TRANSIENT BEHAVIOR OF AN
ADVANCED SODIUM GRAPHITE REACTOR

By
H. H. CAPPEL

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN AVIATION, INC.
P.O. BOX 309 CANOGA PARK, CALIFORNIA

CONTRACT: AT(11-1)-GEN-8
ISSUED: DEC 1 1961

DISTRIBUTION

This report has been distributed according to the category "Reactors Technology" as given in "Standard Distribution Lists for Unclassified Scientific and Technical Reports" TID-4500 (16th Ed.), December 15, 1960. A total of 650 copies was printed.

CONTENTS

	Page
Abstract.	vi
I. Introduction	1
II. Description of Reactor	2
A. General Description	2
B. Reactor Core	2
C. Primary and Secondary Sodium Systems	4
D. Shim and Safety Rods	5
E. Reactor Instrumentation	6
F. Control and Protective System	7
G. Operating Parameters	10
III. Nuclear Characteristics	12
A. Excess Reactivity	12
B. Neutron Lifetime	13
C. Reactivity Coefficients	13
D. Nuclear Parameters	16
IV. Thermal Characteristics	18
A. Thermal Characteristics at Steady State	18
V. Reactor Kinetics	19
A. Reactor Stability	19
B. Dynamic Response to Reactivity Step Functions	20
VI. Reactor Excursion Analysis	27
A. Excursions With Protective System Action	28
B. Excursions Without Protective System Action	34
VII. Conclusions	43
Appendix. Description of Transient Study	44
A. Analog Model	44
B. Nuclear Parameters	44
C. Thermal Parameters	45
D. Hydraulic Parameters	46
Nomenclature	47

TABLES

	Page
II-1. Summary of Tentative Characteristics of the 255-Mwe ASGR.	10
III-1. Nuclear Constants with 2.5 at. % U^{235} Enrichment	17

FIGURES

II-1. General Arrangement, Section Elevation (7519-5238)	3
III-1. Variation of Reactivity with Burnup (2.5% enrichment)	12
III-2. k_{eff} <u>vs</u> Fuel Temperature, Equilibrium Xenon and Samarium Poison (2.5% enrichment).	14
III-3. k_{eff} <u>vs</u> Moderator Temperature (2.5% enrichment).	15
III-4. η , f , and ηf <u>vs</u> Neutron Temperature (2.5% enrichment)	15
V-1. Open Loop Reactor Transfer Function $[ZP(s) PK(s)]$	21
V-2. Total Power <u>vs</u> Time for Step Reactivity Inputs from Full Power . .	22
V-3. Total Power <u>vs</u> Time for Step Reactivity Inputs from 50% Power . .	23
V-4. Total Power <u>vs</u> Time for Step Reactivity Inputs from 5% Power . . .	23
V-5. Maximum Hot Channel Fuel Temperature of the Central Fuel Element <u>vs</u> Time for Step Reactivity Inputs from Full Power	24
V-6. Maximum Coolant Temperature <u>vs</u> Time for Step Reactivity Inputs from Full Power	24
V-7. Maximum Hot Channel Fuel Temperature of the Central Fuel Element <u>vs</u> Time for Step Reactivity Inputs from 50% Power	25
V-8. Maximum Coolant Temperature <u>vs</u> Time for Step Reactivity Inputs From 50% Power.	25
V-9. Maximum Hot Channel Fuel Temperature of the Central Fuel Element <u>vs</u> Time for Step Reactivity Inputs from 5% Power	26
V-10. Maximum Coolant Temperature <u>vs</u> Time for Step Reactivity Inputs from 5% Power	26
VI-1. Power <u>vs</u> Time for Ramp Reactivity Insertions from Full Power with Protective System Action (scram trip point $\Delta P/P_0 = 0.2$; scram delay time = 1 sec)	29
VI-2. Maximum Hot Channel Fuel Temperature <u>vs</u> Time for Ramp Reactivity Insertions from Full Power with Protective System Action (scram trip point $\Delta P/P_0 = 0.2$; scram delay time = 1 sec) . .	30
VI-3. Coolant Channel Outlet Sodium Temperatures <u>vs</u> Time for Ramp Reactivity Insertions from Full Power with Protective System Action (scram trip point $\Delta P/P_0 = 0.2$; scram delay time = 1 sec) . .	30

FIGURES

Page

VI-4.	Power and Maximum Temperature <u>vs</u> Time for a Startup Accident with Protective System Action (scram trip point $P = 0.2 P_0$; scram delay time = 1 sec; reactivity insertion rate $0.0002 \Delta k/sec$) . .	32
VI-5.	Power and Maximum Temperatures <u>vs</u> Time for Loss of Forced Primary Coolant Flow with Protective System Action, Reactor Initially at Full Power (scram delay time = 1 sec)	33
VI-6.	Power and Maximum Temperatures <u>vs</u> Time for Loss of Forced Primary Coolant Flow with Protective System Action, Reactor Initially at 5% of Full Power (scram delay time = 1 sec)	33
VI-7.	Power <u>vs</u> Time for Ramp Reactivity Insertions from Full Power Without Protective System Action (total rod withdrawal time = 20 sec).	35
VI-8.	Maximum Hot Channel Fuel Temperature <u>vs</u> Time for Ramp Reactivity Insertions Without Protective System Action (total rod withdrawal time = 20 sec).	36
VI-9.	Coolant Channel Outlet Sodium Temperatures <u>vs</u> Time for Ramp Reactivity Insertions from Full Power Without Protective System Action (total rod withdrawal time = 20 sec).	36
VI-10.	Power and Temperature <u>vs</u> Time for a Startup Accident Without Protective System Action, Continuous Four-Rod Withdrawal (rod speed = 1 in./sec; reactivity rate = $0.0002 \Delta k/sec$)	38
VI-11.	Power and Temperatures <u>vs</u> Time for a Startup Accident Without Protective System Action, Four-Rod Withdrawal for 50 sec (rod speed = 1 in./sec; reactivity rate = $0.0002 \Delta k/sec$)	39
VI-12.	Power and Maximum Temperatures <u>vs</u> Time for Loss of Forced Primary Coolant Flow Without Protective System Action, Reactor Initially at Full Power and Full Flow	41
VI-13.	Power and Maximum Temperatures <u>vs</u> Time for Loss of Forced Primary Coolant Flow Without Protective System Action, Reactor Initially at 5% Power and 5% Flow	41

ABSTRACT

An analysis of the transient behavior of a 255-Mwe advanced sodium graphite reactor, previously described in NAA-SR-3829, is presented. The reactor and its components are briefly described. Nuclear and thermal characteristics are presented as far as they affect reactor kinetics or are essential in interpreting the results. The study includes an investigation of the inherent kinetic characteristics of the reactor, as well as an analysis of its transient behavior for all conceivable conditions of abnormal operations. Assumed reactor excursions are analyzed with and without ensuing protective system action.

It is shown that the reactor is dynamically stable and that power transients which are followed by normal protective system actions will not lead to potentially unsafe conditions. The conclusion is reached furthermore, that uncontrolled rod withdrawal accidents from source power will be terminated by coolant "choking" and fuel meltdown before extensive coolant boiling occurs, and that the large thermal capacity and long-time constant of the upper plenum will provide protection against pool boiling for other less serious accidents until the reactor can be shut down by external means.

I. INTRODUCTION

The subject of this study is the sodium-cooled, graphite-moderated reactor system described in detail in NAA-SR-3829.* The core is a calandria assembly, consisting of sodium cooled UC fuel elements, surrounded by a stainless steel tank which contains the moderator and reflector graphite. The reactor is designed to produce 256 Mwe from 606 Mw of thermal power at its full power rating.

The objective of this study was to determine the reactor transient behavior for various conceivable abnormal situations. This is the third of three studies concerning this conceptual design. The first presented plant design and economics,* and the second, fuel cycle studies.† The general purpose of these studies is to establish the competitive position of this nuclear power plant as a source of electrical energy.

The reactor, the associated power plant, and routine reactor operations are described only briefly in this report, while emphasis is placed on the analysis of inherent reactor stability, on unusual operating conditions and resulting potential hazards. The study of reactor transients is separated into two parts. Transients resulting from step reactivity insertions, are investigated to determine the reactor's dynamic response only and are included in the section on reactor kinetics. Transients which result from unusual operating conditions are studied separately as part of the reactor excursion analysis.

Several hypothetical reactor excursions, with and without simultaneous failure of the conventional safety system, are investigated and their effects on the reactor and its components evaluated. The analysis was conducted by solving the equations of reactor kinetic behavior by an analog computer. Reactivity-induced transients were studied for various conditions of rod withdrawal accidents from different initial power levels. Loss-of-coolant-flow accidents were investigated assuming an instantaneous change from forced flow to purely natural convection flow. Results of the analog study are presented graphically for the various excursions investigated and are discussed wherever they appear in the text.

It should be noted that, although the type of studies reported herewith are incorporated in Summary Safeguard reports, the contents of this report are limited to transient response only and do not comprise a complete safeguard analysis.

*J. R. Churchill and J. Renard, "An Advanced Sodium-Graphite Reactor Nuclear Power Plant," NAA-SR-3829 (March 1960)

†A. L. Aronson, "Fuel Burnup Studies for a 255-Mwe Advanced Sodium Graphite Reactor," NAA-SR-4529 (June 1960)

II. DESCRIPTION OF REACTOR

A. GENERAL DESCRIPTION

The Advanced Sodium Graphite Reactor of this report, is fueled with uranium carbide, cooled by sodium, and graphite moderated. The graphite moderator and reflector elements are contained in a calandria vessel, which is completely immersed in sodium (Figure II-1). Surrounding the calandria vessel is a neutron shield of iron and titanium hydride. This shield prevents the secondary sodium in the three sodium-to-sodium heat exchangers, located in the annulus between neutron shield and reactor vessel from becoming radioactive.

The fuel consists of clusters of stainless steel clad and sodium bonded cylindrical rods. Process channels containing the fuel rods, and the moderator graphite elements are arranged in a matrix pattern to provide optimal nuclear and thermal conditions. Heat generated by the fissioning process is absorbed by the sodium coolant which is pumped through the core. The process channels are orificed to provide uniform coolant outlet temperatures radially across the core. A constant temperature rise in the core is accomplished by varying the coolant flow with the reactor power.

A plant control system allows automatic operation over the range of 20 to 100% load or manual operation at all loads. The plant is designed to operate on a load following basis. A plant protective system provides rapid detection of any off-normal condition in the reactor plant and automatically takes corrective action. This action includes alarm, setback, and scram. All power plant control is from a central area which integrates both reactor and turbine-generator instrumentation. Monitoring systems with alarms to indicate radiation levels in excess of predetermined values are provided.

B. REACTOR CORE

The reactor core is a fuel, moderator, and reflector assembly contained in a stainless steel calandria tank. The moderator and reflector are graphite in the form of vertical logs of hexagonal cross section. In the active section of the core, each graphite log is pierced by a vertical process channel. These channels contain the fuel elements, the control elements, and a neutron source.

Figure II-1. General Arrangement, Section Elevation (7519-5238)

The calandria tank is a cylindrical vessel 16.5 ft in diameter and 19 ft high. The dished top head and vessel shell walls are 3/4 in. thick, and the bottom head is 1-in. thick. Process tubes are welded into nozzles at the lower head, and bellows, to provide for differential thermal expansion, at the upper head. The calandria tank is designed as a leak-tight vessel. However, provisions for removal of small amounts of sodium which might enter the tank as a result of a minor leak have been incorporated.

The calandria interior atmosphere is helium pressurized to ~10 psig. A small amount of helium is bled intermittently through a detector line which extends to the calandria bottom. In the event of a sodium leak, gas flow through the detector line will be stopped by the sodium, and the resulting pressure difference between vessel and detector line would automatically start the sodium pump-out system.

Each fuel element consists of a cluster of 19 fuel rods, an intermediate hanger rod, and a shield plug. The assembly hangs in the process channel supported by the top shield, which serves as the loading face. The fuel rods are 0.5-in. diameter uranium carbide slugs, sheathed in 0.52-in.-ID by 0.01-in.-thick-wall stainless steel jackets. The rods are 14 ft long with the lower 12 ft containing the fuel slugs and the remaining 2 ft providing space for fission gases. A sodium bond is used between the uranium carbide slugs and the jacket. End caps, welded to the jackets, seal the fuel slugs and sodium in a helium atmosphere.

Surrounding the calandria tank is an annular neutron shield, 17 ft ID by 21 ft OD and 31 ft high, to prevent secondary sodium in the primary heat exchangers from becoming radioactive. The shield is a stainless steel tank of 3/4-in. wall thickness, filled with 10 in. of carbon steel adjacent to the inside wall followed by 8-1/2-in. of titanium hydride and 3/4-in. of 1% boron steel. The iron in the neutron shield serves to attenuate neutrons of energies greater than 1 Mev by inelastic scattering and to absorb thermal neutrons. Titanium hydride is used to moderate epithermal neutrons which are then absorbed in the boron steel.

C. PRIMARY AND SECONDARY SODIUM SYSTEMS

Primary sodium coolant is completely contained within the reactor vessel. The coolant, after absorbing heat from the fuel elements and moderator in the

process tubes, flows upward into the upper plenum. The coolant flow in each fuel channel is orificed to provide a uniform coolant exit temperature distribution across the core. Flow orificing is accomplished by a rectangular opening in the orifice tube, which is located vertically to match a triangular shaped opening in the calandria entry guide. The effective orifice area, and therefore the coolant flow, is varied by rotating the orifice tube from the reactor floor.

After entering the upper plenum, the sodium flows over the neutron shield wall into the annulus between the shield and the reactor vessel wall. Three intermediate heat exchangers, equally spaced in this annulus, transfer heat from the primary sodium to nonradioactive secondary sodium. Below the primary heat exchangers is the suction side of nine electromagnetic pumps, three per heat exchanger. Each pump is designed for pumping sodium at a rate of 3400 gpm against a pressure differential of 40 psi. Cooled sodium is then discharged into the inlet plenum below the calandria core.

The non-radioactive secondary sodium system consists of three parallel but independent loops. In each loop the sodium coolant is pumped from an intermediate heat exchanger to a steam generator. A variable-flow electromagnetic pump in each loop provides flow control to compensate for load changes from the turbine. In case of reactor scram or malfunctioning of a single loop, a throttling valve is used to reduce or shut-off sodium circulation.

Three steam generators, one in each secondary sodium loop, produce 1.835×10^6 lb of steam per hour to the turbine throttle at 1050°F and 2400 psig. Shell-and-tube, once-through steam generators are used with steam and water on the tube side and sodium on the shell side.

D. SHIM AND SAFETY RODS

Eighteen combination shim and safety rods permit an estimated 12% control of reactivity, which is adequate to handle reactivity changes associated with temperature, poisons, and fuel depletion. It is not, however, adequate to control the initial startup; the large amount of excess reactivity in the first core would be reduced by not loading all of the elements initially, or by inserting poison elements in some of the spare fuel channels. Each control element operates in a thin-walled stainless steel thimble which extends from the top

of the loading face down through the core. The poison rings are suspended on a steel pull tube and move vertically by a drive mechanism located in the upper portion of the thimble. When scram action is required, negative reactivity is quickly inserted by de-energizing the motor, and the rods are driven into the core under forces of gravity.

Whenever a shim rod is fully inserted, the entire mechanism is located below the reactor loading face in order to facilitate refueling. An upper thimble is provided to contain the drive mechanism and to maintain a helium atmosphere, since the drive screw is withdrawn above the loading face when the reactor is in operation. In addition, this upper thimble provides a torque-restraint to prevent screw rotation.

The rods can be operated automatically by the plant control and protective system or manually by the operator at the control console. An interlock prevents the shim regulating rods from being raised until all the safety rods are fully withdrawn. The shim rods are equipped with limit switches which stop the drive motors at upper and lower limits of rod position. Upper and lower mechanical stops are also provided. The drive mechanisms are torque-limited in case binding occurs or a limit switch fails.

E. REACTOR INSTRUMENTATION

Reactor instrumentation is provided for the measurement of neutron flux, temperatures, flow rates, sodium level, and shim rod positions.

Neutron flux instrumentation is divided into seven channels: two source range, two intermediate range, and three power range channels. These instruments provide adequate flux measurements over the power range from source level to 150% of design rated power. The seven neutron detectors are located in a circular pattern at the bottom of the loading face shield. A beam tube extends vertically from the top head of the calandria vessel to a position below each detector.

The count-rate channels, which measure neutron fluxes between source level and 0.0025 % of design rated power are used during reactor startup. Two compensated ion chambers, which transmit a signal to logarithmic amplifiers and neutron flux recorders, are used for reactor startup over the range from 0.0001 to 10 % of design rated power. Period amplifiers are tied into the plant

protective system so that if the reactor period drops below pre-set limits, an alarm, a setback, or scram is actuated. The scram circuit is automatically deactivated above 1% of full power to avoid spurious scrams caused by noise in the period circuitry. The three power level channels, which employ uncompensated ionization chambers, cover the range of normal power operation.

Each fuel channel sodium exit temperature is measured by two thermocouples installed at the bottom of each orifice tube. Temperature signals from three preselected fuel channels are used for normal reactor control as well as for activating the plant protective system through coincident circuitry. An alarm, setback, or scram may be initiated in the event of excessive fuel channel sodium exit temperatures. Additional thermocouples are located throughout the reactor vessel and in the biological shielding.

Sodium systems are provided with instrumentation to measure sodium pressures, temperatures, flow rates, levels, and to detect sodium leaks. Pressure sensing elements utilize a NaK filled capsule welded in the sodium line and transmit pressure changes through a capillary tube to an indicator. Sodium temperature sensing elements are removable. Incremental or continuous induction-type transducers are used for measuring sodium levels. Sodium level is measured over the total operating range, and high and low level alarms are provided. Sodium leak detectors are located on primary sodium equipment, which is relatively inaccessible during operation. Each detector consists of a mesh-type electrode which becomes shorted when contacted by sodium. In general, the leak detectors are located in positions where leakage might occur or can readily be detected.

F. CONTROL AND PROTECTIVE SYSTEM

The plant control system provides for automatic operation over the range from 20% to 100% of design rated load, and for manual operation at all loads.

The plant is designed to operate as a load following system, with reactor power level and sodium flow rates varied according to steam demand. Steam flow and pressure vary excitation to the field windings of the secondary electromagnetic pumps, thus controlling the flow of secondary sodium. Primary flow is adjusted to match secondary flow. The average outlet sodium temperature from representative fuel channels is used to control reactor power by initiating changes in shim-rod position.

Instrumentation is provided for the purpose of rapidly detecting any off-normal condition in the reactor plant and, when necessary, automatically reducing reactor power level and coolant flow. This instrumentation and control system constitutes the reactor plant protective system. It is in addition to the plant control system, which serves to regulate power generation during normal operation.

In the event of an off-normal condition which is of such severity that its continuation could cause damage to any part of the reactor plant, the protective system would take corrective action automatically.

The types of protective system action taken in the event of off-normal conditions include:

- 1) Alarm
- 2) Rod interlock
- 3) Setback
- 4) Scram.

By the combination of these various types of protective action, reliability is ensured while maintaining the maximum practicable level of safety.

1. Alarm

Alarm annunciators which do not initiate automatic corrective action are provided to warn the operator whenever an important reactor variable reaches an off-normal condition. They include warnings from reactor auxiliary systems as well as from the principal reactor systems.

An alarm makes the operator aware of off-normal conditions in any circuit, whether caused by an operating signal through a functioning circuit or a signal caused by failure of a circuit. Each alarm consists of an audible annunciator and a visual indicator. The operator must acknowledge each alarm signal by pushing an appropriate button. This silences the audible signal, but leaves a visual indication of the off-normal condition until it has been corrected.

All signals, which may cause automatic corrective action, have alarms associated with them. The alarm system is further arranged to give the operator a signal as to which off-normal condition caused an automatic power reduction, even though several variables may be shown to be out of the normal operating range following the automatic action.

2. Rod Interlock

A rod withdrawal interlock circuit prevents withdrawal of the control rods when conditions exist which would result in an incorrect flux level indication or potentially unsafe reactor startup conditions caused by one of the following:

- a) Short period (intermediate range)
- b) Low detector supply voltage
- c) Nuclear instrument power failure.

3. Setback

Any off-normal condition arising within the reactor as a result of changes in reactor power level, coolant temperature, or coolant flow must be corrected before reactor temperatures or rates of change of temperatures become excessively high. The setback in reactor power level is the first automatic action to correct such conditions and any other off-normal conditions in the reactor short of a complete shutdown.

A setback causes the shim rods to be inserted at a controlled rate resulting in a relatively slow power reduction. Setbacks are initiated whenever a condition is not corrected after indication by an alarm and increase in severity to the set point for a setback. The power reduction is continued until all conditions requiring a setback are corrected.

4. Scram

An automatic reactor scram is a rapid shutdown action, resulting in insertion of all shim and safety rods into the reactor core. It is only initiated if other automatic measures of the plant protective system should fail to correct a condition which is potentially dangerous to the plant and personnel. The reactor may be scrammed manually if the operator is not satisfied that a serious condition is being adequately corrected.

The power, flow, and temperature circuits have scram set points beyond the alarm and other protective system set points, to permit operator or automatic action before a scram. In the cases of a severe earthquake or loss of electric power to all three of the primary sodium pumps, the reactor is scrammed immediately without a prior alarm or lesser action.

Thermocouples measuring exit sodium temperatures from three selected fuel channels are used to initiate a scram prior to reaching temperatures high enough to cause severe overheating of the reactor components. The scram set point is the highest of successive temperature set points which initiate alarm, setback, and scram action.

Loss of power to the relays in the electrical power lines supplying the primary or secondary sodium pumps results in a scram. There is a time delay in this case, in order to avoid scrams from electrical power losses which are brief enough to cause no damage to the reactor. The scram from loss of electrical power normally precedes the signals which would initiate a scram caused by low sodium flow or high fuel channel temperature. The emergency power systems maintain sufficient electric power supply for minimum pumping requirements to remove afterglow heat from the reactor.

G. OPERATING PARAMETERS

Table II-1 summarizes various characteristics of the 255Mw ASGR and the power plant.

TABLE II-1
SUMMARY OF TENTATIVE CHARACTERISTICS OF THE 255-Mwe ASGR

Reactor	
Reactor thermal power (Mw)	613*
Reactor coolant	Sodium
Coolant temperature (°F)	
Inlet	625*
Outlet	1200
Coolant flow (lb/hr)	12.6×10^6
Number of primary loops	3
Secondary coolant	Sodium
Coolant temperature (°F)	
Inlet	600
Outlet	1150
Coolant flow (lb/hr)	12.6×10^6

TABLE II-1 (Continued)

Fuel	19-rod cluster of 1/2-in. diameter x 1-in. UC pellets in stainless steel tubes with sodium bond
Number of fuel elements	151
Number of shim-safety rods	18
Initial enrichment	2.5 at. % U ²³⁵ *
Initial excess reactivity with equilibrium Xe and Sm poison	0.127
Initial conversion ratio	0.505
Average fuel burnup (Mwd/MTU)	17,100
Uranium in core (kg)	16,090*
Active core size (ft)	12 dia by 12 high
Peak-to-average power ratio	3.15*
Specific power, average for new core (kw/kg U ²³⁵)	1400
Maximum fuel temperature (°F)	2250
Average heat flux (Btu/hr-ft ²)	425,000
Average thermal neutron flux in fuel (n/cm ² -sec)	8 x 10 ¹³
Lattice spacing, triangular (in.)	10
Power Plant	
Net electrical power (Mw)	256
Gross electrical power (Mw)	279
Gross cycle heat rate (Btu/kw-hr)	7750
Throttle steam flow (lb/hr)	1.835 x 10 ⁶
Throttle steam conditions	2400 psig, 1050°F
Reheat steam conditions	504 psig, 1000°F
Feedwater temperature (°F)	540

*These characteristics are based on preliminary parametric studies and are therefore somewhat different from the optimized values previously reported (Churchill and Renard, and Aronson, op. cit.). The changes however, are not expected to significantly affect the results and conclusions of this report.

III. NUCLEAR CHARACTERISTICS

The inherent safety of any reactor is primarily dependent upon the nuclear and thermal characteristics of its constituents. Such properties as excess reactivity, temperature coefficients, and neutron lifetime have a strong effect upon the type of operating transients which can be experienced. Each of these parameters is discussed in the following paragraphs.

A. EXCESS REACTIVITY

Excess reactivity available for a power reactor is a function of the initial loading and the operating history. The initial loading is determined by requirements of power output, burnup, equilibrium xenon concentration, and xenon buildup override. Figure III-1 shows the variation of reactivity with burnup.

Figure III-1. Variation of Reactivity with Burnup
(2.5% enrichment)

As can be seen from Figure III-1, the maximum excess reactivity which is available for this reactor is about 11.3% (for the cold, clean reactor). However, after buildup of equilibrium xenon poisoning (a few hours after original startup), the reactivity has dropped to 8.5% and thereafter declines slowly with core exposure. The excess reactivity of initial startup will be less, by a reasonable margin, than the reactivity controlled by the safety and shim rods.

B. NEUTRON LIFETIME

The neutron lifetime is equivalent to the mean delay time between successive neutron generations. Lifetime enters as an important parameter in the analysis of reactor kinetics, since it determines the time rate of change of neutron density for any particular excess reactivity which is introduced into the reactor.

The lifetime for this reactor has been calculated for the 2.5% enrichment considered and found to be 3.96×10^{-4} sec.

C. REACTIVITY COEFFICIENTS

The reactivity response of reactor components to thermal transients is the inherent mechanism which underlies the self-regulating features of a reactor. This response is expressed in the power coefficient, which results from temperature effects on the neutron multiplication for a change in power level. The magnitude and sign of the power coefficient have an important influence on the severity of accidents which can be experienced. A negative prompt power coefficient will quickly reduce any excess reactivity accidentally introduced and restore equilibrium, while a positive coefficient will increase the excess reactivity, thereby adding to the unbalance of the system. The coefficient is a combination of the effects of the reactor component temperature coefficients, which are:

- 1) Fuel temperature coefficient
- 2) Moderator temperature coefficient
- 3) Coolant temperature coefficient
- 4) Structural temperature coefficient.

The coolant and structural temperature coefficients for this reactor were found to be negligible and were therefore not included.

1. Fuel Temperature Coefficient

The fuel temperature coefficient is primarily a function of the Doppler effect which results from resonance broadening with increased fuel temperatures. The coefficient is evaluated by plotting the change in reactivity resulting from a change in resonance escape probability for various average fuel temperatures. (Figure III-2) The slope of this curve is the desired coefficient for this reactor and is equal to $-3.46 \times 10^{-5} \Delta k/^{\circ}\text{C}$.

Figure III-2. k_{eff} vs Fuel Temperature,
Equilibrium Xenon and Samarium Poison
(2.5 % enrichment)

2. Moderator Temperature Coefficient

The moderator temperature coefficient of reactivity results from the effect of a moderator temperature change on the average thermal neutron energy. This effect is brought about by the change of reactor parameters such as fission and absorption cross sections and the cell flux distribution. The coefficient was calculated by computing the excess criticality for different moderator temperatures while keeping the fuel temperature constant. Results of these calculations are shown in Figure III-3.

The moderator temperature coefficient is seen to decrease with increasing moderator temperatures, and to become negative at about 1000°C. This behavior results from the difference in the change of η (the average number of neutrons liberated for each neutron absorbed in the fuel) and f (the thermal utilization) with increasing neutron temperature (Figure III-4). The value of η decreases as the neutron energy increases. At the same time the thermal utilization increases. Eventually, the rate of increase in thermal utilization is smaller than the rate of change of η and the product of these two properties starts to decrease, resulting in a decrease in the effective multiplications and a consequent change in sign of the moderator temperature coefficient.

The full power moderator coefficient is seen from Figure III-3 to be equal to $-2.5 \times 10^{-5} \Delta k/^\circ\text{C}$.

Figure III-3. k_{eff} vs Moderator Temperature (2.5% enrichment)

Figure III-4. η , f , and ηf vs Neutron Temperature (2.5% enrichment)

3. Power Coefficient

The power coefficient of reactivity is the summation of the individual reactivity effects of the reactor component temperature coefficients and is defined as:

$$\frac{dk}{dP} = \gamma \frac{dT_m}{dP} + \delta \frac{dT_f}{dP},$$

where

$\frac{dk}{dP}$ = incremental change in multiplication for an incremental power change.

$\frac{dT_f}{dP}$ = incremental change in fuel temperature for an incremental power change.

$\frac{dT_m}{dP}$ = incremental change in moderator temperature for an incremental power change

γ = moderator temperature coefficient

δ = Fuel temperature coefficient.

Since the contribution of the power coefficient to reactivity is dependent on the temperature feedback from reactor components, it can give no obvious indication of the dynamic reactor behavior other than to indicate the algebraic sign of reactivity contribution. As the temperature coefficients of the fuel and moderator are negative at rated power, it is apparent from the definition above, that the overall power coefficient is negative also.

D. NUCLEAR PARAMETERS

Nuclear calculations were made using a two-group approximation. For the thermal group, a neutron temperature of 1250°F (0.131 ev) was used, based on the average moderator (graphite) temperature in the core. A Maxwellian neutron spectrum was assumed, and average thermal cross sections for materials were computed.

Thermal cross sections were volume and flux weighted in computing group constants. Fast group constants were obtained using the IBM-709 MUFT IV

code, (WAPD-TM-72). With these results, criticality calculations were performed using the IBM-704 WANDA Code (WAPD-TM-28). Initial conversion ratio was calculated from a compatible two-group approximation that accounts for fast fissions in U^{235} . Table III-1 lists the nuclear constants at full power for 2.5 at. % U^{235} enrichments.

TABLE III-1
NUCLEAR CONSTANTS WITH 2.5 at. % U^{235} ENRICHMENT

Constant *	Core	Axial Reflector	Radial Reflector
Σ_{r1}	0.002384	0.003175	0.003522
Σ_{a1}	0.001527	0.000117	0.000013
$\nu\Sigma_{f1}$	0.001115	-	-
D_1	1.2034	1.2412	1.1741
Σ_{a2}	0.004929	0.000972	0.000197
$\nu\Sigma_{f2}$	0.007209	-	-
D_2	0.8831	0.9309	0.8511
L^2	179.182	960.0	4330.0
τ	309.88	381.01	332.30
B^2	0.000149	-	-

$\bar{\phi}_2$ (core) 1.415×10^{14} neutrons/cm²-sec

$\bar{\phi}_2$ (fuel) 6.60×10^{13} neutrons/cm²-sec

$\bar{\phi}_1$ (core) 2.77×10^{14} neutrons/cm²-sec

Fuel Loading: 16.09 MTU, 397.0 kg U^{235}

*Reactor at full power operation.

IV. THERMAL CHARACTERISTICS

A. THERMAL CHARACTERISTICS AT STEADY STATE

In calculating the steady state thermal performance for the reactor, the following assumptions and limitations were used:

- 1) Axial power distribution can be represented by a chopped cosine curve.
- 2) Radial power distribution can be represented by a Bessel function of order zero.
- 3) Coolant flow is throttled in each coolant channel to obtain a constant outlet temperature.
- 4) Heat generated in the fuel is transmitted to the coolant in a radial direction only.
- 5) The volumetric heat source for each fuel rod is constant in a radial direction.
- 6) Thermal conductivity and specific heat do not vary over the temperature ranges considered.
- 7) A maximum fuel temperature of 2250°F for uranium carbide fuel.
- 8) A maximum pressure drop limitation across the fuel elements of 25 psi, excluding end losses.

A peak fuel temperature of 2250°F is obtained in the outer ring of rods of the central fuel elements approximately 1.2 ft above the reactor core centerline based on the following disadvantage factors:

Radial flux peaking factor	1.6
Axial flux peaking factor	1.38
Element flux peaking factor	1.18
Uncertainty flux peaking factor	1.10
Variation in thermal conductivity	1.10
Maximum/average temperature factor*	3.15

*This is defined as the ratio of maximum fuel, average coolant temperature difference to the center fuel temperature, average coolant temperature difference of the average fuel rod.

V. REACTOR KINETICS

As with any dynamic system which is designed for operation at steady state, the requirements for satisfactory performance of a nuclear reactor are smooth and steady operation at conditions for which it is designed. The necessity of providing means for start-up, shut-down or changes in operating level, requires that such actions be achieved smoothly and within the shortest possible time.

All of these conditions demand a high degree of stability or inherent self correcting tendency. Stability implies that reactor power and component temperature are continuous functions of reactivity. This characteristic will be investigated by conventional methods in the following section. Mathematical analysis of stability, however, can give no convenient information about the dynamic response of the reactor to reactivity insertion by external means. In section B therefore, the kinetic reactor behavior is investigated for a variety of arbitrarily selected imaginary step functions.

A. REACTOR STABILITY

The reactor dynamic behavior can be conveniently separated into two parts, high and low frequency dynamics. This is made possible because of the difference in time behavior of temperature changes of the core components during reactor transients. The large heat capacity and low heat transfer coefficient of the graphite moderator make its temperature response to power variations very sluggish, while the fuel and coolant temperatures respond rapidly to power changes. In the analysis of high frequency dynamics, therefore, the reactivity feedback of the moderator is small compared to that of the fuel and coolant, and can be neglected.

A fairly good estimate of the relative time behavior can be obtained from the time constants of the components. These time constants were calculated to be as follows:

fuel	0.86 sec
moderator	0.32 min
coolant	0.218 sec

The reactor kinetic stability was investigated using conventional feedback system techniques, and was found to be stable for all high frequency dynamics.

Theoretical low frequency stability is obtained only at higher power levels. The moderator temperature coefficient of reactivity is negative at temperatures exceeding approximately 1000°C. Below this temperature the moderator coefficient is positive, and as a result the reactor is theoretically unstable. However, the frequencies concerned are in the order of magnitude of the moderator time constant (~32 min). Since this is well within the limit of normal reactor control by manual shimrod operation, these dynamics are not considered adverse to safe reactor operation.

An open loop Nyquist diagram for high frequency dynamics is shown in Figure V-1. It is apparent that the phase shift is always less than 180°, indicating stability at infinite gain margin.

B. DYNAMIC RESPONSE TO REACTIVITY STEP FUNCTIONS

Although the reactor is theoretically stable under normal operating conditions, its degree of stability or the extent to which its inherent self-correcting tendency opposes sudden reactivity insertions, can best be demonstrated by investigating its kinetic behavior when the steady state is disturbed by external means. This analysis is performed by subjecting a mathematical reactor model to various step reactivity functions and recording the dynamic response.

These step reactivity insertions are hypothetical and the resulting power and temperature changes are only indirectly indicative of transients which can be experienced with the actual reactor. Pure step reactivity insertions in a real reactor are in fact impossible to achieve, since reactivity can only be added by either introducing fertile material into the core, withdrawing poison, or decreasing the neutron leakage. All of these methods require the movement and acceleration of relatively large masses of material, and even the most ingenious way of attempting a step would at best result in a fast ramp.

Theoretical step insertions in a mathematical reactor model are useful in demonstrating the inherent response of a reactor to accidental reactivity insertions. Furthermore, it shows the limiting power and temperature conditions for any conceivable accident, since a ramp reactivity insertion can never have consequences more severe than a step of equal magnitude.

Figure V-1. Open Loop Reactor Transfer Function $[ZP(s) PK(s)]$

Figures V-2 to V-4 are graphical representations of power transients for various step reactivity insertions from three initial steady state conditions.* Figures V-5 to V-10 show the associated maximum hot channel fuel temperatures[†] and maximum coolant temperatures.[§] The changes in moderator temperatures were found to be negligible in all cases, and are therefore not shown.

The sharp peaks in the power curves, demonstrate the tendency of the reactor immediately to offset the initial reactivity increase and return to a new stable power level. This tendency results from the prompt negative power coefficient associated with the fuel temperature feedback. The small thermal capacity of the fuel allows its temperature initially to increase rapidly, thereby favoring a prompt negative reactivity feedback, while the relatively large heat transfer coefficient provides sufficient damping to induce the power to level off immediately to a new equilibrium state.

Figure V-2. Total Power vs Time for Step Reactivity Inputs from Full Power

*The curves $\Delta k = 0.0064$ correspond to prompt critical; excess reactivity in this case is approximately equal to the average worth of one control rod.

[†]Defined in Appendix, Section C.

[§]Includes a hot channel factor of 1.016.

Figure V-3. Total Power vs Time for Step Reactivity Inputs from 50% Power

Figure V-4. Total Power vs Time for Step Reactivity Inputs from 5% Power

Figure V-5. Maximum Hot Channel Fuel Temperature of the Central Fuel Element vs Time for Step Reactivity Inputs from Full Power

Figure V-6. Maximum Coolant Temperature vs Time for Step Reactivity Inputs from Full Power

Figure V-7. Maximum Hot Channel Fuel Temperature of the Central Fuel Element vs Time for Step Reactivity Inputs from 50% Power

Figure V-8. Maximum Coolant Temperature vs Time for Step Reactivity Inputs from 50% Power

Figure V-9. Maximum Hot Channel Fuel Temperature of the Central Fuel Element vs Time for Step Reactivity Inputs from 5% Power

Figure 10. Maximum Coolant Temperature vs Time for Step Reactivity Inputs from 5% Power

VI. REACTOR EXCURSION ANALYSIS

A major element of the excursion analysis for nuclear reactor systems is the evaluation of the system kinetic behavior during operating transients. This principle of evaluation is governed by the presumption that reactor transients are the most probable sources of hazardous conditions. The effect of a transient on a reactor is primarily a pronounced and rapid change in the operating power and temperatures. Such changes are undesirable since they may lead to failure of some of the reactor components. Although provisions are made to reduce the occurrence of transients to a minimum, it cannot be guaranteed that they will never occur since the same mechanism by which load changes are made can, by improper or faulty operation, induce an abnormal transient.

In the following sections of this report, an evaluation of the degree of safety afforded with this reactor under specified conditions of transient operation will be presented. In particular it will be attempted to determine and analyze the type of transient behavior which might lead to unsafe conditions.

Unsafe conditions are a single or a succession of events which have as their result a release of radioactivity to the reactor building or surrounding. Unsafe conditions therefore, do not necessarily include events which have as their only consequence a physical damage to the reactor or to some of its components though these conditions are a great concern to the reactor operators.

Transients can be classified into three groups: fast, intermediate, and of slow speed, corresponding in sodium graphite reactors to reactor periods of approximately <0.1 sec, 0.1 to 20 sec, and >20 sec, respectively. A rigorous analysis of a fast transient response would have to include a mathematical description of the reactor system in such detail as to represent accurately the reactor kinetics on the microscopic scale. Fast transients, however, are significant for reactors with short neutron lifetime ($<10^{-5}$ sec) in which large amounts of reactivity can be introduced. In this reactor the neutron lifetime is relatively long (3.96×10^{-4} sec), and possible reactivity additions are limited in amounts and rate by factors which are intrinsic to this particular reactor. Therefore, fast transients have not been considered.

Slow transients are primarily important in evaluating instrumentation and control calibrations, reactor-plant interactions, etc. Such factors as poisoning

effects, decay heat contributions and power plant transients become significant. Slow transients are important only in complete and detailed reactor safeguard evaluations and have not been considered in this report.

Reactor transients, or off-normal conditions leading to transients, can have any of several causes, including the following:

- a) Electrical system malfunction
- b) Plant control system malfunction
- c) Plant component malfunction
- d) Manual misoperation.

In the following discussion it will not be attempted to predict the probability of occurrence of the aforementioned causes. Neither will their individual effect on reactivity be discussed, other than to state that the result of any of these causes will be reactivity changes which can be represented by simple ramp reactivity insertions.

Two types of reactor excursions will be investigated: transients with protective system action, and transients without protective system action. As the designation implies, transients with protective system action are those in which the protective system is the principal factor in terminating the excursion, either by taking automatic corrective action in the form of a reactor power reduction or by completely scramming the reactor. The various forms of protective system action are discussed in section II-F. For the purpose of this analysis, only excursions which are terminated by scrams will be considered.

A. EXCURSIONS WITH PROTECTIVE SYSTEM ACTION

Scrams are brought about by gravity insertion of shim and safety rods, thereby introducing a nuclear poison into the core. Since the flux varies radially and axially in the core and the rod speed is changing with distance of travel, the rate of poisoning will vary with time after initiation of the scram. At the beginning of rod travel, the rate of poison addition is small but increases exponentially, while towards the middle of travel the poisoning increases almost linearly. For practical purposes a scram can be represented by a negative ramp which is started a small time interval after the initiation of the scram.

The different corrective system actions, set-backs, alarms, or scrams, will occur a finite time after an off-normal condition has been detected, depending on which condition causes the correction, and on the set-point, sensitivity, and dead time of the sensing instrument.

The total time interval between detection of off-normal conditions and beginning of correction, is the protective action delay time. This time interval varies with the off-normal conditions and with different protective system action. Conservative estimates range from a few milliseconds up to a few hundred milliseconds. For the purpose of the kinetic analysis, all scram simulations were delayed for one second after a power level trip point of $\Delta P/P_0 = 0.2$, which was arbitrarily assumed to correspond to the point of detection of the off-normal conditions.

1. Rod-Withdrawal Transient from Full Power

Three rates of ramp reactivity insertions were investigated, beginning with a steady state condition equivalent to full power. The three ramps are 0.00005, 0.0001 and 0.0002 $\Delta k/\text{sec}$, corresponding to single rod withdrawals with speeds of approximately 1, 2, and 4 in./sec. The results are presented graphically in Figures VI-1 to VI-3.

Figure VI-1. Power vs Time for Ramp Reactivity Insertions from Full Power with Protective System Action (Scram trip point $\Delta P/P_0 = 0.2$; scram delay time = 1 sec)

Figure VI-2. Maximum Hot Channel Fuel Temperature vs Time for Ramp Reactivity Insertions from Full Power with Protective System Action (Scram trip point $\Delta P/P_0 = 0.2$; scram delay time = 1 sec)

Figure VI-3. Coolant Channel Outlet Sodium Temperatures vs Time for Ramp Reactivity Insertions from Full Power with Protective System Action (Scram trip point $\Delta P/P_0 = 0.2$; scram delay time = 1 sec)

Figure VI-1 shows the resulting power transient and Figure VI-2 and VI-3 the associated maximum hot channel fuel* and process channel outlet sodium temperatures. The reactor power as well as the component temperatures are seen to increase linearly until the protective system action (scram) becomes effective (one second after $\Delta P/P_0 = 0.20$). After the scram, power as well as component temperatures decrease rapidly and no significant overshoot occurs.

Linear power and temperature increase is typical for reactors in which the prompt power coefficient of reactivity correction is faster than the reactivity addition of the ramp. In other words, the reactor is at equilibrium at all times during the transient and ramp reactivity insertions of the rates considered are equivalent to a succession of steady states. In fact, if at any point during the transient the ramp had been stopped, the power and temperatures would have remained at equilibrium at that level.

Maximum hot channel fuel temperatures are seen to be lower by over 1900°F than the melting point temperature of uranium carbide, and the highest coolant temperature reached is less than the coolant boiling point by 245°F. No fuel melting, cladding failure or coolant boiling, therefore, is apparent in this type of excursion.

2. Startup Transient From Source Level

A startup accident with protective system action was simulated starting from source power level. The reactor was assumed to be isothermal at 350°F initially, (the temperature to which the reactor and sodium system is preheated prior to admitting sodium and loading with fuel) and coolant flow was assumed to correspond to that of full power. A slow speed four rod withdrawal was simulated equal to a combined rod speed of 4 in./sec and equivalent to a 0.0002- Δk /sec ramp. The excursion was terminated by a scram, assuming a trip point at 20% of full power and including a one second delay time.

The resulting transient is shown graphically in Figure VI-4. The excursion is seen to be terminated after about 30 sec. Reactor power increases initially with a 3-sec period, and the maximum fuel temperature increases almost by a factor of two. No fuel melting or coolant boiling, however, appears as a result of this transient.

*Defined in Appendix, Section C.

Figure VI-4. Power and Maximum Temperature vs Time for a Startup Accident with Protective System Action (scram trip point $P = 0.2 P_0$; scram delay time = 1 sec; reactivity insertion rate $0.0002 \Delta k/\text{sec}$)

3. Loss of Forced Primary Coolant Flow

Use of three independent heat transfer circuits for removing heat generated in the reactor ensures that failure of any circuit will not leave the reactor without external cooling. Although the reactor may be operated using only two of the three heat transfer circuits, it would never be operated with only one circuit. However, a single circuit may be used to remove afterglow heat from the reactor after shutdown.

In the event of all pumping power loss natural convective flow in the loops is sufficient to remove after glow heat from the reactor after shutdown. Loss of all pumps is highly unlikely, however, since there are three sodium coolant loops and an emergency power source.

Losses of forced primary coolant flow from two different initial power levels are illustrated in Figures VI-5 and VI-6. Coolant flow decay was simulated by assuming an instantaneous change from the initial steady state flow rate to purely convective flow as a function of the time dependent sodium temperature.

Figure VI-5. Power and Maximum Temperatures vs Time for Loss of Forced Primary Coolant Flow with Protective System Action, Reactor Initially at Full Power (scram delay time = 1 sec)

Figure VI-6. Power and Maximum Temperatures vs Time for Loss of Forced Primary Coolant Flow with Protective System Action, Reactor Initially at 5% of Full Power (scram delay time = 1 sec)

The excursions were terminated in both cases by a scram after one second delay time. The only significant change in the loss of flow accident from full power and full initial flow (Figure VI-5) is an increase in the maximum coolant temperature, which reaches a peak of 1490°F, still below its boiling point of 1650°F.

Loss of forced flow from low initial power (5% of full power) actually shows a slight increase in the power output (Figure VI-6). This behavior results from the increase in flow after natural convection predominates (an increase in coolant flow decreases the average fuel temperature and results in an increase of reactivity), and demonstrates the effect of the large convective drive which, during normal operation at low power levels, would actually necessitate a braking of the coolant flow.

B. EXCURSIONS WITHOUT PROTECTIVE SYSTEM ACTION

A transient condition which is not followed by a setback or a scram, is probably the most serious accident which can occur in a nuclear reactor. Such accidents can be brought about only by a succession of independent failures each of which have small probabilities to begin with.

Circumstances required for the occurrence of such an accident are simultaneous failures of the plant control system and the plant protective system, and subsequent failure of the operating personnel to observe the situation and take corrective action. Although it is presumed that the probability of such an accident is small, this type of excursion has been extensively studied, because of its potentially serious consequence.

Transients without protective system action will either result in extensive damage to the reactor, in a shutdown due to limited damage (melting and dropping of fuel elements from effective core region), or will result in attainment of a new higher steady state power level. The extent to which any particular transient without ensuing protective system action will affect properties of the components depends solely on the inherent reactor characteristics as well as the rate and amount of reactivity insertion.

Ramp insertion rates are limited by the speed of the drive mechanisms, while limit switches which are fail safe provide physical stops preventing rod withdrawal beyond a preset point. Determinations of optimal values of rod speed and maximum

allowable extend of travel are objects of a detailed parameter study and are beyond the scope of this analysis. For the purpose of this investigation, the rates and limits of possible rod withdrawal were determined arbitrarily, although within reasonable limits of values to be expected with this reactor, and consistent with such parameters in similar existing power reactors.

The maximum possible rate of reactivity insertion was again assumed to be equal to $0.0002 \Delta k/\text{sec}$. This corresponds to a fast one-rod withdrawal with a speed of approximately 4 in./sec, or conversely, a slow four-rod withdrawal of 1 in./sec. Total time of uncontrolled rod withdrawal was arbitrarily limited to 20 sec at power operation and 50 sec for startup. In addition, however, one uncontrolled continuous rod withdrawal from source level was investigated.

1. Rod-Withdrawal Transients from Full Power

Figure VI-7 shows the variation in power and Figures VI-8 and VI-9 the maximum hot channel fuel and the sodium temperatures of the coolant channel outlets. Again the power and temperatures are seen to increase linearly, attesting to the high degree of stability of this reactor. Maximum fuel temperature for the worst case are still lower by over 1600°F than the melting point of

Figure VI-7. Power vs Time for Ramp Reactivity Insertions from Full Power Without Protective System Action (total rod withdrawal time = 20 sec)

Figure VI-8. Maximum Hot Channel Fuel Temperature vs Time for Ramp Reactivity Insertions Without Protective System Action (total rod withdrawal time = 20 sec)

Figure VI-9. Coolant Channel Outlet Sodium Temperatures vs Time for Ramp Reactivity Insertions from Full Power Without Protective System Action (total rod withdrawal time = 20 sec)

uranium carbide, and the maximum coolant temperature is lower by approximately 60°F than the boiling point of sodium (~1650°F) at the particular static pressure.

The end of each transient is signified by the fact that fuel and coolant temperatures will only slightly overshoot their respective equilibrium values for any particular total reactivity insertion. This demonstrates once more that the reactor during rod withdrawals of the rates considered is at equilibrium at all times. Furthermore, since all reactivity feedback from the fuel as well as from the moderator is negative at those particular power levels, no slow drift to higher power levels will occur.

The close approach of the maximum coolant temperature to its boiling point, and the degree of uncertainty for this analysis suggests, however, that limited coolant boiling is possible for the worst case investigated. Vapor generation would be limited to the channel outlets in such a case, and no significant reactivity increase as a result of sodium void formation should be expected. This type of boiling however, is not considered to lead to pool boiling and consequent pressurization of the reactor, since the time constant of the upper plenum is very large even at full flow (~1 hour), and it would be unreasonable to assume that shutdown could not be accomplished within this time.

On the other hand, sodium bond temperatures are slightly higher than coolant temperatures (~40°F) and therefore should be partially vaporized in the hottest channels. The possibility of hot spots or stress concentration, combined with the increased pressure in the affected fuel rods and decreased strength of the stainless steel cladding at this particular temperature, would make the occurrence of some cladding failure probable. Typical SGR containment prevents release of fission products to the atmosphere, but reactor shutdown and replacement of damaged fuel elements would be required. This is of great interest to the reactor operator.

2. Startup Transient From Source Level

Reactor startup is conducted only after the control and plant protective systems have been thoroughly checked for operability, and is governed by procedures intended to eliminate all possibilities of accidental large reactivity insertions.

The effects of a startup accident such as an uncontrolled rod withdrawal from source level would be particularly serious since the prompt negative power coefficient of reactivity is small at startup. This is because the negative fuel reactivity coefficient is usually an increasing function of temperature. The moderator coefficient is positive at startup (Section III-C), and although small would add to the external reactivity insertion. Reactor power therefore would initially increase exponentially, at a period determined by the excess reactivity, until the fuel temperature has increased sufficiently to make its contribution equal to the external insertion rate. Positive moderator feedback would continue, until its temperature reaches the threshold where the coefficient becomes negative.

Startup accidents have been simulated by assuming an isothermal reactor at 350°F and coolant flow conditions equivalent to that of full flow. Figure VI-10 shows the result of a continuous, slow-speed, four-rod withdrawal. Coolant boiling is seen to start after 100 sec. Figure VI-11 shows a similar accident; in this case total rod withdrawal time was limited to 50 sec. Power and temperatures are now seen to drift in the upward direction after completion of the reactivity ramp. This effect results partly from the contribution of the

Figure VI-10. Power and Temperature vs Time for a Startup Accident Without Protective System Action, Continuous Four-Rod Withdrawal (rod speed = 1 in./sec; reactivity rate = $0.0002 \Delta k/\text{sec}$)

Figure VI-11. Power and Temperatures vs Time for a Startup Accident Without Protective System Action, Four-Rod Withdrawal for 50 sec (rod speed = 1 in./sec; reactivity rate = $0.0002 \Delta k/\text{sec}$)

delayed neutrons and partly from the reactivity effect of the moderator. Coolant boiling for this case is delayed for about 5 min.

At this point the quantitative analysis of the transient has been discontinued for each case because of the complexity in representing analytically a transient condition which includes the effects of coolant boiling. Only a qualitative analysis of the ensuing transient will be given here which, however, is based on information that has been determined for other reactors under similar conditions. In any event, sodium boiling will undoubtedly result in fuel element and core damage.

Coolant boiling in any of the fuel channels would have significant effects on the kinetic reactor behavior. Since vapor formation in the fuel channels decreases the coolant density, such properties as reactivity, coolant flow, and frictional flow characteristics will be drastically changed. The change in reactivity results from a decrease in sodium poisoning and subsequent increase in thermal utilization. Coolant flow is accelerated by the increased buoyancy of the vapor and liquid mixture, while the increased velocity and turbulence increases the frictional pressure drop.

The resultant of these individual effects is difficult to evaluate, especially since the contribution of each effect depends upon the instantaneous power level and coolant flow rate. The onset of boiling is signified by relatively small vapor fraction and void reactivity effects, but large buoyancy forces. Coolant boiling in this case will tend to increase the flow rate, resulting in increased heat transfer, which in turn will reduce the tendency for vapor formation. As power increases, however, boiling will again increase and the resulting change in flow will finally reach a point where the frictional effect begins to predominate. This will now increase the pressure drop, reduce flow, and accelerate coolant boiling. At the same time the accelerated void reactivity effect will increase the power output even more.

The combination of these effects will result in a rapid deterioration of the coolant flow rate to a value where vapor "choking" occurs in one or more of the channels. This "choking" effect, which is characterized by a high vapor velocity but low mass flow rate, will reduce the heat transfer from the fuel to practically zero, and melting of the fuel will follow almost immediately. Since fuel melting would occur in the region where the fuel temperature is a maximum at any instant of time, namely the upper third of the fuel elements, the lower portions would be deprived of their support and subsequently drop out of the active core region. The severed parts of the fuel will fall into the lower plenum, thus introducing a relatively large amount of negative reactivity.

At this point it should be mentioned that although boiling and choking in one or a few of the hottest channels would necessarily increase the coolant flow and thereby tend to suppress boiling in the rest of the channels, the process as described above has to repeat itself until a sufficient number of fuel elements has dropped out of the active core region to render the reactor subcritical.

The events described above take place within a short period of time after the onset of boiling. Choking in fact occurs a few seconds after boiling, and fuel melting follows an equally short time thereafter. Extensive coolant boiling and particularly pool boiling in the upper plenum is improbable for this type of accident.

3. Loss of Forced Primary Coolant Flow

Loss of forced coolant flow without protective system action was studied, assuming again an instantaneous change from the initial steady state flow rate to convective flow only. Figures VI-12 and VI-13 show the resulting transient for

Figure VI-12. Power and Maximum Temperatures vs Time for loss of Forced Primary Coolant Flow Without Protective System Action, Reactor Initially at Full Power and Full Flow

Figure VI-13. Power and Maximum Temperatures vs Time for Loss of Forced Primary Coolant Flow Without Protective System Action, Reactor Initially at 5% Power and 5% Flow

a loss of flow accident from full power and full flow, and from 5% power and 5% flow respectively. The transient from initial full power shows only a slight difference between the maximum coolant temperature and the boiling point of Na. Since this is within the margin of error, similar reasoning as employed in section VI-B-1 would also be applicable for this kind of accident.

VII. CONCLUSIONS

The results of the transient analysis, as presented in the previous section of this report, demonstrate the exceptional stability and self-controlling tendency of this reactor. It was shown that no postulated accident could lead to potential unsafe conditions if followed by normal protective system action, and also that cladding failure or any other damage to the reactor or its components is improbable under the circumstances considered.

The investigation of certain improbable excursions which are not followed by protective system action show that startup accidents from source level are the most serious ones which could be encountered. These accidents would lead to coolant boiling within a relative short time. The immediate appearance of choking and consequent fuel melting would prevent extensive boiling, however, and in fact provides an effective protection against pool boiling by shutting down the reactor before the bulk of the coolant in the upper plenum reaches its boiling point. Severe fuel and core damage would result, but a hazard to the public should not be present.

Limited boiling and fuel cladding failure is seen possible for loss-of-forced-flow accidents or large reactivity insertion from full power without protective system action. The large thermal capacity and long time constant of the upper plenum, however, will provide protection against pool boiling until the reactor can be shut down by external means. A hazard to the public should not exist under these conditions. It should be noted that this study has been concerned with reactivity transients only, and does not comprise a study of the safeguard features of the entire system under all possible conditions. Typical (conservative) conditions have been assumed, to be verified by detailed analysis of plant design, when such design has been completed and the numerous other factors involved in safeguards analysis (site, plant configuration, shielding, etc) are selected.

APPENDIX

DESCRIPTION OF TRANSIENT STUDY

The general objective of the transient study was to observe and record the reactor power and temperature variation for reactivity insertions from different equilibrium states, and for changes in coolant flow without additional reactivity insertions.

A. ANALOG MODEL

The reactor core, as described in Section I, was represented by a unit cell model, suitable for use on an analog computer. The unit cell model consists of an average fuel rod, its associated coolant, and a moderator element in dimensions proportional to the former. The fuel, coolant, and moderator elements were chosen to represent the reactor components for flux and position averaged conditions, and have been coupled by the appropriate heat transfer areas and internal heat generation rates.

B. NUCLEAR PARAMETERS

The analog computer was programmed to solve the kinetic equations of neutron balance for two groups of delayed neutrons. The multiplication factor was coupled to the power coefficients, and the time-dependent functions of step and ramp insertions, as well as reactor scram simulations.

Equations 1, 2, and 3 describe the time rate of change of power, precursors, and reactivity. (See nomenclature at back.)

$$\frac{dP}{dt} = \left[k(t)(1 - \beta) - 1 \right] \frac{P}{\ell} + \sum_1^2 \lambda_i C_i, \quad \dots (1)$$

$$\frac{dC_i}{dt} = k(t) \beta_i \frac{P}{\ell} - \lambda_i C_i, \quad \dots (2)$$

$$k(t) = k_0 + \gamma \int \frac{dT_m}{dt} dt + \delta \int \frac{dT_f}{dt} dt + \int \epsilon(t) dt \quad \dots (3)$$

C. THERMAL PARAMETERS

Fuel and moderator temperature coefficients were flux and volume averaged. The effective heat transfer coefficients of the node elements were volume averaged over the respective components. The volumetric power generation rate in the unit cell model was calculated and adapted to the one-node representation. The mathematical expressions describing the heat transfer process were arranged to represent the model.

The time rate of change of temperatures is as follows:

$$\frac{dT_f}{dt} = \mu \frac{P}{\rho_f C_{pf}} - \frac{U_f A_f}{\rho_f C_{pf} V_f} (T_f - T_c), \quad \dots (4)$$

$$\frac{dT_m}{dt} = \mu' \frac{P}{\rho_m C_{pm}} - \frac{U_m A_m}{\rho_m C_{pm} V_m} (T_m - T_c), \quad \dots (5)$$

$$\frac{dT_c}{dt} = \frac{U_f A_f}{\rho_c C_{pc} V_c} (T_f - T_c) + \frac{U_m A_m}{\rho_c C_{pc} V_c} (T_m - T_c) - \frac{w}{\rho_c V_c} (T_{co} - T_{ci}), \quad \dots (6)$$

where

$$T_{co} = 2T_c - T_{ci}. \quad \dots (7)$$

The effective heat transfer coefficient for the fuel, was calculated from the basic equations of heat transfer in cylindrical rods with internal heat generation, and for heat transfer in concentric cylindrical surfaces in series. The expression for the fuel temperature as a function of radius is as follows:

$$T(r) = T_c + \frac{q}{4K_f} (a^2 - r^2) + \frac{q}{H}, \quad \dots (8)$$

where

$$H = \frac{1}{a \left(\frac{\ln \frac{b}{a}}{K_{NaK}} + \frac{\ln \frac{c}{b}}{K_{ss}} + \frac{1}{ch} \right)}. \quad \dots (9)$$

The average fuel temperature is found by integrating expression 8 over the fuel volume or

$$T_f = \frac{\int_0^a 2\pi r T(r) dr}{\int_0^a 2\pi r dr} \quad \dots(10)$$

the resulting expression is equal to

$$T_f - T_c = q \left(\frac{H^2 + 4K_f}{8HK_f} \right); \quad \dots(11)$$

but since the heat generated in the fuel volume at steady state is equal to the heat transferred across its surface, or

$$\pi a^2 q = 2\pi a V (T_f - T_c), \quad \dots(12)$$

the effective heat transfer coefficient between the average fuel temperature and the coolant, is now found from Equation 11 and 12, and is equal to

$$U_f = \frac{4H}{a \frac{H}{K_f} + 4}. \quad \dots(13)$$

The maximum hot channel fuel temperature was found from the maximum to average temperature factor, which from section IV-A is equal to 3.15 and is defined as:

$$\frac{T_{f(max)} - T_c}{T_{fct} - T_{c(av)}} = 3.15. \quad \dots(14)$$

D. HYDRAULIC PARAMETERS

The computer model was programmed to include the primary flow kinetics. Flow transients were assumed to be similar in each primary loop. Loss of forced flow was simulated by changing the flow parameters from forced flow to purely convective flow for a particular coolant temperature, neglecting the contribution of the initial fluid momentum.

NOMENCLATURE

- a = outside radius of fuel slugs.
- A_c = coolant flow area for unit cell model.
- A_f = fuel surface area for unit cell model.
- A_m = moderator surface area for unit cell model.
- b = inside radius of stainless steel fuel cladding.
- c = outside radius of stainless steel fuel cladding.
- C_{pc} = specific heat of coolant.
- C_{pf} = specific heat of fuel.
- C_{pm} = specific heat of moderator.
- C_i = constant proportional to the concentration of the delayed neutron precursors and reactor power of the two-group approximation.
- g = acceleration of gravity.
- h = surface coefficient of heat transfer of the representative fuel rod.
- H = effective heat transfer coefficient from the surface of the fuel slug to the coolant.
- k_o = effective multiplication factor at start of excursion.
- $k(t)$ = time dependent effective multiplication factor.
- K_f = thermal conductivity of the fuel.
- K_{NaK} = thermal conductivity of NaK bond.
- K_{ss} = thermal conductivity of the stainless steel cladding.
- l = neutron lifetime.
- P = reactor power.
- q = heat generation rate per unit volume in the fuel.
- r = radius.
- T_c = flux and volume averaged coolant temperature
- T_f = flux and volume averaged fuel temperature

T_m = Flux and volume averaged moderator temperature.

T_{ci} = coolant inlet temperature.

T_{co} = coolant outlet temperature.

T_{fct} = fuel center temperature.

$T_f(\max)$ = maximum hot channel fuel temperature.

$T(r)$ = radial fuel temperature

U_f = effective heat transfer coefficient between the average fuel temperature and the coolant.

U_m = effective heat transfer coefficient between the average moderator temperature and the coolant.

v = coolant velocity in fuel channel.

V_c = total coolant volume in unit cell model.

V_f = total fuel volume in unit cell model.

V_m = total moderator volume in unit cell model

ω = mass flow rate.

β = Total delayed neutron fraction.

β_i = fraction of delayed neutrons in the i^{th} group of the two-group approximation.

γ = moderator temperature reactivity coefficient.

δ = fuel temperature reactivity coefficient.

$\epsilon(t)$ = time varying reactivity insertion.

λ_i = delayed neutron precursor decay constant of the i^{th} group of the two-group approximation.

μ = proportionality constant relating the heat generation rate in the fuel to the reactor power.

μ' = proportionality constant relating the heat generation rate in the moderator to the reactor power.

ρ_c = coolant density.

ρ_f = fuel density.

ρ_m = moderator density

ρ_{if} = fuel channel inlet coolant density.

ρ_{of} = fuel channel outlet coolant density.

ρ_{iH} = heat exchanger inlet coolant density.

ρ_{oH} = heat exchanger outlet coolant density.