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GENERAL ELECTRIC

HANFORD ATOMIC PRODUCTS OPERATION - RICHLAND, WASHINGTON

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MONTHLY RECORD REPORT
RESEARCH AND ENGINEERING OPERATION
N-REACTOR DEPARTMENT

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This document consists of 18
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MONTHLY RECORD REPORT
RESEARCH AND ENGINEERING OPERATION
N-REACTOR DEPARTMENT

July 31, 1965

M. C. Leverett

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By DK Hanson, 5-11-94
Verified By JE Savely 5-16-94

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PLANT STATUS

Following completion of the extended outage which started June 15, the N-Reactor was taken to critical on July 8. During this outage, all conversion tie-in work required for the demonstration of 150 pound steam operating capability (Phase A tests) was successfully completed. Prior to the startup on July 8, all necessary pre-operational testing was performed and approvals were secured to permit initiation of the operational testing. Operational testing is continuing with 90 pound steam in the main header and a programmed scram from full power is planned at the first convenient opportunity. Pressure in the main header is then scheduled to be increased to 155 pounds and upon successful demonstration of stable operation at this pressure, the Phase A testing program will be completed. No significant problems have been encountered in this testing program to date.

At the time of this report (July 27) the N-Reactor was operating at 4000 Mw with 83 percent flattening efficiency and 6 mk excess reactivity in the control rods. A loss of about 3 mk excess reactivity is expected by September 1. At this point it will be difficult to maintain a flattening efficiency greater than 80 percent and modest power reductions will be required to comply with the 5000 kw tube power limit. As reported last month, this course is economically preferable to adding more spike enrichment in order to enhance flattening efficiency.

On July 23 a second scram was initiated because of a fuel rupture indication from column 0756. The suspect column was discharged without incident and operation was resumed early on July 25. Examination of the fuel after discharge produced no evidence of a rupture. At the time of writing, it appears that the rupture indication may have resulted from instrument malfunction or from activated debris from prior ruptures. Steps are under way to upgrade rupture monitor operation and to improve the quality of data available.

Further testing of the reliability of the automatic light-off features of the plant service boiler was performed during the June outage. Modifications were completed and the four burner light-off sequence test was successfully performed. Test performance also demonstrated satisfactory operation of the minimum load/pressure control systems.

Based upon test results, a planned reduction in the base load to four main burners in service with a minimum load of 120,000 lb/hr was permitted on July 16.

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TECHNICAL ACTIVITIES

Coproduct Physics Studies

The physics behavior of the base case separate target configuration is being characterized by several calculational techniques. The data indicate that the use of the FLEX program to generate cell constants, followed by either one or two dimensional diffusion theory analyses of the mixed lattices, yields the best information.

Additional calculations with thermal transport theory (THERMOS) yield target cell flux distributions which match very well those obtained from FLEX. These calculations, however, have not yet been incorporated into cell reactivity calculations.

These calculations show that the separate target configuration will exhibit cold-to-hot reactivity decreases within the strength of the control rod system for the cases so far investigated.

Target Melting Calculations - Separate Target Coproduct Concept

Calculations have been completed of the temperature transient followed by the separate target, under the most severe reactor conditions. The results are plotted in Figure 1.

1. Time to target melting - 1690 seconds
2. Graphite temperature when target reaches melting - 1308 F
(target melts at 1215 F)
3. Time at which graphite reaches target melting point - 1076 seconds

Critical Mass Parameters

Critical mass parameters for the base-case separate target design have been calculated up to 2.0 percent enrichment, and an assessment of fuel handling problems is in progress. Calculations of critical mass parameters at still higher fuel enrichments is also under way.

Stack Flooding Calculations

Calculations of the reactivity effect of stack flooding in the coproduct-loaded reactor are under way. Initial calculations indicate that the flooding accident may be within the control range of the rod system alone.

Coproduct Metal Handling

Installation of the equipment for disassembling, decapping, and stripping the coproduct fuel started during the week of July 23. The cask for shipping targets has been received on-site.

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FIGURE 1
TARGET TEMPERATURE TRANSIENT FOLLOWING COOLANT LOSS, SEPARATE
DRIVER-TARGET CONCEPT

RL-NRD-150 7

Coproducer Demonstration Test (1.25 Percent Enriched)

Irradiation of the test block of 1.25 percent enriched driver elements for the coproducer demonstration test (PT-NR-8) was completed. All except four columns were discharged during the June outage. The four remaining columns are equipped with special thermocouple trains from which additional data are desired.

Thermal Hydraulics

The hydraulic inputs to the FLEX-3 computer code are being re-evaluated by actual laboratory measurements to obtain improved information relative to the thermal balance among subchannels at the outlet of the fuel column. The fractions of total flow carried by the three subchannels obtained in the laboratory (under cold, isothermal conditions) are compared with FLEX calculations in the following table.

COMPARISON OF MEASURED AND CALCULATED
FLOW FRACTIONS

<u>Subchannel</u>	<u>Fraction of Total Flow in Subchannel Laboratory</u>	<u>FLEX-3*</u>
Outer Annulus	0.482	0.466
Inner Annulus	0.440	0.458
Center Annulus	0.079	0.076

*With current inputs for fuel support drag coefficients

When analysis of the data is complete the inputs to FLEX will be adjusted to more closely match values deduced from the experimental and in-reactor results. Empirical corrections such as these help to improve the overall reliability of the FLEX-3 computer code.

Primary Cooling System Supply Characteristics

Calculation of the reactor cooling supply characteristics has been completed for anticipated reactor operating conditions. These characteristics are shown in Figures 2 through 4, and include minimum reactor inlet temperatures required to produce Phase II steam pressures over the range of cooling system flow rates corresponding to the conditions shown in Figures 2 and 3. As noted in Figure 2, the hydraulic resistance of the current tube-in-tube fuel would be compatible with cooling system characteristics at 4800 Mw (120 percent of design). Redesign to improve thermal balance should therefore produce a fuel that would operate satisfactorily at the increased power without a sacrifice in weight.

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FIGURE 2
SUPPLY AND DEMAND CURVES

RL-NRD-150 7

FIGURE 3
HEADER-TO-HEADER PRESSURE DROP CURVES

FIGURE 4
REACTOR INLET TEMPERATURES REQUIRED TO PRODUCE 110
AND 150 PSIA STEAM IN THE SHELL SIDES OF STEAM
GENERATORS AT 100 AND 120 PERCENT OF REACTOR DESIGN
POWER

Fringe Tube Orificing

The installation of orifices in 300 fringe zone tubes has provided a better matching of flow distribution and power distribution in the N-Reactor. As a result 110 psia steam pressure in the steam generators was achieved at 4000 Mw without significant difficulty.

Fuel Tube Power Split

Determination of the power split between inner and outer fuel tubes from cesium analysis has been accomplished. These data indicate a power split within approximately two percent of that calculated by FLEX-3 computer code.

Simulated Driver-Target Test

Four columns of simulated driver-target elements have been charged. All were made with the 350 Fe, 350 Si, 800 Al uranium alloy. They will provide a test of this third candidate alloy in conjunction with the concurrent testing of the standard and British (350 Fe and 800 Al) alloys used for the 1.96 percent enriched coproduct test fuel.

Fuel Element Failure Evaluation

Metallographic examination has been completed on failures 2 and 3. Two failure mechanisms have been proposed to explain the observed facts. One postulates that a foreign object lodged against the buggy spring and that these foreign objects then penetrated the cladding by fretting corrosion. The other postulates that failure originated from a flaw in the cladding, probably from a hard inclusion. Steps have been taken to minimize the probability of a recurrence of either mechanism. A thorough search for cladding flaws has been initiated and the capability of detecting them has been improved. The nature and number of foreign objects that are in the primary coolant is under investigation. A large number of non-failed, irradiated fuel elements are being inspected for any evidence of incipient failure.

In no case does the analysis indicate that failure is from a design deficiency.

1.96 Percent Integral-Target Production Test

Initial operating data indicate that the specific power in the 1.96 percent enriched block load is considerably lower than the worst-case calculations had predicted. No special control measures are therefore being applied. Further analysis of operating data is in progress to determine the desirability of adding spike enrichment to boost the 1.96 powers up to those predicted.

Conversion Ratio Tests

Three six-ton batches of N fuel elements have been dissolved, sampled, and are undergoing conversion ratio analysis. Results are expected during early August.

Fuel Examination Facility (CGN-149)

The fuel examination basin was filled with water so that visual inspection and weighing could be started on nine fuel columns recently discharged from the reactor. These consist of five monitor tubes of 0.95 percent enriched fuel, tube 1258 in which a 1.25 percent enriched fuel had ruptured, and three tubes of 1.96 percent enriched fuel. Examination of the 1.96 enriched fuel was completed, and the monitor fuels are currently being examined and weighed for uranium swelling calculations.

Equipment installation in the examination basin was halted when the basin was filled. It is planned to drain the basin early in August to complete this work. As of July 1, the Special Handling and Examination equipment project was 67.5 percent complete.

Process Tube Monitoring

The I.D. probe, TV and depth gauge probe, scrubber probe and ultrasonic hydride probe will be ready to use in the reactor by September. The process tube which is to be removed will be tested in the reactor by these probes prior to its removal. Subsequent sectioning and testing of the process tube in the laboratory will be used to verify the performance of the process tube monitoring equipment.

The strain gauge annulus probe was delivered and has been satisfactorily tested.

Corrosion Monitoring

The 184-N boiler was inspected and found to be in good condition. A thorough study of primary coolant system water treatment and corrosion susceptibility has been initiated to more precisely define actual corrosion levels and causes. Inspection of this system during the outage indicated the presence of considerable rust within the system.

I-131 Concentrations in River Influent Following Ruptures

When fuel element ruptures occur in N-Reactor the primary coolant in the process tubes in which the rupture occurs is diverted to the 1301-N crib, a large (125 foot x 290 foot) rock filled pit designed to allow water to slowly percolate into the ground.

The coolant water of course contains some fission products. The soil under the crib acts to delay the movement of chemicals, thus delaying the movement of fission products so the specific concentration is reduced to low levels by the time fission product bearing ground water reaches the Columbia River. The delay afforded by the filtering process allows short half-lived radionuclides such as iodine-132 to decay away.

Some nuclides are essentially, permanently retained in the soil, but a few, principally iodine and some complex cobalt compounds, move quite rapidly through the soil. Iodine is estimated to take 90 days to travel from the crib to the river. Iodine-131 has a half-life of eight days, long enough so that significant concentrations could reach the river. However, calculations indicate that many fuel element failures could be tolerated without exceeding recommended limits (AEC Chapter 0524 and 10 CFR 20). Iodine-131 is the first biologically significant radionuclide to reach the river following a rupture. A single rupture could possibly result in concentrations that temporarily exceeded the allowable yearly average for I-131 in water solution. However, the concentration may be averaged over a year and unless ruptures occurred almost daily the I-131 concentration would not exceed the average limit.

Iodine-131 Concentration in River Water

	Concentration Averaged over One Year $\mu\text{Ci/cc}$	Peak ⁽¹⁾ Concentration $\mu\text{Ci/cc}$
From one rupture (complete loss - one element)	2×10^{-9}	7.4×10^{-5}
10 CFR 20 Limit (uncontrolled area)	2×10^{-6}	--
AEC-0524 Limit (uncontrolled area)	3×10^{-7}	--

- (1) Peak (maximum) in time. Average over stream lines in ground water. Concentrations in ground water adjacent to the river is at the point of entry.

The quantity of radionuclides discharged to the crib following a single rupture exceeds 10 CFR 20 ground burial limits but does not exceed either 10 CFR 20 or AEC Chapter 0524 release limits to uncontrolled environs provided that coolant water is filtered through the crib before reaching the river.

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No significant quantity of radionuclides was released to uncontrolled areas as a result of N-Reactor operation during the period June 25 to July 28.

Advanced Studies

Production of Np²³⁷ and Pu²³⁸ in N-Reactor

The potential for increasing the production of Np²³⁷ has been analyzed for N-Reactor by recycling the reactor fuel after it has been blended (i.e., U²³⁵ content of the discharge material has been upgraded to the necessary assay) with enriched material from various sources. The results of this study are for an operating mode of 2100 MWD/T exposure using the 21 lb/ft tube-in-tube fuel design.

The yearly, incremental production of Np²³⁷ and Pu²³⁸ by recycling the N-Reactor fuel streams was calculated.

The results for this reactor operating mode (21 lb/ft tube-in-tube fuel at 2100 MWD/T) are tabulated below.

<u>Blend Enrichment % U-235</u>	<u>Source of Blend Material</u>	<u>Incre- mental Np-237 (kg)</u>	<u>Incre- mental Pu-238 (kg)</u>	<u>Incremental Unit Cost Pu-238 (\$/g)</u>	<u>Total Pu-238* (kg)</u>
3.4	Cascade	120	69	270	131
1.7	N.F.S.	163	82	178	144
86.5	1st cycle VIB	169	96	330	158
78.0	2nd " "	210	120	280	182
70.0	3rd " "	263	150	241	212

*If no recycle program were assumed, the total production for the campaign would be: 104 kg Np-237 and 62 kg Pu-238.

Reactor power level - 4000 MWT

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In-Plant Tests

The status of production tests in the N-Reactor is as follows:

PT No.	Document No.	Title, Author, Date	Status			
			Approved	In Progress	Completed	Final Report
2	HW-80369	Routine Graphite Sample Irradiations, DH Curtiss, 1/3/64		X		
3	HW-81478	Routine Monitoring of Graphite Oxidation in N-Reactor, DH Curtiss, 3/26/64		X		
4	RL-NRD-218	Evaluation of Monitor Column Fuel Elements in N-Reactor, AE Guay, 1/20/65		X		
5		(rough draft - not issued)				
6	HW-81339	The Use of Isotope Producing Rods in N-Reactor, RA Chitwood, 3/17/64		X		
7	HW-81029	NPR Shield Evaluation, Shield Plug Measurement During N-2, J Greenborg, RA Bennett, 3/2/64			X	
8	HW-81327	Coproducer Demonstration Test (1.25) TW Evans, 5/4/64		X		
9	HW-81609	Low Goal Irradiation of Fuel Elements with Varying Amounts of White Oxide on Surface, AC Callen, 3/30/64			X	
10	HW-82136	N-Reactor Corrosion Monitoring, RJ Evans and BS Kosut, 5/25/64	X			
12	HW-82241	Copper Base Alloy Corrosion Program, BS Kosut, 6/4/64		X		
13	RL-NRD-299	Coproducer Demonstration Test, (1.95) EG Pierick, 3/26/65				

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PT No.	Document No.	Title, Author, Date	Status		
			Approved	In Progress	Completed Final Report
14	HW-82385	Irradiation of Th-U Crud Monitor Elements in N-Reactor, WK Kratzer, 5/21/64			X
14 SUP	HW-82385 SUP 1	Supplement A to PT-NR-14, Irradiation of 8.47 Inch Long Fuel Element Assemblies, AE Guay, 8/30/64			X
18	HW-82659	Primary Loop Pressure Relief Valves (RV-2-1 and 2) Relief and Reseating Pressure Test, KL Berrett, 6/11/64			Partially Complete
20	HW-82933	Hydraulic Actuators for RWSV-805 Valves, CL Goss, NR Miller, 6/23/64			X
20 SUP	HW-82933 SUP A	Supplement A to PT-NR-20 JL Benson, 10/20/64			X
24 SUP	RL-NRD-107 SUP 2	Supplemental Testing Procedure No. 3 LD Smith, VJ DeJong, 6/29/65			X
27	HW-84238	Steam Generator Moisture Carryover Detection, LD Smith, 9/24/64			X
27 SUP	HW-84238 SUP 1	Supplement 1 to PT-NR-27			X
28	RL-NRD-149	Process Tube Orificing with V-11 Butterfly Valves, JH Fastabend, et al, 11/18/64			X
28 SUP	RL-NRD-149 SUP A	Part I to PT-NR-28 JL Benson, 1/27/65			X
28 SUP	RL-NRD-267	Part II to PT-NR-28 JH Fastabend, et al, 3/15/65		X	
30	HW-84232	Evaluation of Ammonia for Controlling Oxygen in the Graphite Cooling System, WK Kratzer, 10/5/64			X

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30 SUP	RL-NRD-196	Supplement A to PT-NR-30, WK Kratzer, 1/21/65		X		
32	HW-84367	Exposure of Corrosion Coupons in N-Reactor Process Tubes, WK Kratzer, 10/5/64		X		
33	HW-84401	In-Reactor Enthalpy Imbalance Measurements, P Riggle, 10/7/64		X		
33 ADD	HW-84401 ADD 2	Addendum 2: Eccentricity Experiments, RH Shoemaker, 5/18/65	X			
35		(rough draft - not issued)				
36	RL-NRD-55	Simulated Driver-Target Elements, AC Callen, 11/6/64		X		
36	RL-NRD-428	Test Instructions for PT-NR-36, Loading No. 1 M Lewis, 6/15/65			X	
38	RL-NRD-193	Eval. of N Rupture Monitor Modi- fications for Desensitizing the Response of the Gamma Energy Monitor to Activated Corrosion Products, DO Allred, 1/20/65		X		
40		(rough draft - not issued)				
41	RL-NRD-297	Evaluation of the Direct Addition of Ammonia to the Graphite Cooling System, WK Kratzer, 3/26/65	X			
43	RL-NRD-379	Testing of Converted Reactor Plant, DL Renberger, 5/18/65		X		
44	RL-NRD-372	Evaluation of Dump Condenser Condensate Outlet Valve CONDV- 203 Control Characteristics, LD Smith, 5/13/65				RL-NRD-43

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45	RL-NRD-390	Increase in Primary Coolant System Pressurizer Pressure, RE Hall, et al, 5/27/65	X			
46	RL-NRD-427	Primary Drive Turbine Speed Reduction Tests, DL Renberger, et al, 6/15/65	X			Partially Complete
47	RL-NRD-426	Horizontal Rod Coolant Channel Purging, FD Frisch, et al, 6/16/65	X			

PERSONNEL CHANGES

Additions - None

Removals

Sharon Hodgkin, Secretary, terminated effective July 2, 1965.

SECURITY VIOLATIONS

None

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 July 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.

Manager
Research and Engineering

MC Leverett:LCC:vb

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SIGNIFICANT REPORTS

<u>Report Number & Class.</u>		<u>Title</u>	<u>Author(s)</u>	<u>Date</u>
RL-GEN-22 and 15 Supplements	S	Richland Integrated Process Development Program Summary. (15 Supplements)	RL Dickeman, et al	June 30, 1965
RL-NRD-107 SUP2	U	PT-NR-24, Plant Service Boiler Base Load Reduction Program Supplemental Testing Procedure No. 3.	LD Smith, VJ DeJong	June 29, 1965
RL-NRD-379	U	PT-NR-43, Testing of Converted Reactor Plant	DL Renberger	May 18, 1965
RL-NRD-426	U	PT-NR-47, Horizontal Rod Coolant Channel Purging	FD Frisch PK Telford	June 16, 1965
RL-NRD-427	U	PT-NR-46, Primary Drive Turbine Speed Reduction Tests	DL Renberger, RD Crosier	June 15, 1965
RL-NRD-430	U	Evaluation Report, PT-NR-44, Evaluation of Dump Condenser Condensate Outlet Valve CONDV-203, Control Characteristics	LD Smith	June 17, 1965
RL-NRD-434	S	Fuel Element Failure Report Process Channel 1258	DL Renberger	June 21, 1965
RL-NRD-448	U	Progress Report - Horizontal Control Rod Low Coolant Flow Trips After Reactor Scrams	PK Telford	June 28, 1965
RL-NRD-457	U	Procedure for Thermocouple Monitoring of the 1.25 Percent Enriched Coproduct Test Load	P Riggle	July 9, 1965

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-17-

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VISITORS

<u>Name</u>	<u>Company and Location</u>	<u>Contact(s)</u>	<u>Date(s)</u>	<u>Remarks</u>
Frances McBarron	National Secretaries Association International Convention Boston, Mass.		July 8-9	Attend convention
R.E. Hall R.E. Trumble	AEC Contractors Meeting Jackson, Wyoming		July 27	Attend reactor safety conference
R.E. Hall	ANS Conference Jackson, Wyoming		July 28-30	Attend conference on reactor operating experience

VISITORS

F.P. Baranowski et al	AEC-Headquarters Washington, D.C.	R.L. Dickeman	July 8-9	Budget discussions
T.M. Snyder	APED, General Electric Co. San Jose, California	M.C. Leverett	July 22-23	Technical discussions

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-18-

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