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RESEARCH AND DEVELOPMENT PROGRAM

DEVELOPMENT OF EQUATIONS FOR ANALOG COMPUTER
STUDIES TO SIZE THE REACTOR PLANT PRESSURIZER

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WESTINGHOUSE ELECTRIC CORPORATION
ATOMIC POWER DEPARTMENT

PITTSBURGH, 30

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STUDIES TO SIZE THE REACTOR PLANT PRESSURIZER

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by

W. G. Lyman

Large Plant Engineering

For The Yankee Atomic Electric Company
Under Research and Development Subcontract
No. 1 of USAEC-YAEC Contract AT(30-3)-222

October 15, 1958

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I. ABSTRACT

The purpose of this report is to present the assumptions and equations used to conduct reactor plant load transient studies on the analog computer. The study was performed to determine the magnitude of reactor cooling water temperature and volume variations caused by secondary plant load transients, and to establish the size of the pressurizer which would be capable of limiting the cooling water pressure variations caused by the volume surges.

II. INTRODUCTION

During steady state operation of the reactor plant, the heat transferred from the core fuel to the cooling water equals the heat transferred from the water to the steam generator tubes, so the heat content of the water remains constant. A change in electrical load produces a change in steam flow and temperature, resulting in an unbalance in the heat into and out of the cooling water. The resulting change in heat content of the cooling water produces a surge of water into or out of the pressurizer, and the steam volume in the pressurizer acts as a spring to absorb the volume surges.

The change in temperature of the cooling water at the steam generator is felt at the reactor core after a time delay which depends upon the velocity of the cooling water and the length of piping and header volumes between the steam generator tubes and the reactor core. The change in average temperature of the water flowing through the core produces a change in reactivity due to the moderator, or water, negative temperature coefficient, which in turn produces a change in core power in the desired direction. The magnitude of the load change, the piping time delay, and the negative temperature coefficient will determine the magnitude of the change in cooling water heat content or in the volume surge felt in the pressurizer. The volume surge produces a pressure change which depends upon the initial steam and water volumes in the pressurizer vessel.

Knowledge of the expected electrical system load transients and results of the computer studies of various load transients can be used to establish a criterion for which transients must be limited by auxiliary control systems, such as reactor control rod or turbine steam valve controls, and which transients may be limited by the negative temperature coefficient alone. The established criterion and the allowable variations in pressure are used to determine the volume of the pressurizer vessel and the quantities of water and steam occupying the pressurizer volume.

III. CONCLUSIONS

Results of a computer study based on the analysis presented in this report will provide the necessary information for sizing the reactor plant pressurizer. Several of the assumptions made herein are on the conservative side. As an example, the surge volume is calculated on the basis of constant reactor coolant pressure. The coolant pressure change will actually tend to reduce the volume surge, so the maximum volume surge or pressure change for a given load transient should be less than that determined by this analysis. The heat storage of the pressure vessels and piping would also tend to dampen or reduce the rate at which reactor coolant temperature changes during a transient. A pressurizer size based on isentropic compression of the steam volume is conservative, since steam is condensed during the positive surge transient.

The computer study will provide other information that is useful in the design of the reactor plant. Volume surge rates will establish a size for the pressurizer safety valves and the connecting piping. Rates and magnitudes of temperature changes will provide information necessary for thermal stress calculations. Analysis of transient behavior of the uncontrolled reactor plant is useful in the design of the reactor control systems.

IV. EQUATIONS AND ASSUMPTIONS

A. REACTOR NEUTRON KINETICS AND HEAT TRANSFER

The generation of heat in the reactor core is directly proportional to the thermal neutron density, and can be expressed by

$$q_f = \alpha n \quad (1)$$

where q_f = heat generated in the core, Btu/sec

α = proportionality constant, Btu-cm³/neutron sec

n = thermal neutron density, neutrons/cm³

Variations in the core fuel temperature, the reactor water temperature and pressure, and the position of the control rods produce changes in core reactivity which, in turn, change the thermal neutron density. The rate of change of thermal neutron density is expressed by:

$$\frac{dn}{dt} = \frac{\delta k - \beta}{l^*} n + \sum_{i=1}^6 \lambda_i C_i \quad (2)$$

where t = time, seconds

k = change in reactivity

β = fraction of delayed neutrons

l^* = prompt neutron lifetime, seconds

λ_i = decay constant of the i th group of delayed neutrons

C_i = concentration of delayed neutrons emitted of group i .

C_i is defined by

$$\frac{dC_i}{dt} = \frac{\beta_i}{l^*} n - \lambda_i C_i \quad (2a)$$

where β_i refers to the fraction of delayed neutrons of the i th group.

The heat generated in the fuel is transmitted through the fuel and the stainless steel fuel cladding to the water flowing through the core, and is expressed by the following group of equations:

$$q_f = m_f c_f \frac{dT_f}{dt} + H_{fm} [\bar{T}_f - \bar{T}_m] \quad (3)$$

$$H_{fm} [\bar{T}_f - \bar{T}_m] = m_m c_m \frac{dT_m}{dt} + H_{mr} [\bar{T}_m - \bar{T}_r] \quad (4)$$

$$H_{mr} [\bar{T}_m - \bar{T}_r] = m_r c_r \frac{dT_r}{dt} + W c_r [T_{ro} - T_{ri}] \quad (5)$$

where m = mass, lbs

c = specific heat, Btu/lb-°F

$\bar{T}$ = average temperature, °F

T = temperature, °F

H_{fm} = heat transfer coefficient (average), fuel to metal cladding, Btu/sec-°F

H_{mr} = heat transfer coefficient (average), cladding to water, Btu/sec-°F

W = mass flow rate of water through reactor, lb/sec

subscripts: f = core fuel

m = fuel cladding metal

r = water in core volume

ro = water leaving core

ri = water entering core

B. STEAM GENERATOR HEAT TRANSFER

The heat transmitted by the cooling water to the steam generators is transferred through the steam generator tube metal to the saturated

water surrounding the tubes; steam is generated and transmitted to the steam turbine. The following equations express the heat transfer in the steam generator:

$$W_{cb} \int T_{bi} - T_{bo} = m_{bcb} \frac{d\bar{T}_b}{dt} + H_{bg} \int \bar{T}_b - \bar{T}_g \quad (6)$$

$$H_{bg} \int \bar{T}_b - \bar{T}_g = m_{gcg} \frac{d\bar{T}_g}{dt} + H_{gs} \int \bar{T}_g - T_s \quad (7)$$

$$H_{gs} \int \bar{T}_g - T_s = m_{scs} \frac{dT_s}{dt} + \Delta h_s [W_s + W_d] \quad (8)$$

where the subscript b = primary water in steam generator

bi = water entering steam generator tubes

bo = water leaving steam generator tubes

g = steam generator tube metal

s = water and steam on the secondary side

and H_{bg} = heat transfer coefficient (average) primary water to tubes, Btu/sec-°F

H_{gs} = heat transfer coefficient (average), tubes to secondary water, Btu/sec-°F

Δh_s = difference in feedwater and steam enthalpies, Btu/lb

W_s = steam flow to turbine, lb/sec

W_d = steam flow through dump to condenser lb/sec

The vertical steam generator has a recirculation rate of 10 or more, meaning that the flow of saturated water which is removed from the steam by the steam separators is 10 times the flow of feedwater. The saturated water mixes with the feedwater in the annular downcomer which surrounds the tubes, so that the water entering the heat transfer area around the

tubes is practically saturated. On this basis the assumption was made that all of the water surrounding the tubes is saturated (equations 7 and 8), and no feedwater heating effect is included. However, the feedwater temperature is used to determine the heat required to form steam (Δh_s in equation 8).

C. PIPING TRANSPORT DELAYS

The reactor cooling water is transported to and from the core through piping and associated inlet and outlet headers. A change in the temperature of the water leaving the steam generator is felt in the core after a time delay which is dependent on the water flow rate and the magnitude of the transport volume. This delay is approximated by the expression:

$$T_{ri} = T_{bo} - \frac{V_c}{F_c} \frac{dT_{ri}}{dt} \quad (9)$$

for the cold leg, or reactor inlet transport volume, and

$$T_{bi} = T_{ro} - \frac{V_h}{F_h} \frac{dT_{bi}}{dt} \quad (10)$$

for the hot leg, or reactor outlet transport volume,

where V_c = cold leg transport volume, cu ft

V_h = hot leg transport volume, cu ft

F_c = cold leg flow rate, cu ft/sec

F_h = hot leg flow rate, cu ft/sec

D. SURGE VOLUME

In order to determine the surge volume caused by an unbalance in heat addition and removal during load transients, the expression

$$q_s = H_{mr} \sqrt{T_m - T_r} - H_{bg} \sqrt{T_b - T_g} \quad (11)$$

taken from equations (5) and (6), is integrated in time by the computer, resulting in:

$$\Delta Q_s = \int_0^t q_s dt \quad (12)$$

where ΔQ_s is the heat addition to, or subtraction from, the reactor coolant water, expressed in Btu, at any time t during a load transient. The change in average enthalpy of the reactor coolant water is

$$\frac{\Delta Q_s}{M} = \Delta h, \text{ Btu/lb} \quad (13)$$

where M is the mass of reactor coolant water activity involved in heat transfer, and is assumed to remain constant throughout a transient. The specific volume of the water varies linearly with enthalpy at constant pressure, or

$$\Delta v_w = \left. \frac{\partial v_w}{\partial h} \right|_p \Delta h, \text{ cu ft/lb} \quad (14)$$

and the surge volume at the pressurizer is

$$\Delta V = M \Delta v_w, \text{ cu ft} \quad (15)$$

or

$$\Delta V = \left. \frac{\partial v_w}{\partial h} \right|_p \Delta Q_s, \text{ cu ft} \quad (15a)$$

Figure I is a schematic diagram of the reactor plant neutron kinetics and heat transfer equations.

E. PRESSURE VARIATION

The surge volume, ΔV , produces a change in the specific volume of the steam, Δv_s , in the pressurizer, or

$$\Delta v_s = -\frac{\Delta V_s}{M_s} \quad (16)$$

where M_s is the initial mass of steam, in lbs, and the negative sign is used since a positive surge, coming from the primary system, causes

REACTOR PLANT SCHEMATIC DIAGRAM
OF NEUTRON KINETICS AND HEAT TRANSFER EQUATIONS

FIGURE I

a compression of the steam. The pressure increase caused by a positive surge will depend on the rate and magnitude of the surge. Fast, large volume surges will result in compressing the steam isentropically since only a small amount of steam will be condensed during the fast surge. If the positive surge is slow, the compressed steam will remain close to saturated conditions, and the resulting pressure rise will be less than that caused by a faster surge having an equal volume. Since the process is quite complicated, the more conservative assumption of the isentropic compression is used for purposes of sizing the steam volume of the pressurizer. For positive surges, the change in pressure is expressed by

$$\Delta P = \left. \frac{\partial P}{\partial v_s} \right|_s \Delta v_s \quad (17)$$

where ΔP = change in pressure, psi

$$\left. \frac{\partial P}{\partial v_s} \right|_s = \text{constant entropy compression factor, lb}^2/\text{in}^2\text{-ft}^3$$

The negative surge expands the steam in a saturated process. If the water in the pressurizer is saturated at the initiation of a negative surge, part of the water will flash to steam. The mass of steam will increase in this case, such that equation (16) will not be correct, and an artificial expansion factor which corrects for mass changes must be used. Therefore,

$$\Delta P = \left. \frac{\partial P}{\partial v_s} \right|_F \Delta v_s \quad (18)$$

is applicable for negative surges, when the pressurizer water is saturated, where

$$\left. \frac{\partial P}{\partial v_s} \right|_F = \text{flashing water expansion factor, determined from energy balances of the pressurizer during negative surges for various ratios of water to total pressurizer volume, lb}^2/\text{in}^2\text{-ft}^3.$$

F. COEFFICIENTS

The coefficients used in the equations, such as heat transfer, specific heat, flow rate, and mass coefficients, do not vary significantly as temperature change during load transients. These coefficients, evaluated at normal full power and full flow conditions, were allowed to remain

constant. In general, the studies will be conducted to note temperature and volume changes as a function of initial power level, step changes and rates of change of power, and various moderator or reactor coolant negative temperature coefficients. Table I presents the calculated values of constant coefficients for the 482 MW reactor plant operating at 514°F $T_{average}$ and 2000 psia. Table II presents the normal values of the parameters during full load operation at 482 MW.

TABLE I
REACTOR PLANT CONSTANT COEFFICIENTS

H_{bg}	18,450 Btu/sec-°F
H_{gs}	18,450 Btu/sec-°F
M	147,000 lbs
V_c	1220 cu ft
V_h	460 cu ft
W	10,500 lb/sec
c_b	1.177 Btu/lb-°F
c_f	0.079 Btu/lb-°F
c_g	0.15 Btu/lb-°F
c_m	0.15 Btu/lb-°F
c_r	1.177 Btu/lb-°F
Δh_s	895 Btu/lb
λ^*	5.5×10^{-5} sec
m_b	25,430 lb
m_f	60,600 lb
m_g	154,000 lb
m_m	10,000 lb
m_r	6360 lb
$m_s c_s$	114,000 Btu/°F
α	5.2×10^{-3} Btu-cu cm./sec-neutrons
β_i (two group approximation)	$i = 1$ $i = 2$ 0.0053 0.002
λ_i (two group approximation)	0.275 0.0263
$\left. \frac{\partial P}{\partial v_s} \right _S$	- 15,000 psi/cu ft/lb
$\left. \frac{\partial v_m}{\partial h} \right _P$	- 1.95×10^{-5} cu ft/Btu

TABLE II

482 MEGAWATT FULL LOAD PLANT PARAMETERS

F_c	210.4 cu ft/sec
F_h	219.6 cu ft/sec
$T_{average}$	514°F
T_c	495°F
T_h	532°F
T_{steam}	467°F
W_d (5% of full flow)	25.5 lb/sec
W_s	510 lb/sec
q_f	457,000 Btu/sec

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