MASTER

Copy N^o 1

AP Note-430

PATENT CLEARANCE OBTAINED. RELEASE TO
THE PUBLIC IS APPROVED. PROCEEDURES
ARE ON FILE IN THE RECEIVING SECTION.

ALCO PRODUCTS, INC.
NUCLEAR POWER ENGINEERING DEPARTMENT

MAY 19 1965

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

REPORT ON NATURAL CONVECTION
CAPABILITY OF SM-1

By
P. L. Bradley

Approved by:
E. W. Schrader, Acting Project Engineer

Issued: July 11, 1962

Contract No. AT(30-1)-2639
with the United States Atomic Energy
Commission, New York Operations Office

ALCO PRODUCTS, INC.
Nuclear Power Engineering Department
Post Office Box 414
Schenectady 1, N. Y.

AEC LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights:
or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission or his employment with such contractor.

ALCO LEGAL NOTICE

This report was prepared by Alco Products, Incorporated in the course of work under, or in connection with, Contract No. AT(30-1)-2639, issued by U. S. Atomic Energy Commission, NYOO; and subject only to the rights of the United States, under the provisions of this contract, Alco Products, Incorporated makes no warranty or representation, express or implied, and shall have no liability with respect to this report or any of its contents or with respect to the use thereof or with respect to whether any such use will infringe the rights of others.

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

DISTRIBUTION

External Copies

- 1-3 New York Operations Office
 U. S. Atomic Energy Commission
 376 Hudson Street
 New York 14, N. Y.

 Attn: I. M. Adler (2)

 Attn: Library (1)
- 4-6 U. S. Atomic Energy Commission
 Washington 25, D. C.

 Attn: Chief, Water Systems Project Branch,
 (Army Reactors)
 Division of Reactor Development
 Mail Station F-311
- 7 Idaho Nuclear Power Field Office
 P. O. Box 2108
 Idaho Falls, Idaho
- 8-9 Office of the Chief of Engineers
 Department of the Army
 Building T-7
 Washington 25, D. C.

 Attn: Chief, Projects Branch
 Nuclear Power Division
- 10-12 Nuclear Power Field Office
 U. S. Army Engineer Reactors Group
 Fort Belvoir, Virginia

 Attn: Chief, Nuclear Power Field Office
- 13 Nuclear Power Field Office
 U. S. Army Engineer Reactors Group
 Fort Belvoir, Virginia

 Attn: O. I. C. SM-1

External
Copies

DISTRIBUTION (CONT'D)

- 14 Chief, U. S. Army Reactor Group
Fort Greely, Alaska
APO 733
Seattle, Washington

Attn: O. I. C. SM-1A
- 15 Commanding Officer
U. S. Army Polar Research & Development Center
Fort Belvoir, Virginia

Attn: Nuclear Power Officer
- 16 Union Carbide Nuclear Corporation
Oak Ridge National Laboratory
Y-12 Building 9704-1
P. O. Box "Y"
Oak Ridge, Tennessee

Attn: L. D. Schaffer
- 17 Alco Products, Inc.
P. O. Box 145
Fort Belvoir, Virginia

Attn: W. C. Best

Internal
Copies

18	R. Bell	27	P. E. Bobe
19	K. Kasschau	28	S. L. Davidson
20	C. Klotz	29	I. Segalman
21	J. G. Gallagher	30	J. V. Miller
22	J. F. Haines	31	G. Menzel
23	E. W. Schrader	32	W. Flamm
24	W. T. Williams	33	R. E. Neville
25	P. L. Bradley	34-40	NPED File
26	F. T. Matthews		

ABSTRACT

An investigation was made as to the feasibility of operating the SM-1 plant at low powers under the conditions of natural circulation. A steady state analysis using the STDY-3 code was performed and the results given. An analog model was developed to investigate primary loop stability with natural circulation. The ART-04 code was used to investigate the results of a rod withdrawal accident.

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

TABLE CONTENTS

	<u>Page</u>
ABSTRACT	v
1.0 SUMMARY	1
2.0 INTRODUCTION	3
3.0 METHOD OF ANALYSIS	5
3.1 Calculation of Reactor Power and Thermal Driving Head	5
3.2 Steady State Digital Analysis of the SM-1 Core II	9
3.3 Analog Model for Plant Transient Analysis	10
3.3.1 Nomenclature	11
3.3.2 Equations	13
3.3.3 Wiring Schematic, Pot Settings and Plant Constants	17
4.0 RESULTS	23
4.1 Steady State Digital Analysis of SM-1 Core II	23
4.2 Result of Rod Withdrawal Accident	23
4.3 Transients Imposed by Load Changes	28
5.0 CONCLUSIONS	33
6.0 REFERENCES	35
7.0 APPENDIX	37

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

LIST OF FIGURES

<u>Figure</u>	<u>Title</u>	<u>Page</u>
3.1	SM-1 Dimensions Affecting Thermal Driving Head	5
3.2	Core Power Output Vs. Primary Coolant Temperature and Flow Rate	8
3.3	Schematic Diagram of Reactor Plant	15
3.4	SM-1 Natural Circulation Analog Model	18
4.1	SM-1 Reactor Performance Under Conditions of Natural Circulation	26
4.2	SM-1 Core II Theoretical Hot Channel Element Position 65. Plate Surface Temperature Vs. Length Along Channel for Natural Circulation	27
4.3	SM-1 Reactivity Insertion During Rod Withdrawal with Natural Circulation	29
4.4	Transient Performance of SM-1 Reactor Under Natural Circulation Conditions for a Rod Withdrawal Accident. DNBR and Power Vs. Time	30
4.5	SM-1 Nominal Channel Exit Temperature During Rod Withdrawal with Natural Circulation	31
A.1	Estimated Pressure Drop, Free Rotor Pump Q-350-A1	38
A.2	Estimated Pressure Drop, Locked Rotor Pump Q-350-A1	39
A.3	Test Performance Curve, Q-350-A1 Pump 8DF 591-1 . .	40
A.4	Performance Curve Estimated from Test in D/M-8000-A1 Volute	41

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

LIST OF TABLES

<u>Table</u>	<u>Title</u>	<u>Page</u>
3.1	SM-1 Plant Constants	19
3.2	Pot Settings	20
4.1	Results of STDY-3 Analyses	24

1.0 SUMMARY

Steady state analysis was conducted to determine at what power level the SM-1 could be operated to establish a basis for a natural circulation test. The results show that if the same ΔT (20°F) is maintained across the core with natural circulation as with forced circulation, then 0.51 Mwt of power is the maximum that can be removed from the core under natural circulation. The DNBR at these conditions is 104 in the hot channel of element 65.

This value shows extreme safety from burnout during steady state operation, but in the analysis of a rod withdrawal accident, the worst type of accident postulated, the DNBR drops to 7 in 11 sec. A DNBR of 7 is still 2+ times higher than the minimum allowable. The nominal channel outlet temperature would begin to climb rapidly after 11 seconds. However, the high-temperature scram would be expected to actuate before burnout occurred.

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

2.0 INTRODUCTION

Under normal operation of the SM-1 Core II the thermal energy generated in the core is removed and transmitted to the steam generator by forced convection. In the event that the primary pump fails as a result of mechanical or electrical failure, the plant must be shut down immediately and must remain down until necessary repairs or replacement can be made. Some power, probably enough to operate the auxiliary heating and electrical equipment, can be removed from the reactor while operating under conditions of natural circulation. Natural circulation can be introduced by taking advantage of the physical location of the steam generator to develop sufficient thermal driving head to drive fluid by natural convection.

The limiting factors on the power generation of the core during natural circulation operation are the temperature differential across the primary side of the steam generator or across the core, and the elevation between thermal centers of the source of energy and the sink. The temperature differential becomes limiting when thermal stresses developed by large ΔT 's are greater than allowable stresses. Small differences in elevation will not develop sufficient thermal driving head to operate the reactor.

As part of the Army Pressurized Water Support and Development Program⁽¹⁾, Subtask 4.1 was established for examination of the natural convection capabilities of SM-1. This subtask was to develop analytical techniques for examination of SM-1 primary system operation during natural convection cooling, and to experimentally verify these techniques by testing at the site.

The problem has been divided into two analyses. The first uses digital codes to establish core operating power based on minimum burnout ratios (DNBR). The second analysis employs an analog model of the primary system to observe the response of the reactor, with changes in reactivity, to develop sufficient thermal circulation to establish the core power desired.

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

3.0 METHOD OF ANALYSIS

3.1 CALCULATION OF REACTOR POWER AND THERMAL DRIVING HEAD

The thermal driving head available is developed by the temperature differential and the elevation between the thermal centers of the source of energy and the sink. The SM-1 steam generator tube sheet is located at a level of 40-5/8 in. above the core centerline. The highest point in the system is located in the bend of the "U" tubes and is 50-1/2 in. above the tube sheet. From the dimensions of Fig. 3.1 the following derivation of the thermal driving head " ΔP_t " is made for the SM-1 at mean operating temperature of 440°F. Starting at the core midpoint and going in a clockwise direction the equation for ΔP_t becomes

$$\begin{aligned} \Delta P_t = \int \rho dh = & - \left[\frac{\rho_m + \rho_o}{2} \right] h_1 - \rho_o h_2 - \left[\frac{\rho_o + \rho_m}{2} \right] h_3 \\ & + \left[\frac{\rho_m + \rho_i}{2} \right] h_3 + \rho_i \left[h_2 + 2 h_1 \right] - \left[\frac{\rho_i + \rho_m}{2} \right] h_1 \end{aligned} \quad (1)$$

where ρ is the density of the fluid.

$$h_1 = 11 \text{ in.} = 0.92 \text{ ft.}$$

$$h_2 = 29\text{-}5/8 \text{ in.} = 2.47 \text{ ft.}$$

$$h_3 = 50\text{-}1/2 \text{ in.} = 4.21 \text{ ft.}$$

$$T_o = \text{hot leg temp. - reactor outlet temp.}$$

$$T_i = \text{cold leg temp. - reactor inlet temp.}$$

$$T_m = \text{core and steam generator mean temp.}$$

Figure 3.1. SM-1 Dimensions Affecting Thermal Driving Head

Tests on SM-1 at site (Ft. Belvoir) have shown that the primary flow rate is 3975 gpm with the long leg pump (Serial #1) operating. The calculated inoperative pump leg back flow is 285 gpm, yielding a pump flow of 4260 gpm. This latter value establishes the operating point on the pump performance curve, as shown by the intersection of head and loss curves. The 27.5 ft pump head (Westinghouse pump curve #446278) corresponds to the loop drop (less pump) at 3975 gpm.

$$27.5 \text{ ft} = (27.5 \text{ ft}) (12 \frac{\text{in.}}{\text{ft}}) (0.03022 \text{ lb/in.}^3) = 10.0 \text{ psi}$$

$$3975 \text{ gpm} = \frac{(3975 \frac{\text{gal}}{\text{min}}) (0.03022 \text{ lb/in.}^3) (1728 \frac{\text{in.}^3}{\text{ft}^3})}{(60 \frac{\text{sec}}{\text{min}}) (7.481 \frac{\text{gal}}{\text{ft}^3})} = 462 \text{ lb/sec}$$

With the primary loop operating in natural circulation, the loop drops can be considered in two parts. The first is the portion of the loop other than the pump, and can be represented from the above data. Since the 27.5 ft drop is composed of both friction ($H \sim R^{1.8}$) and expansion, contraction ($H \sim R^{2.0}$), H can be closely represented by the relationship $H \sim R^{1.9}$.

The values of the proportionality constant K is

$$K = \frac{\Delta P_{\text{loss}}}{R^{1.9}} = \frac{100}{(462)^{1.9}} = 8.66 \times 10^{-5} \quad (2)$$

Using the "Forward flow" curve of Westinghouse D-375 Estimated Pressure Drop Chart determine the exponent of the pressure drop of the pump as follows:

$$H = KG^x$$

$$H_1 = 4.15$$

$$G_1 = 4$$

$$H_2 = 1.05$$

$$G_2 = 2$$

$$\frac{H_1}{G_1^x} = \frac{H_2}{G_2^x}$$

$$\frac{H_1}{H_2} = \left(\frac{G_1}{G_2}\right)^x$$

$$3.95 = 2^x$$

$$x = 1.98 \text{ or } 2$$

Evaluation of K

$$G_1 = 4,000 \text{ gpm} = 465 \text{ lb/sec}$$

$$H_1 = 4.15 \text{ ft} = 1.5 \text{ psi}$$

$$K = \frac{H}{G^2} = \frac{1.5}{(465)^2} = 0.695 \times 10^{-5}$$

$$\Delta P = (0.6948 \times 10^{-5}) R^2 \quad (3)$$

The total loop pressure drop to be overcome by the thermal driving head to maintain flow is the sum of the loop and the pump pressure losses.

$$\Delta P_{\text{total}} = \Delta P_{\text{pump}} + \Delta P_{\text{loop}} \quad (4)$$

The reactor power can also be expressed as a function of flow rate and temperature differential as

$$Q = 1.0548 \times 10^{-3} R C_c \Delta T \quad (5)$$

Where Q = Reactor power, tMW
 R = Coolant flow rate, lb/sec.
 C_c = Coolant average specific heat, Btu/lb^oF
 ΔT = Bulk coolant temperature rise ^oF

Figure 3.2 shows the family of core power curves obtained from equation (5) for values of coolant temperature differential and flow rate.

Figure 3.2 Core Power Output Vs. Primary Coolant Temperature and Flow Rate

3.2 STEADY STATE DIGITAL ANALYSIS OF THE SM-1 CORE II

The steady state thermal analysis of the core under conditions of natural circulation was conducted using the IBM-704 code STDY-3.⁽⁶⁾ This code, a one-dimensional steady state code prepared by WAPD for pressurizer water reactors, calculates bulk enthalpy, bulk temperature, and steam quality for normal channels from input parameters. Through calculation and comparison of hot and normal channel pressure drops, the code determines a hot channel flow rate. Using this hot channel flow rate, values of bulk enthalpy, bulk temperature, steam quality, plate surface temperature, meat centerline temperatures and DNB (departure from nucleate boiling) ratios are calculated for each axial position from input properties and design equations. Where local or bulk boiling exists in either channel, the code designates the axial position at which the phenomenon starts.

Values of radial and axial power distributions, hot channel, instrumentation tolerances, fuel plate rippling, nuclear uncertainty, core power generation and flow distribution factors used in this analysis were taken from previous analyses of SM-1 Core II.⁽²⁾ Definitions and derivation of these factors can be found in the literature.^{(2),(3),(4),(5)} Core power output factors used were determined in Section 2.1 as a function of the core flow rate under conditions of natural circulation in SM-1.

To determine the flow rate in the individual elements it was assumed that the flow rate under natural circulation was the same percentage of the total flow as under conditions of forced circulation.

The burnout equations used by STDY-3⁽⁶⁾ in this analysis are

$$\frac{\phi_{Bo}}{10^6} = 0.325 (F_p) \left(\frac{H_{Bo}}{10^3} \right)^{-2.5} e^{-0.0012 L/S} \quad (6)$$

$$\text{at } 1200 \text{ psi } F_p = 1.2$$

or

$$\frac{\phi_{Bo}}{10^6} = 0.89 \left(\frac{H}{10^3} \right)^{-0.72} e^{-0.0012 L/S} \quad (7)$$

The code calculates DNBR's using both equations and gives the solution with the lowest value. The second equation becomes the limiting operation

at 1200 psi where $H \leq 630$ Btu/lb. Since at core mean temperature of 440°F, H is approximately equal to 420 Btu/lb, it is obvious the latter equation is used by the code.

The ART-04⁽¹⁰⁾ digital code which includes reactor kinetics and transient hydrodynamic calculations, was used to determine the effect of a rod withdrawal accident during a natural circulation test. The ART-04 code will not account for changes in the secondary system unless supplementary analog data is available to use as input to the digital program. However, since loop transport time is generally greater than 10 sec and since the most critical period falls within these 10 sec, it was assumed that secondary effects could be omitted from this analysis and the core inlet temperature was assumed constant.

ART-04 uses the same burnout equation as STDY-3 and the same input parameters. The use of the ART-04 code is described in detail in WAPD-TM-202, (10) APAE-91(11) and APAE-105. (5)

Element 65 was selected to be analyzed by the ART-04 code at a core power of 0.51 and an inlet temperature of 435°F. It was considered possible that during a natural circulation test the rod drives might jam and continue to withdraw the rods at the maximum drive speed (3 in./min) instead of stopping. The analysis was performed assuming the most reactive time in life at the most reactive rod position (axial). Temperature coefficients were selected at the least reactive temperature to represent the extreme condition.

3.3 ANALOG MODEL FOR PLANT TRANSIENT ANALYSIS

The present analog model used for the representation of SM-1 load perturbations is for a forced circulation system and hence not readily applicable to natural convection representation. A search of the available literature did not uncover a suitable model to represent the SM-1 plant under conditions of natural circulation. Hence one had to be derived from the existing fixed⁽⁷⁾ and variable⁽⁸⁾ flow analog models.

3.3.1 Nomenclature

Symbols

A	Flow or heat transfer area, ft^2
C	Specific heat, $\text{Btu/lb } ^\circ\text{F}$
K	Potential power contribution of delayed neutrons, M/F, Btu/sec
ℓ	Mean neutron lifetime, birth to absorption, sec^{-1}
L	Length, ft
P	Primary system pressure, psia
ΔP	Pressure Drop, psi
Q(t)	Power output of core, Btu/sec
R	Rate of primary system flow, lb/sec
S	Channel spacing, ft
t	Time, sec
T	Temperature, $^\circ\text{F}$
U	Heat transfer coefficient $\text{Btu/sec} - ^\circ\text{F} - \text{ft}^2$
V	Volume, ft^3
W	Weight, lb
α	Fraction of power generated in fuel plates and cladding
β	Delayed neutron fraction of a group
γ	Volume coefficient of primary coolant expansion, $^\circ\text{F}^{-1}$
δ	Excess reactivity of core

Symbols

ρ	Excess reactivity coefficient, $^{\circ}\text{F}^{-1}$ or psia^{-1}
$\bar{\rho}$	Density of fluid, lbs/ft^3
Θ	Temperature difference at design load, $^{\circ}\text{F}$
λ	Decay constant of a delayed neutron group, sec^{-1}
τ	Time lag, sec
ϕ	Amplitude scale factor

Subscripts

C	Mean core coolant condition
D	Design power output condition, steady state
E	Exchanger tubing
F	Mean fuel plate condition
G	Mean steam generator condition, primary (tube) side
i	i^{th} delayed neutron group
L	Liquid in steam generator
p	Refers to pump
o	Reference value
S	Mean steam conditions in generator corresponding to saturation
tot	Total for primary system
T	Core temperature
1-6	Thermo. properties: Condition of location in schematic diagram
1-5	Nuclear parameters: Particular i^{th} delayed neutron group

3.3.2 Equations

The kinetics equations have been previously developed in APAE-38.⁽⁷⁾ A summary of the kinetic equations has been repeated below for convenience.

Neutron population equation -

$$\frac{dP(t)}{dt} = \frac{\delta_r + \rho_T [T_c(t) - T_c(o)] + \rho_p [p(t) - p_D] - \sum_{i=1}^5 \beta_i}{\ell} P(t) + \sum_{i=1}^5 \lambda_i K_i(t) \quad (8)$$

$$\frac{dK_i(t)}{dt} = \frac{\beta_i}{\ell} P(t) - \lambda_i K_i(t) \quad (9)$$

The equations describing the thermal kinetics of the core, the steam generator, plenum chambers and piping have been derived in APAE Memo-198.⁽⁸⁾ These equations are:

Fuel Plate Temperature -

$$\frac{dT_F(t)}{dt} = \frac{\alpha}{W_F C_F} Q(t) - \frac{(U_F A_F)_o}{W_F C_F} \left(\frac{R}{R_o} \right) [T_F(t) - T_A(t)] \quad (10)$$

Average Coolant Temperature -

$$\begin{aligned} \frac{dT_c(t)}{dt} = & \frac{(1-\alpha)}{W_c C_c} Q(t) + \frac{(U_F A_F)_o}{W_c C_c} \left(\frac{R}{R_o} \right) [T_F(t) - T_c(t)] \\ & - RC_c [T_2(t) - T_1(t)] \end{aligned} \quad (11)$$

Steam Generator Primary -

$$\frac{dT_G(t)}{dt} = \frac{1}{\tau_G} \left[T_4(t) - T_5(t) \right] - \frac{(U_G A_G)_o}{W_G C_G} \left(\frac{R}{R_o} \right) \left[T_G(t) - T_S(t) \right] \quad (12)$$

Plenum Chambers -

$$\frac{dT_1(t)}{dt} = \frac{1}{\tau_{6,1}} T_6(t) - \frac{1}{\tau_{6,1}} T_1(t) \quad (13)$$

Piping -

$$\frac{dT_1(t + \tau)}{dt} = \frac{2}{\tau_{6,1}} T_G(t) - \frac{1}{\tau_{6,1}} T_4 - \frac{1}{\tau_{6,1}} T_1(t + \tau_{5,6}) \quad (14)$$

See Fig. 3.3 for location of temperature points.

The rate of change of momentum of an element of fluid is equated to the sum of the forces acting on it. These forces are gravity, friction and fluid inertia. These forces all tend to resist the flow, and correspond to pressures that must be balanced by the pump during steady state operation⁽⁹⁾.

$$\Delta P_p = \sum_i \left[\frac{\Delta X_i}{A_i g} \frac{dR_i}{dt} = (\Delta P_f)_i + P_i \Delta h_i \right] \quad (15)$$

In order to apply this to the reactor model, the following assumptions are made:

The coolant is an incompressible fluid.

The increments of pipe length are sufficiently short so that the integral around the loop may be replaced by a sum over the element of length.

Figure 3.3 Schematic Diagram of Reactor Plant

The density of the fluid at a specific temperature can be measured relative to a reference density (ρ_o) by the following relation involving the volume expansion coefficient of the fluid.

$$\rho_i(t) = \rho_o - \gamma T_i \quad (16)$$

The temperature at any instant between two adjacent points defined in Fig. 3.3 varies linearly with position.

Transposing (15) to solve for change in flow rate:

$$\sum_i \frac{\Delta X_i}{A_i g} \frac{dR_i}{dt} = \sum_i (\Delta P_p) - (\Delta P_f)_i - \rho_i \Delta h_i \quad (17)$$

To evaluate the static pressure term, a simplified view of the system was used (Fig. 3.3) with the convention that flow in an upward direction was positive. Starting at the reactor inlet and performing a sum over the elements of piping calculation, one obtains:

$$\begin{aligned} \sum \rho_i \Delta h_i = & - \frac{\rho_m + \rho_o}{2} h_1 - \rho_o h_2 - \frac{\rho_o + \rho_m}{2} h_3 \\ & + \frac{\rho_m + \rho_i}{2} h_3 + \rho_i (h_2 + 2 h_1) - \frac{\rho_i + \rho_m}{2} h_1 \end{aligned} \quad (18)$$

Substituting (16) into (18), the following relation is obtained:

$$\begin{aligned} \sum \rho_i \Delta h_i = & - T_m \left[\gamma h_1 \right] - T_o \left[\gamma \left(\frac{h_1}{2} + h_2 + \frac{h_3}{2} \right) \right] \\ & + T_i \left[\gamma \left(\frac{3h_1}{2} + h_2 + \frac{h_3}{2} \right) \right] \end{aligned} \quad (19)$$

The term involving friction pressure drop in (17) may be written as:

$$(\Delta P_f)_i = (\Delta P_p)_o \left[\frac{R}{R_o} \right]^2 \quad (20)$$

The exponent 2 stems from basic turbulent flow relations for pressure drops in piping. With this substitution, (15) becomes:

$$\frac{dR}{dt} = \frac{-(\Delta P_f + \Delta P_p)_o \left[\frac{R}{R_o} \right]^2 - \sum_i \rho_i \Delta h_i}{\sum_i \frac{L_i}{A_i g}} \quad (21)$$

3.3.3 Wiring Schematic, Pot Settings and Plant Constants

The analog circuit diagram used in this analysis is shown in Fig. 3.4. A time scale of one computer second equal to real second has been chosen for this simulation. An amplitude scaling factor of one volt equals one physical unit was chosen. Multiples of quantities were generated to keep voltages within the ± 100 volt limit imposed by the analog computer.

The plant constants calculated from as built drawings are given in Table 3.1. The values of potentiometer settings used in this simulation are given in Table 3.2. Both the actual magnitude and the physical quantities are given.

Figure 3.4 SM-1 Natural Circulation Analog Model

TABLE 3.1
SM-1 PLANT CONSTANTS

C_C	=	1.094	$W_{3,4}$	=	586
C_E	=	0.12	$W_{5,6}$	=	1373
C_F	=	0.109	$W_{6,1}$	=	1523
C_L	=	1.03	α	=	0.94
ℓ	=	2.0×10^{-5}	β_1	=	0.85×10^{-3}
p_o	=	1200	β_2	=	2.41×10^{-3}
P_D	=	484 (0.51 MW)	β_3	=	2.13×10^{-3}
R_o	=	22.2	β_4	=	1.66×10^{-3}
V_e	=	4.05	β_5	=	0.25×10^{-3}
V_G	=	11.63	γ	=	9.35×10^{-4}
$V_{2,3}$	=	24.7	ρ_p	=	0.73×10^{-6}
$V_{3,4}$	=	11.3	ρ_T	=	2.4×10^{-4}
$V_{5,6}$	=	26.5	$\phi_{1,2}$	=	20
$V_{6,1}$	=	29.4	λ_1	=	1.61
W_c	=	210	λ_2	=	0.456
W_E	=	2490	λ_3	=	0.151
W_F	=	405	λ_4	=	0.0315
W_G	=	602	λ_5	=	0.0124
W_L	=	2802	$\sum \frac{L_i}{A_i g}$	=	47.33
$W_{2,3}$	=	1279			

TABLE 3.2

POT SETTINGS

Pot No.			
1	$\alpha/W_F C_F$	$= 0.94/405 \times 0.105$	$= 0.0221$
2	$(U_F A_F)_o/W_F C_F$	$= 465.5/405 \times 0.109$	$= 10.545$
3	$(1 - \alpha)/W_C C_C$	$= (1-0.94) / 210 \times 1.094$	$= 0.2611 \times 10^{-3}$
4	$(U_C A_C)_o/W_C C_C$	$= 197.7 / 210 \times 1.094$	$= 0.8605$
5	R_o/W_c	$= 447.4/210$	$= 2.1305$
6	ξ_p/ℓ	$= 0.73 \times 10^{-6}/2.0 \times 10^{-5}$	$= 0.3650 \times 10^{-1}$
7	$\gamma [V_c]$	$= 9.35 \times 10^{-4} \times 4.05$	$= 3.786 \times 10^{-3}$
8	$\gamma [V_{3,4}]$	$= 9.35 \times 10^{-4} \times 11.3$	$= 1.056 \times 10^{-2}$
9	$\gamma [V_{6,1}]$	$= 9.35 \times 10^{-4} \times 2.94$	$= 2.749 \times 10^{-2}$
10	$\gamma [V_G]$	$= 9.35 \times 10^{-4} \times 11.63$	$= 1.087 \times 10^{-2}$
11	$\Delta P_{pos}/\Delta V$	$= 86.0 - 49.0 / 2 - 1$	$= 37.0$
12	$\Delta P_{neg}/\Delta V$	$= (-29.0) - (-14.0) / (-1) - (-0.5)$	$= 30.0$
13	$\sum \beta_n/\ell$	$= 7.30 \times 10^{-3} / 2.0 \times 10^{-5}$	$= 3.65 \times 10^2$
14	NOT USED		
15	β_1/ℓ	$= 0.85 \times 10^{-3} / 2.0 \times 10^{-5}$	$= 0.425 \times 10^2$
16	$\lambda_1 \beta_1/\ell$	$= 1.61 \times 0.85 \times 10^{-3}/2.0 \times 10^{-5}$	$= 0.6843 \times 10^2$
17	λ_1	$= 1.61$	$= 1.61$
18	β_2/ℓ	$= 2.41 \times 10^{-3} / 2.0 \times 10^{-5}$	$= 1.205 \times 10^2$
19	$\lambda_2 \beta_2/\ell$	$= 0.456 \times 2.41 \times 10^{-3} / 2.0 \times 10^{-5}$	$= 0.5495 \times 10^2$
20	λ_2	$= 0.456$	$= 0.456$
21	β_3/ℓ	$= 2.13 \times 10^{-3} / 2.0 \times 10^{-5}$	$= 1.056 \times 10^2$
22	$\lambda_3 \beta_3/\ell$	$= 0.151 \times 2.13 \times 10^{-3} / 2.0 \times 10^{-5}$	$= 0.1595 \times 10^2$
23	λ_3	$= 0.151$	$= 0.151$
24	β_4/ℓ	$= 1.66 \times 10^{-3} / 2.0 \times 10^{-5}$	$= 0.83 \times 10^2$

TABLE 3.2 (CONT'D)

Pot No.			
25	$\lambda_4 \beta_4 / \ell$	$= 0.0315 \times 1.66 \times 10^{-3} / 2.0 \times 10^{-5}$	$= 0.0261 \times 10^2$
26	λ_4	$= 0.0315$	$= 0.0315$
27	β_5 / ℓ	$= 0.25 \times 10^{-3} / 2.0 \times 10^{-5}$	$= 0.125 \times 10^2$
28	$\lambda_5 \beta_5 / \ell$	$= 0.0124 \times 0.25 \times 10^{-3} / 2.0 \times 10^{-5}$	$= 0.155$
29	λ_5	$= 0.0124$	$= 0.0124$
30	$(\Delta P_f + \Delta P_p) / \sum \frac{L_i}{A_i g} R_o$	$= (0.785) / 4733 \times 22.2$	$= 0.7469 \times 10^{-3}$
31	$\gamma \left(\frac{h_1}{2} + h_2 + \frac{h_3}{2} \right) A_{ig} / L_i R_o$	$= 0.208 \times 10^{-2} \times 5.03 / 47.33 \times 22.2$	$= 0.9952 \times 10^{-5}$
32	$\gamma h_1 A_{ig} / L_i R_o$	$= 0.208 \times 10^{-2} \times 0.92 / 47.33 \times 22.2$	$= 0.1821 \times 10^{-5}$
33	$\gamma \left(\frac{3h_1}{2} + h_2 + \frac{h_3}{2} \right) A_{ig} / L_i R_o$	$= 0.208 \times 10^{-2} \times 5.95 / 47.33 \times 22.2$	$= 0.1177 \times 10^{-4}$
34	$2/\tau_{2,3}$	$= 2/2.859$	$= 0.6995$
35	$1/\tau_{2,3}$	$= 1/2.859$	$= 0.3498$
36	$1/\tau_{2,3}$	$= 1/2.859$	$= 0.3498$
37	$12/\tau_{3,4}$	$= 12/1.310$	$= 9.1603$
38	$12/\tau_{3,4}^2$	$= 12/(1.310)^2$	$= 6.9926$
39	$12/\tau_{3,4}^2$	$= 12/(1.310)^2$	$= 6.9926$
40	$6/\tau_{3,4}$	$= 6/1.310$	$= 4.5802$
41	$6/\tau_{3,4}$	$= 6/1.310$	$= 4.5802$
42	R / W_G	$= 447.4 / 602$	$= 0.7432$
43	$(U_G A_G)_o / W_G C_G$	$= 197.7 / 658.6$	$= 0.3002$
44	$P_D L_1 / [W_L C_L + W_E C_E]$	$= 10,055 \times L_1 / (2802 \times 1.03 + 2490 \times 0.12)$	
45	$P_D L_2 / [W_L C_L + W_E C_E]$	$= 10,055 \times L_2 / (2802 \times 1.03 + 2490 \times 0.12)$	
46	$(U_G A_G) / [W_L C_L + W_E C_E]$	$= 197.7 / 3185$	$= 0.0627$

TABLE 3.2 (CONT'D)

<u>Pot No.</u>			
47	$1 / \tau_{6,1}$	$= 1 / 3.404$	$= 0.2938$
48	$2 / \tau_{6,1}$	$= 2 / 3.404$	$= 0.5875$
49	$1 / \tau_{6,1}$	$= 1 / 3.404$	$= 0.2938$
50	$12 / \tau_{5,6}^2$	$= 12 / (3.069)^2$	$= 1.2741$
51	$12 / \tau_{5,6}^2$	$= 12 / (3.069)^2$	$= 1.2741$
52	$6 / \tau_{5,6}$	$= 6 / 3.069$	$= 1.9550$
53	$6 / \tau_{5,6}$	$= 6 / 3.069$	$= 1.9550$
54	$12 / \tau_{5,6}$	$= 12 / 3.069$	$= 3.9101$
55	NOT USED		
56	NOT USED		
57	NOT USED		

4.0 RESULTS

4.1 STEADY STATE DIGITAL ANALYSIS OF SM-1 CORE II

Six elements were selected to be analyzed by the STDY-3 code, based on previous work on Core II. (2)(3) These elements were representative of the hottest, coldest and intermediate elements in Core II. The core positions selected were core positions 16, 35, 44, 62, 65, and 76.

The results of the STDY-3 runs are shown in Table 4.1 and in Fig. 4.1 and 4.2. Core ΔT is the temperature rise of the total coolant from inlet to outlet.

In previous thermal analysis of elements the flow in the hot channel was less than flow in the nominal. Under conditions of natural circulation the STDY-3 analysis shows this to be reversed; the hot channel flow is greater than the nominal. This is caused by the chimney effect where the increased temperatures and the reduction in flow resistance causes more coolant to be drawn into these hot channels.

To permit a test to be scheduled at an early date it was decided to limit the core rise ΔT in the SM-1 to a value of 20°F which is the normal core ΔT at normal operation with forced convection. This limit eliminates any concern in regard to thermal stress. The analysis indicates that with a 20°F rise across the core, the flow rate would be 22.2 lb/sec. The maximum core power obtainable under these conditions is 0.51 Mwt and the minimum burnout ratio is 104 in element position 65. The range of the other 5 elements is 142 to 430.

The STDY-3 code uses a set of equations which are flux dependent. The applicability of these equations to low flows is questionable because of the possibility that laminar rather than turbulent flow may exist. A hand calculation, using more recent flow dependent burnout equations, was made which gave similar results to the STDY-3 results, indicating that the above concern was unwarranted.

4.2 RESULT OF ROD WITHDRAWAL ACCIDENT

If positive reactivity is introduced continuously into the reactor, as during a control rod withdrawal, the core power will change at an even increasing rate. The maximum rate which this excursion attains prior to reaching the overpower trip (scram) setting varies inversely with the power

TABLE 4.1

RESULTS OF STDY-3 ANALYSIS

Loop Flow lbs/sec	Flow Ratio Hot/ Nom.	Hot Channel Mass Velocity, lbs/hr-ft ²	Core Power Mw	Core ΔT, °F	Average Flux Btu/hr-ft ²	Bulk Water Temp, °F	Max Surface Temp, °F	DNBR (STDY-3 Results)	* LB	* BB	Qual. %
Element 65											
10	1.10	2.018×10^4	0.051	4.37	3.350×10^2	452.2	458.8	1061	0	0	0.0
20	1.60	5.872×10^4	0.380	16.4	2.508×10^3	460.8	484.5	141	0	0	0.0
30	1.60	8.808×10^4	1.24	35.5	8.157×10^3	478.9	536.7	43.1	0	0	0.0
40	1.60	1.174×10^5	2.87	61.6	1.886×10^4	503.3	573.0	18.4	0	0	0.0
40	1.70	1.560×10^5	5.51	94.4	3.617×10^4	525.8	574.5	9.5	7	0	0.0
60	1.60	1.762×10^5	9.33	133	6.143×10^4	565.2	575.8	5.4	3	21	0.4
65	1.70	2.028×10^5	11.72	154	7.720×10^4	565.2	576.4	4.3	3	19	1.8
70	1.80	2.313×10^5	14.46	176	9.527×10^4	565.2	577.0	3.5	1	18	3.1
75	1.70	2.339×10^5	17.54	198.5	1.155×10^5	565.2	577.6	2.8	1	13	9.6
Element 16											
30	1.30	3.871×10^4	1.24	35.5	8.157×10^3	508.1	571.0	77.6	0	0	0.0
60	1.90	1.131×10^5	9.33	133	6.114×10^4	565.2	574.8	10.4	1	17	3.3

* Inches from inlet end of the fuel plate at which phenomenon starts.

TABLE 4.1 (CONT'D)

RESULTS OF STDY-3 ANALYSIS

Loop Flow lbs/sec	Flow Ratio Hot/ Nom.	Hot Channel Mass Velocity, lbs/hr-ft ²	Core Power Mw	Core ΔT , °F	Average Flux Btu/hr-ft ²	Bulk Water Temp, °F	Max Surface Temp, °F	DNBR (STDY-3 Results)	* LB	* BB	Qual. %
Element 35											
30	1.10	7.227×10^4	1.24	35.5	8.157×10^3	449.9	468.1	181	0	0	0.0
60	1.10	1.445×10^5	9.33	133	6.114×10^4	464.3	550.3	24.9	0	0	0.0
Element 44											
30	0.90	5.545×10^4	1.24	35.5	8.157×10^3	484.4	562.9	58.7	0	0	0.0
60	0.90	1.109×10^5	9.33	133	6.114×10^4	565.2	574.8	7.4	1	16	4.0
Element 62											
30	1.30	5.765×10^4	1.24	35.5	8.157×10^3	468.8	508.8	105	0	0	0.0
60	1.30	1.153×10^4	9.33	133	6.114×10^4	532.3	574.2	13.8	4	0	0.0
Element 75											
30	1.70	5.020×10^4	1.24	35.5	8.157×10^3	482.5	531.0	89.9	0	0	0.0
60	1.70	1.004×10^5	9.33	133	6.114×10^4	565.2	574.5	11.4	1	18	2.7

* Inches from inlet end of the fuel plate at which phenomenon starts.

Figure 4.1 SM-1 Reactor Performance Under Conditions of Natural Circulation

Figure 4.2 SM-1 Core II Theoretical Hot Channel Element Position 65.
Plate Surface Temperature Vs. Length Along Channel for
Natural Circulation

level. Thus during the proposed natural circulation test a rod withdrawal could result in the maximum rate of power increase.

Figure 4.3 shows the reactivity change resulting from this type of accident in the SM-1. Reactor power increase as a function of time is shown in Fig. 4.4 along with departure from nucleate boiling ratios (DNBR). The nominal channel outlet temperature has been plotted vs. time in Fig. 4.5.

As can be seen in Fig. 4.5 the DNBR drops to 7 after 11 sec. This is a factor of 2+ above the minimum allowable. The nominal outlet temperature of element 65 has risen some 40° in 4-1/2 sec (7 sec to 11.5 sec) and is continuing at a rapid rate. However, there exists the high temperature alarm and scram system which will actuate to suppress power generation when the reactor outlet temperature reaches 480°F (scram). Since this analysis has been conducted on the hottest element it is not known how far the reactor outlet temperature lags behind the hot element. It is assumed that the alarm circuit will be actuated before the hot channel element reaches boiling or burnout.

This is considered the worst accident that the reactor could undergo under conditions of natural circulation.

4.3 TRANSIENTS IMPOSED BY LOAD CHANGES

Because of lack of time and analog computer scheduling, no transient load testing of the SM-1 primary system using analog model simulation could be performed as part of this task.

Figure 4.3 SM-1 Reactivity Insertion During Rod Withdrawal with Natural Circulation

Figure 4.4 Transient Performance of SM-1 Reactor Under Natural Circulation Conditions for a Rod Withdrawal Accident. DNBR and Power Vs. Time

Figure 4.5 SM-1 Nominal Channel Exit Temperature During Rod Withdrawal with Natural Circulation

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

5.0 CONCLUSIONS

1. If the core temperature rise is limited to 20°F (the normal ΔT of the SM-1 core), 0.51 Mwt is the maximum that can be removed from the core during natural convection operation.
2. Burnout ratios under conditions of natural circulation at 0.51 Mwt show extremely safe operation even when a rod withdrawal is imposed upon core operation (Minimum DNBR = 7.0).
3. Based on these results the SM-1 can be safely tested under natural circulation conditions at the recommended power level.

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

6.0 REFERENCES

1. Alco Products, Inc., "Fiscal Year 1962 Program Plan for Engineering Support and Development of Army Pressurized Water Reactor Power Plants", AP Note-378, September 6, 1961.
2. Lee, D. H., et al "Experiments and Analysis for SM-1 Core II with Special Components", APAE-85, April 26, 1961.
3. Coombe, J., et al "Hazards Report for SM-1 Core II with Special Components", APAE-84, April 30, 1961.
4. Bradley, P. L. "Investigation of Local Boiling of SM-1", APAE-88, June 20, 1961.
5. Davidson, S. L. and Segalman, I., "Thermal Analysis of Type 3 Elements in the SM-1, SM-1A and PM-2A Cores", APAE-105, March 30, 1962.
6. Pyle, R. S., "STDY-3, A Program for the Thermal Analysis of a Pressurized Water Nuclear Reactor During Steady State Operation", WAPD-TM-213, June 1960.
7. Brondel, J. O. and Tomonto, J. R., "Plant Transient Analysis of the APPR-1 by Analog Computer Methods", APAE-38, October 1, 1958.
8. Tomonto, J. R., "Analysis of the Decay Heat Removal Problem for the SM-2 by Analog Computer Methods", APAE Memo-198, June 1, 1959.
9. Tomonto, J. R., "PM-2A Reactor Analysis of Auxiliary Decay Heat Removal by Analog Computer Methods", APAE Memo-207, October 1959.
10. Meyer, J. E., Peterson, W. D., "ART-04 - A Modification of the ART Program for the Treatment of Reactor Thermal Transients on the IBM-704", WAPD-TM-202, July 1960.
11. Segalman, I., Bradley, P. L., "Steady State and Transient Thermal and Hydraulic Analysis of SM-2 - Termination Report", APAE-91, September 8, 1961.

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

7.0 APPENDIX

ESTIMATED PRESSURE DROP
FREE ROTOR
PUMP Q-350-A1

Figure A.1 Estimated Pressure Drop, Free Rotor Pump Q-350-A1

Figure A.2 Estimated Pressure Drop, Locked Rotor Pump Q-350-A1

Figure A.3 Test Performance Curve, Q-350-A1 Pump 8DF 591-1

Figure A.4 Performance Curve Estimated from Test in D/M-8000-A1 Volute