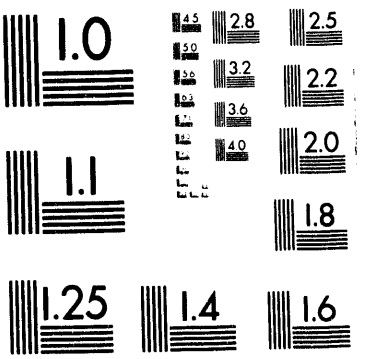
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DR-REACTOR HEAT DECAY TEST AT HIGH OUTLET WATER TEMPERATURES

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DR-REACTOR HEAT DECAY TEST AT HIGH OUTLET WATER TEMPERATURES

by

S. S. Jones

March 6, 1961

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PRODUCTION TEST IP-338-A

DR-REACTOR HEAT DECAY TEST AT HIGH OUTLET WATER TEMPERATURES

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OBJECTIVE

This test is designed to obtain up-to-date information on heat decay values following a reactor scram. More specifically its purpose is to determine the effect of increased outlet water temperatures upon this heat output value.

BASIS AND JUSTIFICATION

A recent review[1] of the "last ditch" cooling system at the Hanford reactors indicated that at some reactors this backup coolant supply did not meet our coolant adequacy criteria at certain critical times after the loss of electric and steam pumping power. On the basis of this review, a high priority program is underway to bring the adequacy of reactor backup cooling systems into conformance with the criteria.

An important part of this program lies in improving the accuracy of certain basic elements of reactor cooling conditions following a power loss.[2] Primarily these are:

1. Accurate reactor low flow values
2. Accurate reactor heat output values

A program is already in progress to improve both the amount of last-ditch cooling flow and to improve the accuracy with which this flow is determined.[3,4]

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(1) Jones, S. S., "Review of the Status of Hanford Reactor Emergency Cooling," HW-64417, June 1, 1960.

(2) Greager, O. H., "Recommended High Tank Temperature for Maintenance of High Tank Backup Support," HW-67080, October 11, 1960.

(3) Jensen, H. F., S. S. Jones, and N. R. Miller, "Production Test IP-337-AP Chemical Descaling of 105 Hi-Tank Piping," HW-65726, June 17, 1960.

(4) Benson, J. L., "Production Test IP-326-I Low Flow Calibration Tests at the Old Reactors," HW-65715, June 20, 1960.

This production test is planned to provide the more accurate heat output values needed for this program. It will not provide the experimental confirmation of the extreme situation of 90°C bulk outlet temperature during the shutdown transient. Instead, heat decay will be measured at two different outlet water temperatures. A combination of extrapolation and theoretical methods will be used to carry the test results up to the 90°C outlet water temperature limit established by the criteria.(

It is important to complete this test as soon as possible for the following reasons:

1. Our present values are based upon experimental tests conducted over six years ago at a time when the reactor power level, gas composition, and moderator properties were all considerably different than they are today.
2. No tests have been conducted at elevated outlet water temperatures. Thus, the effect of a change in the moderator sensible heat contribution to this total heat output has never been measured.
3. At the critical times of hi-tank take over from the pump fly-wheel decay and export take-over from the empty high tanks, the moderator heat contribution represents 20 to 40 percent of the total heat load; thus it is quite important.
4. It is expected that this test will result in a 5 to 10 percent improvement in the known cooling adequacy of the Last Ditch cooling system.

Thus, in addition to a basic improvement in reactor technology, there may well be significant economic incentive for performing this test.

Before going to the test procedure, however, it is in order to examine the nature of the reactor heat output problem. Following a reactor shutdown the total heat output comes from several sources:

1. Heat generated from the fission of U-235 by delayed neutrons.

(5) Trumble, R. E., "Performance Criteria for Hanford Production Reactors," HW-66929, September 1, 1960 (Unclassified).

[redacted]

2. Heat generated from radioactive decay of the fission products,
3. Sensible or stored heat flow from the fuel elements, dummy charge, piping, and miscellaneous reactor components to the coolant,
4. Sensible or stored heat flow from the graphite moderator to the coolant.

There are several reports that indicate the value of the first two effects.[6,7] These are believed accurate to within ±10 percent in the critical region of 5 to 1,000 seconds following the loss of reactor coolant pumping power.

There are also several theoretical reports that indicate the expected values of total heat output following a reactor scram.[8,9] In general, the analytical procedures used in these reports must make a number of basic but inaccurate assumptions to simplify the problem. Thus, they introduce significant error in the final solution. The purpose of this test is to obtain experimental values that will serve as a basis for the analytical solutions to this problem and permit the extension of the analytical solutions to other cases with a greater degree of confidence.

The completion of these tests has a direct bearing upon plant project action since it will affect cooling adequacy. The tests will be performed at the DR Reactor in order to make use of the considerable instrumentation already installed and in order to make more complete use of the data already

---

(6) Jones, S. S., "Heat Generation and Total Heat Output from the Pile After Shutdown," HW-33870, November 23, 1954.

(7) Untermeyer, S., and J. T. Weills, "Heat Generation in Irradiated Uranium," ANL-4790, February 1952.

(8) Larson, M. B., "A Method of Solution of Reactor Temperature Transients as Applied to the 100-K Plant," HW-38944, September 7, 1955.

(9) Tippets, F. E., "Analysis of the Transient Conduction of Heat in Long Solid Cylindrical Fuel Elements for Nuclear Reactors," HW-41896, May 1, 1956.

obtained under Production Tests IP-278-A[10] and IP-301-AI,[11] but the test results will be applied to the B through K Hanford reactors.

TEST PREPARATION

This test involves manual reactor flow reductions very soon after a reactor scram. This is a critical time and poor flow control could result in fuel damage. Therefore, extreme care must be taken to assure complete reactor instrumentation and complete test programming and personnel briefing before running the actual tests.

The instrumentation requirements for this test are shown in Table I. The majority of this is already installed. The rest should be so arranged that instrumentation can be completed on four hours notice. Then, if an event occurs indicating the desirability of reactor shutdown, every effort should be made to conduct the test, provided the required procedure write-ups and personnel briefings have been completed and approved.

SUMMARIZED TEST PROCEDURE

A detailed, unclassified procedure has been prepared as a separate appendix to this report. The general procedure is outlined as follows:

1. Preliminary Procedure

   a. To ensure adequate reactor flow backup, the high tank and export lines should be flushed and the strainers cleaned. Their Groves valves must be set for prompt manual trip upon demand by the test coordinator* and the flow balance valve must be open. In addition,

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(10) Jones, S. S., "Production Test IP-278-A Verification of Transient Bulk Temperature Surges at the DR Reactor," HW-61456, August 18, 1960.

(11) Agar, J. D., and S. S. Jones, "Production Test IP-301-I Dual Area Trip-Out of D and DR Reactor Process Pumps," HW-63586, January 29, 1960.

* The manual trip operation of these valves is a non-standard procedure and several time checks of this arrangement should be made to determine the opening reliability and time delay of this system.

TABLE I

TEST INSTRUMENTATION

No.	Point No.	Item	Function	Remarks
1	T_i	Reactor Inlet Temp.	ΔT	Install accurate inlet temp. recorder
2	T_o	Reactor Bulk Outlet Temp.	ΔT	
3	AT 1,3	Rear Riser Bulk Outlet Temp.	ΔT_R	Record on visicorder
4	AT 2,4	Rear Riser Bulk Outlet Temp.	ΔT_R	Calibrate and speed up control room recorder
5	AS 1,2	Signal Lights	Time of scram and 1XX trip	Lights to go from dim to bright as power loss and 1XX relays open
6	20 T_{t_o}	Tube Outlet Temps.	Central zone tube temps.	Monitor 20 high power high* exposure power on left-hand Brown at increased chart and scanning speed
7	20 T_{t_o}	Fringe Zone	Fringe zone tube temps.	Monitor 20 high power high* exposure power on right-hand Brown at increased chart and scanning speed
8	T_G	Graphite Stringer Temps.	Moderator temp. decay	Record stringer temps. on Geminol recorders
9	Δp_c	Cross-header Press.	$G\Delta p_c$	Monitor front header to rear header Δp on 4 selected headers
10	p_{R_i}	Heise Gauges and Movie Cameras	p_T	Install calibrated Heise gauges, timers, lights, and cameras - 105 DR
11	p_{R_i}	Heise Gauges	p_T	190 Control Room
12	(VSR)	Vertical Rod Drop	Poison insertion	Install camera and lights on top of unit along with trip, or 1XX, signal light, and clock
13	SCM	Subcritical Monitor	Neutron flux decay	Set to record at high speed during first 0-2 minutes of test (Install time trace)
14	PRM	Power Rate Meter	Tube temp. decay	Record on visicorder
15	IE	I and E Probe Temps.	Tube outlet temp.	Record on visicorder

* Install additional temperature recorder for monitoring individual high tubes as desired.

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detailed 105 and 190 buildings test procedures must be prepared and reviewed by all parties concerned with the test prior to its performance.

b. It should be determined that all reactor pumps and power supplies are normal and no reactor cross-header valving will be done during the course of this test.

2. Test Procedure

Test No. 1: Low Temperature Water: $T_i + 15 \pm 5^\circ$ C

With all preliminary preparations completed, adjust the reactor pumping system to yield flow from the electric pumps backed up by 8 steam pumps operating at full speed. After the test count-down, scram the reactor and 30 seconds later, after assuring that at least 25 VSR rods have seated in the reactor and a rapid decay of neutron flux has occurred, the electric pump PD valves will be closed simultaneously by means of their master controller. This valve closing will take about 4.5 minutes and will cause a slower flow decay than would a BPA failure.*

After 5 minutes, the electric pump bypass line valves will be closed in succession on order from the test coordinator. He will maintain continuous watch of the bulk outlet, 20 central zone, and 20 fringe zone tube temperatures and will not ask for a further test step until after all of the following conditions are met at each step:

a. The bulk outlet temperature is $\leq T_i + 15^\circ$ C**

b. The tube temperatures are $\leq T_i + 20^\circ$ C

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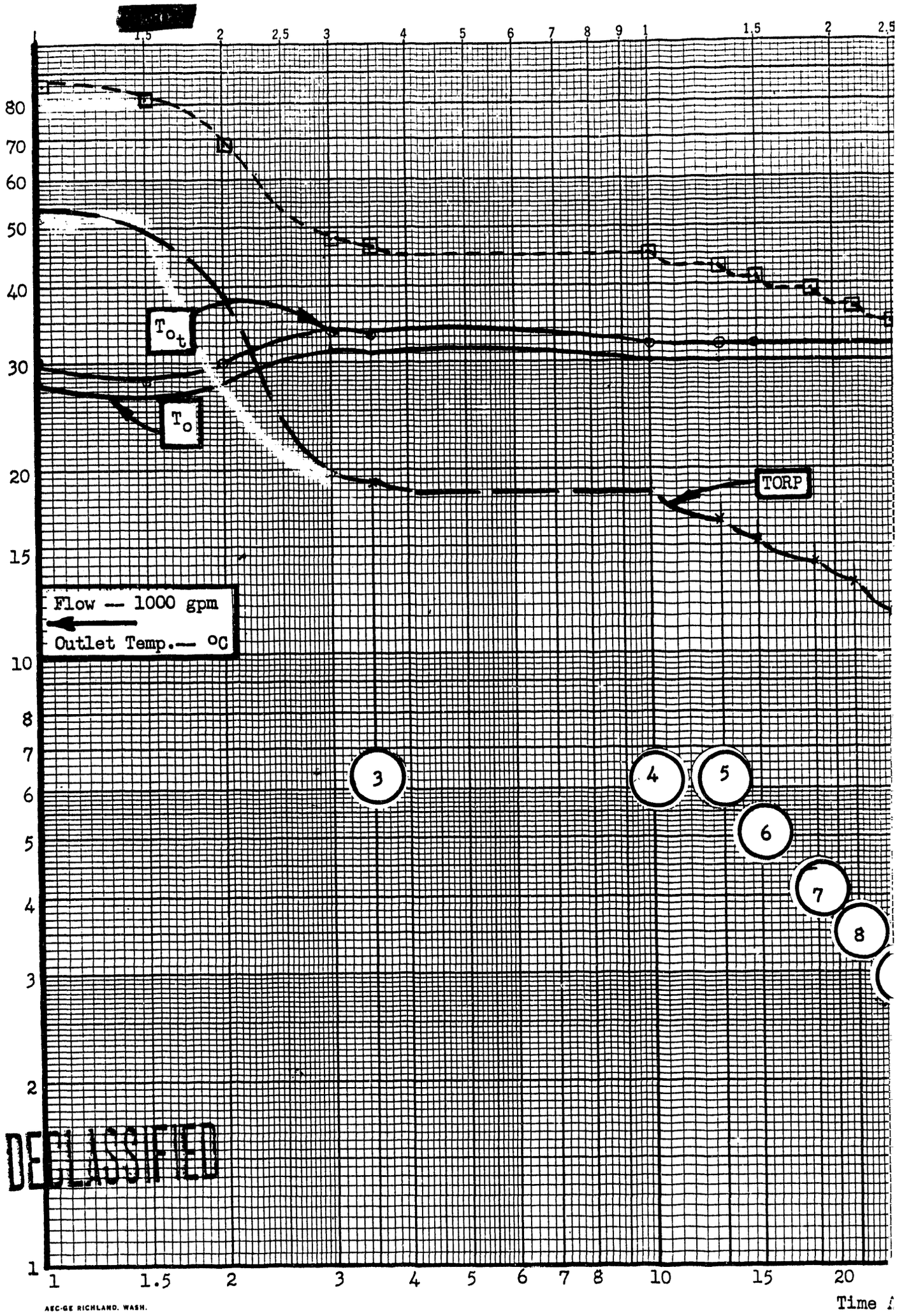
* This condition was established as a part of the CG-558 acceptance tests.

** The rear riser temperatures, AT-1, 2, 3, 4, will serve as the bulk outlet temperature monitor in this test.



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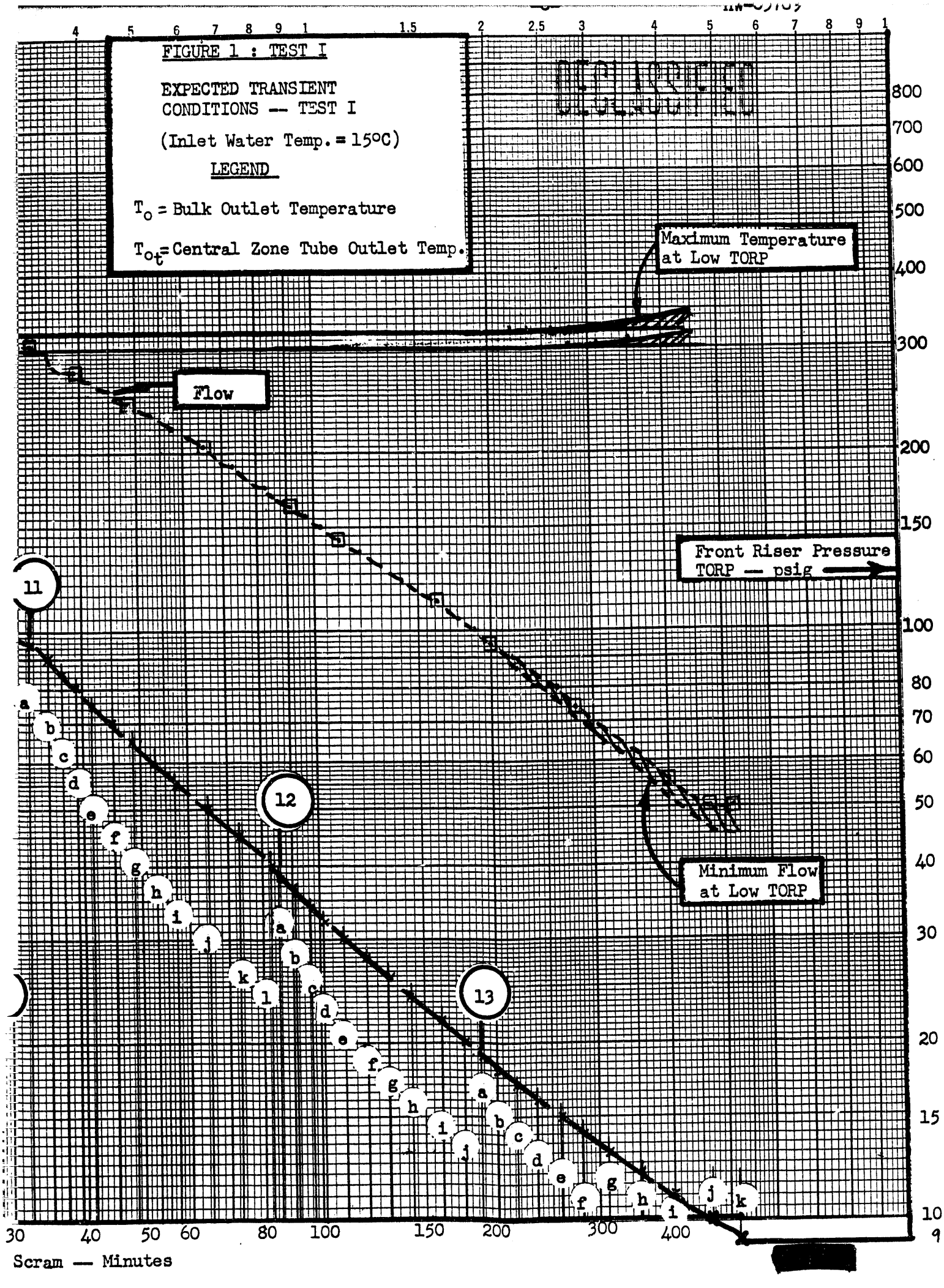
FIGURE 1 : TEST I
EXPECTED TRANSIENT CONDITIONS -- TEST I
(Inlet Water Temp. = 15°C)
LEGEND
T_o = Bulk Outlet Temperature
T_{o_t} = Central Zone Tube Outlet Temp.
Maximum Temperature at Low TORP
Flow
Front Riser Pressure TORP — psig
Minimum Flow at Low TORP
11
12
13
Scram — Minutes

As the electric pump bypass line valves are closed, the 8 steam pumps will gradually and automatically take over the pumping load. At the end of an estimated elapsed time of 30 minutes, the reactor will be down to 30,000 gpm and an expected bulk temperature rise through the reactor of $15^{\circ} \pm 2^{\circ}$ C (See Figure 1).

At this time, further incremental reductions in reactor flow will be made by reducing the speed of the steam pumps, at the direction of the test coordinator,* by means of the master control valve in the 190 Building. Accurate Heise gauges installed both in the 105 and 190 control rooms and all set to read TORP will be used to measure reactor flow, and plainly visible bulk and tube outlet temperature recorders will be used to monitor the course of the test.

This is the most critical part of the test and the man who controls the speed of these steam pumps will be the 190-DR control room operator. The same operator should control these pumps in both tests. This is essential for the continuity of test performance required between the two tests. Each pressure reduction step will be done only at the direction of the test coordinator and only of the magnitude given in the table shown on pages 10-14 of the Appendix.

After achieving stable flow and starting on these stepwise pressure reductions the following precautions will be observed:

a. If bulk outlet temperatures exceed $T_i + 20^{\circ}$ C or tube outlet temperatures exceed $T_i + 26^{\circ}$ C, accelerate all 8 steam pumps, and reduce TORP to previous setting + 5 psi as soon as bulk temperatures drop to $T_i + 17^{\circ}$ C. If tube temps also drop to $T_i + 22^{\circ}$ C, continue test.

b. <u>If bulk outlet temperatures continue to increase</u> and exceed $T_i + 35^{\circ}$ C or tube outlet temperatures exceed $T_i + 45^{\circ}$ C, immediately open the

* This is the most critical part of the two tests and the men who control the speed of these steam pumps must be fully qualified and intimately familiar with the test procedures. This is essential for the continuity of test performance.

[redacted]

high-tank and export Groves valves; and conduct a rapid investigation of the reasons for the elevated temperatures.

c. If there is a BPA failure or a critical "Y", accelerate 8 steam pumps to full flow and cancel the test.

d. If there is a failure of one to two steam pumps, the 190 Operator is to immediately start one or two of the 190 idling steam pumps.

e. If there is total steam system failure, immediately advise the test coordinator who will direct the opening of the high tank and export Groves valves and starting the electric pumps by opening their PD valve bypass lines.

At the completion of this test, return 3 steam pumps to full reactor flow and place the others in the manual off position. Measure reactor TORP for ~~this 3-steam~~ pump condition and repeat this test with 5 and with 7 steam pumps on full flow to the reactor. This will provide us with an accurate calibration of steam pump reactor flow for use in Test No. 2. This Test No. 1 will be completed at 4 hours after the reactor scram by bringing up the steam pumps to full speed, opening the electric pump bypass valves and returning the system to normal shutdown conditions.

Test No. 2 - High Temperature Water: $T_1 + 40 \pm 3°$

Modify the expected outlet temperature curves for this test (Figure 2) and modify the Time:TORP data of Table A-III which will be used to control this test. This modification to come from the results of Test No. 1, and from the reactor low flow calibration test that is required prior to running this second test.

With all preliminary conditions completed, as for Test No. 1, close the electric pump bypass line valves and accelerate 8 steam pumps

* Any leakage thru the electric pump P.D. valves should also occur in test #2 and tend to be self compensating.

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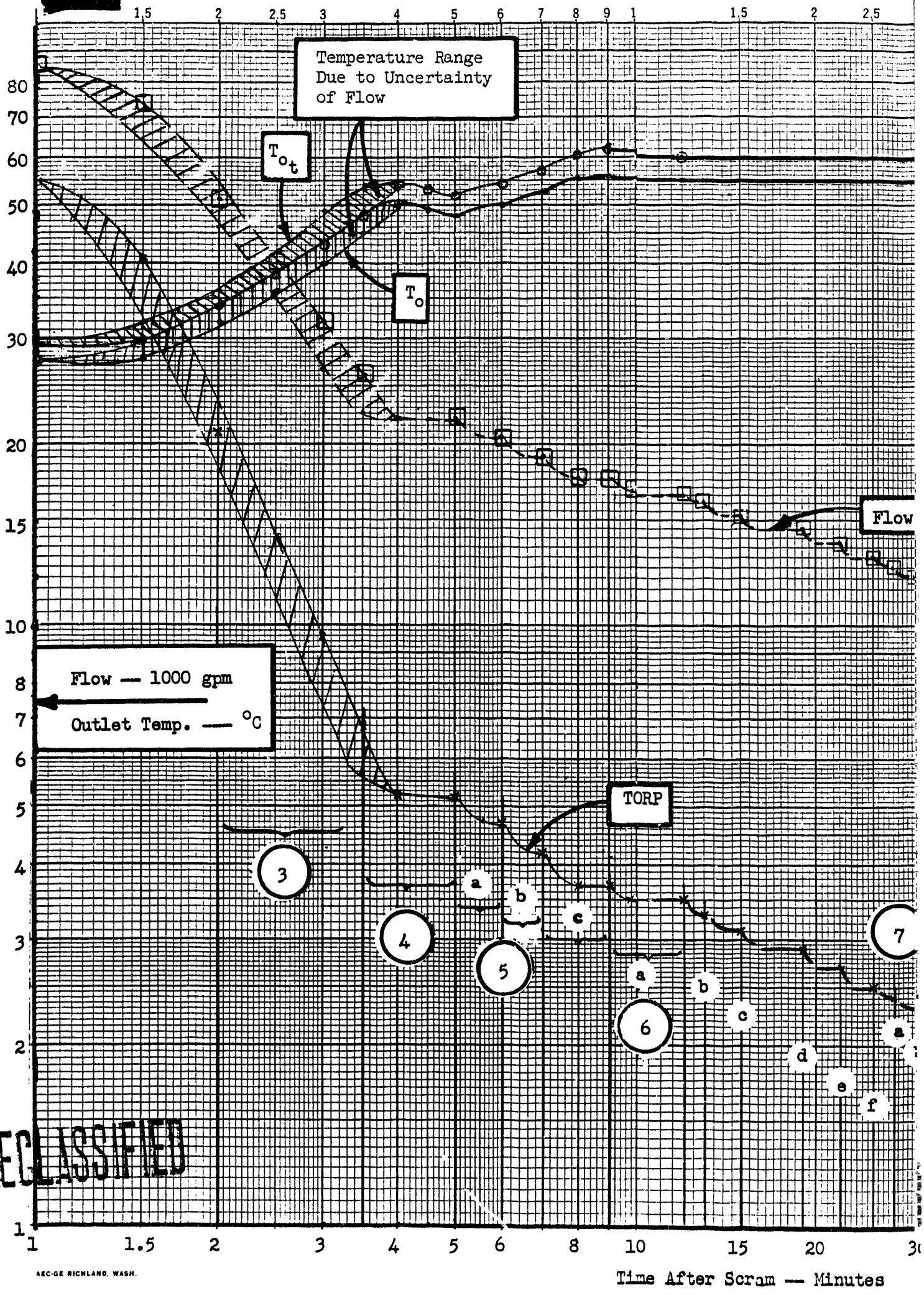

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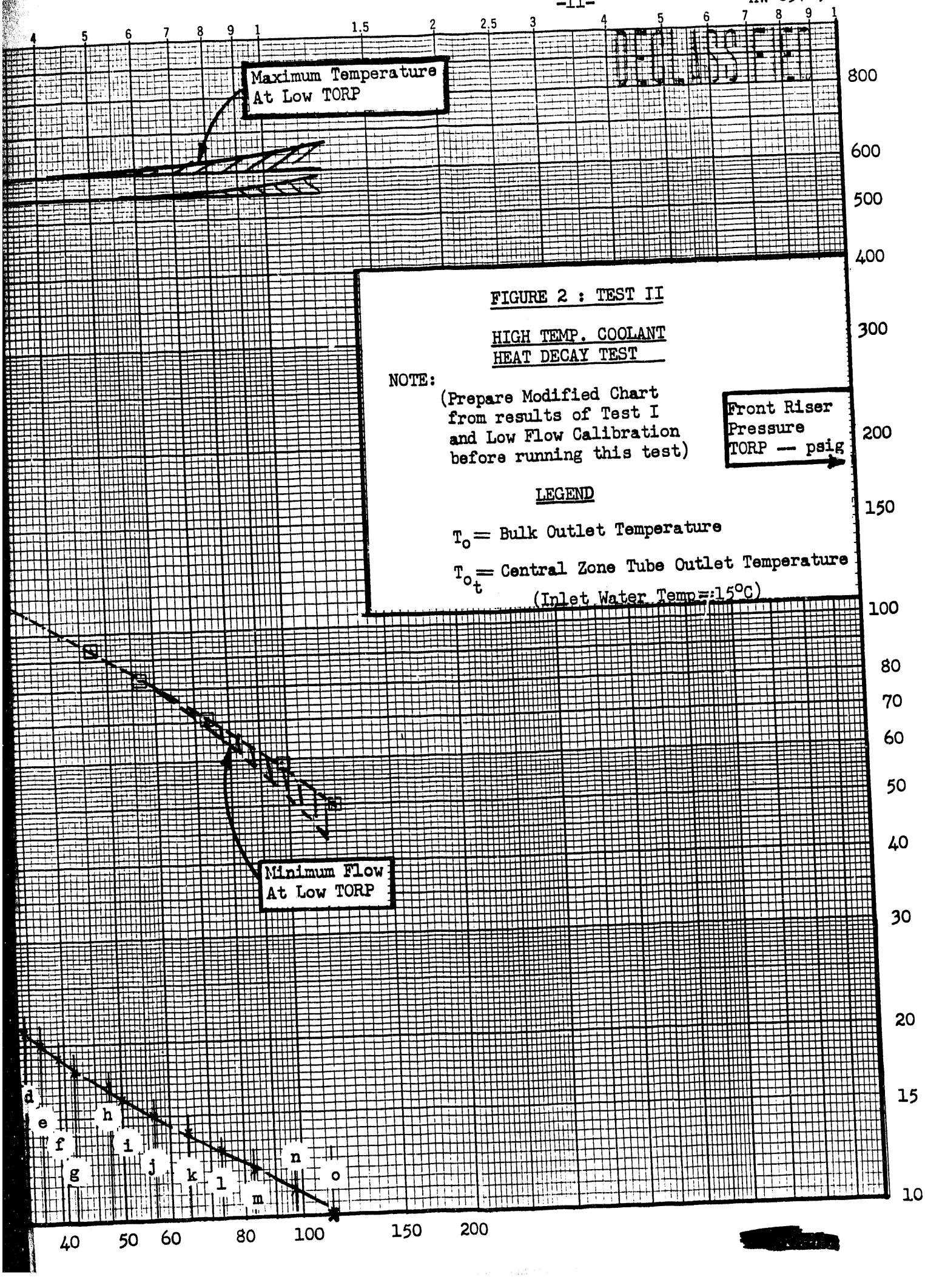

FIGURE 2 : TEST II

HIGH TEMP. COOLANT HEAT DECAY TEST

NOTE:

(Prepare Modified Chart from results of Test I and Low Flow Calibration before running this test)

LEGEND

T_o = Bulk Outlet Temperature

T_{o_t} = Central Zone Tube Outlet Temperature

(Inlet Water Temp = 15°C)

[redacted]

to full speed, their FD valves open. Allow this flow condition to stabilize and check readiness for start of test.* After the test count-down, scram the reactor and observe that at least 25 VSR rods have seated in the reactor and the neutron flux has decayed. At 30 seconds after scram, at the direction of test coordinator, close the electric pump FD valves simultaneously by means of their master controller. This will cause a flow decay similar to that in Test No. 1.** At two minutes following the scram, turn two of the backup steam pumps to the manual off position. At 3.5 minutes turn off another steam pump. This will leave five steam pumps supplying a reactor flow of 22,000 gpm at 4 minutes after the scram. The bulk outlet temperature will be $T_i + 33 \pm 2°$ C.

During this time, as well as the rest of the test, the coordinator will maintain a continuous watch of the bulk outlet, 20 central zone, and 20 fringe zone tube temperatures. He will not request a further flow reduction until the following conditions are met at each step.

a. The bulk temperature $\leq T_i + 40°$ C

b. The central zone tube outlet temperature $\leq T_i + 53°$ C

Five minutes after the scram, with the electric pump FD valves essentially closed, the reactor flow should be further decreased by step-wise reduction in TORP at the direction of the test coordinator. This will be effected by throttling the steam pump master control valve in clearly specified TORP pressure reduction steps. This procedure will continue for up to two hours after the reactor scram. At this point 8 steam pumps will be returned to full speed and 30,000 gpm reactor flow and the reactor system will be returned to normal shut-down conditions.

---

* The high-tank limitorque valves will be kept in the 1/4 open position during this test This will ensure adequate high tank cooling upon opening of the high tank Groves valve.

** The flow will decay down to a lower value but should follow the time transient determined in Test No. 1.

During the course of this test, the same emergency procedures and precautions will be observed as in Test No. 1. These are outlined as follows:

a. If bulk outlet temperatures exceed T_1 + 48° C or tube outlet temperatures exceed T_1 + 62° C during the course of this test, accelerate all 8 steam pumps to full speed and flow. If the temperature drops to T_1 + 40° C continue the test by returning to the previous TORP + 5 psig.

b. If bulk outlet temperatures continue to rise and exceed T_1 + 50° C or tube outlet temperatures exceed T_1 + 65° C, immediately open the high tank and export Groves valves and conduct a rapid investigation of the reasons for these elevated temperatures.

c, d, and e. These safety steps for BPA and steam system difficulties are the same as in Test No. 1.

EXTENDED SHUTDOWN HEAT OUTPUT TEST (Tests 1A & 2A)

At the end of each shutdown which is started with one of these heat output decay tests, an additional test will be conducted. This will require no recording equipment and a minimum of personnel coverage. The test details are given in the Appendix as Test 1A & 2A. Basically, it involves having the reactor valving and tubes returned to normal operating conditions and then gradually reducing reactor flow to 6,000 gpm while monitoring the bulk and tube outlet temperatures. Hold the reactor at low flow for 20 minutes* and then return to normal full flow conditions in preparation for reactor startup. This should give a valuable measure of extended reactor downtime heat output.**

PROCESS STANDARDS

This production test authorizes exceeding the following Process Standards:

1. A-030.01 which specified minimum shut-down riser pressure and flows.
2. A-080.01 which specified by intent that high tank flow be available automatically during operation in case of a loss in water supply from other sources.

* This test authorizes 20 minutes at this condition. However, it will be stopped 2 minutes after reaching steady temperature conditions.

** The anticipated flows and temperatures for this test are shown in Figure 3.

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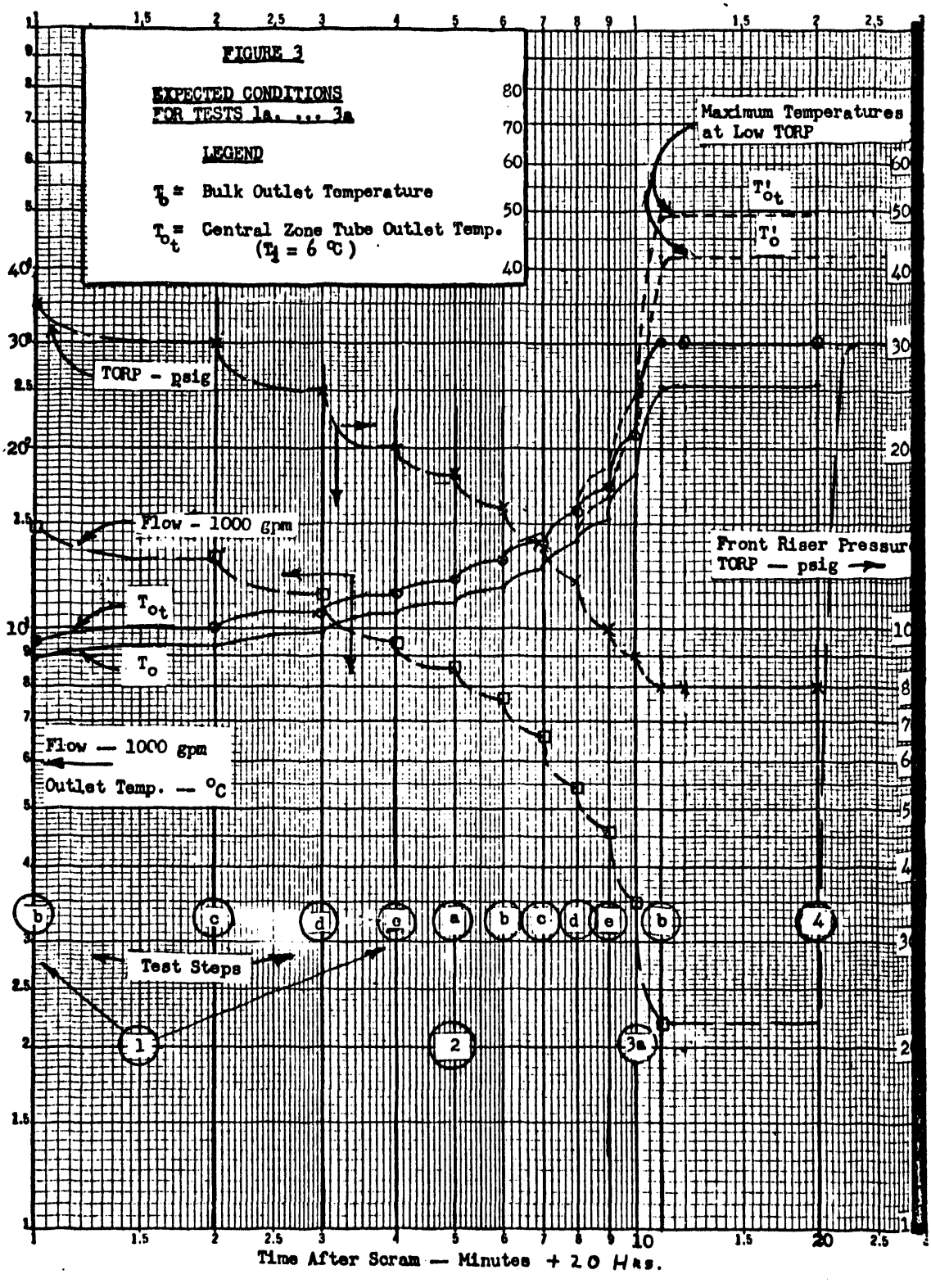

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3. 190-B-070.03 (Water Plant Standards) which specifies that the export Groves valve be kept on automatic during reactor operation.

HAZARDS REVIEW

The only hazards associated with this test are those associated with gross violation of procedures. Scramming the reactor at full reactor flow is almost a normal shut-down procedure; closing the pump discharge valves after 30 seconds causes a water flow reduction transient that is slower than the pump flywheel decay. Thus, we are imposing a flow transient after 30 seconds of down time that is slower than the flywheel transient that could be safely handled at zero shut-down time. However, after this first, almost automatic transition, valving changes are made that could conceivably lead to difficulty. Therefore, the test outlines a step-wise procedure and the mishandling of any one step would not cause sufficient flow change to raise the reactor outlet temperature even close to the boiling point. Furthermore, both the bulk and 40 selected tube outlet temperatures will be followed continuously and any increase in temperatures above prescribed limits will be reason for returning the system to full water flow. Because of the decrease in riser pressure below steam sweep-out values, no reactor cross-header or other valving is permitted during the course of these tests.

In the very remote event that neither the steam nor the electric pumps respond to this demand for increased flow, the high tanks will deliver 13,000 gpm within 7 seconds of demand with the further backup of the export system.

The expected changes produced during these tests upon TORP, flow, and temperatures are summarized in Figures 1 and 2. This illustrates the small effects

of each individual step. These changes will be monitored and the next step will not be ordered by the test coordinator until or unless the preceding step shows cooling adequacy to exist within the test limits.

Included in these figures are the increased temperature effects of maximum decreases in reactor flow attributable to low flow errors. In every case, the test is stopped before these errors become significant.

In the case of Test No. 2, a new Figure 2 must be prepared prior to running this test. This figure to include the results of the low flow calibration test and the results of Test No. 1 to yield a more accurate prediction chart for the more critical No. 2 test.

If for any reason a critical "Y" or other situation of doubtful reliability exists in the BPA or steam power system, then the test will be stopped and the reactor returned to maximum backup flow conditions.

The critical nature of the test is indicated by the curves in Figure 4 which show the maximum time that water can be off of a tube before it reaches 100° C. It is estimated that high-tank water would be available to the reactor within 7 seconds after the request for emergency coolant.

SCHEDULE

Test #1(T_i+15°C) will be run at DR Reactor as soon as can be scheduled after the production test is approved and all detailed procedures are written. Test #2 (T_i + 40°C) will be run at DR Reactor the first scheduled shutdown following the first test. In addition, a low-flow calibration test (P.T.-IP-326-I)[12] will be

(12) Benson, J. L., "Production Test IP-326-I Low Flow Calibration Tests at the Old Reactors," HW-65715, June 20, 1960, SECRET.

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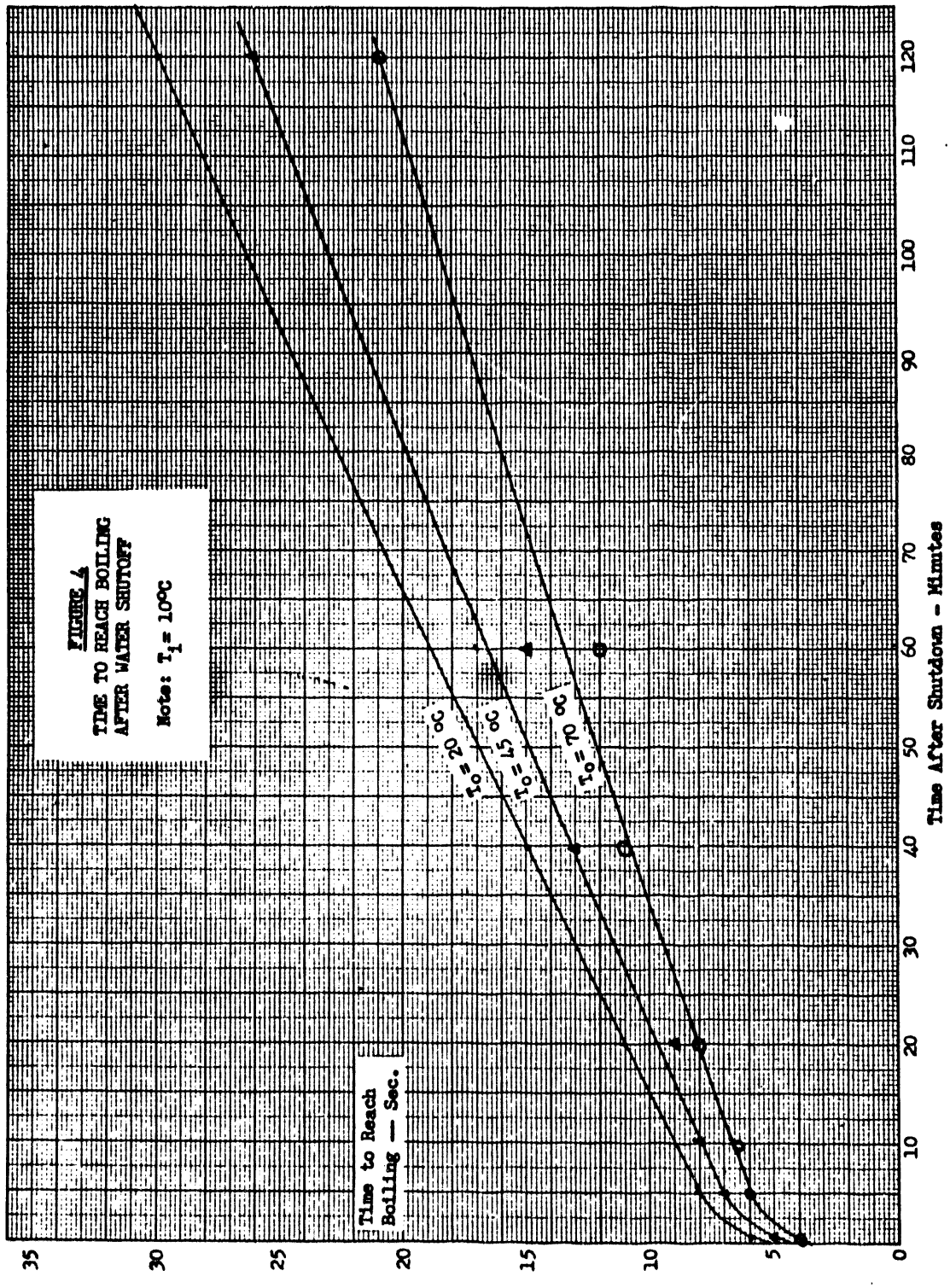

[redacted]

AEC-GE RICHLAND, WASH.

required before the Test #2 is run. This is necessary in order to avoid all possible errors in the 55° C test, which might help cause a dangerous overshoot in temperature, and to provide an accurate flow-pressure calibration curve with which to interpret the test results. This can be run either in conjunction with the 30° C test or before it, whichever is more convenient.

COSTS

A total of 8 hours of outage time is authorized by this test. This is to include the time required for:

1. Test No. 1, for reactor controlled flow scram with 30 ± 5° C outlet water temperature, and subsequent low-flow test at end of shutdown.
2. Reactor Low Flow Calibration Test as per PT IP-326-I.*
3. Test No. 2, for reactor controlled flow scram with 55 ± 5° C outlet water temperature, and subsequent low-flow test at end of shutdown.

It is estimated that this will cost 700 MWD of lost production.

TABLE II

ALLOCATION OF TEST COSTS

Item	Approximate Cost	Unit Incurring Cost	Subsection Responsible For Costs
Instrumentation	$ 500		Process and Reactor Development
Test Operation	500		Process and Reactor Development
Test Operation	100	Photographic	Process Technology

All other costs incidental to running the test will be borne by IPD - Manufacturing Operation on the basis of the anticipated gain in the production capability within the coolant backup criteria at all reactors.

AUTHORIZATION

A low-flow calibration test is authorized by this production test under the conditions specified in "PT-IP-326-I, Low Flow Calibration Tests at the Old Reactors," *

* This low flow test has been completed.

[redacted]

J. L. Benson, June 20, 1960, SECRET.[4] A detailed procedure to cover the performance of this test will be written by the Process Engineering Unit. In addition, this test authorizes not more than three reactor scrams, and subsequent heat output tests.

RESPONSIBILITY

1. D-DR Reactor Operation

   DR Reactor Operation will be responsible for scheduling and overall coordination of this test, for assisting in the collection of data, for the safety and continuity of reactor operation, and for providing the test coordinator.

2. DR Power Operation

   DR Power Operation will be responsible for preparing a detailed test procedure for power personnel, for adequate briefing of personnel, and for supplying the requested reactor riser pressures at the time requested.

3. Process and Reactor Development Subsection

   Process and Reactor Development Subsection will be responsible for technical coordination of the test, for the technical safety of the test procedure, and for reactor safety and continuity of operation during the testing program. They will be responsible for the final report on this production test and for the integration of the test results into the last-ditch program and analysis.

4. Process Technology Subsection

   Process Technology Subsection will be responsible for preparing a detailed operations test procedure, for coordinating the test preparations, assisting in the collection of test data, and for the conduction of a reactor low-flow calibration test and the calibration of steam pump flows.

5. Plant Engineering Operation

Water Plant and Utilities will be responsible for review of power procedures to acertain that proposed throttling will be compatible with individual pump and motor drives as related to current FED programs. Related observations will be made as appropriate.

Equipment Application Engineering will be responsible for installation, calibration, and operation of all appropriate test equipment not a routine reactor instrument or otherwise specified. A detailed list of instruments and their specific relation to the test will be established.

6. Photographic Unit

This unit will be responsible for taking 35 mm movies of the Heise riser pressures in the 105 control room and for taking any required VSR drop movies.

S S Jones

S. S. Jones
Reactor Engineering
Process and Reactor Development
RESEARCH AND ENGINEERING OPERATION

DECLASSIFIED

APPROVALS

J. H. Brown, Manager
Process and Reactor Development
RESEARCH AND ENGINEERING

R. T. Jessen, Manager
FACILITIES ENGINEERING

R. W. Reid, Manager
Process Technology
RESEARCH AND ENGINEERING

D. S. Lewis, Manager
DR Processing
D-DR Reactor Operation
MANUFACTURING

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F. P. Britson, Manager
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W. W. Windsheimer, Manager
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O. C. Schroeder, Manager
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C. A. Pinode, Manager
Production
MANUFACTURING

DATE
FILMED
6/9/94

END