CONF-850570--4

MASTER

ORNL ANALYSES OF AVR PERFORMANCE AND SAFETY*

J. C. Cleveland

Engineering Technology Division
Oak Ridge National Laboratory
Oak Ridge, Tennessee

CONF-850570--4

DE85 011934

To be presented at

IAEA Specialists' Meeting on Safety and Accident Analyses
for Gas-Cooled Reactors
Oak Ridge, Tennessee

May 13-15, 1985

Notice: This document contains information of a preliminary nature. It is subject to revision or correction and therefore does not represent a final report.

---

*Research sponsored by the Division of HTR Development, Office of Converter Reactor Deployment, U.S. Department of Energy under contract No. DE-AC05-84OR21400 with Martin Marietta Energy Systems, Incorporated.

ORNL ANALYSES OF AVR PERFORMANCE AND SAFETY*

John C. Cleveland
Engineering Technology Division
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37831

Introduction

Because of the high interest in modular High Temperature Reactor performance and safety, a cooperative project has been established involving the Oak Ridge National Laboratory (ORNL), Arbeitsgemeinschaft Versuchs Reaktor GmbH (AVR), and Kernforschungsanlage Jülich GmbH (KFA) in reactor physics, performance and safety. This project has been established within the frame of the HTR Implementing Agreement between the U.S. Department of Energy and the Ministry of Research and Technology of the Federal Republic of Germany. The effort focuses on examination of AVR neutronics and thermofluid dynamics with objectives of further insights into pebble bed reactor behavior.

Benefits to AVR and KFA from the cooperative effort include an independent computation by ORNL of core physics parameters with an examination of the effect of new nuclear data libraries on computed results, and independent thermofluid dynamic investigations of selected performance tests and of a hypothetical depressurized core heatup accident, including consideration of how the AVR staff might conduct a depressurized core heatup test. Benefits to ORNL and to the DOE Program from the cooperative effort include improving pebble bed reactor analysis capabilities in the U.S., access to data from certain AVR performance and safety tests, and the opportunity for code validation through comparison with test results.

Many AVR design parameters and safety features are similar to those of current modular HTR concepts. Specifically the AVR core diameter (3 meters) and power density (2.5 W/cc) combine with the large heat capacity of the pebble bed core and the very good retention of fission products by the fuel to high temperatures such that the AVR has the capability for safe removal of afterheat through the vessel wall. Also, for transients involving changes in core flow rate, the negative temperature coefficient causes the power generation to closely follow the rate of heat removal from the core. With flow reductions or complete loss of forced convection, high heat capacity and natural convection combine to limit increases in fuel temperatures to safe levels. Finally, the large negative temperature coefficient leads to compensation of reactivity insertions with only small to modest changes in fuel temperature.

This paper presents initial results of ORNL's examination of a hypothetical depressurized core heatup accident and consideration of how a depressurized core heatup test might be conducted by AVR staff. Also presented are initial analyses of a test involving a reduction in core flow and of a test involving reactivity insertion via control rod withdrawal.

---

*Research sponsored by the Division of HTR Development, Office of Converter Reactor Deployment, U.S. Department of Energy under contract No. DE-AC05-84OR21400 with Martin Marietta Energy Systems Incorporated.

.Brief Description of the AVR Reactor

The AVR is a 15-MW_e HTR steam cycle demonstration plant in Jülich, West Germany. The AVR began generating electricity in December 1967. Its purpose is to demonstrate the feasibility of an HTR with pebble fuel elements and high operating temperatures. The operating utility group is Arbeitsgemeinschaft Versuchs-Reaktor (AVR) GmbH of Düsseldorf. The constructor was Brown-Boveri-Krupp Reaktorbau GmbH.

The AVR is shown in Fig. 1 and technical information is summarized in Table 1. The core is fueled with ~100,000 graphite pebbles containing coated fuel particles. The pebbles are 6-cm diameter. During operation, pebbles are continuously withdrawn from the bottom of the reactor core and pebbles are added at the top of the core so that a circulation takes place. After several passes through the core, the spent pebbles, which are removed from the cycle when their target burnup is achieved, are replaced by new pebbles.

Helium flows upward through the pebble bed and then across the steam generator tubes to produce steam. The reactor can operate at full power with a gas outlet temperature ranging from 770°C to 950°C. The steam generator is located above the core in the steel reactor vessel. It is shielded from core radiation by a 50-cm-thick graphite top reflector and two additional 50-cm-thick carbon brick layers. The helium is circulated by two blowers located in the lower part of the vessel. The power of the reactor is controlled by varying the coolant flow.

The AVR has four control rods in reflector "noses" which protrude into the core region. The rods are driven from the bottom by a counter weight device and are used to achieve cold shutdown. During normal operation, the control rod position is adjusted to control the core temperature and the coolant outlet temperature.

The inner vessel is enclosed concentrically in a second reactor vessel. A biological shield is located between the vessels. The space between the vessels is cooled by natural convection of helium at a slightly higher pressure than the primary coolant. About 2.4% of nominal thermal power is removed by the interspace coolers.

The reactor is surrounded by a steel containment vessel and by a 1.5-m-thick concrete building. A water sprinkle system is available to cool the outer surface of the steel containment shell in an extreme accident using water stored above the steel containment. This system can provide containment cooling for several hours until further measures can be taken.

Table 1. Technical data for AVR

Thermal power rating, MW	46.0
Core power density, MW/m^3	2.5
Core inlet temperature, °C	275
Core outlet temperature, °C	950
Primary system pressure, bar	10.8
Core diameter, m	3.0
Steam pressure, bar	73
Steam temperature, °C	505
Absorber rods	4

The AVR achieves a high degree of safety through the following features:

- low excess reactivity (due to its on-line refueling)
- low power density
- high heat capacity
- high retention of fission products in the fuel to high temperatures
- large negative temperature coefficient
- capability to transfer heat from the core to the steam generator by natural convection
- ability for afterheat removal through the vessel wall

Analyses of a Hypothetical Depressurized Core Heatup Accident

ORNL has analyzed a hypothetical depressurized core heatup accident to provide the AVR staff with an independent analysis of the AVR response and with observations which the AVR staff may wish to consider as they plan a depressurized core heatup test.

As discussed in another paper (Ref. 1) presented at this conference by G. Ivens and K. Krüger of the AVR staff, the AVR is planning experiments simulating a depressurized core heatup accident to demonstrate important features which provide modular HTR concepts as well as the AVR with a high degree of safety. Such experiments would also provide data on fission product performance under actual conditions and data for verification of reactor analysis tools with irradiated materials under actual conditions.

The analyses reported here assume an instantaneous depressurization and loss of forced convection at the AVR with the reactor initially operating at full power (46 MW_t) with a gas outlet temperature of 950°C. A scram is assumed to occur at accident initiation.

The most appropriate tool for analyzing the reactor thermal response to a hypothetical depressurized core heatup is the THERMIX-KONVEK code which has been developed by KFA. Analyses performed by KFA with THERMIX for a hypothetical depressurized core heatup accident at the AVR are presented in Ref. 2.* To provide AVR and KFA with an independent analysis, ORNL's initial computations have been performed with the HEATING-6 general purpose heat transport code. HEATING-6 treats heat conduction and radiation, but does not consider heat transport by natural convection as does THERMIX-KONVEK. Certainly under depressurized conditions, heat transport by natural convection is small, and neglecting it is conservative from the viewpoint of fuel temperatures; however, vessel temperature is also of concern and neglecting natural convection is not conservative for predictions of vessel temperature (Ref. 3).

The HEATING-6 analyses were performed in two dimensions to account for heat transport to the cooler (lower) regions of the core and the side and bottom reflectors. Various material properties were input as temperature dependent; the fluence effects on reflector graphite conductivity were treated by defining regions of

---

*ORNL has obtained THERMIX-KONVEK from KFA and has installed it on the ORNL computer. Work is underway to set up a model of the AVR for use in further analyses of core heatup transients.

different material properties. An implicit differential equation solution technique, which can range from a Crank-Nicholson to a classical implicit procedure, was employed. The system of equations is solved by point successive overrelaxation iteration and includes procedures to estimate the optimum acceleration parameters.

Of key interest in this analysis was the maximum fuel temperature and the time the fuel experiences above normal temperatures. The effect of certain parameters on computed fuel and vessel temperatures was also investigated.

In the radial direction, the analytical model included the core, reflector, core barrel, thermal shield, inner pressure vessel, biological shield, and outer pressure vessel. In the axial direction, the model represented the region from the upper reflector thorough the core and bottom reflector.

The core heat removal processes considered were:

- conduction and radiation through the core and conduction through the side reflector
- radiation from the core barrel to the thermal shield, from the thermal shield to the inner pressure vessel, from the inner pressure vessel to the first biological shield, and from this biological shield to the outer pressure vessel
- conduction through the core barrel, the thermal shield, the inner and outer vessels, and the biological shield
- radiation from the upper surface of the core to the top reflector.

Adiabatic boundary conditions were applied to the top surface of the top reflector and to the lower surface of the bottom reflector. Neglecting axial heat loss is conservative from the standpoint of maximum fuel and vessel temperatures.

A boundary condition of a constant temperature of 100°C was applied to the outer surface of the outer pressure vessel to approximate the effect of the containment cooling. Supplementary calculations were performed to explore the effect of this assumption.

The low levels of heat which would be removed by natural circulation in the interspace cooler under depressurized conditions were neglected. This is conservative relative to both the fuel and the vessel temperatures.

Results are shown in Figs. 2, 3, and 4. In Fig. 2, the radial profiles are plotted for the axial plane in which the highest core temperature occurs at that point in time. Initially, the highest temperature is near the top of the pebble bed and moves downward toward the core midplane as the transient progresses.

The response of the core temperatures is very slow with the maximum fuel temperature increasing only 25°C in the first half hour following accident initiation and only 100°C in the first two hours. The highest temperature reached in the fuel is 1320°C and occurs 17 hrs after accident initiation.

Significant quantities of heat can be removed through the outer vessel only after the temperature of the thermal shield, the inner vessel, and the biological shield have increased sufficiently to establish temperature differences supporting heat transport. During the first several hours, the primary effect is a temperature redistribution within the reactor with those regions which are initially relatively cool (the lower core region, the side and lower reflectors, the biological shield and pressure vessels) undergoing a slow temperature increase. Figure 3 shows that only after 200 hrs does the rate of heat loss through the inner vessel exceed the decay heat generation rate.

The effect on maximum fuel temperatures of the assumption that the outer vessel wall can be kept at 100°C was investigated by performing an adiabatic heatup analysis assuming no heat loss through the outer vessel. Figure 4 shows that for about the first 200 hrs (about 8.3 days), the rate of heat loss through the outer vessel has no effect on the maximum fuel temperature. However, it clearly will effect the time fuel experiences above normal temperatures as well as the inner vessel temperature. With an adiabatic heatup, the inner vessel reaches 650°C at 330 hrs (compared to 530°C for the case shown in Fig. 2) and continues to increase.

The results in Fig. 2 show that the inner vessel would reach its highest temperature at a time when the fuel temperature has long been decreasing. (In fact, a depressurized core heatup test would necessarily be terminated prior to this time to prevent vessel damage.) The biological shield conductivity is an important parameter in determining the vessel temperature since heat removed through the outer vessel must first pass through this 75 cm thick shield. Decreasing the conductivity of the biological shield from 2.0 W/mK to 1.0 W/mK increased the maximum temperature of the inner vessel by 100°C. This change in biological shield conductivity had essentially no effect (less than 5°C) on maximum fuel temperature.

In Ref. 1, the AVR staff describes how they might perform an actual depressurized core heatup test. Their procedure would be to conduct a normal shutdown and then intentionally depressurize - a procedure which requires three days. The reactor would then be taken critical and fuel temperature would be increased to normal levels. A small flow of feedwater would be supplied to the steam generator to prevent overheating of the steam generator tubes. The decay heat would then be simulated by varying the nuclear heat with time with the reactor critical. ORNL's analyses of the hypothetical depressurized core heatup accident led to the following items which may warrant evaluation as the AVR staff develops detailed plans for the test:

- Power densities higher than the AVR reference power density could be simulated by scaling the nuclear power which simulates the decay heat generation rate.

- The core heatup tests could possibly be conducted in a staged approach with initial tests at lower temperature and/or only partially depressurized and with some heat removal by the interspace cooler. Good agreement between prediction and experiment would provide confidence in predictions for tests involving more extreme conditions.

- Temperature measurements are made in the middle and upper regions of the reflector noses, at several locations through the side reflector, at the inner and outer vessel, and the biological shield. Fuel temperatures are not measured. In order to increase the usefulness of the tests for code validation purposes, methods for indirect measurement of fuel temperatures may warrant evaluations. An indirect measurement of the change in nuclear average fuel temperature could be obtained by equating the change in reactivity due to control rod motion to maintain criticality with the product of the temperature coefficient and the change in nuclear average fuel temperature.

  A second indirect temperature measurement technique would involve the use of monitor pebbles similar in concept to those currently employed in the AVR. Special graphite pebbles would be loaded into the reactor prior to the test.

These pebbles would enclose capsules containing flakes (or wires) of metal or eutectic alloys. A few different materials with melting temperature spanning the appropriate temperature range would be enclosed in different capsules in each pebble. The pebbles would be examined by X-ray techniques after discharge to determine which metals exceeded their melting point, thereby giving an indication of the highest temperature reached.

- Further examination of the capability to remove heat through the outer vessel may be warranted. This depends on containment cooling capability as well as on the conductivity of the biological shield. If it would be necessary to determine the thermal properties of this shield more accurately than they are currently known, consideration could be given to performing dynamics tests with the interspace cooling system.

Analyses of Flow Reduction and Reactivity Insertion Tests

Other features important to safety are illustrated by the reactor response to large flow reductions and to reactivity insertions. To examine reactor response to such conditions, ORNL is analyzing selected experiments from a series of tests performed at the AVR during 1982-1983. The AVR staff performed these tests to examine the change in reactor performance as the core composition was changed from an all HEU/Th core to a mixed HEU/Th and LEU core.

Analyses of these experiments provides the opportunity to:

a. investigate the important core characteristics which influence the reactor response
b. examine the modeling complexity necessary to predict the reactor response to "at power" transients involving flow changes and/or small, slow reactivity insertions

To predict the reactor behavior, ORNL has prepared a dynamic model of the AVR core. The approach is to use fairly simple models to examine various effects and to obtain early results. The modeling detail will be improved as necessary. Model features are summarized below:

- point (space independent) neutron kinetics with six groups of delayed neutron precursors.
- a coarse-structure thermal model with heat conduction dynamics and heat convection in each axial section approximated by a model of the "average pebble" in that section.
- nuclear importance (flux squared) weighting of solid temperatures in the axial direction to determine the effective temperature-to-reactivity feedback to the neutron kinetics model.
- computation of reactivity effects due to changing ^{135}Xe concentration using coupled, first order, time dependent equations for the core average ^{135}I and ^{135}Xe concentrations based on the core average thermal flux level.
- a quasi-static, one-dimensional representation of the helium temperature and flow.
- for forced convection conditions, helium flow is computed from measured circulator speed, core inlet temperature and pressure assuming volumetric flow is proportional to speed. For natural convection conditions, helium flow is computed by balancing unrecoverable losses through the primary loop against the density difference driving head.

- computation of the decay power contribution to total power as the output of a series of lead-lag filters with prompt power (as determined by the point kinetics equation) as an input and with filter coefficients and time constants selected to match afterheat generation following a step decrease in flux.

Figures 5 and 6 show initial results and the measured power for a flow reduction test performed on April 16, 1982. The reactor was initially at full power with a core inlet gas temperature of 271°C and a gas outlet temperature of 807°C. The test was initiated by reducing the speed of each circulator from 4000 rpm to 2000 rpm over 68 seconds. The speed was held constant at 2000 rpm until shutdown of both circulators was initiated at 1085 sec. During the test there was no control rod motion. When the circulator speed was reduced, the negative temperature coefficient (-5.9×10^{-5}/°C) and the increasing fuel temperature caused the power to closely follow the flow reduction as shown in Fig. 5. The large heat capacity prevented excessive increases in fuel temperature. With the decrease in flux, the ^{135}Xe burnout rate decreased resulting in a transient increase in ^{135}Xe concentration and a resulting negative reactivity contribution as shown in Fig. 6). About 150 sec after initiation of the reduction in speed of the circulators the core reactivity returned to zero with the negative contribution due to the increasing ^{135}Xe concentration being balanced by a positive fuel temperature contribution resulting from operation at the reduced power and flow.

With shutdown of the circulators (initiated at 1085 sec) the fuel goes through a slight heatup (note in Fig. 6 the dip in the reactivity contribution due to the fuel temperature increase) driving the reactor subcritical. The rate of increase in ^{135}Xe concentration increases when the reactor goes subcritical (due to the decrease in ^{135}Xe burnout rate) and the resultant additional contribution of negative reactivity is sufficient to hold the reactor subcritical to about 1600 sec even with cooling of the fuel by natural convection (which the model estimates to be about 8.5% of full flow). (Note: this recriticality could be delayed by several hours by closing the main circuit valves, as has been done in AVR tests described in Ref. 1, thereby blocking natural convection gas flow through the main loop). The prediction of nuclear power during the transient compares well with the measured flux.

Summarizing the flow reduction test, the influence of core characteristics on reactor response is:

- the negative temperature coefficient causes the power to closely follow the rate of heat removal from the core.
- as the reactor power (neutron flux) changes, the resulting change in ^{135}Xe reactivity is an important component in the overall reactivity balance.
- with a large reduction in core flow (e.g. by 50%) and with no control rod motion, the high heat capacity of the fuel and the negative temperature coefficient combine to produce only moderate changes (on the order of 30°C) in maximum fuel temperature.
- the reactor can be driven subcritical by stopping the circulators. The natural convection through the main gas flow loop limits the increase in fuel temperature.

ORNL has also analyzed a control rod withdrawal experiment (performed on May 8, 1982) which involved introduction of ~10.6 cents reactivity in 25.6 sec with the reactor initially at ~82% power. The reactor was then allowed to stabilize at a new power level. Figures 7 and 8 show initial results. Figure 7 compares the predicted core power transient with the measured power. Again, due to the high heat capacity and large negative temperature coefficient, only small fuel temperature changes result.

These two experiments illustrate reactor performance characteristics which are important to the safety of the AVR reactor. Results of these analyses suggest that such transients can be modeled with the fairly simple analysis method described above.

Conclusion

In conclusion, the predicted response of the AVR to a core heatup accident and to the performance tests analyzed here illustrate important features which contribute to its safety. In the event of a depressurized core heatup accident, fuel temperatures should remain below 1350°C.

Access to data from AVR performance and safety tests provides the opportunity to benchmark pebble bed reactor analysis techniques and tools against actual conditions. Results reported here illustrate that simple modeling techniques can provide accurate predictions of the reactor power response for "at power" transients involving changes in core flow and/or slow reactivity changes.

Acknowledgment

The author very much appreciates the interests of AVR and KFA in the cooperative project with ORNL; specifically the help of Mr. Krüger of the AVR and Dr. Teuchert of KFA in these evaluations is appreciated. Also appreciated is the contribution of S. J. Ball of ORNL in jointly developing the AVR core dynamics model with the author.

References

1. G. Ivens and K. Krüger, "Safety-Related Experiences with the AVR-Reactor," presented to IAEA Specialists' Meeting on Safety and Accident Analysis for Gas-Cooled Reactors, Oak Ridge, Tennessee, U.S.A., 13-15 May, 1985.

2. K. Petersen, Zur Sicherheitskonzeption des Hochtemperaturreaktors mit natürlicher Wärmeableitung aus dem Kern im Störfall, Kernforschungsanlage Jülich GmbH, Institut für Reaktorentwicklung, Jül-1872, Oktober 1983.

3. R. M. Harrington, S. J. Ball, "Simulation of Thermal Response of the 250 MW_t Modular HTGR During Hypothetical Uncontrolled Heatup Accidents," Submitted to the IAEA Specialists' Meeting on Safety and Accident Analysis for Gas Cooled Reactors, May 13-15, 1985.

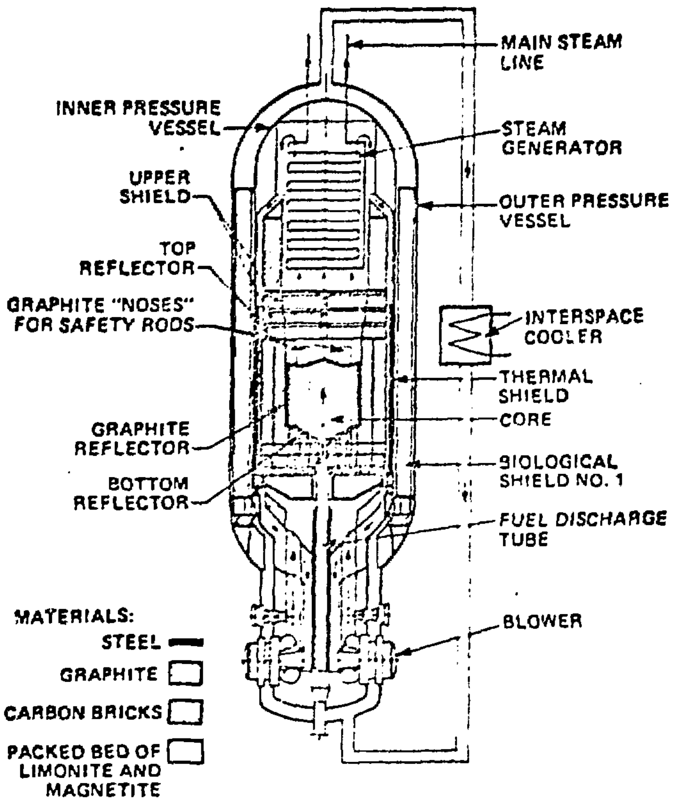

Fig. 1. AVR.

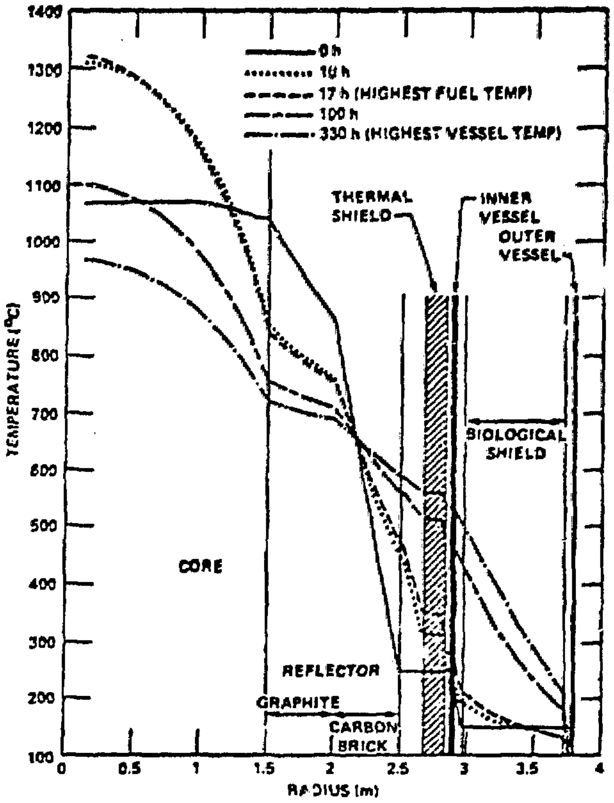

Fig. 2. Temperature distribution during depressurized core heatup in AVR.

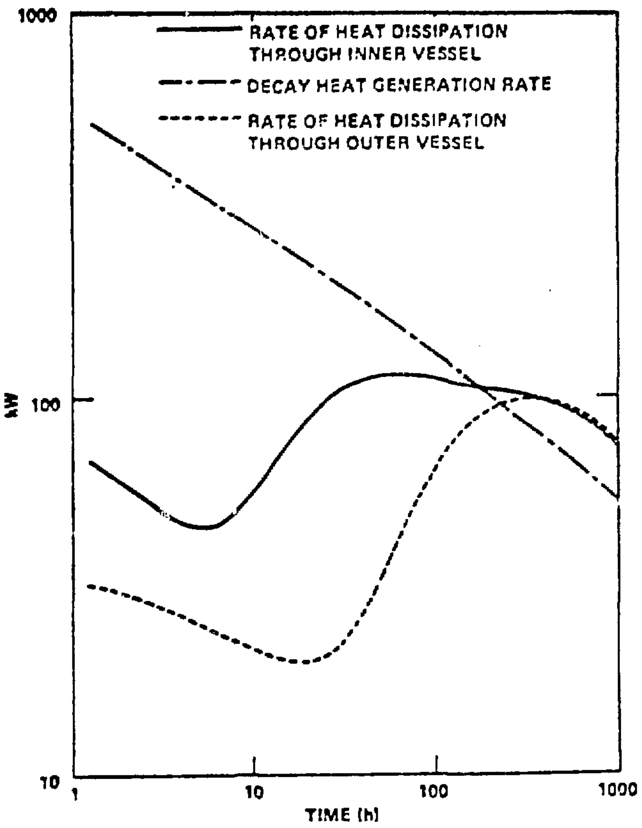

Fig. 3. Heat generation and dissipation rates during depressurized core heatup in AVR.

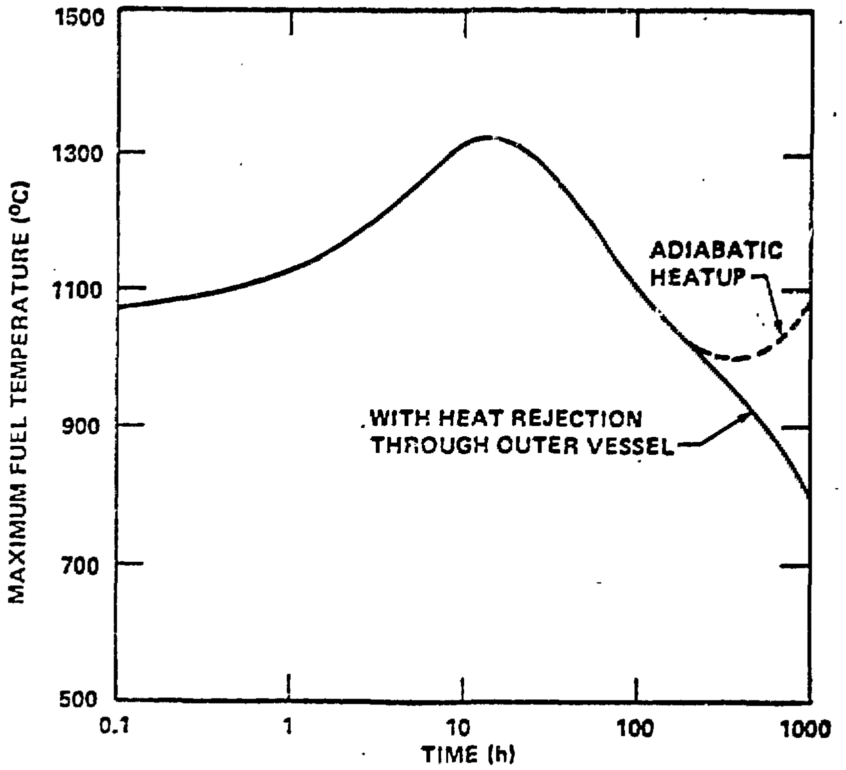

Fig. 4. Peak fuel temperature in depressurized core heatup in AVR.

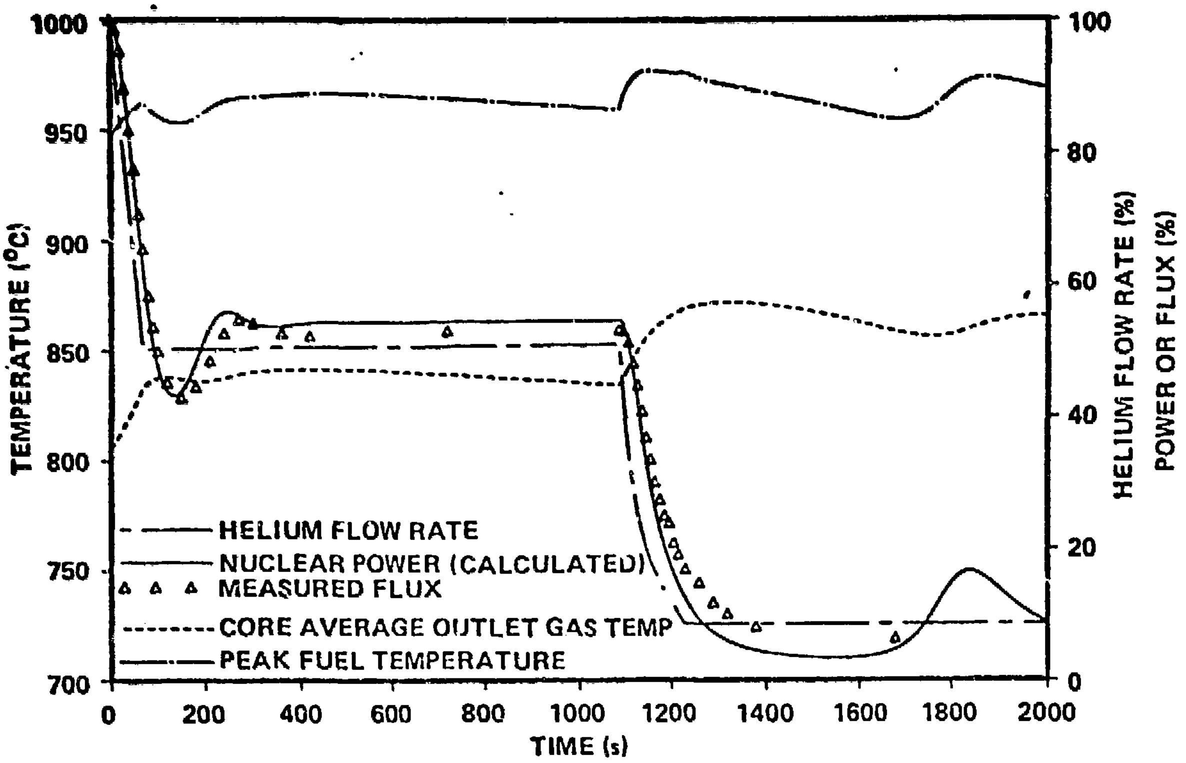

Fig. 5. AVR response to flow reduction of 4.16.82.

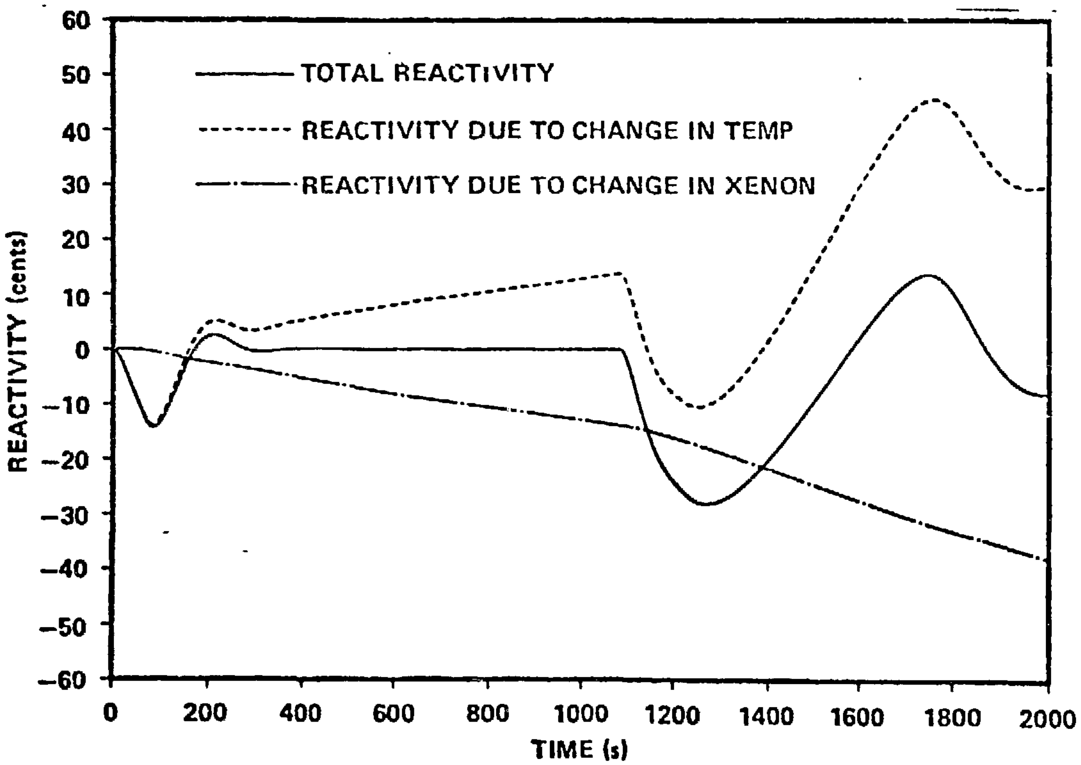

Fig. 6. Reactivity during AVR flow reduction test of 4.16.82.

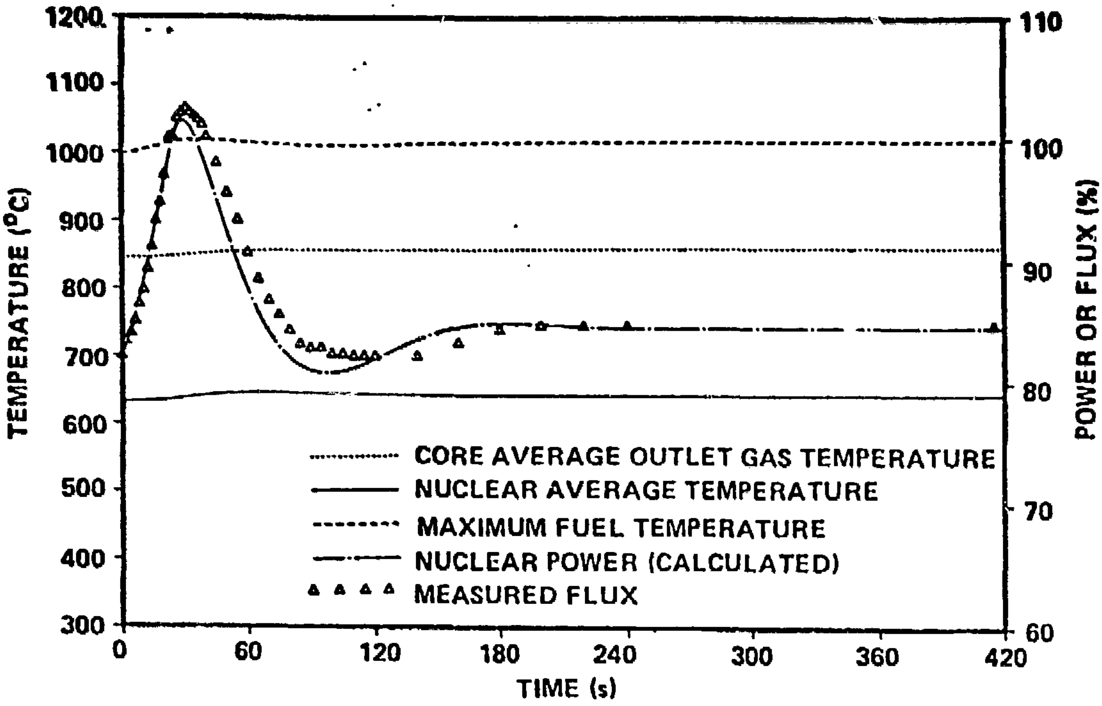

Fig. 7. AVR response to control rod withdrawal transient of 5.8.82.

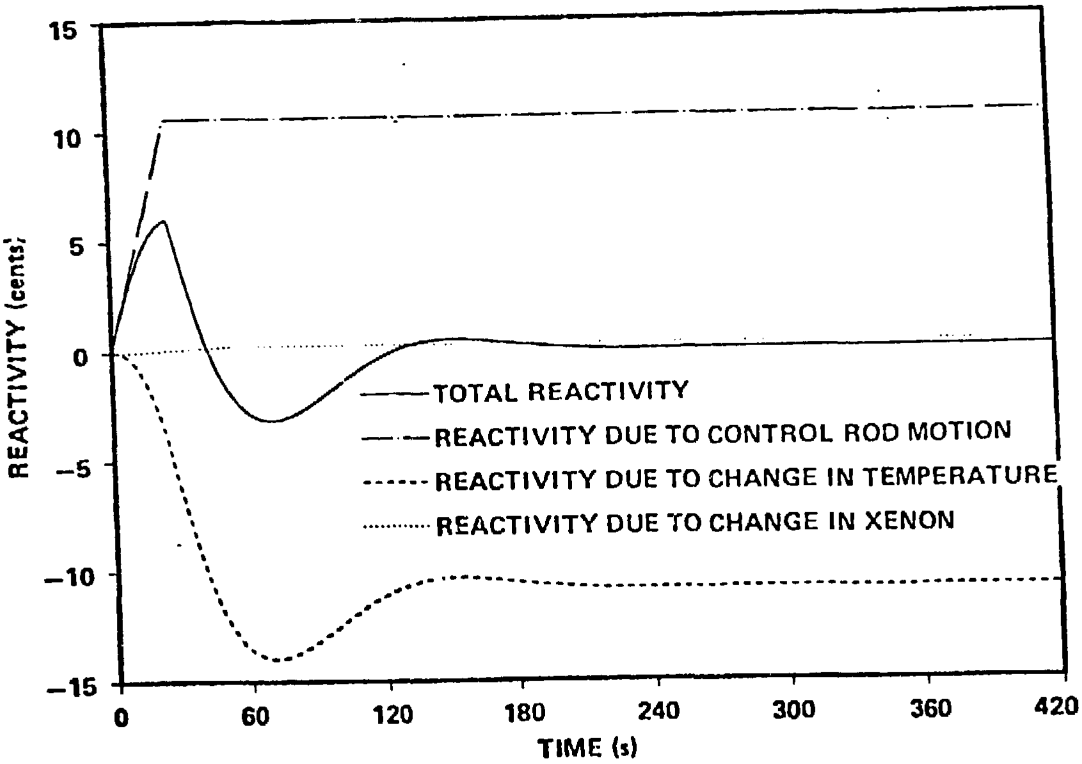

Fig. 8. Reactivity during AVR control rod withdrawal test of 5.8.82.