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RL-NRD-150 8C

This document consists of 19 pages.

MONTHLY RECORD REPORT

RESEARCH AND ENGINEERING OPERATION

N-REACTOR DEPARTMENT

September 30, 1965

M. C. Leverett

Classification Cancelled and Changed To

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By Authority of D. S. Lewis

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By DK-Hanson, 5-11-94

Verified By J. E. Sauerly 5-16-94

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PLANT STATUS

The reactor operated without interruption from September 11 through September 26 at a power level of about 3600 Mw. On September 26 rupture indications in tube 2557 required that the reactor be scrammed. Some difficulty was experienced in obtaining proper operation of the diversion valve and as a result contamination levels in the primary loop are considerably higher than had been experienced in earlier ruptures where the diversion valves worked properly on first attempt.

The ruptured fuel element was easily identified. It had occupied the fourth position from the downstream end of the column and the outer member of the fuel element was very obviously the site of the rupture. It is estimated that this fuel element had accumulated an exposure of 2170 MWD/T. The ruptured element is presently in the fuel storage basin and will be moved to the 300 Area for radiometallurgical examination.

The occurrence of the fuel rupture on September 26 led to activation of the plans which had been made for an 18 day outage starting on September 27. Charge-discharge, maintenance work and WPPSS tie-in work will be accomplished during this outage period.

Reactor Physics

On September 24 at 8:00 a.m. the reactor power level was 3600 Mw; the reactivity held in the rods was 3.25 mk; the flattening efficiency was approximately 80 percent; the hottest graphite was 1036 F. The average fuel exposure at that time was 1581 MWD/T.

The modification to the equilibrium rod pattern reported last month resulted in stable, cycle-free operation during September. The ratio of peak-to-average power along the fuel columns was about 1.55 at equilibrium, permitting operation at full design level within specific power limits.

TECHNICAL ACTIVITIES

Coproduct Fuel Element Comparisons

A ceramic (aluminum oxide, lithium oxide target) has been selected for coproduct use. As indicated in Table I, four basic types of targets had been under consideration. Three of these contained a lithium aluminum alloy as the fertile material, and the fourth contained aluminum oxide, lithium oxide mixtures. Three of the targets were to be clad in Zircaloy, and one in Niobium, which it is believed would be strong enough to contain any conceivable gas pressure resulting from melting of the contained lithium-aluminum target. Three integral and one target configurations were being considered. Table I summarizes the configurations and materials

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considered and lists the criteria by which the various designs and targets were compared. The aluminum oxide, lithium oxide target was selected primarily on the basis that it offers maximum security against the target loss accident and that it does not preclude reaching 4800 Mw reactor power. None of the other candidates were as acceptable in these two respects.

The ceramic target has been under consideration for approximately two years, but only within the last two months has it become apparent that recovery of the tritium from the target can be accomplished by dissolving the target in molten sodium tetraborate in the Savannah River extraction equipment. Certain modifications of the Savannah River plant equipment and the establishment of the actual process parameters and conditions by the Savannah River Laboratories would be required to effect the program for the processing of these targets.

Table II compares the various targets in regard to the technical problems which they present. The key problems of the selected alternative are to further develop a process for tritium recovery for ceramics, and to develop uranium with irradiation temperature capabilities of 600 C. The 600 C uranium program has been underway for approximately two years, and there are no indications that it will not be successful. The basics of the tritium recovery process are now known although refinements and exploration of variations in key parameters will still be necessary.

Ceramic Targets - Extraction Studies

A material balance has been achieved for the first time in an extraction run on irradiated lithium aluminate. The sample was very small (≈ 1 gram) and the GVR was very low (≈ 1). The work, however, was carefully done and the results were internally consistent.

Operation	Percent Release		Product
	T	He	
Irradiation	29	75	98% T ₂
325 C Anneal	5	10	90% T ₂ O, 10% T ₂
Extraction*	65	15	88% T ₂ O, 12% T ₂
Residue	1	-	100% T ₂ O

* The extraction was carried out by dissolution in sodium tetraborate at 800 C for ten hours.

The rate of release during dissolution was not measured.

TABLE IComparison of Targets by Technological Aspects

<u>Configuration</u>	<u>A</u> <u>Integral</u>	<u>B</u> <u>Integral</u>	<u>C</u> <u>Integral</u>	<u>D</u> <u>Separate</u>
Clad Material	Zr	Zr	Nb	Zr
Core Material	Al-Li	Al ₂ O ₃ -Li ₂ O	Al-Li	Al-Li
Freedom from Susceptibility to Target Loss Accident	Poor	Good	Fair	Good
State of Technology of Target Perf. and Mfr.	Good	Fair	Fair	Fair
State of Technology of Product Recovery	Good	Fair	Good	Good
T Production Penalty	None	None	2-5%	2-5%
Time Required to Put Technology on Firm Footing, Months	None	6-9	2	2
Ability to meet HAPO Nuclear Safety Criteria	Poor	Good	Fair	Good
Freedom from Dependence on 600 C Uranium	Poor	Poor	Poor	Fair
Fuel Basin Poisoning Required	No	No	No	Yes
Status of Physics	Good	Good	Good	Fair
Reactor Orificing Required	No	No	No	Yes

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TABLE II
COMPARISON OF TARGETS IN RE KEY PROBLEMS

<u>Problem</u>	<u>Configurations and Composition*</u>			
	<u>A</u>	<u>B</u>	<u>C</u>	<u>D</u>
Devise Process for T Recovery from Ceramic		X		
Prove Target Loss Incredible or Harmless	X		X	
Show unnecessary to meet HAPO Total Control Criteria	X		X	
Develop 600 C Uranium	X	X	X	
Poison Fuel Basin				X
Develop "New" Physics				X

* As shown under A, B, C and D in Table I.

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Coproducer Test (1.25% Enriched)

Four columns of this material remain in the reactor. The exposure of the remaining columns is now about 1600 MWD/T (2400 maximum). One column is scheduled for discharge during the current outage. It will be weighed and examined before the next startup. The thermocouple data indicate: no change in the thermal hydraulic balance has occurred. Pertinent results are shown in Figure 1.

Coproducer Test (1.96% Enriched)

The 15 remaining columns of PT-NR-13 coproducer elements have reached an average exposure of about 500 MWD/T.

Fuel Element Performance Evaluation

The main effort during September has been to interpret and refine uranium swelling values calculated from post-irradiation weighing data. After applying appropriate corrections the data shown in Figure 2 were obtained. These data show that up to about 2000 MWD/T fuel swelling follows the KER loop experience (gained before the reactor was put into operation). Fuel now in the reactor will provide the principal evidence for the amount of swelling to be expected at higher exposures.

Four fuel columns were discharged during the month because of rupture indications. Of these, only one has been confirmed. In the other three cases, the ruptures, if present, are so small that with present techniques it is not possible to identify which of the fuel elements in the column has the rupture. More refined techniques are being developed.

Fuel Element Integrity Tests

During previous heating and quenching tests on irradiated N fuel elements, two irradiated inner and two irradiated outer fuel elements were heated to 1800 F (982 C) to simulate the loss of primary coolant and then quenched in 20 C water to simulate the introduction of emergency raw water.

Two more heating tests have now been completed. Both tests have shown that the threshold temperature for fuel element rupture in a thermal cycle is 1030 C or higher. It has been found that the uranium, once exposed, reacts rapidly with the available steam. No water quenching tests on failed elements at high temperature have yet been done.

Simulated Driver-Target (SDT) Elements

The campaign for extrusion of both SDT and standard outer fuel tubes from 8.1 inch liners was started September 7. The SDT material was extruded first and included all the alloys under consideration, though predominantly

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Percent of Enthalpy Balance

Figure 1
Thermocouple Train Measurements in 1.25 Percent Test Fuel

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Figure 2

Comparison of NPR Inner Uranium Swelling with Inner
Uranium Swelling in KER Tests

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the "British" composition (800 ppm Al, 400 ppm Fe). Primary uranium billet fabrication was varied to include standard processing, pierced ingots, upset forged material, and direct cast stock.

Inner and Outer Tube Power Split (Standard N Fuel)

Results from this experiment which were based on cesium-137 concentration determinations have been reported by BNW. The ratio of power per unit weight of the inner to the outer element is reported as $771 \pm .027$ (3.4%) for six separate sets of samples. The FLEX calculated value is 0.76. The 1 σ limits on the experimental data are $\pm .0205$, while the FLEX value differs from the experimental data by -0.01. In view of these small differences, it will not be necessary to modify the FLEX program for closer agreement with the empirical data.

Fuel Element Spacer Misalignment

Indications of low flow in isolated process tubes which have occurred from time to time in the past have been definitely traced to improper seating of the charge in the process tube. The downstream spacer (dummy) is in this case susceptible to misalignment by the hydraulic forces in the exit nozzle, and if misaligned can partially block the exit port. This has happened on a number of occasions and has caused fuel columns to operate at temperatures which would normally indicate unacceptable fuel element swelling. In some cases such columns have been discharged and subsequent post irradiation measurements have shown that the level of swelling in the fuel elements was actually well within the acceptable ranges. The downstream spacer is also responsible for about half the pressure drop across the outlet nozzle. The spacers are therefore being redesigned to prevent such misalignment and to improve their flow characteristics.

Systems Analysis

Analogue computer simulations were made in support of the Dual Purpose (N-4) test program to determine the effect of reactor scrams on the change in pressurizer level in relation to primary flow after coastdown and the number of inoperable steam generator cell bypass valves. The results are summarized in Figure 3. With no bypass valve failures, the level change could be reduced by about four to five feet, if the percent primary coolant flow after a scram could be changed from the present 33 percent to the design value of 25 percent. Recent scram results indicated that percent primary flow after a scram tends to be increasing and this adversely affects the primary coolant contraction rate. A production test has been written which authorizes the testing of methods for reducing the primary drive turbine speed in order to improve the over-all flow balance.

Figure 3 also illustrates the effect of cell bypass valve failures. For every valve failure, the pressurizer level will drop an additional 2.5 feet

following a scram. Usually, such failures are temporary since the valve will be repaired in a subsequent reactor outage. However, in the interim, it is necessary to reduce the reactor power level to offset additional volumetric contraction induced by the valve failure. As indicated in Figure 3, for one bypass valve failure at a pump coastdown to 33 percent, the pressurizer level just reaches the "zero" indicated level. While the actual bottom of the pressurizer is about four feet below the indicated zero level, the actual volume of water is not very large because of the space occupied by the heaters. The zero indicated level should therefore be considered to be the absolute limit for a pressurizer minimum level.

The estimated power levels for one, two, and three bypass valve failures to give a post-scram pressurizer level change equivalent to a scram from 4000 Mw and to valve failures are given in Figure 4.

Process Tube Monitoring Equipment

Testing of the ultrasonic hydride probe has shown that it can effectively detect 275 ppm of zirconium hydride. After minor modifications the probe will be transferred from BNW to N-Area for in-reactor testing.

Graphite Coolant Tube Removal

Two graphite cooling tubes were removed from the reactor during the current outage. Laboratory evaluation of these tubes is underway.

Process Tube Removal

The tooling for process tube removal has been developed and tested. Detailed work procedures for removal and installation are being prepared.

Environmental Effects of Melting of Fuel in the Core Fringes

The quantity of fuel in the fringes of the core which would melt after a complete loss of both the primary and emergency raw water coolant systems has been calculated. The moderator system was assumed intact and functioning normally for the purposes of these calculations. Melting of the fringes of the core would begin about 1-1/2 hours after the onset of an accident and would continue for a period of three days. The dose at the exclusion boundary from noble gases released promptly during an accident was calculated as 25 rem. At that time the dose from noble gases released during melting of the core fringes was not included. The dose at the exclusion boundary, calculated as the cumulative dose for the first two hours after onset of an accident, would not be increased by fringe melting.

The added whole body dose from melting of the core fringes is calculated to be less than 2 rems at 9 miles (low population zone boundary) accumulated over a period of one week. This is less than the conservative margin in the

original estimate. Thus, neither the exclusion zone boundary distance nor the low population zone boundary would be changed by including the effects of fringe melt.

Technical Bases for Nuclear Safety Specifications

Critical mass parameters for all currently used fuels have been tabulated, and are being documented. A follow-on effort now in progress involves a recalculation and confirmation of the critical mass parameters for exposed fuel.

Boiler Load Reduction Program

Based on the results of testing performed under production test PT-NR-24, a base load reduction to a minimum of two firing burners with three available for standby service and a minimum base load of 60,000 lb/hr was achieved on September 14. The boiler is being operated on a manual load-following basis with a minimum of five serviceable burners prepared to accept load following manual ignition should the reactor plant scram. Special procedures and process change authorizations have been issued to permit plant operation down to a condition of two burners in service with the plant service boiler set for manual operation following a reactor scram. No significant difficulties have been experienced with this mode of operation. The relaxation in boiler availability requirements which resulted from this study did eliminate the need for a reactor plant scram on September 15 when a critical Y power condition was declared as a result of boiler difficulties. The current operating restrictions permitted restoration of the boiler before a scram was required.

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Figure 3

Pressurizer Level Change Following Scram
From Full Power Versus
Percent Primary Flow After Coastdown for
Zero, One, and Two Steam Generator Cell
Bypass Valve Failures

Figure 4

Maximum Power Level to Give Equivalent
Pressure Level Change Versus
Number of Inoperable Steam Generator Cell
Bypass Valves

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PT No.	Document No.	Title, Author, Date	Status			
			Approved	In Progress	Completed	Final Report
2	HW-80369	Routine Graphite Sample Irradiations		X		
3	HW-81478	Routine Monitoring of Graphite Oxidation in N-Reactor		X		
4	RL-NRD-218	Evaluation of Monitor Column Fuel Elements in N-Reactor		X		
6	HW-81029	NPR Shield Evaluation, Shield Plug Measurement During N-2			X	BNWL-135
8	HW-81327	Coproducer Demonstration Test (1.25)		X		
9	HW-81609	Low Goal Irradiation of Fuel Elements with Varying Amounts of White Oxide on Surface			X	
10	HW-82136	N-Reactor Corrosion Monitoring	X			
12	HW-82241	Copper Base Alloy Corrosion Program		X		
13	RL-NRD-299	Coproducer Demonstration Test, (1.96)		X		
14	HW-82385	Irradiation of Th-U Crud Monitor Elements in N-Reactor			X	
14 SUP SUP 1	HW-82385	Supplement A to PT-NR-14, Irradiation of 8.47" long Fuel Element Assemblies.			X	
18	HW-82659	Primary Loop Pressure Relief Valves (RV-2-1 and 2) Relief and Reseating Pressure Test			Partially Complete	
20	HW-82933	Hydraulic Actuators for RWSV-805 Valves			X	
20 SUP SUP A	HW-82933	Supplement A to PT-NR-20			X	
27	HW-84238	Steam Generator Moisture Carryover Detection			X	

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PT No.	Document No.	Title, Author, Date	Status			
			Approved	In Progress	Completed	Final Report
27 SUP	HW-84238 SUP 1	Supplement 1 to PT-NR-27			X	
30	HW-84232	Evaluation of Ammonia for Controlling Oxygen in the Graphite Cooling System			X	
30 SUP	RL-NRD-196 SUP	Supplement A to PT-NR-30			X	
32	HW-84367	Exposure of Corrosion Coupons in N-Reactor Process Tubes		X		
33	HW-84401	In-Reactor Enthalpy Imbalance Measurements		X		
33 ADD	HW-84401 ADD 2	Addendum 2: Eccentricity Measurements	X			
36	RL-NRD-55	Simulated Driver-Target Elements		X		
38	RL-NRD-193	Eval. of N Rupture Monitor Modifications for Desensitizing the Response of the Gamma Energy Monitor to Activated Corrosion Products			X	
41	RL-NRD-297	Evaluation of the Direct Addition of Ammonia to the Graphite Cooling System		X		
43	RL-NRD-379	Testing of Converted Reactor Plant		X		
45	RL-NRD-390	Increase in Primary Coolant System Pressurizer Pressure	X			
46	RL-NRD-427	Primary Drive Turbine Speed Reduction Tests	X			Partially Complete
47	RL-NRD-426	Horizontal Rod Coolant Channel Purging	X			
48	RL-NRD-466	Corrosion Test Samples in End Caps of the Graphite Cooling System		X		

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PT No.	Document No.	Title, Author, Date	Status			
			Approved	In Progress	Completed	Final Report
49	RL-NRD-492	100-N Area A Bus Power Supply Staged Transient Loading	X			
50	RL-NRD-504	V-12 Diversion Valve Actuation Timing			X	RL-NRD-544
52	RL-NRD-511	Fast Indexing Drive Modification to Rupture Monitor Turrets		X		

PERSONNEL CHANGES

Additions

Belva Pleake, Secretary for Advanced Technology, September 28.

J. R. Bolliger, Engineer, transferred from Relations, September 15.

L. L. Sims, Engineer, new hire with Process Engineering, September 7.

J. J. Dorgan, Engineer, new hire with Process Engineering, September 13.

T. V. Tung, Engineer, new hire with Thermal Hydraulics, September 16.

Water Quality Laboratory transferred from Plant Section to Research and Engineering; including the following personnel:

K. L. Fowler, Supervisor

Chem. Technologists

Chem. Analysts

C. O. Harvey, Chemist, Analytical

B. J. Beehler

D. V. Archer

W. L. O'Daniel

M. L. Hadley

B. E. Pieterston

H. Latal

R. W. Sanders

J. R. Kerby

D. L. Moore

Internal Transfers

L. D. Smith transferred from Process Engineering to Thermal Hydraulics, September 1.

Reductions

F. D. Frisch transferred to Plant Section, September 15.

V. J. DeJong, Engineer, terminated to return to school, September 24.

SECURITY VIOLATIONS

R. J. Shields, open file, September 17.

L. C. Clossey, open file, September 18.

INVENTIONS

All Research and Engineering personnel engaged in work that might reasonably be expected to result in inventions or discoveries advise that to the best of their knowledge and belief, no inventions or discoveries were made in the course of their work during September 1965. Such persons further advise that for the period herein covered by this report, notebook records, if any, kept in the course of their work have been examined for possible inventions or discoveries.

L C Clossey for
Manager
Research and Engineering

MC Leverett:LCC:vb

SIGNIFICANT REPORTS

<u>Report Number and Class.</u>		<u>Title</u>	<u>Author</u>	<u>Date</u>
RL-NRD-349	C	Hazards Evaluation, Use of Nitrogen in N-Reactor Gas Atmosphere	EE Leitz	May 3, 1965
RL-NRD-466	U	PT-NR-48, Corrosion Test Samples in End Caps of the Graphite Cooling System	WM Scott	August 18, 1965
RL-NRD-466 APP	S	Appendix to PT-NR-48, Prototype Cap Testing	WM Scott	August 18, 1965
RL-NRD-492	U	PT-NR-49, 100-N Area A Bus Power Supply Staged Transient Loading	LD Smith RC Hoffman	August 6, 1965
RL-NRD-505	C	N-Reactor Heat Balance and System Parameters, July 22, 1965	JV DeJong et al	August 12, 1965
RL-NRD-513	U	FLEX-3R, A Fuel Element Design/Analysis Code	EE Mills RJ Shields	August 27, 1965
RL-NRD-527	U	DAWN - A One Group Diffusion Code for the Calculation of Critical Parameters from Material Buckling	DG Stenstrom	September 3, 1965
RL-NRD-528	U	PT-NR-24, Plant Service Boiler Base Load Reduction Program, Process Change Authorization	RE Hall	August 30, 1965
RL-NRD-532 PT 1	S	Feasibility Study No. 6502, Process Control Equipment	CF Poor and DW Leiby	September 3, 1965
RL-NRD-532 PT 2	U	Feasibility Study No. 6502, Appendices	" "	" "
RL-NRD-543	S	Fuel Element Failure Report, Process Channel 2747	AJ Ebens	September 9, 1965
RL-NRD-551	U	Trip Report, GERL, Schenectady, New York and U.S. Naval Research Laboratory, Washington, D.C.	WM Scott	September 16, 1965
RL-NRD-556	C	Primary Loop Coolant Shrinkage Following a Scram	EE Leitz	September 20, 1965

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VISITS

<u>Name</u>	<u>Company and Location</u>	<u>Contact(s)</u>	<u>Date(s)</u>	<u>Remarks</u>
RL Dickeman MC Leverett	AEC-Washington, D.C. Budget Division		9/1-2/65	Budget discussions
RL Dickeman MC Leverett	GE Missile and Space Division, Valley Forge, Pennsylvania		9/3/65	Reliability discussions
TW Evans	AIME, Asheville, No. Carolina	RF Dickerson	9/7-10/65	Attend symposium on U swelling and radiation effects
TW Evans	Savannah River Laboratories	WR McDonell	9/10/65	Swelling of U alloy fuel elements
MC Leverett	APED, General Electric Co. San Jose, California	OH Greager	9/13-16/65	Safety Review group
M Lewis	Savannah River Laboratories, Augusta, Georgia	AA Johnson	9/15-17/65	Discuss separations technology

VISITORS

ACRS Full Committee Meeting		RLOO-AEC	9/9-10/65	NRD presentation to ACRS on September 10.
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