JUN 20 1967

ACNP-67514

MASTER

PATHFINDER ATOMIC POWER PLANT

RESPONSE TO SCRAM AND RUNBACK TESTS (431)

Submitted to
U. S. ATOMIC ENERGY COMMISSION
NORTHERN STATES POWER COMPANY
and
CENTRAL UTILITIES ATOMIC POWER ASSOCIATES
by
ALLIS-CHALMERS MANUFACTURING COMPANY
ATOMIC ENERGY DIVISION
Bethesda, Maryland

Ref: AEC Contract No. AT(11-1)-589

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HQ \$3.00; MN-65

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by T. G. Shultz and J. T. Stone

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NORTHERN STATES POWER COMPANY

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by

ALLIS-CHALMERS MANUFACTURING COMPANY

under

Agreement dated 2nd Day of May 1957, as Amended
between

Allis-Chalmers Manufacturing Company and Northern States Power Company
under

AEC Contract No. AT(11-1)-589

April 1967

Classification - UNCLASSIFIED

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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, this particular report deals with the Response to Scram and Runback (Test 431). The Pathfinder Plant is located at a site near Sioux Falls, South Dakota and reached criticality early in 1964. Owners and operators of the plant will be 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 Water type with Nuclear Superheater.

ABSTRACT

During power escalation of the Pathfinder Atomic Power Plant, a test program was conducted to evaluate response of the power plant to safety actions. This test program, entitled Response to Scram and Runback (Test 431), was part of the overall Pathfinder testing program. The safety actions involved in Test 431 were: reactor scram, control rod runback, turbine trip, and turbine load dump.

This document is the final report on this phase of the testing program. The initial sections of the report give the objectives and method of the testing program. Descriptions of expected plant response to each safety action are included. Individual tests are discussed in detail. Reproductions of the recorded results of each test are included. Section 4 is devoted to an unusual occurrence which was consistently observed throughout the testing program; this occurrence consisted of an oscillation in steam line pressure following isolation scram, turbine trip, and turbine load dump. The source, characteristics, and effects of the oscillation are discussed. As a result of Test 431, it is concluded that the reactor and its associated control system respond as expected to plant safety actions.

CONTENTS

1.	<u>INTRODUCTION</u>	1
2.	<u>TEST DESCRIPTION</u>	1
2.1	SCRAM	1
2.2	RUNBACK	2
2.3	TURBINE TRIP	3
2.4	TURBINE LOAD DUMP	3
3.	<u>TEST RESULTS</u>	4
3.1	ISOLATION SCRAM FROM 38 MWT	4
3.2	SCRAM FROM 38 MWT	5
3.3	SCRAM FROM 62 MWT	5
3.4	ISOLATION SCRAM FROM 60 MWT	5
3.5	SCRAM FROM 112 MWT	6
3.6	ISOLATION SCRAM FROM 140 MWT	6
3.7	CONTROL ROD RUNBACK FROM 76 MWT	7
3.8	TURBINE TRIP FROM 38 MWT	7
3.9	TURBINE LOAD DUMP FROM 38 MWT	8
3.10	TURBINE LOAD DUMP FROM 95 MWT	8
3.11	TURBINE LOAD DUMP FROM 140 MWT	9
3.12	TURBINE LOAD DUMP FROM 160 MWT	9
3.13	TURBINE LOAD DUMP FROM 168 MWT	10
3.14	TURBINE LOAD DUMP FROM 60 MWT	11
4.	<u>STEAM LINE PRESSURE OSCILLATIONS</u>	12
4.1	SOURCE	12
4.2	CHARACTERISTICS	12
4.3	DAMPING	13
4.4	EFFECTS	13
5.	<u>SPATIAL EFFECTS OBSERVED DURING SCRAM AND RUNBACK</u>	15
5.1	SPATIAL EFFECTS DURING SCRAM	15
5.2	SPATIAL EFFECTS DURING RUNBACK	16
6.	<u>CONCLUSIONS</u>	17
6.1	REACTOR RESPONSE	17
6.2	PLANT CONTROL SYSTEM RESPONSE	17
7.	<u>FIGURES</u>	20

1. INTRODUCTION

Test 431, entitled Response to Scram and Runback, was initiated on June 20, 1966 and was completed on January 24, 1967.

The procedure for Test 431 was designed to permit determination of the response of the Pathfinder reactor to scram, runback, turbine trip, and turbine load dump at various levels during the Reactor Operation and Power Escalation test series (Test 278).

Test 431 had the following three specific purposes:

- (1) to confirm that all pertinent control systems were adjusted to respond properly and in a safe direction during a reactor scram, runback, turbine trip, and turbine load dump;
- (2) to observe reactor and turbine behavior in response to turbine trip and turbine load dump; and
- (3) to observe whether local hot spots in the superheater could result from unfavorable control rod programming configurations.

A simplified plant schematic, indicating positions in the plant where the various parameter signals were obtained, is contained herein as Fig. 17.

2. TEST DESCRIPTION

Response of the reactor and turbine-generator to each test disturbance was tested by manually initiating the disturbance so that "controlled" conditions would prevail during the test. This approach was taken, rather than permitting a process of nuclear system action which would result in a reactor scram. Some results are included which were obtained after non-intentional system disturbances resulted in automatic safety system action. The various safety actions are described in the following sections.

2.1 SCRAM

A reactor scram consists of an interruption of power to the boiler control rod drive clutches which results in rapid insertion of the boiler control rods into the core. The time required for total insertion of the boiler control rods is less than 1 sec. Coincident with a scram, the four superheater control rods are driven into the core at a rate of 1.2 in./sec.

Following a reactor scram without turbine-generator operation (i.e., when all steam is dumped directly to the main condenser), the automatic pressure control system positions the regulating dump valve in an attempt to maintain steam line pressure at 540 psig. Dump valve closure is limited (the valve will remain about 9 percent open) to prevent the control system from terminating steam flow and thereby prevent cooling of the superheater fuel.

Following a reactor scram with turbine-generator operation, the normal mode of operation is as follows. The regulating dump valve is already closed, and the automatic pressure control system starts to close the turbine governor valves to maintain steam line pressure. Five seconds after scram, the turbine stop valves are automatically tripped, thus stopping turbine steam flow and effecting fast opening of the regulating dump valve to a position corresponding to the interrupted steam flow. The pressure control system then positions the dump valve to the minimum valve position required to maintain steam line pressure.

During most of Test 431 performance, an unusual mode of pressure control was used due to the marginal performance of the turbine inlet valve positioner. Under the mode of control used, the dump valve was partially open to control pressure, and the turbine inlet valves were fixed in position by the turbine load limit servo.

An isolation scram is a reactor scram followed by reactor building isolation which results in the following simultaneous actions:

- (a) emergency condenser outlet valve is opened, allowing steam flow to the emergency condenser;
- (b) main steam isolation valve begins to close (it is fully closed within 17 sec);
- (c) main steam isolation bypass valve begins to close (it is fully closed within 120 sec); and
- (d) pressure control system is automatically transferred to HAND operation which, in effect, causes constant steam throttling downstream of the isolation valves during closure of the isolation valve.

2.2 RUNBACK

During a runback, the control rods are inserted at rates corresponding to their radial location. These rates are:

- (a) 1.2 in./sec for the eight outer boiler rods (1 min elapses from the time that the rods are fully out to the time that they are fully in);
- (b) 0.43 in./sec for the eight inner boiler rods (2.8 min full out to full in); and
- (c) no insertion of the four superheater rods.

During performance of Test 431, control rod runback was continued until all boiler control rods were fully inserted.

2.3 TURBINE TRIP

Turbine trip causes rapid closure of the turbine stop valves (i.e., closure occurs in less than 0.3 sec), a reactor scram, and rapid opening of the regulating dump valve (i.e., opening occurs in less than 0.25 sec) to a position sufficient to pass existing steam flow.

2.4 TURBINE LOAD DUMP

Turbine load dump (opening of generator circuit breaker) results in the following actions:

- (a) rapid closure of the turbine inlet valves (i.e., closure occurs in less than 0.4 sec);
- (b) rapid opening of the regulating dump valve (i.e., opening occurs in less than 0.25 sec) to a position sufficient to pass existing steam flow;
- (c) initiation of a 5-sec time delay;
- (d) re-opening of the turbine inlet valves (after 5 sec) to the THROTTLE MINIMUM STOP position set by the pressure control system (about 11 percent open);
- (e) partial closure of the regulating dump valve to compensate for turbine inlet valve opening; and
- (f) turbine speed governor control at about 102 percent of normal speed.

The purpose of turbine load dump is to keep the reactor at power and to permit the turbine generator to carry the plant load.

3. TEST RESULTS

An Offner eight-channel strip chart recorder (Type R Dynograph) was employed to record reactor system variables during the tests. Reproductions of applicable Offner traces are included to indicate effects of disturbances on plant variables. The traces are arranged in such a way that time proceeds from right to left and the increasing direction for all recorded variables is towards the top of the page, except for those marked differently. These traces are discussed in detail elsewhere in this report. To obtain maximum information, different variables to be recorded were selected before each test. This difference in the variables focused upon is reflected in the discussion by the change in emphasis for the different tests.

3.1 ISOLATION SCRAM FROM 38 MWT (FIG. 1)

The isolation scram test was conducted without turbine-generator operation.

One out-of-core and two in-core ion chamber power measurements were recorded. These indicate that reactor power decreased below 10 percent of initial power in 0.7 sec. The indicated power increase on the out-of-core ion chamber (which occurred approximately 20 sec after the scram) was caused by operator action in switching the micro-microammeter to a more sensitive range.

Steam flow through the Main Steam Bypass Isolation Valve (MSBIV) was found to be zero during the monitoring preceding the isolation scram, since nearly all of the steam was being passed by the Main Steam Isolation Valve (MSIV). Following an isolation scram, the MSIV begins to close, and it is fully closed 17 sec later. Bypass flow increases as the MSIV is closing; this is indicated by the recorded results. Indicated flow is constant during the initial 6 sec of MSIV closure because the steam flowmeter is not yet on scale, and during the final 6 sec of MSIV closure because the steam flow exceeds the full scale reading of the flowmeter. Following isolation scram, the MSBIV also begins to close, and it is fully closed in 120 sec. Indicated bypass steam flow comes on scale again after 36 sec (full scale equals 75,000 lb/hr), and it indicates zero flow after 58 sec. Therefore, steam flow to the main condenser following isolation scram was terminated (or at least reduced to less than 10,000 lb/hr) in approximately 1 min. rather than in the expected 2 min. This condition was investigated and was found to be ultra-conservative from the standpoint of superheater temperatures after shutdown (superheater temperatures were lower than predicted); therefore, adjustments to the valve closing time were deemed unnecessary.

Reactor dome pressure and steam line pressure both decrease following isolation scram. This pressure decrease is due to the decreased steam generation and to the absence of automatic steam pressure control. (The automatic pressure control system was not in service preceding the isolation scram; however, it is automatically transferred to HAND by an isolation scram signal; automatic pressure control is therefore not in service following a reactor building isolation scram.)

The oscillation in steam line pressure is discussed in Sec. 4.

3.2 SCRAM FROM 38 MWT (FIG. 2)

The scram test was conducted without turbine-generator operation.

Automatic pressure control was employed during the test but had little effect since the regulating dump valve opening was initially only 3 percent greater than the valve-minimum-position-limit setting. Automatic system action consisted of closing the dump valve from 12 percent open to 9 percent open in an attempt to maintain steam line pressure at 540 psig.

Reactor power decreased to nearly zero power in 0.6 sec.

3.3 SCRAM FROM 62 MWT (FIG. 3)

The scram test was conducted without turbine-generator operation.

A fast recorder chart speed was used. This trace illustrates the smooth decrease in reactor power as the control rods enter the core. Power decreases to less than 10 percent of initial power in 0.6 sec. The large "spike" on most recorder channels, just as scram was initiated, was due to noise pickup.

A comparison of steam line temperature with superheater fuel temperature indicates that the response time of the steam line thermocouples is about the same as that of the superheater thermocouples. This information is significant, since the superheater thermocouple responses are known to be very fast (time constant of about 0.1 sec), and the steam line thermocouples are used in the automatic safety system. Experimental evidence indicating fast response of the steam line thermocouples gives confidence in the effectiveness of the high steam line temperature scram.

3.4 ISOLATION SCRAM FROM 60 MWT (FIG. 4)

The isolation scram test was conducted without turbine-generator operation.

The isolation scram was initiated by reducing the high radiation scram set point on the air ejector radiation monitor below the existing radiation level.

Two superheater thermocouple indications were recorded. One of these, thermocouple O-10, is located 10 in. from the bottom of the core and is welded to an outer fuel tube in superheater fuel location A-18. Thermocouple I-39 is located 39 in. from the bottom of the core and is welded to an inner fuel tube in superheater fuel location E-9. The thermocouples indicated that superheater fuel temperature is essentially constant and is very near the boiler saturation temperature within 9 sec following scram.

The automatic pressure control system was in service preceding the isolation scram. Automatic transfer of the control system to HAND control occurs with an isolation scram signal; this results in constant dump valve position.

Reduction in main steam flow resulted from MSIV closure.

Indicated steam line temperature decreases to about 500 F and remains constant because 500 F is the low end of the thermocouple re-transmitter range.

The oscillation in steam line pressure is discussed in Sec. 4.

3.5 SCRAM FROM 112 MWT (FIG. 5)

The scram test was conducted with the turbine governor valves fixed in position by the turbine load limit servo at 30 percent open and with the regulating dump valve approximately 8 percent open and controlling the steam line pressure.

The Offner results provide a good comparison between axially spaced in-core ion chambers and also between axially spaced superheater thermocouples. In-core ion chambers No. 7, No. 8, and No. 9 are located respectively 54 in. from the bottom of the core, 36 in. from the bottom of the core, and 18 in. from the bottom of the core. The decay of reactor power varies, as would be expected with the control rods entering the core from the top.

The superheater thermocouples also display different drop-off times. Thermocouple I-39 (located 39 in. from core bottom) indicates vessel saturation temperature within 5 sec; thermocouple O-10 (located 10 in. from core bottom) indicates vessel saturation temperature within 7 sec; and thermocouple I-00 (located at core bottom) indicates vessel saturation temperature within 9 sec. This behavior in measured temperatures is due both to the fact that steam coolant enters the top of the superheater, and also to the change in axial power shape during scram.

The oscillation in steam line pressure which begins at the time of turbine trip (5.4 sec after scram) is discussed in Sec. 4.

3.6 ISOLATION SCRAM FROM 140 MWT (FIG. 6)

Preparations were made for an isolation scram test. The turbine governor valves were fixed in position by the turbine load limit servo; the dump valve was approximately 6 percent open and was controlling pressure. The Offner recorder had been calibrated and was running at slow speed when a low steam temperature runback was initiated by the automatic safety system. Seven seconds later, isolation scram was manually initiated.

The steam line thermocouple temperature information which is sent to the pressure control system temperature bias circuit was being recorded on the Offner at the time of the test. During the 52 sec preceding initiation of the runback, this temperature indication decreased 100 F and then suddenly returned to normal (with a little overshoot). The pressure control system does not act on indications of decreasing temperature, but it does cause steam flow to increase (by opening the valve controlling line pressure at the time) on indications of increasing steam temperature. Since temperature was decreasing, no valve action occurred until the sudden increase in temperature indication caused the dump valve to open, thereby increasing steam flow and decreasing steam temperature, with a resultant low steam temperature runback. Since superheater fuel temperature, steam flow, steam pressure, and

reactor power were all constant during the initial fluctuation in the steam line temperature indication, it is certain that the low steam line temperature was a spurious indication.

Steam line pressure is seen to suddenly decrease about 20 psi as steam flow increases. This results from increased pressure drop across the superheater with increased steam flow. Steam flow increased about 25 percent, which corresponds to a 50 percent increase in superheater pressure drop, or about 20 psi at the 75 percent power level.

Isolation scram was manually initiated about 7 sec after control rod runback. The odd disturbances on the Offner recording at the time of scram result from an increase in chart speed made an instant before the scram.

The plant responded normally to the isolation scram.

The oscillation in steam line pressure is discussed in Sec. 4.

3.7 CONTROL ROD RUNBACK FROM 76 MWT (FIG. 7)

The runback test was conducted without turbine-generator operation.

The results shown in Fig. 7 were plotted from the Offner results, which were too lengthy to reproduce. This is the only Test 431 runback during which the control rods were run all the way in without a reactor scram.

Reactor power decreased smoothly as the control rods entered the core. The superheater thermocouples indicated decreasing fuel temperature. No rapid changes or increases in fuel temperature were observed.

3.8 TURBINE TRIP FROM 38 MWT (FIG. 8)

The turbine was tripped by operation of the turbine trip pushbutton located on the control console. Turbine trip action results in stop valve closure, reactor scram, and regulating dump valve opening within about 0.5 sec of the trip. After the valve motions are complete, the system behaves the same as if scram had occurred without turbine-generator operation.

The recorder trace indicates that power, steam pressure, and steam temperature responded as expected, and also as previously experienced during scram test without turbine operation.

Also shown is the regulating dump valve command signal. It rises to the required level immediately following the trip. Actual dump valve position is not shown, but the valve response is well known; and it is certain that the valve was at the indicated demand level in less than 0.2 sec.

The odd behavior of the dump valve command signal and the reactor power indication on the adjacent recorder channel resulted from overlapping of the recorder pens.

3.9 TURBINE LOAD DUMP FROM 38 MWT (FIG. 9)

The load dump test was conducted with a generator load of 7 Mwe and with the regulating dump valve about 2 percent open and controlling the steam line pressure.

Dumping the turbine load consists of tripping the main generator breaker, thereby disconnecting the electrical load from the main power system. Plant electrical requirements (station load) continue to be supplied by the generator; the reactor is not scrammed.

Immediately following the load dump, the turbine inlet valves quickly close and the dump valve opens to a position sufficient to pass the total existing steam flow. The dump valve command signal is shown on the recorder trace.

The turbine inlet valves remain closed for 5 sec and then re-open to the THROTTLE MINIMUM STOP position set by the pressure control system, from which the turbine speed governor takes control of the valves to regulate turbine speed. Re-opening of the inlet valves after 5 sec is reflected in the dump valve command signal, which causes the dump valve to partially close, thus compensating for the turbine inlet valve opening.

No deleterious effects on reactor power, steam pressure, and steam temperature result from the load dump action.

3.10 TURBINE LOAD DUMP FROM 95 MWT (FIG. 10)

The load dump test was conducted with a generator load of 18 Mwe and with the regulating dump valve about 6 percent open and controlling the steam line pressure.

Actual dump valve position, turbine inlet valve positions (not command signal), and total steady flow are shown on the recorder trace. These three parameters clearly illustrate the events following the generator load dump.

The inlet valves close fully in about 0.2 sec, and the regulating dump valve opens in 0.1 sec. Indicated steam flow increases to a maximum of 60,000 lb/hr above equilibrium flow and then quickly returns to about 15,000 lb/hr above equilibrium. The momentary large increase in indicated steam flow (60,000 lb/hr) is caused by the faster response of the dump valve, and possibly by some steam flowmeter overshoot. The more stable 15,000 lb/hr increase in steam flow is caused by the mismatch of flow capacity between the regulating dump and inlet valves.

After 5 sec, the inlet valves re-open to the throttle minimum stop position, and the dump valve partially closes to compensate. Steam flow response is similar to initial response for the same reasons. This time, however, the magnitude of the flow change is smaller; also, the flow change is in the decreasing rather than increasing direction because the valves are moving in directions opposite to their initial motions.

After 13 sec, the inlet valves close slightly. This is caused by the fact that the turbine speed governor takes control of the valves to regulate turbine speed.

The irregularities in indicated steam line temperature and reactor dome pressure are caused by the signal retransmitters. These are internally balanced devices which can become "sticky" and thereby can have difficulty in generating a smoothly changing output.

The oscillation in steam line pressure is discussed in Sec. 4.

3.11. TURBINE LOAD DUMP FROM 140 MWT (FIG. 11)

The load dump test was conducted with a generator load of 39 Mwe and with the regulating dump valve about 6 percent open and controlling the steam line pressure.

Valve positions were not recorded, since the steam flow trace is quite descriptive and representative of system actions. Since the dump valve responds faster than the turbine inlet valves, a momentary large increase in steam flow exists immediately after the load dump. Also, part of this indication (155,000 lb/hr) may be attributed to overshoot of the steam flowmeter. After 1.6 sec, the initial surge in the steam flowmeter indication is over, and indicated flow is 45,000 lb/hr greater than initial flow. This increase results from the mismatch in flow capacity between the regulating dump valve and the turbine inlet valves. In approximately 5 sec, steam flow decreases about 20,000 lb/hr, indicating that the turbine inlet valves have re-opened and that the dump valve has closed to compensate for this valve opening. Once more, the change in steam flow is caused by the valve mismatch.

The change in steam flow due to the valve mismatch (net change of about 25,000 lb/hr) is corrected by the pressure control system. The control system adequately handled the transient, since only a 17 F change in steam line temperature occurred.

Although the mechanics of the load dump went quite well, the test was terminated in a low feedwater temperature runback about 1.5 min. after initiation of the disturbance. This was expected, since calculations had indicated that the No. 14 high pressure feedwater heater is not capable of supplying 320 F feedwater (runback temperature set point) at this power level without the heating capacity of the three low pressure feedwater heaters. These heaters receive only turbine extraction steam, which decreases to an insignificant level (and therefore become unavailable) following a load dump.

The oscillation in steam line pressure is discussed in Sec. 4.

3.12 TURBINE LOAD DUMP FROM 160 MWT (FIG. 12)

The load dump test was conducted with a generator load of 44 Mwe and with the regulating dump valve about 6 percent open and controlling the steam line pressure.

Immediately after the generator breaker was tripped, reactor power increased and leveled off about 13 percent above initial power. Less than 1 sec later, reactor scram was initiated by the two-of-four high flux level scram circuitry. This behavior was anticipated due to an earlier occurrence which is discussed in Sec. 3.13.

The increase in reactor power is attributed to an increase in electrical system voltage and frequency, which increased recirculation pump speed and flow, and thereby increased reactor power. Voltage and frequency increased due to turbine speedup following the generator breaker trip. Tripping the breaker resulted in generator load reduction from 44 Mwe to about 3 Mwe (station load) and, although closure of the turbine inlet valves to prevent overspeed is very fast, it is not instantaneous. Indications were that turbine speed increased about 5 percent and generator output voltage increased about 10 percent.

The increase in reactor power is about 13 percent of initial power, and the flow increase represents about a 5 percent increase in recirculation flow. Test 433* results indicate that the power rise associated with a recirculation flow increase is two to three times as great in terms of the percentages mentioned above. Therefore, the magnitudes of the power and flow increases correspond quite well.

In order to attribute a power increase to a recirculation flow increase, the power and flow increases must be nearly simultaneous, and the magnitudes must correspond. From the above discussion, the magnitudes correspond. From the recorder trace, the flow increase appears to lag somewhat behind the power increase (about 0.6 sec). However, the sluggishness associated with the recirculation flowmeters is sufficient to account for the observed lag.

The increase in feedwater flow resulted from feedwater pump speedup following the system frequency increase.

The oscillation in steam line pressure is discussed in Sec. 4.

3.13 TURBINE LOAD DUMP FROM 168 MWT (FIG. 13)

The load dump test was conducted with a generator load of 46 Mwe and with the regulating dump valve about 6 percent open and controlling the steam line pressure.

This test was conducted prior to the test discussed above in Sec. 3.12. It was during this load-dump test that the reactor power increase following a turbine load dump was first experienced. The reason for the increase in power was not immediately evident, but it was believed that an increase in electrical system frequency due to turbine speedup could increase recirculation pump speed and flow and thereby increase power. This belief led to repeating the load dump test and recording recirculation flow.

Reactor scram was initiated due to high power-to-recirculation flow about 1 sec after the generator breaker was tripped. The load dump test was repeated with a slightly lower initial power to gain a larger margin between initial conditions and safety system set points, so that the test could be completed without reactor shutdown.

*Test 433 will be reported in ACNP-67515, now in preparation.

3.14 TURBINE LOAD DUMP FROM 60 MWT (FIG. 14)

This test is the first and only load dump that has been performed with the new electrical drive unit installed in the inlet valve controller. This test was done with the inlet valves on AUTO, with the dump valve initially closed, and with the generator supplying about 16 Mwe to the system.

As in the load dumps at 85 percent power and at 90 percent power, the reactor power, as recorded by both in-core and out-of-core channels, increased in about 0.4 sec to about 20 percent of its initial value. Since no safety system set points were approached in this test, the reactor did not scram. The difference between this test and the 85 percent and 90 percent load dumps is that the inlet and dump valves were matched so that a 35,000 lb/hr deficit in steam flow occurred in the 30 percent load dump, whereas a surplus in steam flow occurred in the higher power load dumps. In addition, a negligible amount of recirculation flow increase is observed in the 30 percent test, while approximately a 3000 gpm increase occurred in the higher power tests. Thus, it is concluded that the mechanism for power increase in the 30 percent case is different than for the 85 percent and 90 percent cases.

In the 30 percent case, the reactor dome pressure increases 2 psi at the time of the power increase, giving support to the theory that this power rise was caused by a pressure or pressure rate effect on the core voids. In further support of this theory, it was noted that in this case, and for the first time, the in-core power (on in-core channel No. 7, located at the top of the core) follows the steam line pressure oscillations; in addition, when the turbine inlet valves re-opened in the prescribed 5 sec and increased steam flow, reactor power follows the decreasing dome pressure.

4. STEAM LINE PRESSURE OSCILLATIONS

A 3-5 cycle per sec oscillation in indicated steam line pressure has been consistently observed at the Offner recorder following steam flow disturbances caused by isolation scram, turbine trip, and turbine load dump. The source, characteristics, and effects of the oscillation have received considerable attention and are discussed below. A summary of the observed oscillations is presented in Table 1.

4.1 SOURCE

Initially, the oscillation was believed to originate in the Bailey pressure transmitter or associated equipment. Inability to reproduce the oscillation by means of various test inputs to the pressure transmitter resulted in the installation of a pressure transducer upstream of the pressure transmitter (between the steam line and the pressure transmitter). The load dump test results shown in Fig. 11 include the outputs of both the Bailey pressure transmitter and the pressure transducer. Existence of the oscillation at both points eliminates the possibility of the oscillation originating in or downstream of the Bailey pressure transmitter.

The next possible source of the oscillation appeared to be an actual oscillation in steam pressure or of water in the tubing which connects the steam line pressure taps to the Bailey pressure transmitter. Due to the relatively high frequency of the oscillation, steam pressure appeared most likely.

Further evidence that an actual steam system oscillation does exist appears in the results of a reactor scram test at 112 Mwt shown in Fig. 5. The steam line thermocouple temperature indication follows the pressure oscillation. Also, in Fig. 14, load dump from 60 Mwt shows that in-core chamber No. 7 at the top of the core follows the steam line pressure oscillation.

The conclusion is that sudden changes in steam flow rate result in an acoustical-type resonance of the steam system.

4.2 CHARACTERISTICS

The oscillation is initiated by sudden disturbances to steam flow rate. These disturbances consist primarily of the transfer of steam flow from the turbine to the regulating dump valve (following turbine trip and load dump) and also of disturbances caused by emergency condenser operation following isolation scram.

Following a turbine trip, the oscillation is initiated once, damps out, and is not repeated. Following load dump, the oscillation is initiated twice due to two sudden valve movements. Following isolation scram, however, the oscillation continues to be initiated intermittently due to continued flow disturbances caused by emergency condenser operation.

Also, the frequency (cps) of the oscillation decreases somewhat with time due to valve closure and other effects which increase effective resistance and thereby decrease resonant frequency of the steam system.

4.3 DAMPING

Table 1 is a summary of all observed steam line pressure oscillations. This summary shows that the peak-to-peak magnitude of the steam line pressure oscillations increases as power increases up to 140 Mwt; after that the peak-peak magnitude remains constant at about 33 psi.

The damping of the train of pressure oscillations is independent of power, but is different for the different disturbances. For example, λ (which is the damping constant in the time dependent pressure, $P_2 = P_{20} e^{-\lambda t} \sin(\omega t)$) is constant at 0.67 for the isolation scram for all power levels at which this disturbance was studied; λ is 0.50 for the load-dump transients studied up to 170 Mwt. A study of these data indicates that the nature of the transient depends on the cause of the disturbance and that, up to a point, the magnitude of the disturbance depends on steam flow rate.

4.4 EFFECTS

The steam system oscillation is considered to be of an acoustical nature, not involving oscillatory bulk movement of steam. As such, it is also believed that the steam pressure wave is reflected off the bottom of the superheater, thereby having no effect on either the boiling or superheating regions of the core. This belief is confirmed by the fact that reactor power, as indicated by the superheater ion chambers, does not reflect the oscillation; and reactor power through the pressure reactivity coefficient would be affected by the oscillation.

One notable exception has been observed: the power response to a load dump test at 60 Mwt. The results of this test are shown in Fig. 14. In-core ion chamber No. 7 indicates that the pressure effect was seen in the reactor vessel, which would seem to be in conflict with the above discussion. As the following will indicate, however, these test results can be so interpreted as to strengthen the conclusion that the oscillation is of an acoustical nature.

When the pressure control system is in automatic service, it receives an input signal from the Bailey pressure transmitter. System operation is such that (in this case) the regulating dump valve receives a position command signal nearly proportional to the pressure signal. Further, the dump valve is fast enough to follow, and in fact did follow, the command signal. Therefore, an oscillation in bulk steam flow did occur due to dump valve movement, and a power effect was to be expected. Therefore, although a power effect was noted in this one instance, it would not be expected to occur without dump valve movement and a resultant change in bulk steam flow. Also, the fact that power is capable of

being affected by the oscillation (as seen from the 60 Mwt load dump results) coupled with the fact that it is consistently not affected by the oscillation (as seen in all other results) indicates that the steam system oscillation is not seen by the reactor power unless there is dump valve movement.

5. SPATIAL EFFECTS OBSERVED DURING SCRAM AND RUNBACK

In the original design phases of Pathfinder, questions were raised concerning its dynamic performance, since it contains a central section with no power coefficient of reactivity. Calculations utilizing analog and digital methods showed that the superheater section should "follow" the peripheral boiling section of the core during operational transients and during hypothetical accident conditions. In order to verify this theory, the core was instrumented with nine miniature fission ion chambers and with a number of superheater fuel thermocouples for use in the test program. Presented in this section are conclusions reached by analyzing the data for spatial effects that may have occurred during the large reactivity insertions of scram and runback. Locations of the in-core ion chambers are shown in Fig. 15.

In each of the three superheater locations shown, three ion chambers were axially positioned 45.7 cm, 91.4 cm, and 137 cm from the bottom of the core. Instrumentation requirements limits simultaneous ion chamber reading to only three of the nine ion chambers, but a coaxial switching scheme provides an easy means of selecting any three of the nine chambers for observation and recording. Each string of three ion chambers is provided with a flux wire thimble which is periodically used to calibrate the ion chambers. By means of this calibration, differences in in-core ion chamber sensitivity may be accounted for.

Specifically examined here, is how the superheater follows the boiler core, whether any notable differences could be detected between in-core and out-of-core neutron detector responses, and how the magnitude of the disturbance is reflected in ion chamber response.

5.1 SPATIAL EFFECTS DURING SCRAM

Planned reactor scrams were initiated at various power levels. Of interest from a spatial dynamics standpoint was what the axial and radial flux shapes would do during the scram. The responses of in-core ion chambers 7, 8 and 9 were analyzed and showed that as the rods fell into the core the transient flux shape usually was pushed into the bottom of the core, as might be expected. This effect is demonstrated in the lower curve of Fig. 16, which shows the ratio of in-core chamber 8 to in-core chamber 9 current decreasing to a minimum at 500 msec, at which time the scramming rods were approximately just above the center of the core. The ratio then increases to 1.0 at about 1.2 sec, indicating that the rods were fully seated at that time. (Chamber 8 is at the core midplane, while chamber 9 is below it in the same fuel channel, 18 in. from the bottom of the core.)

In certain rod configurations, however, the same ratio of in-core currents increases before decreasing to 1.0 after the rods have been seated in the core. The top curve of Fig. 16 illustrates this condition. When the axial flux peak was approximately just above the center of the core, and with a fairly strong group of control rods just above the core midplane (where in-core chamber 8 is located), the axial flux shape was initially pushed toward the center of the core, as this rod group scrambled. This phenomenon caused the ratio of chamber 8 current to chamber 9 current to increase before decreasing.

Note that this top curve in Fig. 16 does not imply that a power peak higher than the initial power peak occurred as scram occurred. At 500 msec, the power (and all in-core chamber currents) had decreased to about 25 percent of the initial power. The ratio of the in-core currents, and thus the axial power shape, changed during the scram.

5.2 SPATIAL EFFECTS DURING RUNBACK

Another large disturbance studied was control rod runback, during which all rods, except the superheater rods, are driven into the core. The outer boiler rods, Groups IV and V, are driven the entire 73 in. in about 1 min, while the inner boiler rods, Groups II and III, are driven the 73-in. core length in about 3 min.

Figure 7 shows the response to runback of in-core chambers 4, 5, 6, and out-of-core channel 5, with Group II rods at 0 in. Chambers 4, 5, and 6 are located in the center of the superheater core, while out-of-core channel 5 is located outside the reactor vessel in the biological shield near the bottom of the core. Figure 7 shows that the out-of-core chamber response closely followed the response of the in-core chambers, demonstrating the predicted close coupling of the core. Presentation of this same data, similar to the data shown in Fig. 16, shows that the ratio of chamber 5 to chamber 6 currents decreases with no minima such as appeared in Fig. 16. In each case of runback or reactor scram, it was noted that the out-of-core neutron detector response fell off faster than any of the in-core ion chambers during the same transient.

In tests similar to this, a radial set of in-core ion chambers was observed and showed the same sort of behavior, with no radial flux shape shifting being evident.

6. CONCLUSIONS

A considerable amount of information has been generated during Test 431 testing, the bulk of which has been presented in previous sections of this report. This information, as well as a certain amount of additional information derived during the testing program, warrants certain conclusions concerning performance of the reactor and of the associated plant control systems following safety actions. (Refer to Fig. 17 for a simplified schematic of the reactor plant systems.)

6.1 REACTOR RESPONSE

Response of the reactor to each of the safety actions has been satisfactory, with the possible exception of load dump. Steps are being considered to modify the pressure control system so that load dump will be an acceptable plant action.

Reactor scram has consistently resulted in the reduction of reactor power to a value less than 10 percent of initial power within 1 sec of initiation. This time includes both instrumentation lag time and power drop-off time.

Superheater thermocouples have indicated that superheater fuel temperature is at or near boiler saturation temperature within about 9 sec of scram.

Reactor response to turbine trip has been essentially the same as response to reactor scram, which is as predicted.

6.2 PLANT CONTROL SYSTEM RESPONSE

Throughout the Test 278 power escalation series, the plant control systems have been adjusted to obtain desired performance both during normal reactor operation and following reactor safety actions.

The pressure control system positions the turbine inlet and regulating dump valves to maintain constant steam line pressure and to quickly transfer steam flow from the turbine to the dump valve following turbine trip and load dump. The pressure control system provides satisfactory performance following each of the safety actions.

The reactor water level control system positions the feedwater control valve to maintain constant water level in the vessel. Reactor scram and the resultant reduction in steam flow imposes the most significant disturbance on this control system. The level control system maintains reactor level satisfactorily following reactor scram and under all other operating conditions.

The feedwater temperature control system positions a valve to control steam flow to the high pressure feedwater heater. Performance of this system has been satisfactory under all operating and test conditions.

TABLE 1

SUMMARY OF STEAM LINE PRESSURE OSCILLATIONS OBSERVED DURING TEST 431

Figure No.	System Disturbance	Initial Power Level	Max. Peak-Peak of ΔP_2 Oscillations	Frequency of Oscillations	λ , sec^{-1} Damping Time $P_{20} e^{-\lambda t} \sin \omega t = P_2$	Times Oscillation Train Repeated	Comments
1	Isolation Scram	38 Mwt	30 psi	4.5 cps	0.67 sec^{-1}	10	Oscillations begin 5 sec after scram
2	Scram	38 Mwt	10 psi	--	--	--	Small scale on pressure
4	Isolation Scram	60 Mwt	16 psi	4.5 cps	0.71 sec^{-1}	3+	--
5	Scram	112 Mwt	35.5 psi	4.2 cps	0.39 sec^{-1}	1	Inlet valves initially at 30%; dump valve at 8%
6	Isolation Scram	140 Mwt	32 psi	4.0 cps	0.67 sec^{-1}	4+	Largest oscillations 17 sec after scram
8	Turbine Trip	38 Mwt	2.5 psi	18 cps	--	1	Very small response on P_2 trace
9	Turbine Load Dump	38 Mwt	2.5 psi	17 cps	--	1	--
10	Turbine Load Dump	95 Mwt	20.8 psi	4.2 cps	0.50 sec^{-1}	1	--
11	Turbine Load Dump	140 Mwt	36 psi	4.2 cps	0.63 sec^{-1}	1	Inlet valves initially at 50%; dump valve at 6%

Table 1 -- Summary of Steam Line Pressure Oscillations Observed During Test 431 (cont'd)

Figure No.	System Disturbance	Initial Power Level	Max. Peak-Peak of ΔP_2 Oscillations	Frequency of Oscillations	λ, sec^{-1} Damping Time $P_{2o} e^{-\lambda t} \sin \omega t = P_2$	Times Oscillation Train Repeated	Comments
12	Turbine Load Dump	162 Mwt	33.5 psi	4.5 cps	0.53 sec ⁻¹	1	--
13	Turbine Load Dump	169 Mwt	33.5 psi	4.2 cps	0.46 sec ⁻¹	1	--
14	Turbine Load Dump	57 Mwt	10 psi	3.5 cps	0.50 sec ⁻¹	2	Second train of oscillations occurred when inlet valves re-opened at 5 sec

7. FIGURES

1. Reactor Isolation Scram, 20% Power (38 Mwt), Manual Pressure Control, Manual Feedwater Temperature Control
2. Reactor Scram, 20% Power (38 Mwt), Auto Pressure Control with Dump Valve, Auto Level Control, Manual Feedwater Temperature Control
3. Reactor Scram, 30% Power (62 Mwt), Auto Pressure Control with Dump Valve
4. Isolation Scram, 30% Power (60 Mwt), Without Turbine
5. Reactor Scram at 58% Power (112 Mwt), Inlet Valves Fixed at 30% Open, Dump Valve on Auto at 8% Open
6. Isolation Scram, 75% Power (140 Mwt), Auto Pressure Control with Dump Valve
7. Response to Control Rod Runback, Rod Groups I, III, IV, and V at 73 In., Group II at 0 In.
8. Turbine Trip, 20% Power (38 Mwt), Auto Pressure Control with Dump Valve
9. Turbine Load Dump, 20% Power (38 Mwt), Auto Pressure Control with Dump Valve
10. Turbine Load Dump, 50% Power (95 Mwt), 18 Mwe
11. Turbine Load Dump, 75% Power (140 Mwt), Inlet Valves 50% Open, Dump Valve 6% Open and Controlling Pressure
12. Turbine Load Dump, 85% Power (162 Mwt)
13. Turbine Load Dump, 89% Power (168 Mwt)
14. Turbine Load Dump, 30% Power, 16 Mwe, Inlet Valves Controlling Pressure
15. Pathfinder Core, Plan View
16. Ratio of In-Core Ion Chamber Currents Following Reactor Scram
17. Simplified Pathfinder Plant Schematic

REACTOR SCRAM, 20% POWER (38 MWT), AUTO PRESSURE CONTROL
WITH DUMP VALVE, AUTO LEVEL CONTROL, MANUAL FEEDWATER
TEMPERATURE CONTROL

REACTOR SCRAM, 30% POWER (62 MWT), AUTO PRESSURE CONTROL WITH DUMP VALVE

FIG. 3

ISOLATION SCRAM, 30% POWER (60 MWT), WITHOUT TURBINE

REACTOR SCRAM AT 58% POWER (112 MWT), INLET VALVES FIXED AT 30% OPEN, DUMP VALVE ON AUTO AT 8% OPEN

FIG. 5

ISOLATION SCRAM, 75% POWER (140 MWT), AUTO PRESSURE CONTROL WITH DUMP VALVE

RESPONSE TO CONTROL ROD RUNBACK, ROD GROUPS I, III, IV, AND V AT 73 IN., GROUP II AT 0 IN.

FIG. 7

TURBINE TRIP, 20% POWER (38 MWT), AUTO PRESSURE CONTROL
WITH DUMP VALVE

TURBINE LOAD DUMP, 20% POWER (38 MWT), AUTO PRESSURE CONTROL WITH DUMP VALVE

TURBINE LOAD DUMP, 50% POWER (95 MWT), 18 MWE

FIG. 10

TURBINE LOAD DUMP, 75% POWER (140 MWT), INLET VALVES 50% OPEN, DUMP VALVE 6% OPEN AND CONTROLLING PRESSURE

FIG. 11

TURBINE LOAD DUMP, 85% POWER (162 MWT)

TURBINE LOAD DUMP, 89% POWER (168 MWT)

FIG. 13

TURBINE LOAD DUMP, 30% POWER, 16 MWE, INLET VALVES
CONTROLLING PRESSURE

FIG. 14

PATHFINDER CORE, PLAN VIEW

Ion-chamber 8 is 36 inches from bottom of core (mid-plane).

Ion-chamber 9 is 18 inches from bottom of core.

Group III rods controlling at 48 inches from bottom of core. Power peak initially near center of core.

Core I rods controlling at 48 inches from bottom of core. Power initially in bottom of core.

RATIO OF IN-CORE ION CHAMBER CURRENTS FOLLOWING REACTOR SCRAM

FIG. 16

FIG. 17