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PATHFINDER ATOMIC POWER PLANT

ANALOG SIMULATOR RESULTS

MASTER

Submitted to
U. S. ATOMIC ENERGY COMMISSION
NORTHERN STATES POWER COMPANY
and
CENTRAL UTILITIES ATOMIC POWER ASSOCIATES
by

ALLIS-CHALMERS MANUFACTURING COMPANY
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Submitted to

U. S. ATOMIC ENERGY COMMISSION
NORTHERN STATES POWER COMPANY

and

CENTRAL UTILITIES ATOMIC POWER ASSOCIATES

by

ALLIS-CHALMERS MANUFACTURING COMPANY

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PATHFINDER ATOMIC POWER PLANT
ANALOG SIMULATOR RESULTS

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FOREWORD

One of a series of reports on research and development in connection with the design of the Pathfinder Atomic Power Plant, specifically, this report presents the results of the dynamics testing of the Pathfinder plant as simulated on an analog computer. The simulation model is described in Report No. ACNP-62001, entitled "Pathfinder Atomic Power Plant Analog Simulator".

The Pathfinder plant will be located at a site near Sioux Falls, South Dakota, and is scheduled for operation in 1964. Owners and operators of the plant will be the Northern States Power Company of Minneapolis, Minnesota. Allis-Chalmers is performing the research, development, and design as well as being responsible for plant construction.

The U.S. Atomic Energy Commission, through Contract No. AT(11-1)-589 with Northern States Power Company, and Central Utilities Atomic Power Associates (CUAPA) are sponsors of the research and development program. The plant's reactor will be of the Controlled Recirculation Boiling Reactor type with Nuclear Superheater.

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NOMENCLATURE

C_F	- Specific Heat of Fuel in the Superheater Core
C_P	- Specific Heat of Steam in the Superheater Core
db	- Decibels ($20 \log_{10}$)
Δh	- Subcooling
h_{FW}	- Feedwater Enthalpy
h_f	- Enthalpy of Saturated Water
K	- Excess Reactivity ($K = K_R - K_D - K_V$)
K_D	- Reactivity in Fuel Temperature (Doppler Effect)
K_R	- Reactivity in Control Rods
K_V	- Reactivity in Moderator Voids (Density Effect)
M_F	- Mass of Fuel in the Superheater Core
M_S	- Mass of Steam in the Superheater Core
n	- Reactor Power Level
P_1	- Reactor Dome Pressure
P_2	- Superheater Exit Steam Pressure
ΔP_{SH}	- Superheater Pressure Drop
ΔP_{SL}	- Steam Line Pressure Drop
Q_B	- Heat Transferred to the Reactor Coolant in the Boiler Core
Q_g	- Heat Generated in the Boiler Core
T_2	- Superheater Exit Steam Temperature
T_F	- Superheater Hot-Spot Fuel Temperature
T_{FW}	- Feedwater Temperature

NOMENCLATURE (Continued)

- T_S - Temperature of Steam in the Superheater Core
 T_{SF} - Temperature of Fuel in the Superheater Core
 W_{FW} - Feedwater Weight Flow Rate
 W_S - Steam Line Weight Flow Rate
 W_{SV} - Safety Valve Steam Weight Flow Rate
 W_V - Recirculation Weight Flow Rate
 $\bar{x}$ - Average Quality in the Boiler Core
 $\bar{\alpha}$ - Average Void Fraction in the Boiler Core

SYMBOLS

- Δ - This prefix indicates a change from initial conditions, with the exception of Δh , ΔP_{SH} and ΔP_{SL} . In these cases the prefix δ is used.
- $()_{iv}$ - This indicates the initial value of a variable.
- $\dot{P}_2 = dP_2/dt$ - The first order time derivative.

ABSTRACT

The results of exhaustive studies concerning the dynamics of the Pathfinder Atomic Power Plant as simulated on an analog computer are presented. The presentation is in the form of five distinct categories.

The categories or sections are:

1. Operational studies
2. Accident studies
3. Parameter studies
4. Stability studies
5. Special simulations and studies

The section on operational studies answers the following two questions; "What disturbances will be considered allowable for normal operation?" and "What will be the response of the reactor to these allowable disturbances?" These two questions were asked of each of the major systems that make up the reactor system.

The section on accident studies verifies that the integrity of the reactor system is maintained by the protection system during extreme conditions. Extreme conditions initiated by coincident causes are not reported.

The section on parameter studies determines the effect of variations in system parameters on the response of the reactor. The parameters varied were physical constants and calculated design parameters.

ABSTRACT (Continued)

The section on stability studies demonstrates the stability of the reactor system as determined by frequency response techniques.

The section on special simulations and studies presents the results of studies that fall into the classification of being over and above the normal requirement for a pre-operational understanding of the system.

1.0 SECTION ONE - OPERATIONAL STUDIES

1.1 Introduction

The scope of this section is to report the results obtained from a particular group of experiments that were performed on the Pathfinder Transient Simulator. These experiments answer the following two questions: 1) What disturbances will be considered allowable for normal operation and control of the reactor... and, 2) What will be the response of the reactor to the allowable disturbances. The definition of an allowable disturbance is one that will not result in damage and/or shutdown of the reactor. These disturbances are initiated by the major systems necessary for the operation and control of the reactor. The systems investigated are the control rod, feed-water, recirculation, and the pressure-temperature control. All the experiments have a physical basis because they simulate only the system adjustments that are available to the plant operator. The results of these experiments will afford an insight into the permissible range of adjustment for each of the reactor systems. The results also provide a means of comparing the initial design techniques and the data received from actual operational experiments. This (theoretical-experimental) comparison will improve subsequent designs and, in general, add to nuclear power plant technology.

1.2 Control Rod System

The power level of the Pathfinder Reactor will be varied by manually

moving the control rods. Also, the automatic shutdown of the reactor depends primarily on the control rods. Shutdown can be automatically accomplished by either of two methods initiated by the automatic reactor safety system. They are runback and scram.* Therefore, the control rods serve two purposes: 1) power level control... and, 2) automatic reactor shutdown.

1.2.1 Control Rod Withdrawal: Figures 1-1 and 1-2

The effect of control rod movement on the power level was investigated by an experiment designed to increase the power level from initial steady-state conditions. Since rod withdrawal is the most critical, it was felt that controlled rod insertion experiments were unnecessary. The experiment studies the effect of control rod withdrawal at various power levels and various recirculation flow rates. The three flow rates used correspond to one, two, and three pump operation.

The variation in power level and flow rate is reflected in the parameter γ . The definition of γ is the ratio of the initial power level to the initial recirculation flow rate ($\gamma = (n)_{iV}/(W_V)_{iV}$), with three pump operation defined as 100 per cent flow rate. One hundred per cent power is rated power for the three pump flow rate. (These definitions of power level and

*Not studied in this report is a third means of reactor shutdown - controlled shutdown.

Figure 1-1...CONTROL ROD WITHDRAWAL AT 5¢/SEC FOR VARIOUS INITIAL VALUES OF n AND n/W_v ; $(n/W_v)_{iv} = \gamma$; (Exp. 62-07)

Superheater Fuel Hot Spot Temperature

Superheater Steam Outlet Temperature

Figure 1-2...CONTROL ROD WITHDRAWAL AT 5¢/SEC FOR VARIOUS INITIAL VALUES OF n AND n/W_v ; $(n/W_v)_{iv} = Y$; (Exp. 62-071); Dwg. 43-025-007

recirculation flow rate hold throughout this entire report.) The power levels for one, two, and three pump operation are respectively, 47, 80, and 100 per cent; corresponding values for γ are 1.12, 1.04, and 1.00. These power and flow conditions are stated for operating the recirculation line butterfly valve(s) in a full open condition.

The control rod withdrawal was simulated by adding reactivity to the system at a linear rate of 5 cents/sec. This corresponds to the maximum rate that reactivity can be inserted at voided conditions by the control rods. The 5 cents/sec. corresponds to an original design criteria, which these experiments showed to be acceptable. In fact, the experiments demonstrated that the reactor system could actually tolerate a considerably faster rate. Nevertheless, since the control rod drives were chosen to give a maximum reactivity insertion rate of 5 cents/sec., it was decided that a change to an increased rate is unwarranted. In each case the control rods were withdrawn for 10 seconds resulting in a total reactivity addition of 50 cents. After the 10 seconds the control rods were held stationary. The results of the experiment show that the smaller the value of γ , the greater will be the power transient when expressed as a percentage of the initial power level. This tendency for the transient to increase with a decrease in the γ ratio is also demonstrated by the transients, particularly, the steam

temperature. The reactor's ability to damp out transients is a direct function of the γ ratio, with decreased damping corresponding to decreasing γ .

1.2.2 Runback: Figure 1-3

As previously mentioned, the two safety actions are runback and scram. Both of these mechanisms center around the insertion of control rods. The differences between the two are basically: a) the rate of control rod insertion... and, b) extent to which the reactor is shut down. For runback the rods are inserted at the rate the control rod drive motors operate (inner boiler rods, 0.43 in/sec; outer boiler rods, 1.20 in/sec.). Runback is terminated when the condition causing runback has been corrected or when all rods are fully inserted into the core. Therefore, the extent of shutdown depends on the severity of the hazard. With this in mind, runback was studied by an experiment that simulated two of the actual conditions, each of which will precipitate runback. The runback conditions simulated were: 1) the feedwater temperature is less than 310 F... and, 2) the power level is greater than 110 per cent but less than 115 per cent. The first condition was achieved by decreasing the enthalpy of the feedwater at a rate of 6 Btu/lb/sec for 45 seconds from an initial feedwater temperature of 356 F. Runback was initiated when the feedwater dropped below 310 F and continued until all the rods were inserted.

Figure 1-3...REACTOR RESPONSE DUE TO RUNBACK CAUSED BY FEEDWATER TEMPERATURE
 $\leq 310^\circ F$; $\Delta h = -6$ BTU/lb-sec for 45SEC (Exp. 62-40)

43-025-413

The feedwater temperature decreased to 92 F in 45 seconds and remained at that level throughout the entire experiment. This is equivalent to a complete loss of feedwater heating in 45 seconds. The second condition, excessive power level, was achieved by continuously withdrawing a control rod with a reactivity insertion rate of 5 cents/sec. For a complete discussion of the second condition the reader is referred to Section Two - Withdrawal of the Most Valuable Rod. The purpose of the experiments was to determine how runback actually would affect the reactor power level. The plot of reactor power for the case of runback due to low feedwater temperature exhibited a plateau. The plateau means that the reactor is in a state of being just critical or zero excess reactivity, since the negative reactivity inserted by the control rods being run back is compensated for by an equal decrease in the amount of reactivity being held down by voids. The steam flow will remain constant during this period of constant power. This will cause the superheater temperature to seek the steady-state value for that particular power level and steam flow. The temperature plot exhibits this effect and, as expected, the temperature peaked and then started to drop as the reactor went subcritical. The experiment showed that the low feedwater temperature and high power level runbacks result in very similar temperature transients. In both cases, the superheater

hot-spot temperatures peaked within 50 F of the initial steady-state operating temperature.

1.2.3 Scram: Figure 1-4

The other safety action is reactor scram. In order to properly evaluate the capability of the scram action, experiments were conducted at extremes in core lifetime. The experiments simulated a full power or 100 per cent scram with a clean core and a core at maximum lifetime. For the clean core experiment the amount of control rod (negative reactivity) available for insertion is a minimum. Although at maximum lifetime the amount of negative reactivity insertion will be maximum, the removal of decay heat becomes a problem.

The clean core scram was simulated by inserting a reduced amount of negative reactivity with the control rods. The amount of reactivity inserted was approximately 50 per cent of the total worth of the control rods in the hot condition. (This amounts to a scram reactivity of approximately -10 dollars.) The total time to insert the control rods was assumed to be 2 seconds. (This will be the value used throughout this report.) The 2 seconds is conservative because a control rod drop test showed the insertion time varied from 1.0 to 1.5 seconds. The results of the computer experiment show that the reactor was shut down in less than 2 seconds. Since with a clean core there is not a

ΔK rods (\$)

-10
- 8
- 6
- 4
- 2
0

Scram Reactivity Curve

β (\$)

100
80
60
40
20
0

Reactor Power Level

T_F ($^{\circ}F$)

1270
1070
870
670
470
270

Superheater Fuel Surface Temperature at Hot Spot

0 10 20 30 40 50 60 70 80 90
Time, (Seconds)

Figure 1-4...REACTOR SCRAM FROM FULL POWER 43-024-584

significant amount of decay heat, the fuel temperatures declined monotonically. This is shown in the plot of superheater hot-spot temperature vs. time.

The only change in the results due to increased core lifetime is that the response of the fuel temperature changes. The changes are demonstrated in the response of the superheater hot-spot temperature. This temperature initially dropped due to the power shutdown and then began to rise. The rise is due to a lack of coolant (steam) flow to remove the decay heat. The temperature and pressure control system for the Pathfinder Nuclear Power Plant is designed to contain this rise within safe limits. The superheater hot-spot temperature, after the initial dip, rose to a steady-state temperature of 900 F.

1.3 Feedwater System

The feedwater system provides make-up water to replace the generated steam. This water is added to the reactor in the lower part of the downcomer. Besides maintaining the water level in the reactor, the feedwater provides the desired amount of inlet subcooling. These two items, water level and subcooling, are then functions of the feedwater flow rate and temperature. Therefore, in order to understand the effects of the feedwater system on the reactor, temperature and flow rate changes must be investigated. For this purpose, experiments were designed to give reactor response as a function of feedwater

temperature and feedwater flow rate changes. The feedwater temperature is altered by either changing the temperature setpoint on the feedwater temperature controller or by a malfunction in the feedwater heating system. The latter will be discussed in Section Two. A change in the feedwater flow rate occurs by manually moving the feedwater control valve, or by altering the reactor water level setpoint on the feedwater controller which subsequently moves the feedwater control valves.

1.3.1 Temperature Control: Figure 1-5 and 1-6

The experiment designed to study the effect of feedwater temperature change was performed in the following manner. The feedwater temperature was increased from the normal operating value to 400 F by a series of equal step changes in the temperature setpoint. The subcooling was consequently varied, which caused the reactor power level to change. The initial power levels used were 100, 30, and 15 per cent, all with 100 per cent recirculation flow. The feedwater temperature step changes were 15 F for all cases. The results show that the power level decreased in every case. This was due to the increase in void content resulting from the decrease in subcooling caused by the rise in the feedwater temperature.

The steam flow exhibited different effects at low powers and at high powers. At low powers, an increase in steam flow and a decrease in power level followed a feedwater temperature increase.

Feedwater Temperature Setpoint

Figure 1-5...MANUAL CHANGE OF FEEDWATER TEMPERATURE SETPOINT (Exp. 62-08);
Dwg. 43-025-060

Figure 1-6...MANUAL CHANGE OF FEEDWATER TEMPERATURE SETPOINT (Exp. 62-08);
Dwg. 43-025-058

This is due to the coupled thermodynamic and nuclear consequences of a change in subcooling. The thermodynamic effect is increased steam generation due to decreased subcooling. The nuclear effect is decreased power due to increased void content. The decrease in power level will in turn reduce steam generation. The results at the lower power levels show that the thermodynamic effect is greater than the nuclear effect. At higher initial power levels both power and steam flow decreased when feedwater temperature was increased.

The conclusion of this experiment is that increasing the feedwater temperature is a safe maneuver. Decreases in the feedwater temperature are discussed in Section Two.

1.3.2 Flow Rate Control: Figures 1-7 and 1-8

The effect of feedwater flow rate is twofold: 1) It will alter the reactor water level... while, 2) simultaneously affecting the inlet subcooling. The dominating effect from the standpoint of reactor response is the effect on the subcooling. The experiment designed to investigate this effect was conducted at three initial power levels, 100, 30, and 15 per cent. The input in each case was an increase equal to 3.67 lb/sec^2 for 4.66 seconds. The rate and time corresponds to the actual speed for manually opening the feedwater control valve. The outcome of the experiment was as expected, with the reactor power level rapidly

Figure 1-7... REACTOR RESPONSE TO SLOW FEEDWATER FLOW CHANGE $\Delta W_{FW} = 3.67 \text{ lb/sec}^2$ FOR 4.33 SEC AT VARIOUS POWER LEVELS; (Exp. 62-08)

Figure 1-8... REACTOR RESPONSE TO SLOW FEEDWATER FLOW CHANGE $\Delta \dot{W}_{FW} = 3.67 \text{ lb/sec}^2$ FOR 4.33 SEC. AT VARIOUS POWER LEVELS; (Exp. 62-08)

increasing to new steady-state values. The steam flow and the superheater hot-spot temperature increased in each instance. The increases were all within safe limits. From the curves, it can be seen that the system is less damped at lower powers. For further discussion, see Section Five.

To develop a fuller understanding of this effect, experiments were devised to vary the speed of the valve (within actual physical limitations) and the amount of change in the flow rate. Of these two parameters, it was shown that the one having the greatest effect was the amount of variation. This becomes especially important at low power levels because at these levels the possible amount of variation is the greatest. It was found from subsequent experiments that it was possible to reach power and temperature levels which required safety action. For example, doubling the feedwater flow at 20 per cent power resulted in a reactor scram caused by excessive steam temperature. But in all cases, reactor integrity was maintained.

1.4 Recirculation System

The Pathfinder Recirculation System consists of three recirculation lines, with a total rated flow capacity of 7250 lb/sec (65,406 gpm). Each recirculation line is made up of a pump and manually controlled butterfly valves. Since Pathfinder is a boiling water reactor it

will respond strongly to recirculation flow rate changes. The reactor response was investigated with regard to both the rate of change of the flow rate and the amount of flow change.

1.4.1 Two to Three Pump Operation: Figure 1-9

One of the experiments performed studies the effect of starting up the idle recirculation pump with the other two pumps already operating at their rated capacities. This experiment was run at initial power levels of 78.5 (approximately rated power level for two pump operation), 50, and 20 per cent. In the latter two cases, the lower power level was achieved by control rod insertion. The rate of increase in the flow rate simulated the actual valve speed, which resulted in the valve opening in 210 seconds. This is the maximum opening rate physically possible with the existing design. The greatest superheater hot-spot temperature rise was the 78.5 per cent case. The average of the three cases for rise in power level was 21 per cent of the initial power level. Similarly, the average peak excess reactivity, was 4.33 cents (the 78.5 per cent case peaked at 3.50 cents). The reason for the low power level resulting in the highest excess reactivity peak (5.50 cents at 20 per cent) is that at the lower power level, the increase in the non-boiling length is greater than at the higher power levels. This, in turn, means that the amount of reactivity inserted by a certain increase in recirculation flow rate is inversely

Neutron
Density
 $n(\%)$

Superheater
Hot - Spot
Temperature
 $T(^{\circ}F)$

Excess
Reactivity
 $K(\beta)$

Superheater
Exit Steam
Pressure
 $P_2(\text{psia})$

Figure 1-9...INCREASE IN RECIRCULATION FLOW RATE -TWO PUMP TO THREE PUMP
OPERATION $(n)_{iv} = 78.5\%$ $(W_v)_{iv} = 78.5\%$ EXPERIMENT: 63-03

proportional to the initial power level. The responses for the 50 and 20 per cent cases are very similar to the 78.5 per cent case, therefore, these results are not presented.

The simulator results also demonstrated that the power control of Pathfinder might be optimized by including automatic recirculation flow control. Other experiments were performed to study this effect but are not presented in this report. This information will be used to help develop the test program for the Pathfinder Nuclear Power Plant, in order to physically verify the analog results.

1.5 Pressure and Temperature Control System

In order to examine the experiments on the pressure and temperature control system, a brief discussion of the plant control philosophy is necessary. The Pathfinder Atomic Power Plant is to be operated as a base load plant with approximately constant electrical output. An increase or decrease in the electrical output is to be achieved by adjusting the reactor power level. The control system automatically senses any change in the reactor exit pressure and temperature. These two steam properties determine the system's required capacity. The capacity of the steam system is made up of the turbine inlet valves total opening, and the opening of the regulating dump valves. Since the regulating dump valve is normally closed, the full demand is usually taken up by the turbine inlet valves. But, whenever the

control system's demand for steam capacity becomes excessive, the regulating dump valve will be automatically opened to account for the excess. This condition will occur whenever the demand signal from the control system is not instantly cancelled by the "capacity" feedback signal from the turbine inlet valves.

The control system has two modes: 1) manual, and 2) automatic. The system is normally operated in the automatic mode which means that the position of the turbine inlet valves is automatically determined by the control system. The manual mode permits manual positioning of the turbine inlet valves.

1.5.1 Manual Mode

The manual mode is used principally during the startup of the reactor (see Section Five). The experiments performed were designed to determine the response of the system during operational maneuvers with the control system in manual. The maneuvers discussed are: a) control rod withdrawal... b) recirculation pump dropout... and, c) closure of the turbine inlet valves.

1.5.1.1 Control Rod Withdrawal: Figure 1-10

The input for the control rod withdrawal experiment was reactivity insertion at a rate of 2 cents/sec for 10 seconds. The results showed that the reactor will achieve a new steady-state condition with an increase

43-025-46.9

Figure 1-10...REACTOR RESPONSE WITH MANUAL PRESSURE CONTROL INPUT: ROD
WITHDRAWAL (2¢/SEC FOR 10 SECONDS)(Exp. 62-33)

in the reactor dome pressure. The superheater hot-spot temperature peaked about 60 F above the initial temperature of 1270 F, and returned to approximately the initial value. The initial power level and recirculation flow rate were 100 per cent. The power level peaked at 109 per cent. This is below the 110 per cent limit at which runback is initiated.

1.5.1.2 Recirculation Pump Dropout: Figure 1-11

The results obtained from the recirculation pump dropout experiment demonstrated that this is a safe maneuver with regard to power and temperature transients. The initial condition for this experiment was 100 per cent power and 100 per cent recirculation flow (three pumps operating). The system stabilized at a power level lower than the initial value with a corresponding decrease in steam flow.

1.5.1.3 Closure of Turbine Inlet Valves: Figure 1-12

This experiment investigated the response of the reactor to manual closure of the turbine inlet valves while at conditions of 100 per cent power and 100 per cent recirculation flow. The input for the experiment was closure of the turbine inlet valve at a rate of 10 per cent/sec until complete closure is attained. The results

Figure I-II...REACTOR RESPONSE WITH MANUAL PRESSURE CONTROL INPUT: ONE PUMP DROPOUT; (Exp. 62-33)

43-025-468

NEUTRON
DENSITY
 $\rho(\%)$

STEAM
FLOW
 $W_s(\%)$

SUPERHEATER
HOT-SPOT
TEMPERA-
TURE
 $T_F(^{\circ}F)$

43-025-470

Figure 1-12...REACTOR RESPONSE WITH MANUAL PRESSURE CONTROL INPUT: CLOSURE OF TURBINE INLET VALVE (10%/SEC.)(Exp. 62-33)

showed that the power level rose sharply while the steam flow dropped, resulting in a superheater temperature rise. The rise in power coupled with the inlet valves being closed, results in increased reactor exit pressure and temperature. Therefore, the control system senses a need for increased steam capacity. The control system then demands higher steam capacity. Since closing the turbine inlet valves creates a demand for increased steam flow capacity (as previously explained), the regulating dump valve is automatically opened. The steam flow reaching a minimum and starting to increase indicates the opening of the regulating dump valve. The increasing steam flow means greater superheater cooling causing a corresponding maximum in superheater hot-spot temperature. The results also show that runback would actually have been initiated when the power level reached 110% to further contain the transients.

1.5.2 Automatic Mode

The automatic mode of the control system is to maintain constant reactor exit steam conditions automatically. The control system consists of two controllers. One is the temperature controller which takes the temperature error signal and applies basic proportional plus rate control. The temperature error signal

is the difference between a temperature setpoint signal and the output of the temperature sensor. The output of the temperature controller is added to the pressure error signal and results in a temperature compensated pressure error signal. The pressure error signal is the difference between a pressure setpoint signal and the output of the pressure sensor. The compensated pressure error signal is fed to a pressure controller, consisting of proportional plus rate plus reset control. The output of this controller is the so called steam demand signal which is fed to the actuators of the turbine inlet valves. The control system thus has two separate setpoints, one for exit steam pressure and one for exit steam temperature. These setpoints can be individually changed. The experiments in this part give the response of the reactor to changes in these setpoints.

1.5.2.1 Pressure Setpoint Changes: Figures 1-13 and 1-14

The change-in-setpoint experiments had the following inputs: a) step changes in the pressure setpoint of + 5 psi and + 20 psi...b) step change in the temperature setpoint of + 5.5 F. For both the 5 psi and the 20 psi change, the control system proved to be stable and limited the transients to within acceptable design limits. Also, the possibility of changing the power level to a small degree by means of the exit pressure

Figure 1-13...REACTOR RESPONSE TO A STEP CHANGE IN P_2 SET POINT. $P_2 + T_2$ CONTROL. $(n)_{iV} = 10\%$, $(W_V)_{iV} = 100\%$, $(P_2)_{iV} = 555$ psia, $\Delta P_2 = 5$ psi; Dwg. 43-024-931

Figure 1-14...REACTOR RESPONSE TO A STEP CHANGE IN P_2 SET POINT, $P_2 = 20$ psi;
Dwg. 43-025-494

was shown to be feasible for the Pathfinder Atomic Power Plant.

1.5.2.2 Temperature Setpoint Change: Figure 1-15

The experiment that examined the response of the system to a change in the temperature setpoint, was run at two initial power levels, 100 and 80 per cent. The negative change in temperature setpoint generates a demand for greater steam capacity. This causes the valves to open in order to satisfy the increased demand. The action of the valves had the following effects: 1) dropped the pressure which causes a negative insertion of reactivity and, therefore, caused the power level to drop... and, 2) increased the steam flow which caused the superheater temperature to drop. Reduction in pressure provided additional control to help limit the transient. The case of increasing the temperature setpoint causes the opposite effects. In both of the cases the control system proved stable and returned the system to steady-state conditions.

1.6 Conclusion

All the experiments presented and discussed in this section simulated the actual reactor system conditions as closely as possible. Therefore, decisions concerning both reactor design and reactor operation were

Figure 1-15...PATHFINDER RESPONSE TO STEP CHANGE OF T_2 SET POINT IN CONTROL SYSTEM: (Exp. 62-10); Dwg. 43-025-062

made based on the information presented in this section. The information gathered from these experiments shows that the pressure control system of the Pathfinder Atomic Power Plant should meet the operational requirements that will be demanded of it.

2.0 SECTION TWO - ACCIDENT STUDIES

2.1 Introduction

The purpose of the experiments described in this section is to verify that the protection system for the Pathfinder Atomic Power Plant will maintain the integrity of the system under extreme conditions.

Information leading to the proper choice of protection action, whether scram or runback, was obtained from some of these studies. The initiating cause of all the accidents was assumed to be either equipment malfunction or operator error.

2.2 Withdrawal of the Most Valuable Rod: Figure 2-1

The physical basis of this accident is that the control rod OPERATE switch becomes stuck in the withdrawal position while a rod is being withdrawn. The control rod being withdrawn is assumed to have the maximum reactivity worth of any rod with the reactor at operating temperature. The protection scheme was designed to have runback occur at 110 per cent power and scram at 115 per cent power. The initial power level was 100 per cent. The rate of reactivity insertion was 5 cents per second. The results of the experiment showed that the runback mechanism held the power level to 110 per cent while the rod was being withdrawn. Physically, the stuck switch is continuously withdrawing the control rod, while the runback mechanism is inserting the other control rods to limit the power excursion to 110 per cent of full power. During the entire time of the transient

Figure 2-1...REACTOR RESPONSE DUE TO RUNBACK CAUSED BY POWER EXCEEDING 110%. INITIAL DISTURBANCE WAS CONTROL ROD WITHDRAWAL; (Exp. 63-02); Dwg. 43-025-414

the maximum rise in the superheater hot-spot temperature was 60 F. This temperature peaked early in the transient and then dropped off when the steam flow began to catch up with the power level. At no time did the power level exceed the 110 per cent limit for runback. The experiment demonstrated that this accident would be contained by the protection system until further corrective action could be taken by the responsible plant personnel.

2.3 Recirculation Pump Trip: Figures 2-2, 2-3, and 2-4

The trip of a recirculation pump will occur automatically if pump motor current becomes excessive. These conditions will cause the pump circuit breakers to open (trip) which, in turn, will cut off the electrical power to the pump motor. The safety system will initiate controlled shutdown (this amounts to automatically inserting all rods until they have reached the bottom of the core) in the event that all three of the recirculation pumps are tripped out. The immediate effect of a pump trip is, of course, an immediate decrease in the recirculation flow to a lower steady-state flow, causing the reactor power to decrease and level out at a lower steady-state power. The area of concern is, therefore, not a nuclear excursion but excessive superheater temperatures and excessive boiler heat fluxes. The experiments for investigating this accident utilized the simultaneous trip of one, two and three pumps as inputs. Although under actual circumstances, the three pump trip would initiate controlled shutdown, the experiment was performed without that protection. Figures 2-2, 2-3, and 2-4 are

Figure 2-2...PATHFINDER RECIRCULATION PUMP TRIPOUT; (Exp. 62-17)

ONE PUMP TRIPOUT

43-025-107

Figure 2-3...PATHFINDER RECIRCULATION-PUMP TRIPOUT; (Exp. 62-17)
TWO PUMP TRIPOUT

43-025-108

Figure 2-4...PATHFINDER RECIRCULATION PUMP TRIPOUT; (Exp. 62-17)
THREE PUMP TRIPOUT

43-025-109

the results for the trip of one, two, and three pumps, respectively. In all three cases, the superheater temperature transients did not exceed the normal operating temperature, nor did excessive boiler heat fluxes develop. In the case of the three pump trip, the recirculation flow rate decreased to about 12 per cent of rated flow, the calculated maximum natural circulation for Pathfinder. This calculation is described in Section 1.2.1.6 of Report No. ACNP-62017.

2.4 Feedwater System Accidents

2.4.1 Loss of Feedwater Heating - Figures 2-5 and 2-6

The feedwater system consists of four heat exchangers that are connected in series. Three of these are low pressure heaters (Numbers 11, 12, 13) and the other (Number 14) is a high pressure heat exchanger. The No. 14 heat exchanger incorporates an automatic control system for controlling the feedwater final temperature. The main source of heat for the heaters is steam that is extracted from the turbine. In addition, No. 14 heater utilizes steam tapped from the main steam line for its automatic temperature control function. The exchanger that adds the greatest amount of heat is the high pressure heater No. 14. The two types of accidents investigated are: 1) The loss of heating (steam flow) to No. 14 exchanger (Fig. 2-5) and, 2) the simultaneous loss of heating to all the exchangers (Fig. 2-6). The loss of steam flow to heater No. 14 was

Figure 2-5...LOSS OF HEATING IN THE HIGH PRESSURE FEEDWATER HEATER
 (#14); $(n)_{iv} = 100\%, 50\%, 20\%$ (Exp. 62-38); Dwg. 43-025-488

Figure 2-6...PATHFINDER RESPONSE TO COMPLETE LOSS OF FEEDWATER HEATING
AT TWO POWER LEVELS (Exp. 61-27)

performed at three initial power levels: 100, 50, and 20 per cent. Although no scram action was simulated in this experiment, the 20 per cent case would have resulted in a scram because of either an excessive power level at 29 per cent power or an excessive rise in the superheater exit steam temperature. The 100 and 50 per cent cases did not initiate any corrective safety action. In each case the superheater hot-spot temperature did not go high enough to cause damage to the system even with no corrective action being initiated.

The other experiment, Figure 2.6, which studied the total loss of heating to the feedwater heat exchangers, showed that reactor power exceeds the scram level in both cases and would actually be shut down by a scram at 115 per cent of the selected range. The experiment was performed without any protective action to show results of the accident with no safety action.

2.4.2 Increasing the Feedwater Flow Rate: Figure 2-7

The subject of increases in the feedwater flow rate was previously discussed in Section One (Operational Studies). It is discussed again in this section because of the nature of the cause that precipitated the increase. In this case, the increase was equivalent to manual opening of the feedwater regulator valves at their maximum rate until the full capacity of the system is reached. This disturbance is considered an operator error and,

Figure 2-7...REACTOR RESPONSE TO INCREASING FEEDWATER FLOW RATE AT MAXIMUM RATE MANUALLY

Scram when $T_2 \approx 775F$
 a. Initial Power = 100%
 b. Initial Power = 20%

therefore, an accident. The results from this investigation further verify the conclusion reached in Section One, i.e., that the severity of the transient is a direct function of the amount of flow increase. The amount of flow increase depends, of course, upon the initial flow rate. The system has been designed to prevent the operator from inserting excessive feedwater into the reactor while it is critical.

2.5 Turbine Trip: Figure 2-8

Turbine trip will occur whenever the turbine speed exceeds 110 per cent rated and will initiate three distinct actions; 1) the turbine stop valves will close, stopping all steam flow to the turbine, 2) the regulating dump valve will open rapidly to the previous inlet valve capacity in order to maintain steam flow, and 3) the reactor will scram. The latter two actions are interlocked with the turbine stop valves to hold reactor pressure and superheater temperature within safe limits.

In the case of turbine trip, a problem may arise resulting from the time delay between the complete closure of the turbine stop valves and the complete opening of the regulating dump valve. If the valve characteristics and dynamics of these two valves are not perfectly matched, a transient in pressure and steam flow will occur. In addition, there is a delay between the action of the stop valves and the effect of the scram on reactor power. The time delay between the action of the stop valves and dump valve will result in a momentary

SUPERHEATER FUEL HOT SPOT TEMPERATURE

Figure 2-8...PATHFINDER TURBINE TRIP AT VARIOUS INITIAL POWER LEVELS (Exp. 62-14); Dwg. 43-025-104

decrease in steam extraction and will decrease steam flow, resulting in increasing temperatures in the superheater. The experiment was run at initial power levels of 50, 80, and 100 per cent. The results show that in each case the superheater fuel temperature did not exceed 1320 F.

2.6 Stuck Regulating Dump Valve: Figures 2-9, 2-10, and 2-11

This experiment was run to determine if the Pathfinder protection system is capable of maintaining the integrity of the reactor under the severe condition of total stoppage of steam flow. In this accident the no-steam-flow condition results from closing the turbine inlet valves with the regulating dump valve stuck in the closed position. This would result, as previously explained, in increasing superheater temperatures and increasing system pressure. The reactor main steam line is equipped with four safety valves which will pop open whenever their individual pressure setpoint is exceeded. In addition to the mechanical popping mechanism, one of the safety valves (#4S.V.) is driven open whenever both the turbine inlet and regulating dump valves are simultaneously closed. Besides this protection, a reactor scram is initiated whenever a safety valve pops and/or the main steam line pressure exceeds 565 psig.

Experiments were designed to investigate various combinations of these protective actions for use in the event of total loss of steam flow. The first experiments (Fig. 2-9) included the high pressure scram

43-025-473

Figure 2-9...REACTOR RESPONSE TO CLOSING OF THE TURBINE INLET VALVE WITH A STUCK DUMP VALVE. SCRAM INITIATED BY 25 PSIA RISE IN TURBINE INLET PRESSURE WITH DUMP AND INLET VALVES CLOSED. (Exp. 62-13)

43-025-355

Figure 2-10...REACTOR RESPONSE TO STUCK DUMP VALVE (S_b), SCRAM AT $n \geq 115\%$ WITH INLET VALVES CLOSED; (Exp. 61-41)

Figure 2-11... REACTOR RESPONSE TO STUCK DUMP VALVE (S_b) SCRAM AT $n > 115\%$ WITH INLET VALVE CLOSED (Exp. 61-41)

43-025-350

coupled with the one safety valve being driven open. This experiment was performed with initial power levels of 100, 50, and 15 per cent.(but only the 50 per cent case is presented). The greatest superheater hot-spot temperature was experienced in the 50 per cent case. The curve of superheater hot-spot temperature shows the temperature rising to a peak of 1600 F. The steam flow curve demonstrates the action of the safety valve in maintaining steam flow throughout the transient.

The next experiment (Figures 2-10 and 2-11) compared various safety valve actions. The curve designations stand for the following protection schemes: 1) #4SV ONLY - This symbol designates the #4 safety valve being driven open when both the turbine inlet and regulating dump valves are closed at the same time. 2) W_{SV} ONLY - This designation pertains to (a maximum of) four of the safety valves popping open when their individual setpoints are exceeded. 3) W_{SV} and #4SV - This symbol designates the popping action of the four valves in addition to the fourth valve being driven open. The results show that the combined action (W_{SV} and #4SV) provided the best protection. This was expected because the combined action will result in maintaining the greatest amount of steam flow, giving lower superheater temperatures.

2.7 Changes in the P₂ Setpoint: Figure 2-12 and 2-13

Changes in the P₂ setpoint are discussed in two sections of this report (in Section One under Pressure and Temperature Control System

Figure 2-12...DRIFT OF P_2 SET POINT WITH SAFETY VALVES, $\dot{P}_2$ SET = 2 psi/sec.
LIMITED AT 638 psia. Dwg. 43-024-578

Figure 2-13...DRIFT OF P_2 SET POINT WITH SAFETY VALVES $\dot{P}_2$ (SET) = 2 psi/sec.
 LIMITED AT 638 psia, Dwg. 43-024-577

and in this Section) because of the difference in how the setpoint is varied. The setpoint was changed as a part of the operational procedure for the experiments in Section One, whereas, in this section the change is of an accidental nature.

One type of change that will precipitate an accident is an upward drift of the setpoint. A rate of 2 psi/sec was chosen to simulate a malfunction of the setpoint controller. Also the drift was continued until 638 psia was reached, thereafter it remained constant. The protection used was the popping of the safety valves; the results (Fig. 2-12) show that the popping of the safety valves was sufficient to maintain the integrity of the reactor. In the actual case, there will be the added protection of scram signals generated when each safety valve opens and a high pressure scram at 565 psig.

A sudden setpoint change was also investigated. This was a 50 psi increase occurring in 1 second. The experiment was performed with two superheater control rod configurations. They are: 1) The superheater control rods being full in and... 2) the superheater control rods full out. For each case the protective action was a high main steam line temperature ($T_2 > 775$ F) scram. The results of this experiment are not presented but are discussed in the following paragraph.

The case of rods full out demonstrated the most severe transient. This conclusion was made by comparing the superheater hot-spot

temperature responses. For the case of superheater rods full out, the maximum rise was 280 F; whereas, for the superheater rods full in, the maximum rise was held to 140 F. If the entire protection system is assumed to operate, the 280 F temperature rise would have been reduced to approximately a 150 F rise since a high neutron flux scram would have occurred previous to the high steam temperature scram. This experiment demonstrated that the system is adequately protected against accidents of this type.

2.8 Reactor Isolation: Figures 2-14 and 2-15

Reactor isolation is a maneuver during which the steam flow through the superheater and turbine is reduced to a lower value. (However, temperatures in the superheater will rise to high levels if steam flow is not maintained.) The mechanism of reactor isolation is the simultaneous closing of the main steam isolation and main steam isolation bypass valves. These valves are located upstream from the turbine inlet and regulating dump valves. After these valves close completely, steam flow is maintained by opening the condensate return stop valve. This valve dumps the steam into the emergency shutdown condenser which condenses steam generated in the reactor by the decay heat. The closing time for the main steam line valves is different for each valve. The main steam isolation valve takes 17 seconds to close, while the main steam isolation bypass valve requires two minutes. In this analysis, all the valves start closing

Figure 2-14...REACTOR ISOLATION WITH SCRAM (MAIN STEAM SCRAM)
 MAXIMUM $\gamma + \beta$ HEAT
 SCRAM REACTIVITY = $- \$10$ IN 2 SECONDS
 EMERGENCY CONDENSER OPERATES

Figure 2-15...REACTOR ISOLATION WITH SCRAM MAIN STEAM LINE (Dwg. 43- 024-382)
 MAXIMUM $\gamma + \beta$ HEAT
 SCRAM REACTIVITY = $- \$10$ IN 2 SECONDS
 EMERGENCY CONDENSER OPERATES

at the same time. Any one of the following conditions can initiate this isolating mechanism:

1. Loss of both of the condenser circulating water pumps.
2. High radioactivity, as monitored, at the turbine building exhaust vent.
3. High radioactivity in the main steam line.
4. High radioactivity in the condenser off-gas.
5. Manual scram signal from the control room.

The experiments designed to study the response of the system to isolation have scram protection included. Also included was the maximum possible amount of decay heating (6 per cent of full power at time zero) to simulate conditions of maximum severity. The initial conditions for which the experiment was performed are 100 per cent power level with all variables at their rated values. The results of the experiment show that the reactor would be fully shut down without any rise in the superheater hot-spot temperatures over its rated value of 1270 F. The plot of steam flow vs. time consists of two curves. They represent: 1) the total steam flow (W_S) and, 2) the portion of the total steam flow that is being dumped into the emergency condenser (W_C). The amount of steam flowing through the main steam isolation and main isolation bypass valves (through the steam line) is therefore the difference between the two curves. As expected, the flow in the main steam line ($W_S - W_C$) went to zero in two minutes which is the time required to fully close the main steam isolation

bypass valve and thereby complete the isolation.

2.9 Reactor Startup Accidents: Figures 2-16 and 2-17

The two accidents considered in this section can be specifically considered as startup accidents. The first accident is starting up the reactor using only two of the three recirculation pumps and then, upon reaching operating temperature, placing the idle pump into operation. The second accident is caused by mistakenly withdrawing a control rod with the core flooded at a low power level.

The first accident is commonly classified as a cold water accident because the water in the idle recirculation leg, upon reaching operating temperature, will be at a temperature of 126.8 F and heating at a rate of 8.7 F/hr. The heating effect is provided by the reverse leakage in the idle recirculation leg. When the idle pump is started, the cold water will increase subcooling at the inlet of the core for a period of time equal to the time required to completely empty the leg of cold water. The time required to empty the cold leg is 24.8 seconds. The "leg" is defined as the portion of the recirculation loop that extends from the outlet of the downcomer to the inlet of the lower plenum.

This experiment (Fig. 2-16) simulated the changing recirculation flow rate and inlet subcooling according to the calculations for the actual conditions. The experiment was performed at an initial power level of 20 per cent, with scram on high steam temperature. The

Figure 2-16...TWO TO THREE PUMP OPERATION WITH COLD WATER IN IDLE RECIRCULATION LOOP. SCRAM INITIATED BY 25F RISE IN STEAM TEMPERATURE;
(Exp. 63-05)

43-025-471

43-024-511

Figure 2-17...REACTOR STARTUP ACCIDENT

$\Delta K = 5\%$ /second for 10 seconds - Total $\Delta K = 50\%$

Case 1. No scram

Case 2. Scram at 11.5% full power

resulting temperature rise at the superheater hot-spot was nearly 400 F. This temperature rise was considered unacceptable so the experiment was rerun with scram at 29 per cent power. The results of the rerun showed that the temperature transient was reduced but still unacceptable. The reason for the difference in the temperature transients of the two runs is the existence of a time lag between the superheater exit and the thermocouple that measures the steam temperature. This time lag does not exist in the second case. This time lag is especially important on low power levels because it is a function of the steam flow rate. For this work, the variation with the steam flow rate was $W_s^{-.8}$ with a time lag of 2 seconds at 100 per cent steam flow.

Since the results of both runs were unacceptable, it was decided to interlock the recirculation pumps so that they cannot be started while the reactor is critical. This will prevent the cold water accident from occurring even though a cold leg condition may exist.

A second startup accident experiment (Fig. 2-17) was performed both with and without scram protection. The accident condition is a control rod withdrawal at a rate of 5 cents/sec. for 10 seconds, for a total reactivity insertion of 50 cents. The scram protection was a power scram which was initiated at 11.5 per cent power (115 per cent of the selected range in Channels 5 and 6). In the case without scram one safety valve alternately popped and released

but did not affect the power vs. time curve. The "no-scam" and "scram" cases are presented as Case 1 and Case 2 in Figure 2-17.

2.10 Conclusion

The purpose of this section on accidents was to demonstrate that the Pathfinder protection system will safeguard the integrity of the system against credible excursions, whether nuclear or otherwise. The results presented clearly indicate that the system is well safeguarded. Whenever an experiment was performed without either runback or scram included as part of the protection scheme, the control system was able to adequately contain the accident. In addition, due to the wide range of circumstances and conditions over which the accidents were performed, it is possible to study variations of these accidents using the information given above.

3.0 SECTION THREE - PARAMETER STUDIES

3.1 Introduction

The information presented in this section shows the effect of variations in system parameters on the response of the reactor. The parameters studied on the simulator fall into two categories; first, physical constants, such as void coefficient of reactivity; and second, calculated design parameters, such as the pressure drop across the superheater. The reason for this investigation was to verify that uncertainties which exist in most design and physical parameters will not adversely affect the dynamic response of the Pathfinder Plant. The knowledge obtained from the experiments performed in this section will be used in comparison of the design calculations with the results of the operational test program.

3.2 Variation in the Void Coefficient of Reactivity: Figures 3-1 and 3-2

The purpose of this investigation was to determine the effect of a variation in the void coefficient of reactivity. This was accomplished by conducting rod withdrawal experiments with various values of the void coefficient of reactivity. Besides a variation in the void coefficient, the initial power level was also varied. In all the runs, the total reactivity inserted was 50 cents added at a linear rate of 5 cents/sec. The initial power levels studied were 100 and 20 per cent. The variation in the void coefficient of reactivity was 1/2 and 2 times the rated values. The term "rated value" corresponds

43-025-475

Figure 3-1...REACTOR RESPONSE TO ROD WITHDRAWAL WITH A VARIATION IN THE REACTIVITY IN VOIDS; (Exp. 62-35)

— $K_{\text{VOIDS}} = 2$ (RATED)
 - - - $K_{\text{VOIDS}} = 1/2$ (RATED)
 Rated = 19.24% Ave. Void Fraction

Figure 3-2...REACTOR RESPONSE TO ROD WITHDRAWAL WITH A VARIATION IN THE REACTIVITY IN VOIDS;

(Exp. 62-35); Dwg. 43-025-474

———— $K_{VOIDS} = 2$ (RATED)
 - - - - $K_{VOIDS} = 1/2$ (RATED)

to the calculated value for the Pathfinder reactor, which is 19.2 cents/per cent change in average core volume void fraction. This is stated in ACNP-62001. The results of the experiment were as expected, the larger the value of the void coefficient of reactivity (larger feedback gain), the smaller the transient for the same initial power. The reason for this is that the greater the void coefficient of reactivity the smaller the amount of voids that are necessary to offset the positive reactivity inserted by the withdrawal of the control rods. This will always be the case for the Pathfinder reactor because it has been designed to have a negative void coefficient of reactivity at all reactor power levels. Comparison of the temperature curves show that the magnitude of the transients become more severe with a decrease in the initial power level. The reason is that in order to achieve a certain increase in steam flow, the per cent change in power level is inversely proportional to the initial power level.

3.3 Variation in Superheater Fuel Heat Capacity: Figure 3-3

The Pathfinder superheater is examined by calculating the fuel hot-spot temperature. This experiment studied the effect on the hot-spot temperature of a variation in the physical parameter fuel heat capacity. The experiment was designed to study the transients resulting from control rod withdrawal. The control rod was withdrawn for 10 seconds at a rate of 5 cents/sec, resulting in a total insertion of reactivity of 50 cents. The experiment consisted of

Figure 3-3...EFFECT OF PATHFINDER FUEL HEAT CAPACITY VARIATION
 RATED $C_h = .0985 \text{ BTU/ft}^2 \text{ } ^{\circ}F$

INPUT: CONTROL ROD WITHDRAWAL AT 5¢/SECOND FOR 10 SECONDS;
 (Exp. 62-05)

three runs, each one with a different value for the fuel heat capacity. The values corresponded to 1/2, 1, and 2 times the rated value. The rated value ($C_p = 0.0985 \text{ Btu/ft}^2\text{-F}$) is the value used in the existing design of the Pathfinder reactor. All the runs were conducted at an initial power level of 100 per cent. For relatively slow transients such as the one shown in Figure 3-3 the variation in fuel heat capacity has little effect on the temperature transients. For very fast changes, however, effect of higher superheater fuel capacity is much more noticeable. The higher capacity will cause a slower fuel temperature response to reactor power changes and, therefore, lower temperature transients.

3.4 Variation in the Superheater Power Fraction: Figures 3-4 and 3-5

The superheater power fraction is the ratio of the power generated in the superheater to the total reactor power. The total reactor power is the power generated in the superheater plus boiler regions. In steady-state operation (constant subcooling), the amount of steam flow through the superheater is determined by the amount of power generated in the boiler. The steam temperature rise or amount of superheat added to the steam by the superheater depends upon the power generated in the superheater and the amount of steam flow. This, in turn, means that operating temperatures in the superheater are subject to the ratio of the superheater power to total power (superheater power fraction).

Figure 3-4...PATHFINDER SUPERHEATER POWER FRACTION VARIATION, BOILER POWER CONSTANT; (Exp. 62-06) ΔK_{rods} @ 5¢/sec for 10 seconds

43-025-105

43-025-091

Figure 3-5...PATHFINDER SUPERHEATER POWER FRACTION ($\bar{p}$) VARIATION TOTAL
 POWER CONSTANT, ΔK_{rods} @ 5¢/sec for 10 seconds (Exp. 62-04)

In order to study the effect of superheater power fraction, two sets of experiments were performed, which can be described by the following sets of conditions:

- 1) Variable superheater power fraction with constant boiler power (Fig. 3-4).
- 2) Variable superheater power fraction with constant total power (Fig. 3-5).

The case of the constant boiler power is, therefore, a constant steam flow rate condition. The input for all of the experiments was the insertion of reactivity in the form of control rod withdrawal with reactivity being inserted at the rate of 5 cents/sec for 10 seconds. It was confirmed that the initial or steady-state operating temperature is very sensitive to variations in the power split between superheater and boiler. This fact is seen from the inspection of the temperature curves in Figures 3-4 and 3-5. Additionally, in the case of constant boiler power, the rise in power level was not extensively affected by the variation in the superheater power fraction. Likewise, the steam flow rate also remained relatively unaffected.

3.5 Variation in the Superheater Pressure Drop: Figure 3-6

The superheater pressure drop has an important effect on the control of the reactor because it exists between the reactor dome and the point at which the pressure is being controlled, i.e. the exit of the superheater. Control of exit pressure is necessary in order to

Figure 3-6...EFFECT OF SUPERHEATER PRESSURE DROP ON SYSTEM
RESPONSE TO A 10-SECOND CONTROL ROD WITHDRAWAL;
Dwg. 43-024-495

prevent the safety valves from operating during normal system maneuvers. Since the pressure at the exit of the superheater is maintained at a constant value, the reactor dome pressure will be determined by the pressure drop through the superheater. The reactor dome pressure directly affects the response of the reactor because of the positive pressure coefficient of reactivity. For the Pathfinder system, the analysis showed that variations in superheater pressure drop will result in variations in the dynamic response of the reactor. The input for all cases of the experiment was again the 10 second withdrawal of a control rod at 5 cents/sec. The nominal pressure drops used were 70 psi and 10 psi at rated conditions $(\eta)_{iV} = 100$ per cent, $(W_S)_{iV} = 100$ per cent. The experiment shows that transients become more severe as the pressure drop increased.

For the analysis of the transient effect, it must be understood that the pressure drop through the superheater is approximately a direct function of the square of the steam flow rate. Because the superheater exit pressure is held constant, the reactor dome pressure will vary with the steam flow rate. It should be noticed that the rate of increase in the steam flow is greater for the low pressure drop case. This follows since the superheater is a resistance to flow between the steam dome and the turbine. The smaller this resistance, the more closely will steam flow follow power, thus giving smaller temperature transients.

3.6 Variation in the T_2 Measuring Time Lag (W_s): Figure 3-7

T_2 is the bulk temperature of the superheated steam near the superheater exit and is measured at that point by a thermocouple. The exact location of the thermocouple is 42 feet downstream from the exit of the superheater. Therefore, whenever a temperature transient occurs in the superheater, a transport time lag plus a thermocouple response time lag will cause the measured temperature to lag the actual exit steam temperature. Thus, the magnitude of this total time lag (termed the T_2 Measuring Time Lag) is a function of the steam velocity and the distance the thermocouple is removed from the point of the transient. Since the distance remains a constant, the lag will only be a function of the velocity of the steam. The steam velocity is a function of the steam flow (which is a function of the power level). Therefore, the time lag (T_2 measuring time lag) will vary with reactor power level. The transport time lag at a constant power level was varied. The extent to which this variation will effect the response of the reactor (through the pressure and temperature control system) was investigated by experiments performed with two different nominal time lags, each with two different input disturbances. The inputs were: 1) two pump dropout, and 2) control rod withdrawal. The two nominal time lags used in the experiments were 0.5 and 1.0 seconds. The results of the experiments are presented in terms of the response of the superheater hot-spot temperature. The control rod withdrawal rate was 5 cents/sec for a period of

Figure 3-7...PATHFINDER SUPERHEATER FUEL TEMPERATURE RESPONSE TO VARYING T_2 MEASURING LAGS; (Exp. 62-16)

43-025-094

10 seconds. The results show that the control of the reactor will be relatively unchanged by variations in the time lag of this magnitude . Greater variations in the time lags would be considerably more noticeable.

3.7 Variation of the Kinetics Parameter β/l^*

The β/l^* parameter arises from the solution of the neutron kinetic equations. This parameter is made up of the two physical constants, total delayed neutron fraction (β) and effective neutron lifetime (l^*). This experiment studies the effect by varying the kinetic parameter β/l^* . The experiment showed that the transient response of the system would be virtually unchanged by a ± 50 per cent deviation in the effective neutron lifetime. Therefore, it is believed that the lifetime calculation has the necessary accuracy and that a variation in this parameter will not adversely affect the operation of the system. See Section Four for a discussion of the effect on reactor stability of various values of β/l^* .

3.8 Conclusion

The purpose of the work carried out for this section, as stated in the introduction, was verification that reasonable uncertainties in a parameter will not adversely affect the Pathfinder Atomic Power Plant. Besides fulfilling this purpose, such design criteria as the Superheater Measuring Time Lag were established. The accumulation of information for comparison with data from the Pathfinder post-

construction test program, to be conducted, should prove to be valuable. A deeper insight into the operation and response of the system was also obtained through these experiments.

4.0 SECTION FOUR - STABILITY STUDIES

4.1 Introduction

The purpose of this section is to present and discuss the results of the stability experiments which were run on the Pathfinder Transient Simulator. The experiments were conducted to determine if the entire Pathfinder system is stable. This information was taken in the form of the frequency response of the system. From this study, the maximum possible stable power level was obtained. Also, the technique that will be used to analyze the data received during the experimental testing of the Pathfinder plant was studied. This was done in order to verify the technique.

4.2 Transfer Function: Figures 4-1, 4-2, 4-3 and 4-4

The experiments discussed in this section give the frequency response of the zero-power, open-loop, and closed-loop transfer functions.

The method of obtaining the frequency response data was as follows: The sinusoidal input was provided by a variable amplitude frequency generator. The output for the zero-power and closed-loop transfer functions was the neutron density $n(s)$. The output for the open-loop was the feedback reactivity. In each case, the input and output were fed to the X and Y axes of an oscilloscope and from the resulting Lissajous figure the phase and gain of the transfer functions were calculated.

— $\lambda = 10^{-4}$ SEC. - SCHULTZ, CONTROL OF
 NUCLEAR REACTORS AND POWER PLANTS, PP 115
 □ PATHFINDER TRANSIENT SIMULATOR -
 LISSAJOUS FIG.

43-025-715

Figure 4-1...ZERO POWER FREQUENCY RESPONSE - NOTE: THE VALUE OF NEUTRON
 LIFETIME USED ON THE PATHFINDER TRANSIENT SIMULATOR IS
 $9 \cdot (10^{-5})$ sec.

43-O25-716

Figure 4-2...PATHFINDER OPEN - LOOP FREQUENCY RESPONSE-LISSAJOUS FIGURE METHOD.

—————	$(n)_{ V} = 100\%$	$(W_V)_{ V} = 100\%$
- - - - -	$(n)_{ V} = 80\%$	$(W_V)_{ V} = 100\%$
- . - . -	$(n)_{ V} = 80\%$	$(W_V)_{ V} = 77\%$

PATHFINDER TRANSIENT
SIMULATOR - LISSAJOUS FIG.
(n)_{iv} = 100% (W_v)_{iv} = 100%

GRAPHICAL TECHNIQUE

(n)_{iv} = 100% (W_v)_{iv} = 100%

(n)_{iv} = 80% (W_v)_{iv} = 100%

(n)_{iv} = 80% (W_v)_{iv} = 77%

Figure 4-3...CLOSED-LOOP FREQUENCY RESPONSE; Dwg. 43-025-717

Figure 4-4...COMPLEX PLANE PLOT OF OPEN-LOOP FREQUENCY RESPONSE

The zero-power frequency response phase and gain information (Figure 4-1) closely agreed with the literature.¹ This adequately verified the computer setup of the neutron kinetics. Also, since the computer results matched the calculated transfer function very well, the use of the computer in the frequency range .01 to 10 cps was justified.

The open-loop data was used for two purposes. First of all, a complex plane plot (Figure 4-4) was made using the data for the initial conditions of $(n)_{iV} = 100$ per cent and $(W_V)_{iV} = 100$ per cent. This plot made it possible to determine the phase and gain margins of the Pathfinder system. The gain margin was calculated to be 12.0db.² The phase margin was found to be 132 degrees. From the literature³ "A resonable degree of stability is generally considered as being provided by a phase margin of approximately 50 degrees and a gain margin varying between 10 to 15 db". Thus, it is concluded that the Pathfinder Reactor system will be very stable.

1. M. A. Schultz, Control of Nuclear Reactors and Power Plants, McGraw-Hill Publishing Co., pp. 115
2. The oscillator experiments conducted on VBWR and EBWR showed gain margins of 12.3 db and 13.0 db respectively. It should be noted that the average core void fraction in the VBWR experiment was 20 per cent; this is equal to the void fraction in Pathfinder at full power.
3. J. G. Gille, M. J. Pelegrin, P. Decauline, Feedback Control System, McGraw-Hill Publishing Co., pp. 281

The open-loop information was also plotted on a Nichols Chart. The Nichols Chart is devised to convert open-loop frequency response information to the frequency of closed-loop times feedback or vice versa. A Bode plot of the closed-loop times feedback and feedback responses was made in order to determine the closed-loop frequency response. The resulting closed-loop information along with actual simulator results are shown in Figure 4-3. The feedback transfer function frequency response was determined from a Bode plot of the zero-power and open-loop data.

The closed-loop frequency response was determined by two methods:

- 1) a graphical technique using Nichols Charts and Bode plots... and,
- 2) an actual closed-loop frequency response taken using the Pathfinder Transient Simulator. Figure 4-3 presents both sets of the closed-loop data, with the curves being determined by the graphical technique and the data points being the actual simulator response. It can be seen from inspection of Figure 4-3 that the results of the two methods are in agreement. Therefore, this graphical technique will be used to evaluate the results of the experimental frequency response of this transfer function for the Pathfinder system.

Briefly, the technique will be used in the following manner. The transfer function test will give the closed-loop frequency response. From a Bode plot of the experimental closed-loop frequency response and a model feedback transfer function frequency response, the

closed-loop times feedback response is obtained. The open-loop frequency response is obtained from a separate Bode plot. This Bode plot is made up of the experimental zero-power frequency response and the same model feedback frequency response. If through the proper use of the Nichols Chart, good agreement exists between these two newly formed frequency responses, it can be concluded that the model feedback transfer function physically exists. From the analytical form of the model feedback transfer function, it will now be possible to determine important physical parameters (i.e., power coefficient, void coefficient of reactivity, fuel element heat transfer time delay, etc). The above technique for finding the feedback transfer function is somewhat similar to that employed at EBWR (ANL-5850).

During all the above analog experiments, the inputs and outputs were also fed to the Pathfinder pile oscillator instrumentation. This instrumentation is designed to calculate the phase and gain of the experimental Pathfinder transfer functions. The phase and gain results from the pile oscillator instrumentation was compared to those calculated from the Lissajous figures. The comparison showed excellent agreement between the two sets of results. This, plus good repeatability tests, demonstrated that the Pathfinder pile oscillator instrumentation will give reliable results during the experimental testing of Pathfinder.

4.3 Conclusion

The purpose of this section was to discuss the results of the investigation into the stability of the Pathfinder system with the aid of the Pathfinder Transient Simulator. By assuming a linear relationship of gain and phase shift with power level, the maximum stable power for Pathfinder was determined to be 400 per cent of full power.

5.0 SECTION FIVE - SPECIAL SIMULATIONS AND STUDIES

5.1 Introduction

The purpose of this section is to describe the special studies that were carried out incorporating the Pathfinder Transient Simulator. Special studies are those over and above the normal requirement for a pre-operational understanding of the system. The special studies are involved with the development of new techniques of simulation and analytical attack which will result in a more precise understanding of the actual physical situation. Besides the above material, this section contains some curves and simulator circuits that were not normally incorporated as part of the Pathfinder Transient Simulator. The need for these special circuits and curves arose from an attempt to very closely simulate the actual system.

5.2 Special Superheater Study: Figure 5-1

The purpose of this special superheater study was to determine the limitations or the adequacy of the simplified Pathfinder simulation as presented in ACNP-62001. The study essentially consisted of dividing the superheater into a series of equal axial sections.

The equations used for the study were:

$$\begin{aligned} \text{Eq. 1. } C_F M_F \frac{\partial T_{SF}(z,t)}{\partial t} &= P(z,t) - (H)_{iv} \left[\frac{W_S(t)}{(W_S)_{iv}} \right]^N [T_{SF}(z,t) - T_S(z,t)], \\ \text{and Eq. 2. } C_P M_S \frac{\partial T_S(z,t)}{\partial t} &= (H)_{iv} \left[\frac{W_S(t)}{(W_S)_{iv}} \right]^N [T_{SF}(z,t) - T_S(z,t)] \\ &\quad - \frac{C_P W_S(t)}{A} \frac{\partial T_S(z,t)}{\partial z}, \end{aligned}$$

43-025-175

Figure 5-1...MULTI-SECTION SUPERHEATER; COMPARISON OF A NUMBER OF AXIAL SECTIONS FOR A SPECIAL LOOP DUMP ACCIDENT

where: $P(z,t) = \frac{P(t)}{(P)_{iv}} P(z)$

The simulation solved these equations in their perturbed form (i.e., $P = (P)_{iv} + \Delta P$). Also a set of lumped parameters was calculated for each section. The following assumptions were used in the model.

1) The axial superheater power profile ($P(z)$) remained constant during a transient.

2) The heat transfer coefficient is only a function of steam flow.

$$h = (H)_{iv} \left[\frac{W_s(t)}{(W_s)_{iv}} \right]^N ; \text{ with } N = .8,$$

3) The steam conditions at the inlet of the superheater remain constant.

4) The steam temperature in each section is equal to the exit temperature of that section.

5) The only heat transfer mechanism is forced convection.

6) The specific heat of the steam (C_p) varies linearly with steam temperature.

7) The specific heat of the fuel and cladding are the same (C_F).

The number of sections was varied in order to determine the optimum number necessary for analyzing transients. The transient investigated using this special analog simulation was a turbine load dump with stuck dump valve (ref. Section Two, Figures 2-9, 2-10, and 2-11). This is a very rapid transient.

The reason for using a rapid transient is that if a limitation exists in the simulation, it will be evident in the case of a rapid transient. The multi-section simulation has the added advantage of indicating fuel and steam temperatures at various axial positions.

The number of sections in the simulation was varied from ten to one (simplified simulation). A portion of the results of the investigation are shown in Figure 5-1. This figure shows the peak channel temperature reached in the rapid transient plotted vs. number of axial sections used in the analysis of the transient. The plot shows that with this multisection model, the results do not change with the use of more than six sections. Also, the results with a decreasing number of sections tend to move in the conservative direction. As a final check, to see that the multisection technique was giving valid results, the identical transient was run using the IBM 704 digital computer with a ten section model. The code used was ART-04. The peak temperature from the 10 section digital run compared quite favorably with that from the ten section analog simulation. The result is also shown in Figure 5-1.

In conclusion, it can be said that the purpose of undertaking the study was accomplished; it was demonstrated that the Pathfinder Transient Simulator lumped parameter one section superheater is quite adequate.

5.3 Reactor Pressurization Procedure: Figures 5-2, 5-3 and 5-4

The pressurization of the reactor by nuclear heating was investigated.

43-025-719

Figure 5-2...REACTOR PRESSURIZATION SCHEME

Figure 5-4...REACTOR PRESSURIZATION - 310 to 375 PSIA

This experiment utilized the following startup procedure: The system will be pressurized from 310 psia to 560 psia (P_2 - superheater exit pressure), while increasing the feedwater temperature from 200 F to 340 F. The entire procedure was designed to take place at a constant power of 20 per cent. Because of the constant power level requirement, it will be necessary to frequently reposition the control rods. The experiment also determined the scheme of control rod movement necessary to maintain the 20 per cent power level.

The procedure consisted of dividing the total pressure increase and the feedwater temperature increase into four steps. The feedwater temperature steps were each 35 F. The pressure steps were 65 psi, 65 psi, 60 psi and 60 psi. Each pressure step was achieved by a series of 5 psi increments. Figure 5-2 is a graphical representation of the entire pressurization procedure. This figure shows that the procedure consists of the following two independent processes. First, an increase in the feedwater temperature at a constant pressure; and second, an increase in pressure at a constant feedwater temperature.

Since the Pathfinder Transient Simulator circuit parameters were derived for operation near 560 psia, it was necessary to modify some of the parameters for a somewhat wider pressure range. The modification was achieved by special simulations of the following set of equations:

Doppler Reactivity

$$\text{EQ(1)} \quad \Delta K_D = \frac{dK_D}{dP_1} (\Delta P_1) + \frac{dK_D}{dT_{AV}} \left(\frac{1}{C} \right) \int_0^t \left[(1-\gamma)(\dot{Q}_g)_{iv} \left(\frac{n}{(n)_{iv}} \right) - \left(\frac{\dot{Q}_B - \gamma(\dot{Q}_g)_{iv} n}{(n)_{iv}} \right) \right] dt,$$

Void Reactivity

$$\text{EQ(2)} \quad \Delta K_V = \frac{dK_V}{d\bar{\alpha}} \left(\frac{d\bar{\alpha}}{d\bar{x}} \Delta \bar{x} - \frac{d\bar{\alpha}}{dP_1} \Delta P_1 \right),$$

Steam Line Pressure Drop

$$\text{EQ(3)} \quad \delta \Delta P_{SL} = \frac{dP_{SL}}{dW_S} \Delta W_S - \frac{dP_{SL}}{dP_2} \Delta P_2,$$

Superheater Pressure Drop

$$\text{EQ(4)} \quad \delta \Delta P_{SH} = \frac{dP_{SH}}{dW_S} \Delta W_S - \frac{dP_{SH}}{dP_2} \Delta P_2,$$

The results of the experiment were as expected. After each increase in feedwater temperature, the resulting power level was < 20 per cent, and after each increase in pressure the resulting power level was > 20 per cent. The reason for the decrease in power level is that raising the feedwater temperature decreases subcooling which increases the void content of the core. Since the reactor has a negative void coefficient of reactivity, an increase in void content inserts negative reactivity. The reason for the increase in power level is that increasing the pressure decreases the void content of the core which inserts positive reactivity. Figures 5-3 and 5-4 are the results of increasing the feedwater temperature from 200 F to 235 F, followed by

the increase in pressure from 310 psia to 375 psia (in a series of 5 psi steps). These are the processes (1-2) and (2-3) as shown on Figure 5-2.

During all of the transients the highest power level experienced was 23 per cent. This occurred during process (8-9) of Figure 5-2. The superheater fuel temperature had a maximum transient rise of 40 F. Every transient during the entire experiment was identical to those shown in Figures 5-3 and 5-4. These results indicate that this pressurization procedure can be safely utilized.

5.4 Special Circuits and Curves: Figures 5-5, 5-6, 5-7, 5-8 and 5-9

The control system for Pathfinder was previously discussed in Section One. The analog schematic of the system will be presented here. The schematic, Figure 5-5, shows the system being made up of two controller units (A) and (C). Controller unit (A) takes the temperature error signal (ΔT_2) and applies proportional plus rate control action. The temperature error signal is the deviation of the superheater exit temperature from a setpoint value. The output of controller unit (A) is used to compensate the pressure error signal (ΔP_2). The pressure error signal is similar to the temperature error signal in that it is the deviation of the superheater exit pressure from a setpoint value. The compensated pressure error signal (output of Summing Unit (B)) is fed to controller unit (C). Controller unit (C) applies proportional plus reset plus rate action.

43-025-409

Figure 5-5:..ANALOG SCHEMATIC OF THE PRESSURE CONTROL SYSTEM

43-025-099

Figure 5-6...INTEGRAL ROD WORTH CURVE MEASURED WITH MOCKED-UP OPERATING
VOIDS IN CRITICAL FACILITY

Figure 5-7...PATHFINDER TURBINE TRIP CAPACITY VS. TIME;
Dwg. 43-025-106

43-025-171

Figure 5-8...PATHFINDER RECIRCULATION PUMP COAST DOWN

43-024-509

Figure 5-9... SIMULATION OF PATHFINDER SAFETY VALVES

The output is the command signal which moves the regulating dump valve and/or the turbine stop valves so that superheater exit setpoint conditions are maintained. In summary, the Pathfinder control variables are temperature and pressure and the controlled variable is main steam line flow rate (W_s).

In order to achieve the maximum degree of analogy between the Pathfinder Transient Simulator and the Pathfinder system, it was necessary to simulate information taken from experiment and equipment performance data. Therefore, it was necessary to simulate the following curves:

- a) Integral Rod Worth - Figure 5-6. The information contained on this curve was obtained from a series of physics experiments, conducted on the Pathfinder Critical Facility, with the Pathfinder core mocked-up, including void content, at various operating levels.
- b) Pathfinder Turbine Trip Capacity - Figure 5-7. This curve has the information necessary to determine the steam flow to the turbine after a turbine trip has occurred. The steam flow is cut off by closing the turbine stop valves whenever a turbine trip occurs.
- c) Pathfinder Recirculation Flow Coastdown - Figure 5-8. This figure shows the calculated flow coastdown at any time after any number of the recirculation pumps have been shut down.
- d) Simulation of Pathfinder Safety Valves - Figure 5-9. This figure is the operating scheme of the safety valves in terms

of the steam flow rate through the valve or valves and the main steam line pressure.

5.5 Conclusion

The purpose of this section was to present information gathered from the special studies which were conducted with the aid of the Pathfinder Transient Simulator. It was also the purpose of this section to present special circuits and curves. The ones presented are those used most frequently. There were others, not presented, which were used in conjunction with a single experiment; these are discussed as part of the individual experiment.

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