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REACTOR DESIGN ANALYSIS MONTHLY RECORD REPORT
JUNE - 1966

June 21, 1966

R. K. Robinson
Reactor Design Analysis Unit
Process Design Operation
N-Reactor Project Section

This document is classified by:

R K Robinson

HANFORD ATOMIC PRODUCTS DEPARTMENT
INTERNAL REPORT

This document contains information of a preliminary nature, prepared primarily for the use of the Hanford Atomic Products Department and is subject to revision on further checking or collection of additional data.

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Process Design Operation
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REACTOR DESIGN ANALYSIS MONTHLY RECORD REPORT
JUNE - 1966

TECHNICAL ACTIVITIES

A. N-Reactor Characteristics and Behavior

1. Drive Turbine Surface Condenser CRWS Flow Rate - J. Muraoka

Surface condenser CRWS flow rates were calculated which would permit the drive turbines to operate with seven stages. A problem exists because the Elliott Co. contends damage to their drive turbines may be attributable to low vacuum in the surface condensers. They recommend a vacuum which will not allow steam velocities in the seventh stage to exceed Mach 1.5 and have provided a curve relating the steam load to the allowable condenser vacuum. In these calculations the relationship between river temperature and condenser water flow was determined which would comply with the Elliott criteria.

The analysis was performed from a heat balance written for the condenser. The following relationship results:

$$q = w c_p \Delta T = UA \frac{\Delta T}{\ln \frac{T_s - T_i}{T_s - T_o}} \qquad (1)$$

where q = heat load - B/hr

w = CRW flow rate - lb/hr

c_p = specific heat - B/lb F

ΔT = CRW temperature rise - F

T_i = CRW inlet temperature - F

T_o = CRW outlet temperature - F

T_s = condenser saturation temperatures - F

U = overall heat transfer coefficient - B/hr ft^2 F

A = heat transfer area - ft^2

[redacted]

The important term in equation (1) is the overall heat transfer coefficient, U, which in these computations were obtained from engineering data contained in BPF 11276.

1. Surface area - 6000 ft^2
2. No. tubes - 1310
3. Tube dimensions - 7/8" ODX 18 BWG x 20' (Admiralty Metal)
4. Heat load - 127,719,000 B/hr
5. CRWS flow rate - 6800 gpm
6. Inlet water temperature - 60 F
7. Condenser vacuum - 3.50" Hg abs.

Based on these data the design coefficient of 549 B/hr ft^2 F was calculated. Assuming only the water film varies,* the coefficient was expressed as a function of water flow in the following form:

$$U = \frac{1}{.000576 + \frac{1.459}{Q^{.8}}} \qquad (2)$$

where U = overall coefficient - B/hr ft^2 F

Q = CRW flow rate - gpm

From equation (1) and (2), the river temperature (T_i) can be expressed in terms of the condenser vacuum or saturation temperature, water flow and steam load. The Elliott criteria relates the steam load to the condenser vacuum; therefore, the river temperature is actually a function of only two independent variables. The variables used in the analysis are water and steam flows. The procedure used to determine the relationship between the dependent variable and the independent variables was to hold water flow constant and then calculate the river temperature as a function of steam load. The results of these calculations indicate that the maximum allowable river temperature for a given flow occurs at the lower steam loads. In other words the Elliott criteria is most restrictive at low steam loads. The calculations indicate the following river temperature to flow ralationship which complies with the Elliott criteria for all steam loads of interest.

---

*Effectively, it is assumed the steam condensation, film, tube wall and scale resistances are independent of steam flow, water flow and temperature. This assumption is commonly used in this type of analysis and is believed reasonable.

[redacted]

TABLE I

River Temperature F	Water Flow, gpm
35	3000
39	4000
43	5000
47	6000
>50	6800

Table II contains the predicted operating condenser vacuum for these flows and temperatures.

TABLE II

Steam Flow lb/hr	Elliott Vacuum Inches-Hg abs.	Calculated Condenser Vacuum - Inches Hg abs.					
		3000 gpm 35 F	4000 gpm 39 F	5000 gpm 43 F	6000 gpm 47 F	6800 gpm 50 F	6800 gpm 80 F
20×10^3	.33	.38	.39	.42	.46	.50	1.37
40	.61	.69	.62	.62	.64	.67	1.80
60	.88	1.19	.97	.90	.89	.92	2.33
80	1.15	1.99	1.47	1.28	1.21	1.22	3.00
100	1.42	3.20	2.18	1.79	1.58	1.60	3.84
120	1.70	5.03	3.18	2.49	2.17	2.21	4.85

Since the analysis used calculated values for the heat transfer coefficient, the results may not be exact; however, the conclusions drawn from the analysis are believed correct. Thus, with experimental tests and/or operating experience precise operating conditions are believed attainable which would comply with the Elliott criteria.

Finally, computations were made to determine the change in vacuum after loss of CRWS river pump(s). Two cases were examined: First, a transient from four to three river pumps without a scram and second, a transient from four to two pumps with a scram. The worst case occurs during 3000 gpm condenser flow with four pumps in service and 35 F river temperature. In both cases the condenser pressure increases after the transient; however, in neither case is a vacuum trip indicated. If normal steam load is maintained after the loss of two pumps, a trip will probably occur (trip setting 15 inches Hg abs.); however, when two river pumps are lost, a scram is actuated and the steam load will drop immediately to 25 percent. Thus a vacuum trip

 [redacted]

is not indicated. The surface condenser vacuums with four, three and two CRWS river pumps or the equivalent of the normal loss of one pump and loss of two pump cases are shown in Table III.

TABLE III

Steam Flow lb/hr	Condenser Vacuum - Inches Hg abs. Four Pumps (3000 gpm)*	Three Pumps (2250 gpm)*	Two Pumps (1500 gpm)*
20×10^3	.38	.45	.62
40	.69	.94	1.66
60	1.19	1.77	3.90
80	1.99	3.35	8.45
100	3.20	5.88	16.81
120	5.03	9.89	29.92

*Drive turbine condenser flow.

2. Drive Turbine Header Pressure Control - R. K. Robinson

The drive turbine header pressure is currently controlled at approximately 75 psig by controlling the steam flow from the main steam header through the HPV-6210 valves. This pressure is greater than the 55 psig design turbine throttle pressure; however, operating experience has shown that about 75 psig in the turbine header is required to assure safe and reliable operation of the N-Reactor plant. This increased operating pressure is the result of a nuclear of interrelated items.

The turbine header is not, as the name implies, a simple steam header. Instead it is a complex piping arrangement which supplies steam to the 109-N drive turbines, 184-N T-G set, and process and heating steam throughout the plant. The distance and piping lengths connecting these users are quite long, and the pressure drops become significant. For example, the 184-N T-G set and the 184-N deaerator obtain their steam a considerable distance downstream from the 109-N drive turbines. In order to insure sufficient steam pressure at the T-G throttle, it is necessary to increase the pressure at the 109-N drive turbine throttles about 5 psi to compensate for the pressure drop.

The controlling pressure signal for the steam supply from the main steam header originates within the 109-N Building. The pressure signal to the valves from the boiler originates in the 184-N Building. Because of the 5 psi pressure differential between these two points and allowing for another 5 psi control range, it has been found necessary to set the boiler low pressure turbine header pressure override setpoint 10 psi below the controlling setpoint pressure currently attainable with the existing instrumentation is 60 psig. In addition, instability in the 184-N deaerator has been observed at steam pressures below 60 psig. Therefore, the minimum turbine header pressure in the 109-N Building is 60 plus 10 or 70 psig because of control instrumentation limitations.

In March, 1966, the damaged seventh stages were removed from the drive turbines in cells one through five. In order to continue reactor operation at full power it was found necessary to increase the drive turbine throttle pressure to 72 psig to maintain a 3600 rpm pump speed. Allowing for pressure drop and control variations, the 109-N turbine header pressure is controlled at a minimum of 75 psig. It is this item which currently determines the minimum turbine header pressure.

In the event of a reactor scram the turbine header pressure increases approximately 5 psi due to the reduced primary pump drive turbine steam demand. However, this pressure increase is reduced shortly after the system transients have stabilized out.

The minimum turbine header operating pressure as measured in the 109-N Building during normal reactor operation will be limited as follows:

a) In order to maintain 3600 rpm primary pump speed with only six stages in the drive turbines, the minimum turbine header pressure is 75 psig.

b) If seventh stages are added to each drive turbine, the minimum turbine header pressure will be 70 psig. A further reduction is not possible because of current control instrumentation limitations.

c) Modifications to the control instrumentation in conjunction with seventh stages and correcting the deaerator instability at lower pressure would allow an ultimate possible minimum turbine header pressure of 65 psig.

A turbine header pressure below 65 psig is not felt possible because of the pressure differences between the drive turbine and T-G throttle pressures and the need for some margin between the controlled turbine header pressure and boiler override pressure setpoints.

3. <u>Partial Inlet Riser Break</u> - J. Muraoka

Potentially a partial or small rupture to an inlet riser upstream of the check valve could result in fuel-melting. However, to cause melting the hole size must satisfy two conflicting requirements:

a) The hole must be large enough to prevent normal primary coolant flow beyond the break.

b) The hole must be small enough to prevent a rapid depressurization of the primary loop which would permit ERW cooling.

Previous studies by D. D. Stepnewski have indicated that the normal trips may not actuate a dump in this instance. Ultimately, a low pressure trip would occur; however, there is considerable doubt whether a low flow trip would occur simultaneously. Therefore, it was recommended that a containment pressure trip be added to the dump circuit. In this analysis it was assumed the certainment pressure occurs seconds after the rupture; and thus, when the loop low pressure trips, the dump valves begin to open. Twenty seconds were assumed to open the V-4 dump valves.

Presently the loop low pressure trip is set at 375 psia. However, preliminary blowdown calculations for this incident indicate that the setting should be increased to 600 psia minimum. The calculations indicate the pressure decay is fairly rapid to about 500 psia, but then the decay rate falls off appreciably. Thus, with the trip setting at 375 psia, potential fuel melting may occur before a dump is actuated.

A parallel study is currently underway to examine the cooling achieved by flow from the intact risers to the affected riser via the "safety flow leg."

4. HCR Temperature Transient - J. D. Agar

The preliminary results of a temperature transient study of a horizontal control rod indicates that the originally calculated four minutes to melting of the aluminum after a complete loss of coolant without a reactor scram may be excessive.[1] The present results show that it is possible to reach the melting point of aluminum within three minutes in the event of a hose crimp or other sudden downstream coolant flow blockage of the rod. Melting is reached within 1.5 minutes in the extreme case of instant loss of coolant by a break of both up and downstream connectors. Both of these cases assume no reactor scram.

Calculations are currently being made to determine the effect of a reactor scram after loss of rod coolant. The effect of various time delays before a reactor scram is initiated are also being investigated.

---

(1) Letter to D. L. Condotta from M. C. Fraser, "Scram Instrumentation on the NPR Control Rods," June 24, 1960.

5. Increased Reactor Power - R. K. Robinson

In order to permit continued operation of the export plant at 800 MW_e it is necessary to generate steam at 135 psig in the main steam header even at higher reactor power levels. Therefore, the addition of the sixth cell contributes very little to increased power as it is required to generate 135 psig steam pressure at 4000 MW_t.

Assuming that the 800 MW_e generating capability must be maintained, increased reactor power levels can be achieved only by improvements in reactor operating parameters such as the fuel channel enthalpy unbalance, system operating pressure and coolant flow rate. The effect of each of these items is shown below:

Improvement	Percent Gain
a) Increase operating pressure to 1500 psig in the pressurizer and reduce the operating tolerance to ± 75 psi. (This has already been done. Power levels approaching 4400 have been possible except for a specific power limit with the Mark I fuel.)	8-10
b) An improvement in the enthalpy unbalance from 22 percent to 10 percent. (This appears realistic from in-reactor test data.)	10
c) Increased flow from sixth loop over and above that necessary to generate 135 psig steam pressure.	2
d) Increased primary pump speed to 3740 rpm.	3

It can be seen that increased reactor power level up to 4800 MW_t can be achieved without the increased pump speed.

6. Environmental Hazards Study - C. A. Mansius

Dose rates for fission product release from Zone I via the stack were previously reported (RL-NRD-742 5) for line source geometry as a result of light wind conditions. Subsequently, calculations were made for no wind conditions, assuming a 200 foot diameter source. The dose rates calculated for maximum fission product release rates of 1000 second duration for six cases are shown in Table IV below:

TABLE IV

CALCULATED DOSE RATES FOR MAXIMUM RELEASE RATES OF 1000 SECOND DURATION IN 200 FOOT DIAMETER SPHERICAL GEOMETRY

	Distance From Surface of Sphere, R/hr			
	One Ft	100 Ft	600 Ft	1000 Ft
12" Inlet Header Break	13 350	2620	117	36
Maximum Break 2000 cfm Release	3 642	714	32	9.8
Maximum Break 1000 cfm Release	607	119	5.3	1.6
10" Inlet Header Break	18 200	3570	159	49
7.3" Inlet Header Break	10 320	2020	90	28
Maximum Break 10,000 cfm Release	5 460	1020	48	15

The values reported in Table IVare maximum values because maximum release rates were chosen. These are specific cases with low probability of occurrence; but if the condition should arise, a rapid removal of personnel from the area is required to keep accumulated exposure to a minimum.

7. Decontamination Piping Pressure Drop Calculation - J. A. White

Pressure drop calculations have been performed on the decontamination piping leading from the mixing pump located in the 109-N Building to the flushing header located on the rear face of the reactor. The pressure drop across the piping was calculated to be 1.5 feet. The low pressure drop across the piping is a result of the low flow rate which is required (20 gal/min).

CONVERSION PROJECT

1. Transient Analysis - W. G. Conn

A document RL-GEN-1025[2] has been prepared which describes expected N-Plant response to turbine trips and reactor scrams during N-4 tests 5.3, 5.4 and 5.6 operating at steam pressures of 150 psia in the steam generator.

(2) W. G. Conn, "N-4 Test 5.3, 5.4 and 5.6 Transient Predictions," RL-GEN-1025, June 6, 1966 (Unclassified).

This document reports the results of analysis done, using the system Dynasar model and earlier conversion test results, on the transient effects of these system perturbations. Previous documents [3,4,5] have predicted the resulting transients from these system perturbations. However, this work was done prior to any of the conversion tests and hence, was very much on the conservative side. The analysis presented in RL-GEN-1025 was made from a model as up to date as present data allows.

It should be emphasized that instrumentation such as transducers, transmitters, etc. have a finite response time to any change in a system parameter. These response times tend to reduce the rate and magnitude of change of any system parameter during transient conditions. A good example of this was experienced during earlier conversion testing when the indicated steam flow to 185-N following a turbine trip would take in excess of 30 seconds to reach a minimum. The expected shut-off time was predicted to be no more than a few seconds. Considerable time was spent trying to find out where the steam was going following the turbine trip. When attention was finally turned to the flow indicator it was found to have a very slow response time and that what was indicated to be a very slow steam rejection rate following a turbine trip was actually slow instrument response.

2. Failure Sequence Analysis - J. A. White

A failure sequence analysis of the control instrumentation added for conversion has been performed. The analysis covers equipment which has been installed for control of the export steam to the turbines and control of the condensate return from the 185-N Building. A description of the system and potential failures are discussed. It has been determined that failures of a serious nature are highly improbable and that the control instrumentation as designed will perform adequately. A document (RL-GEN-1034)[6] has been issued showing the results of the investigation.

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(3) F. J. Mollerus, Jr., "N-Reactor Conversion Studies of Secondary Cooling System Performance During Phase II Operation," HW-77195, August 6, 1963.

(4) F. J. Mollerus, Jr., "N-Reactor Secondary Cooling System Transients Phase II Operation Conditions," RL-NRD-386, May 25, 1965.

(5) W. G. Conn, "Secondary Coolant System, Phase II Transient Analysis," RL-GEN-909, April 12, 1966.

(6) J. A. White, "N-Reactor Conversion Equipment Failure Sequence Analysis, RL-GEN-1034, June 8, 1966.

[redacted]

WORK FOR OTHER CONTRACTORS

1. C-Reactor Overbore - J. A. White

A study has been completed on replacement of the overbore channels in the block pattern of the production test facility at C-Reactor with standard size process tubes. There were two bore sizes of interest in the investigation. One was the present 1/2-inch overbore, and the other was a one-inch overbore.

The results of the investigation show that graphite sleeves will have to be placed around the process tubes to reduce graphite temperatures within the filler block below the accepted limit (1355 degrees F). The tolerance between the graphite ring and the process tube, and the tolerance between the graphite ring and the overbore channel must be kept below 25 mils.

R K Robinson

Supervisor
Reactor Design Analysis

RK Robinson:jw

Attachment

ADMINISTRATIVE SUMMARY

Personnel Changes

Name	Component	Disposition	Effective Date
Paul J. Waibler	New Hire (Summer Professor)	Reactor Design Analysis	June 20, 1966

Visits - None

Visitors - None

Significant Reports

Title	Title	Author	Date
RL-NRD-742 6 Confidential	"Reactor Design Analysis Monthly Record Report, June, 1966"	RK Robinson	6-21-66
RL-GEN-1025 Unclassified	"N-4 Test 5.3,5.4 and 5.6 Transient Predictions"	WG Conn	6-6-66
RL-GEN-1034 Unclassified	"N-Reactor Conversion Equipment Failure Sequence Analysis"	JA White	6-8-66

Safety and Security - None

Patent Applications - None

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