TM-5021, Vol. IV
Sept. 30, 1975	ORNL/TM-5128
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Sept. 30, 1976	ORNL/NUREG/TM-66
Dec. 31, 1976	ORNL/NUREG/TM-96
Mar. 31, 1977	ORNL/NUREG/TM-115

Related Topical Reports Published During the Current Fiscal Year

J. C. Cleveland, CORTAP: A Coupled Neutron Kinetics—Heat Transfer Digital Computer Program for the Dynamic Simulation of the High Temperature Gas-Cooled Reactor Core, ORNL/NUREG/TM-39 (January 1977).

FOREWORD

HTGR safety studies at Oak Ridge National Laboratory (ORNL) are sponsored by the Division of Reactor Safety Research, which is part of the Office of Nuclear Regulatory Research of the Nuclear Regulatory Commission.

This report covers work performed from Apr. 1 to June 30, 1977. Previous quarterly reports are listed on p. v, along with the topical reports published as a result of work performed for this program during the current fiscal year. Copies of the reports are available from the Technical Information Center, Energy Research and Development Administration, Oak Ridge, Tenn. 37830.

HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF REACTOR SAFETY RESEARCH QUARTERLY PROGRESS REPORT, APRIL 1—JUNE 30, 1977

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ABSTRACT

HTGR safety studies work this quarter concentrated on detailed investigations of postulated Fort St. Vrain rod-withdrawal accidents.

General Atomic Company's TAP program was implemented on the local IBM 360 computers, and the results of sample problem runs appear to be in reasonable agreement with those published by General Atomic.

1. HTGR SYSTEMS AND SAFETY ANALYSIS

S. J. Ball

Work for the Division of Reactor Safety Research (RSR) under the HTGR Systems and Safety Analysis Program began in July 1974, and progress is reported quarterly. Work during this quarter consisted mainly of investigations of reactivity insertion accidents and refinements to the ORTAP-FSV code.

1.1 Development of the FSV Nuclear Steam Supply System Simulation Code (ORTAP-FSV)

J. C. Cleveland   S. J. Ball   J. C. Conklin

Development work was continued on the ORTAP-FSV code,[1] a simulator of the Fort St. Vrain (FSV) nuclear steam supply system dynamics.

The code was modified to account for the 152-sec rod bank insertion time on a scram and for several variations of the minimum assumed scram reactivity available. A "standard" GAC-supplied scram power vs time curve had been used previously. Presently, the reactor core power derived from the kinetics equations is used until the power level falls to the point where it is equal to the standard scram curve, which accounts for afterheat. Thereafter, the scram curve is used. This modification has resulted in higher predicted peak fuel temperatures.

The code was also modified to simulate turbine trip transients in which the reheat steam temperature control system, which uses reactor power level as a dependent variable, functions to keep the plant operating. Several successful turbine trip runs have been made.

The circulator-turbine model (ORCIRT) was modified so that the turbine will operate on wet steam. The main steam bypass system subroutine was also modified to dampen the "noise" and oscillations in the calculated bypass flow seen during parts of the shutdown transients. This problem has not yet been solved completely, however, as there are some conditions for which the noise is still present.

Simulation of plant operation at low-power (~25%) conditions has been achieved. Substantial trimming of control system parameters resulted in fairly good agreement with most plant operating characteristics. However, further investigation of low-flow conditions is necessary for satisfactory plant stability at low power.

Improvements to several steam property subroutines and the feedwater heater subroutine have resulted in a 35% decrease (since the previous quarterly report) in computer running time for certain transients. The CPU running time for certain rampdown transients has been decreased by 50% since the beginning of the year. Further means for decreasing running time have been identified and will be implemented.

A report[1] that describes the ORTAP computer models and the code was completed.

During the quarter, a paper on the Fort St. Vrain (FSV) reactivity analyses was accepted for presentation at the ANS Topical Meeting on Thermal Reactor Safety in Sun Valley, Idaho, on Aug. 1–4, 1977, and will be published in the proceedings. The paper is entitled "Simulation of the Response of the Fort St. Vrain High Temperature Gas Cooled Reactor System to a Postulated Rod Withdrawal Accident," by J. C. Cleveland, S. J. Ball, R. A. Hedrick, J. G. Delene, and J. C. Conklin. The abstract is as follows:

Abstract

Transients resulting from postulated accidental withdrawal of a control rod pair from the Fort St. Vrain HTGR core have been analyzed with a nuclear steam supply system simulation. Various cases have been investigated to determine what conditions and assumptions lead to the most severe core temperature transients. Results indicate that the most severe temperature transient occurs if the accident initiates from full power at beginning of equilibrium cycle conditions.

The paper[2] on the ORTAP code for the 1977 Summer Computer Simulation Conference was also completed.

1.2 Analysis of FSV Postulated Rod-Withdrawal Accidents Using the ORTAP-FSV Code

J. C. Cleveland
S. J. Ball
J. C. Conklin

In the FSV plant, there are six lines of defense which are sequentially effective in limiting the consequences of a rod pair withdrawal accident in the power range. These are:

1. normal action of the neutron flux controller,
2. a rod-withdrawal prohibit at 120% power,
3. a scram at 140% of rated power,
4. a scram when the measured reheat steam temperature reaches 42°C (75°F) above rated temperature,
5. manual scram of the control rods at any time,
6. manual insertion of the reserve shutdown system at any time.

In the analyses of rod-withdrawal accidents, it has been assumed that the first two systems do not function and that the operators do not shut the reactor down manually. Several postulated accidents were analyzed to determine what conditions lead to the most severe temperature transients. Analyses were made for various times in the core loading history, including the beginning- and end-of-cycle conditions for both the initial and equilibrium cores. Cases were also run for the runaway rod pair initially in the fully inserted position and initially half inserted. In all cases, the scram rods were conservatively assumed to be fully withdrawn.

Normally, the hot fuel region calculations are done by simulating a fuel stick, the surrounding graphite, and coolant channels in the refueling region which has the greatest radial power peaking factor. Due to the regional flow orificing in FSV, these regions also have the highest flow. For rod pair withdrawal transients, however, this may not be the region which reaches the highest fuel temperatures. Flux peaking in the region from which the rod pair is being withdrawn and in adjacent regions causes the local power density to increase more rapidly than the core average power density. Furthermore, the region flow orifice settings, initially adjusted to match regional flow to regional power, are assumed unchanged during the transient since the orifice positions are set manually. This results in a further imbalance in the power-to-flow ratio. To account for these effects, time-dependent peaking factors obtained from the FSV Final Safety Analysis Report (FSAR)[3] for the refueling region assumed to be experiencing rod pair withdrawal were superimposed on the rise of the average core power density. These time-dependent peaking factors were obtained by General Atomic Company (GAC) using two-dimensional static diffusion theory calculations. These calculations neglected any negative temperature feedback and are therefore believed to be conservative. For additional conservatism, these peaking factors were applied to each axial segment of the region from the time of accident initiation even though the rod pair is actually being removed from the axial segments sequentially.

Peak fuel and region gas outlet temperatures were determined for both the region with the greatest initial radial power factor and for the region experiencing rod pair withdrawal. For conservatism, conduction from these hot regions to adjacent regions was neglected. Furthermore, conservatively low values for the conductances of the fuel stick, graphite moderator, and fuel-graphite gap were used. Results for the transients from 100% power are summarized in Table 1, along with comparisons with GAC results (in parentheses). Case 2 represents the conditions determined by GAC to lead to the most severe temperatures. Note that less severe fuel and helium temperatures are reached due to the earlier scram and lower integrated core power for the withdrawal from the initially half-inserted position (case 1). The same is true for the case where the scram signal is at 140% power (case 3). The FSV FSAR states that data for the impact of time and temperature on fuel particle integrity indicate that failure can be expected for any fuel that reaches 2500°C, that is maintained above 2000°C for about 1 hr, or that remains at 1600°C for several hundred hours. Table 1 shows the percent of the fuel in the region experiencing rod withdrawal and in the region with the greatest initial radial power peaking factor with centerline temperature above 2500 and 1600°C. From the FSAR statement, any fuel exceeding 2500°C may fail during the transient while fuel remaining below 1600°C will not. Peak fuel temperatures in Table 1 which are above 2500°C should be considered only on a relative basis. The fuel stick thermal model is a homogenized stick model and does not treat the fissile and fertile coated particles and the graphite filler separately. This thermal model does not account for the latent heat of fusion of the UC_2 particle kernels at 2500°C but simply extrapolates specific heat beyond this temperature.

In order to explore GAC's claim that the most severe temperature conditions are reached if the accident occurs at the end-of-equilibrium cycle for a maximum rod pair worth of 0.012, three additional cases

Table 1. Results of rod pair withdrawal transients from 100% power

	Case								
	1	2[a]	3	4	5	6	7	8	9
Rod worth, Δk	0.012	0.012	0.012	0.016	0.016	0.016	0.016	0.015	0.010
Initial position of rod pair	Half in	Fully in	Fully in	Fully in	Fully in	Half in	Half in	Fully in	Fully in
Kinetics parameters	EOC-EQ	EOC-EQ	EOC-EQ	BOC-EQ	BOC-EQ	BOC-EQ	BOC-EQ	EOC-1	EOC-EQ
Scram signal	RHST[b]	RHST	140%	140%	RHST	RHST	140%	RHST	RHST
Time at scram initiation, sec	76.2	102.5 (105)[c]	39.2	38.4	103.1	76.4	9.1	105.5	108.4
Power level at scram initiation	210%[d]	282	140	140	248	183[e]	140	242	253
Maximum core average fuel temperature, °C	1098[f]	1195 (1225)[c]	861 (870)[c]	863	1144	1060	865	1139	1146 (1116)[c]
Maximum mixed mean core outlet temperature, °C	927[f]	994 (1062)	801 ($\leq$796)	809	969	893	800	968	962 (1002)
Region experiencing rod withdrawal:									
Peak centerline temperature, °C	2082	3057[g] (2870)	1137 (1183)	1194	3305[g]	2206	1014	3204[g]	2685[g] (2364)
Peak region outlet temperature, °C	1224	1654 (1650)	862 ($\leq$914)	888	1766	1219	801	1737	1496 (1414)
Fraction with centerline above 2500°C	0.0	0.5	0.0	0.0	0.8	0.0	0.0	0.7	0.3
Fraction with centerline above 1600°C	0.8	1.0	0.0	0.0	1.0	0.8	0.0	1.0	0.8
Region with highest initial power density:									
Peak centerline temperature, °C	1637[f]	1831	1151	1148	1724	1545	1155	1701	1728
Peak region outlet temperature, °C	1045[f]	1131	832	847	1006	996	832	1103	1097
Fraction with centerline above 1600°C	0.3	0.5	0.0	0.0	0.5	0.0	0.0	0.5	0.5

[a]Reference case.

[b]Measured reheat steam temperature 42°C above normal.

[c]Values derived from FSAR.

[d]Peaked at 226% prior to scram.

[e]Peaked at 209% power.

[f]At initial steady-state conditions the core average fuel temperature = 816°C; mixed mean coolant outlet temperature = 774°C; peak fuel centerline temperature = 1064°C; and region outlet helium temperature = 785°C.

[g]Calculation neglected latent heat of fusion of UC_2 kernels at 2500°C by extrapolating a homogenized fuel stick heat capacity beyond 2500°C.

were analyzed:

1. Withdrawal of a rod pair from the fully inserted (cases 4 and 5) or half inserted (cases 6 and 7) position at 100% power, beginning-of-equilibrium cycle conditions. For these conditions the maximum rod pair worth is greater than at end-of-cycle conditions, while the temperature coefficients are more negative and the delayed neutron fraction larger.

2. Withdrawal of a rod pair from the fully inserted position at first-cycle conditions (case 8). The maximum rod pair worth at 100% power operating conditions during cycle 1 is 0.015. End-of-cycle-1 temperature coefficients and neutron precursor data were used.

3. Withdrawal of a rod pair of worth 0.010 from the fully inserted position at 100% power, end-of-equilibrium cycle conditions (case 9).

The last case was analyzed to determine if the slower increase in power and the resulting longer time available for circulator speed reduction would limit the increase in reheat steam temperature, delaying the scram signal to the extent that the integrated power and the resultant fuel temperatures would be greater than in the reference case.

During rod withdrawal transients, the helium flow rate is reduced by the plant control system in an attempt to maintain the main steam temperature at its set point. In the reference case, the helium flow decreased to ~75% of its initial value at the time the scram was initiated. This reduction in helium flow prior to the scram results in a slower increase in reheat steam temperature and therefore delays initiation of the scram signal. To determine how much delay is introduced and how much increase in core temperatures results from this delay, a calculation was performed holding the circulator speed request signal constant during a rod-withdrawal transient from the fully inserted position with initial core conditions corresponding to case 2. The constant circulator speed resulted in a scram ~11 sec earlier at a power level of 273% as opposed to 282%. The peak fuel centerline temperature remained below the 2500°C limit, and the peak region coolant outlet temperature was reduced by ~150°C. However, although the core temperatures were reduced, the reheater and steam generator temperatures were significantly increased due to the more rapid rate of heat removal from the core and the higher heat transfer coefficients in the steam generator. Maximum reheat and main steam temperatures were 746 and 850°C, respectively, compared to 658 and 620°C in the reference case.

This is a good example of how a postulated accident which is not a "worst-case" accident from the core maximum temperature standpoint may be a worst-case, or at least a more serious, accident for the steam generator. The assumption of a constant circulator turbine speed request is not impossible either, considering that the main steam temperature control system (which has the requested circulator speed as a dependent variable) has been and could be left in the manual control mode.

In the analysis of the rod-withdrawal accident, low conductivity values of fuel, gap, and bulk moderator were used because they typically lead to higher fuel temperatures. However, low conductances result in greater fuel temperature increases, yielding a larger negative Doppler feedback reactivity and thus limiting the power increase prior to scram. In order to determine if more severe hot region temperatures would ultimately result if higher conductivities were used in the coupled heat transfer—neutron kinetics average region simulation, two cases were run to compare with the reference case (case 2, Table 1). The gap conductance was doubled in the first case, while the fuel, gap, and bulk moderator conductivities were each increased by 50% in the second. For both cases, as in the reference case, the CORTAP calculations for the hot region were performed with low conductances; the power history determined by the system simulation is an input to these calculations. The results showed that each increased conductance case resulted in an earlier scram due to the increased rate of heat removal from the core and the resultant more rapid rate of increase of reheat steam temperature, leading to lower peak temperatures in the hot region. Power levels at scram initiation, however, were increased due to the reduced Doppler feedback.

Finally, calculations were performed for rod pair withdrawal transients that occur at lower power levels (~4 and ~25%) to determine if these could lead to more severe temperatures than transients initiating from 100% power. Analyses of these transients should be considered as scoping in nature, however, because variations in core inlet temperature and flow resulting from response of other system components and control system action were not taken into account. For this analysis, the CORTAP core simulation code was used with constant core inlet temperature and flow conditions. Maximum expected rod pair worths were obtained from GAC information presented in Ref. 4. Since no indication could be obtained from this analysis as to the time at which a plant protection system scram signal would result, fuel and mixed-mean helium outlet temperatures are reported both at 140% power and at 102 sec, assuming that a manual scram could be initiated at this time. The time of 102 sec was chosen because this was the time at which the reheat steam temperature scram occurred for the reference 100% power case. The results indicated that no conditions more severe than those reached if the accident occurs at 100% power are expected for the accidents initiating from lower power.

The following conclusions can be reached based on this analysis:

1. In a rod pair withdrawal accident, several plant control and safety systems must be inoperative for fuel temperatures to exceed 1600°C.

2. More severe core temperature conditions can be expected for transients initiating from the 100% power level than from intermediate (~25%) or low (~4%) power levels.

3. If a scram is initiated at 140% power for a maximum-worth rod pair withdrawal accident, none of the fuel in the reactor will reach the 1600°C fuel failure temperature limit. Considerably more time is required to reach the reheat steam temperature scram set point than to reach the 140% power scram set point. No significant difference exists between the calculated time of scram initiation and GAC's reported value for the reference case.

4. If the scram is signaled by the rise in reheat steam temperature, transients resulting from the accidental withdrawal of a maximum-worth rod pair from the half-inserted position result in earlier scrams and less severe temperature conditions than transients resulting from the accidental withdrawal of the same rod pair from the fully inserted position.

5. For the reference case, the use of low core heat conductance values results in prediction of more severe core temperatures even though the rate of power increase is initially larger, due to reduced Doppler feedback, than if larger conductance values are used.

6. The action of the plant control system in reducing helium flow in an attempt to maintain the main steam temperature delays the reheat steam temperature scram signal since it reduces the rate of increase of the reheat steam temperature. This delay results in more severe peak temperatures in the core but less severe temperatures in the reheater and steam generator.

7. For transients resulting from accidental withdrawal of a maximum-worth control rod pair from 100% power conditions and with a scram resulting from a 42°C (75°F) rise in measured reheat steam temperature, conservative calculations indicate that some of the fuel in the hottest core region will reach centerline temperatures in excess of 2500°C. This is the temperature indicated by GAC at which immediate failure of the fuel particle coatings may occur. No determination was made as to the total amount of fuel exceeding this failure temperature since detailed flux peaking information for regions other than the hottest region is not presented in the FSAR. Based on these analyses, higher temperatures are reached in the hot region if the accident occurs at beginning-of-equilibrium cycle conditions than at other core operating conditions.

1.3 University of Tennessee Subcontract

T. W. Kerlin

1.3.1 Introduction

Work continued on gas-cooled reactor modeling and model evaluation. Individual project activities are described below.

1.3.2 Evaluation of modeling assumptions for an HTGR steam generator (Ioannis Anastasiou)

A report on the comparative study of three different models simulating transients for the FSV steam generator is nearly complete. The three models under comparison are (1) BLAST — a nonlinear, multinode, fixed-boundary model developed at ORNL; (2) UTSGRHM — a nonlinear, six-node on each side, moving-boundary model developed at the Nuclear Engineering Department of the University of Tennessee; and (3) LAP — a linear, three-node, moving-boundary model developed at General Atomic Company (GAC).

The transients simulated by all the above models are initiated by three perturbations: (1) a step increase in helium inlet flow rate by 1 lb_m/sec per steam generator module (~1.5% perturbation); (2) a step increase in helium inlet temperature by 5.5°C (10°F) (~1% perturbation); and (3) a feedwater flow rate step increase by 0.05 lb_m/sec per tube (~5% perturbation).

Despite the large differences among the models, calculated quantities such as steam and helium exit temperatures show remarkably good agreement. Also, it can be shown that the different structure of the models, such as moving vs fixed node boundaries, can account for the differences that appeared in the tube and helium temperature responses along the steam generator.

Differences in behavior among the models, such as the faster response of the UTSGRHM model, can be explained through the choice and structure of the conservation equations employed. Finally, merit is given to each model with respect to the computer time consumed and the particular problems and needs where each might be most useful.

1.3.3 Detailed nonlinear model of the FSV steam generator (Ming-Huei Lee)

Last quarter, the effort was devoted to testing the transient program for the FSV steam generator model. Two types of perturbations were used — a step change of the inlet enthalpy and a step change of the mass flow rate. These two types of disturbances were applied both to primary and secondary coolants. Since outputs were desired quickly for making corrections and modifications for the transient program, runs were performed using short CPU time.

Reasonable transient behavior was obtained for the cases with enthalpy perturbations. However, for the cases with mass flow rate perturbations, the transient responses oscillated although they were tending toward convergence. The oscillation problem was investigated, and it was found that a complicated algebraic loop was involved in the system equations for determinations of the moving-boundary lengths. Current effort is being spent on overcoming the trouble induced by the mass flow rate perturbations.

In order to obtain a comparison and to facilitate trouble-shooting, two types of models were run. One model with an assumption of no storage term for mass is applicable to incompressible fluid; the other is for a system with part of the fluid being compressible. The problem for the former model is simpler than that of the latter model, since modeling for compressible fluid involves acoustic phenomena which should be removed for the present purposes for dynamic modeling.

Planning for this quarter includes overcoming the problems in mass flow rate perturbation and dynamic studies for a long observation time.

1.3.4 Nonlinear model for FSV (J. G. Thakkar)

The main effort in this subtask during this quarter has been on developing a nonlinear model for the FSV core and modifying the nonlinear FSV steam generator model to calculate the transient response of the once-through steam generators used in pressurized-water reactors (PWRs) (to check this model against previously published PWR results).

The FSV core dynamic behavior is described by a model consisting of nonlinear differential and algebraic equations. The neutronic behavior is described by point kinetics and six groups of delayed neutrons. Reactivity feedbacks due to changes in the fuel and the moderator temperatures are included in the reactor power equation. The core heat transfer model uses nodal approximations for the fuel, graphite, and helium temperatures. The core is divided into six axial nodes. The fractional power generator in each node is calculated from the axial power peaking factors reported in the FSV FSAR.

The transient response of the isolated FSV core was calculated for the following perturbations: (1) a 1¢ step change in control rod reactivity, (2) a 5.5°C (10°F) step change in the helium inlet temperature, (3) a 10-lb/sec step change in the helium inlet flow rate, and (4) a reactor trip simulation from 100% power.

The FSV core model was also used to calculate the sensitivity of the system dynamic response to selected design parameters. The sensitivity analysis results were obtained by calculating the transient response for a 1¢ step change in reactivity for reference values of parameters and for (1) a 20% change in the fuel temperature coefficient of reactivity; (2) a 20% change in the moderator temperature coefficient of reactivity; and (3) a 20% change in the fuel-to-moderator and the moderator-to-coolant effective heat transfer coefficients. The results were compared with the reference case.

The nonlinear FSV steam generator model was modified to calculate the system response of the Integral Economizer Once-Through Steam Generator (IEOTSG) reported in Chen's dissertion.[5] Calculations will be performed and results compared with those of Chen as a further test of the approach used in the FSV model. The modification included (1) a change from helium to pressurized water as a heating medium, (2) a change to secondary fluid in the shell side and primary fluid in tubes, and (3) a change in heat transfer and pressure drop correlations.

The model has three nodes — one each for the economizer, evaporator, and the superheater sections. The model is being tested for steady-state calculation and some of the assumptions used in Ref. 5.

2. COMPUTER CODE IMPLEMENTATION

J. P. Sanders

The TAP[6] computer program that has been made operational on the IBM 360 computer has been exercised during this quarter to accomplish two objectives. Two sets of sample input, which had been supplied with the original transmission of the code, represented essentially steady-state calculations for the 1160-MW(e) HTGR plant at 100% and at 25% of design power.

The first objective was to formulate a set of input data that would exercise all or most of the various subroutines of the program so that it could be determined that all segments had been properly modified for the IBM 360 system. This was accomplished by running a case in which the reactor was scrammed from 100% power, followed by a turbine trip 2 min later. Only two input variables had to be changed to make the run. The simulation ran smoothly until terminated by the circulator turbine routine 28 min after the scram due to lack of sufficient steam flow. No auxiliary steam supply had been assumed to be available.

The second objective in the execution of the TAP program was to generate output that could be used for comparison purposes with computations executed using the ORTAP-FSV program. While no direct comparisons of output can be made (the two codes are set up to simulate different HTGRs), the general features of the response are comparable. For example, in both cases the helium flow decreases to a minimum value after the turbine trip and later increases (by action of the control system) because the main steam temperature is below the set point value. Subsequently, the helium flow decreases due to the dwindling steam supply. It was also noted that for comparable problems, the execution time required for the TAP code was considerably less than that required by ORTAP.

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