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TITLE

MONTHLY RECORD REPORT
RESEARCH AND ENGINEERING
N-REACTOR DEPARTMENT

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MONTHLY RECORD REPORT

RESEARCH AND ENGINEERING OPERATION

N-REACTOR DEPARTMENT

Classification Cancelled and Changed To
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By Authority of CG-PR-2,
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By Jessie Maley 5-10-94
Verified By PK Schultz 5-11-94

November 30, 1964

M.C. Leverett

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PLANT STATUS

Power Ascension Program

AEC approval was received November 6 to proceed with the ascension to 100 per cent power with certain restrictions pertaining to:

- Duration of operation at intermediate plateaus
- Operation within Operating Safety Limits
- Reduction in the boiler firing rate
- Reports from planned scram tests
- Maintenance of 10 mk shutdown margin under any credible operating condition

These restrictions were incorporated into the power ascension plans and the reactor was started up on November 7.

The reactor continued in operation at 3600 megawatts until November 17 when a planned scram test from this power level was successfully executed. The reactor was started up again on November 21, reaching 3600 megawatts on November 22. Operation at this power level was in effect at the close of the report period.

A careful reassessment of the performance of the plant and its capacity for going to full power was made. This assessment showed that the plant has the capacity for going safely somewhat beyond the full power without exceeding any existing process standards.

Dump Condensers

Examination of the internals of number one dump condenser was completed. Close examination of the steel fingers and the girdle which have been provided to restrain oscillation of the dump condenser tubes shows no indication of fretting or chafing, nor any other adverse indications. The remedial measures taken have evidently been highly effective.

Valve Stems

Further consideration of the 17-4 PH valve stems in the cell isolation valves, of which two have failed, has led to the decision to replace all 17-4 PH valve stems with Inconel-X valve stems. The replacement will be made as necessary over the next several months.

Physics Status

Equilibrium Conditions (November 23, 1964):

Reactor Power Level	3600 MW
Reactivity in Rods	7 mk*
Flattening Efficiency	83 %
Axial power distribution	Peak-to-average ratio is equal to or lower than a cosine distribution. This was lowered from a ratio about 20% higher than a cosine by a rod pattern change on November 21.
Horizontal Rod System Worth	67 ± 4 mk
Spike column loading	144 columns (no change)
Reactivity contribution of spike enrichment	6 mk

* The reactor is being operated with the inlet temperature approximately 15 F lower than the design value. This results in approximately 1 mk more reactivity than expected.

The power ascension sequence is being controlled to provide minimum shutdown margins of 10 mk or greater as required by Commission directive.

TECHNICAL ACTIVITIES

Nuclear Health and Safety

All documentation has been prepared in support of the power ascension program and the loading of 1.25 per cent enriched coproducer production test. Presentations were made before the subcommittee and full committee of the ACRS on November 6 and 12, respectively, to describe production test criteria and the 1.25 per cent coproducer test plans.

Coproduct Program

Physical Components Test Reactor Program: The PCTR experiments are proceeding on schedule and initial test data was received late in the month. Spline measurements, k_{∞} and dry lattice experiments were completed for the 1.25 per cent U^{235} enriched fuel elements.

Physics Design - 1.95 Per Cent Coproduct Test: Several alternate physics designs are being studied for this test loading. The use of an unshielded 21-tube coproduct block in the reactor would cause a flux depression in the base load. At the same time the outer edges of the block would have an excessively high power-density. One means of balancing the load would involve the introduction of a buffer layer of 24 natural uranium channels around the coproduct block. Another alternate involves the addition of a second ring of natural uranium to completely suppress power peaking. Other expedients, such as the loading of lithium in the coproduct block, are also being studied.

Critical Mass Evaluations - 1.25 Per cent Enriched Uranium: The exponential experiments on 1.25 per cent fuel are nearing completion. Three experimental points have been obtained for both the outers alone and the spike assemblies. These points correspond to lattice spacings of 2.81, 3.1 and 3.4 inches. It appears that all three lattices are overmoderated and it has been decided to perform the experiment on the outers for a close pack geometry. This experiment is scheduled for completion by December 1.

Fuels Fabrication: Fifteen tons of billets containing 1.95 per cent U^{235} have been received. Extrusions are underway and will be complete by December 3. The extruded uranium will be used to fabricate fuel elements for exponential experiments, critical mass experiments and for production test loadings. The same fuel elements will be used sequentially for the multiple purposes noted.

Melt-Down Experiments: Seven melt-down experiments have been completed. These experiments showed that the target does not become fluidized and that the lithium is largely retained in the alloy -- even at the high temperature of 1100 C. The aluminum and zirconium apparently react exothermally to form a high melting solid phase. Reactions of this type could not be found in the technical literature. Experimental work and analysis of the data is continuing.

Simulated Driver-Target Test: An important element of the in-pile fuel testing program during the next year will be the Simulated Driver-Target Test (SDT). In this test various uranium alloys will be tested in the same configuration -- that of the coproduct driver. Difficulty was experienced in the coextrusion of the first non-heat treated SDT billets. Additional billets, both heat treated and not, are on order and two to four columns of SDT fuel elements will be ready for charging during December or early January.

Coproducer Hazards Studies: Fuel melting time calculations made earlier on the 1.25 per cent test fuel and the 1.95 per cent twenty pound per foot case indicated that lighter weight fuels would melt at earlier times on total loss of cooling. Subsequent computer runs made to define lower weight limits based on melting time criteria for use in parametric studies contradict the earlier calculations.

The principal factor contributing to the apparent anomaly is the effect of heat capacity of the target and the geometry of the assembly. For a given active zone pressure drop varying weight of the driver requires changing the size of the target to maintain approximately the same flow area in the inner annulus. Thus the lighter weight driver designs involve larger targets having greater heat capacities which increase nearly in direct proportion to the increase in volume. In addition, the increase in target diameter results in a decrease in annular gap (thus a decrease in conductive resistance) and an increase in radiation heat transfer area.

Re-Evaluation of Reject Fuel

As a first step in re-evaluating fuel elements rejected during fabrication, those with large clad thickness variation are being considered. The variations occur mainly as localized clad thickening. The areas with greater than the specified 12 mils variation are no greater than 2 square inches and the variation is gradual. No step transitions in thickness have been observed in the fuels being considered for reactor charging.

Acceptance of outer fuel tubes with inner clad thickness variation in the 12 to 21 mil range would effectively increase the first and second load fabrication yield by two to three per cent. It is estimated that FY-1965 savings of about $110,000 would result.

Studies indicate that these fuel elements could be safely irradiated in the two upstream and two downstream positions of each fuel column in the reactor. Thermal gradients in fuels in these positions are less than one-half the maximum, and the uranium temperatures and exposures would be sufficiently low to avoid excessive swelling. Efforts in fabrication development to eliminate clad thickness variation as a reject category will continue unabated however. Evaluations are continuing.

Graphite Program

An $84,000 increase in the cost of graphite irradiations will require compensating reductions in other R & D programs. The R & D program report to the AEC, which covers the first six months of the fixcal year, will reflect the projected changes which will be required during the balance of the fiscal year.

Tests were conducted by Equipment Development to determine the manner in which the keys on the tube blocks in the reactor might fail as a result of longitudinal contraction of the graphite bars. The test setup simulated in-reactor conditions using full-size tube and filler blocks. In the four tests performed, the keys failed in shear at a 45 degree angle about as predicted. In one test, the filler block failed first and the corresponding tube block key failed by wedging rather than shear. In each test considerable separation of the tube block was observed. Although not conclusive, these test results tend to confirm the theory that the reactor stack would rise rather than fall as a result of individual bar contraction.

Graphite Cooling System Water Quality

During the October outage the graphite cooling system was modified to permit the addition of deoxygenated makeup water to the top of the surge tank to prevent the accumulation of explosive concentrations of hydrogen and oxygen in the gas space at the top of the tank. The surge tank gas will be sampled during the next shutdown to determine the effectiveness of this modification.

Production test NR-30, Evaluation of Ammonia for Controlling Oxygen in the Graphite Cooling System was approved November 9. This test authorizes addition of ammonia to the graphite cooling system in order to determine its ability to remove radiologically generated oxygen. This step is being taken as a possible means for improved corrosion control.

Primary Coolant System Crud Measurement

The four crud monitor thermocouple trains authorized by PT-NR-14 were charged in the reactor October 16 to provide warning of catastrophic crudding that could damage the fuel load. The monitor trains have been performing without difficulty since reactor startup on October 24.

Irradiated Fuel Storage Criticality

Variations of the three basic storage modules were examined during the past month. The variations resulted primarily from desires to increase handling efficiency, or from the realization that CPD may not be able to dump 26 inch fuel into the dissolvers. In general the new variations increased handling efficiency but reduced fuel storage density when compared with the canister stacking method reported last month. It is believed that this work is now complete.

Fuel Element Enrichment Analyzer

A test has demonstrated the ability to differentiate between enrichments of N type fuel elements. This method measures the .184 mev gamma activity by analyzing each element for 15 seconds. The elements can be analyzed as they are loaded into each fuel magazine to insure proper enrichment loading, thereby enhancing nuclear safety. Cost of installation and fabrication would by $10,000.

Control Rod Enhancement

Increases in the reactivity worth of the present NPR control system, through the use of B^{10} in place of Boron in B_4C may be impractical. AEC-Oak Ridge prices are \$3.00 per gram for B^{10} and will not be supplied in quantities in excess of one hundred pounds. N-Reactor control rods contain approximately 3000 pounds of B_4C or approximately 2400 pounds of natural boron. Based upon these numbers the material costs of modifying the control rods to incorporate this material would be approximately 3.3 million dollars.

This high cost, coupled with inherent uncertainties in calculations of control rod worths, further supports the need for the control rod tests proposed for the PCTR.

This PCTR program will be started late in December and the use of other absorbers in place of B^{10} will be considered as a part of the program.

Reactor Power Level Capabilities

Projections have been made from the eighty and ninety per cent levels to evaluate the ability to reach full design power level with currently applied process tube outlet temperature limits. These projections have shown that full power level can be achieved by raising operating rear riser pressures to 1450 psig, correcting the hot channel situation in tube 1155 and in controlling the next hottest tube, 1262. The validity of these projections is contingent on several principle factors.

a. the ability to limit pressure surges after scrams to 140 psig or less.

b. the ability to operate with the minimum reactor inlet temperature consistent with the secondary system steam requirements, 362 F.

c. the ability to maintain flattening efficiencies achieved during operation at the eighty and ninety per cent levels.

The results obtained from the projections show that most of the available margin will be expended in reaching one hundred per cent power under current secondary system steam pressure requirements. If it becomes necessary to increase steam pressure, or, if it becomes desirable to gain more operating latitude, it will be necessary to either orifice the reactor or to increase primary loop flow capacity. A joint study to further delineate pertinent factors is being carried out by Research and Engineering Section and the N-Reactor Project Section.

In-Reactor Thermocouples

The thermocouple fabrication and testing will be completed during early December and charging will be completed during the first available outage. These thermocouples will provide important empirical information conditions within the reactor. Such information will be used to establish important guidelines for effective and safe operation, process tube temperature limits and other heat balance considerations. They will also help to determine the reactor power level capabilities discussed above.

Steam Flow Distribution Study

The possible pressure and flow differences in the secondary steam system resulting from unbalances in steam input and steam outlet flows is under study. The model has been programmed for the digital computer and results have been obtained for selected cases. The results indicate that under the worst possible conditions, steam pressure differences in the order of twenty pounds per square inch can exist. Additional cases will be studied to cover the expected conditions of operation with combinations of steam generator cells in service and the export generators in operation.

Potential Methods for Po^{210} Production

A review has been made of the possible techniques for Po^{210} production in N-Reactor.

Four methods are identified. The most readily available method uses the 68 peripheral channels to irradiate bismuth. This could produce as much as 725 grams of Po^{210} per year at a cost of less than $1000 per gram, as previously reported.

A single tube (or coproduct) configuration with a Bi_2O_3 flow splitter would produce about 250 grams of Po^{210} per year, and would be neutronically superior to using an aluminum flow splitter. The cost of the Po^{210} thus made has not been estimated.

The reactivity loss of the proposed single tube with an aluminum flow splitter is 8 mk. Potentially this could be reduced by 4 mk if Bi_2O_3 replaced the aluminum. As a result of this change in fuel design, N-Reactor would produce about 250 grams of Po^{210} per year. Thus, the irradiation of Bi_2O_3 rather than aluminum in the flow splitter would reduce reactivity losses by $200,000 per year and produce Po^{210} against which about $100,000 of the remaining cost could be allocated; thus contributing to the advantages of the single tube design and N-Reactor's Po^{210} production capability.

A third possible technique is the substitution of bismuth for uranium in the normal process channels. This approach will produce polonium at a somewhat higher cost due to the sacrifice of plutonium production and reactor power level. If the reactor were operated at 4800 megawatts with maximum tube power limits of 6000 kilowatts, 250 tubes could be used for Po^{210} production. This would reduce the reactor power level to 3500 megawatts. Po^{210} production would be about ten grams per process tube per year. The maximum which could be produced by this technique would be about 2500 grams. The "cost" of the Po^{210} would depend on the value assigned to the plutonium not produced.

A fourth technique with the potential of producing large quantities of Po^{210} is the utilization of the 640 cross coolant channels. Conceptually these channels could be overbored to a 2-inch diameter. In the overbored graphite channels about 90 tons of bismuth could be irradiated to produce about 2000 grams of Po^{210} per year without any reduction in uranium inventory or reactor power. The reactivity losses due to bismuth in the cross coolant channels would be about 10 mk.

The study to further refine the various technical parameters and economic bases is continuing.

Alternate Fuel Cycles for Power-Only Operation of N-Reactor

The first phase of this program, comparing the fuel cycle costs of uranium metal and thorium oxide cycles to the reference uranium oxide cycle is essentially complete. The economic calculations show that the reference uranium oxide fuel could compete successfully with either the thorium-oxide or uranium metal cycles. The uranium metal cycle holds potential for improvement which would provide for its use during an interim period.

Work on the plutonium enrichment case is continuing. Physics data has been acquired for the 1.0, 1.5 and 3.0 w/o PuO_2 cases. These data will be analyzed and compared to the UO_2 fuel case.

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	Com-pleted	Final Report
2	HW-80369	Routine Graphite Sample Irradiations, D.H. Curtiss, 1/3/64	X	X		
3	HW-81478	Routine Monitoring of Graphite Oxidation in N-Reactor, D.H. Curtiss, 3/26/64	X	X		
4		(rough draft - not issued)				
5		(rough draft - not issued)				
6	HW-81339	The Use of Isotope Producing Rods in N-Reactor, R.A. Chitwood, 3/17/64	X	X		
7	HW-81029	NPR Shield Evaluation; Shield Plug Measurement During Startup Test N-2 J Greenborg, RA Bennett, 3/2/64	X		X	
8	HW-81327	Coproducer Demonstration Test(1.25) T.W. Evans, 5/4/64	X			
9	HW-81609	Low Goal Irradiation of Fuel Elements with Varying Amounts of White Oxide on Surface, A.C. Callen, 3/30/64	X	X		
10	HW-82136	N-Reactor Corrosion Monitoring, RJ Evans and BS Kosut, 5/25/64	X			
11	HW-82074	105-N Fog Spray Control Valves, DR Resner and JL Benson, 4/30/64	X		X	
12	HW-82241	Copper Base Alloy Corrosion Program, B.S. Kosut, 6/4/64	X	X		
13		(rough draft - not issued)				

PT No.	Document No.	Title, Author, date	Status: Approved	Status: In Progress	Status: Com- pleted	Status: Final Report
14	HW-82385	Irradiation of Thorium-Uranium Crud Monitor Elements in N-Reactor, W.K. Kratzer, 5/21/64	X	X		
18	HW-82659	Primary Loop Pressure Relief Valves (RV-2-1 and 2) Relief and Reseating Pressure Test, K.L. Berrett, 6/11/64	X	X		HW-83175 (Interim Repor
20	HW-82933	Hydraulic Actuators for RWSV-805 Valves, C.L. Goss, N.R. Miller,6/23/64	X		X	
21	HW-83200	Optimization of Secondary Coolant Supply Operational Modes, F.J. Mollerus, Jr. 7/8/64	X	X		
21 SUP	HW-83216	Supplemental Testing Procedure, PT-NR-21, Surge Tank Pressure Control Test, F.J. Mollerus, Jr., 7/8/64	X		X	
21 SUP	HW-83442	Supplemental Testing Procedure, 184-N Turbine Generator Tests, F.J. Mollerus, Jr., 7/27/64	X		X	
21 SUP	HW-83857	Supplemental Testing Procedure, PT-NR-21, Pressurizer Level Control Test, E.E. Leitz, 8/28/64	X		X	
23	HW-83389	Horizontal Rod Scram No. 2, Power Ascension Program, N-3, E.E. Leitz, 7/22/64	X		X	HW-84066
26		(not issued)				
27		(not issued)				
28		(rough draft - not issued)				
29	HW-84239	Full Load Dump Condenser Characteristics, F.D. Frisch, 9/29/64	(Test cancelled by RLOO-AEC)			

PT No.	Document No.	Title, Author, Date	Status: Approved	Status: In Progress	Status: Completed	Status: Final Report
30	HW-84232	Evaluation of Ammonia for Controlling Oxygen in the Graphite Cooling System, W.K. Kratzer, 9/24/64	X	X		
31		(cancelled)				
32	HW-84367	Exposure of Corrosion Coupons in N-Reactor Process Tubes, W.K. Kratzer, 10/5/64	X			
33	HW-84401	In-Reactor Enthalpy Imbalance Measurements, P Riggle, 10/7/64	X			
34	RL-NRD-26	Horizontal Rod Scram No. 5, Power Ascension Program, N-3, D.L. Renberger, 10/26/64	X		X	

PERSONNEL CHANGES

Additions

V. J. DeJong, Engineer, transferred to Process Engineering, effective November 1.

Removals

None

SIGNIFICANT REPORTS

HW-83892	Copper-Base Alloys in the N-Reactor Graphite Cooling System, W.E. Gurwell and B.S. Kosut, 11/30/64 (Unclassified)
HW-83828 SUP1	N-Reactor Summary Scram Reports, E.E. Leitz, 9/30/64 (Confidential)
RL-NRD-59	Predicted N-Reactor Plant Heat Balances, V.J. DeJong, A.J. Ebens and L.D. Smith, 11/6/64 (Confidential)

RL-NRD-23	Secondary System Instrumentation Modifications for Conversion, A.A. Maupin and D.W. Leiby, 10/21/64, (Unclassified)
RL-NRD-20	Comments: Target Element Thermal Stress Tests, A.E. Guay, 10/20/64 (Secret)
RL-NRD-28	Need for Hydrogen Addition to N-Reactor Primary System Coolant, W.K. Kratzer, 10/26/64 (Confidential)
RL-NRD-32	Preliminary Results and Recommendations on Copper Base Alloys in the N-Reactor Primary System, B.S. Kosut, 10/28/64 (Unclassified)
RL-NRD-83	Confinement System Stack Monitoring Modification Requirements, J. W. Vanderbeek, 11/23/64 (Unclassified)

VISITORS

H. S. Isbin, Consultant, from University of Minnesota, visited Hanford November 12 and 13 to discuss heat transfer studies; R.H. Shoemaker - contact.

W. G. Wehmeyer and H. Hovila, of Engineered Industrial Systems, Seattle, Washington, visited Hanford November 18 to discuss bridge cranes under the C and D elevators with A. A. Maupin.

H. C. Neilson of the Boeing Company, Seattle, Washington, visited Hanford November 23 to discuss the radiological consequences of possible release of tritium with M. M. Hendrickson.

TRIPS

B. S. Kosut attended the ASTM Officers Meeting held October 27, in conjunction with the Canadian Institute of Mining and Metallurgy and Pacific Northwest Metals and Minerals Conference at Vancouver, B.C.

D. W. Leiby, M. C. Leverett, R. E. Hall and R. E. Trumble attended meetings at Burns and Roe, Inc., Hempstead, L.I., New York, November 11. Contact was Ivan Gabel. Discussions were on N-Reactor conversion.

D. W. Leiby and R. E. Hall visited Bailey Meter Co., Cleveland, Ohio on November 12 to discuss N-Reactor conversion.

R. E. Hall visited the Philadelphia Electrical, Philadelphia, Pennsylvania, November 11.

J. R. Bolliger and R. E. Trumble visited the Bonneville Power Authority, Portland, Oregon, October 29 to discuss the Phase II hazards review.

M. C. Leverett and R. E. Trumble made presentations before the ACRS in Washington, D.C. on November 5 and on November 12 on N-Reactor production test criteria and the coproduct production test.

M. C. Leverett, J. L. Benson and J. W. Riches attended the joint ANS-AIF meeting in San Francisco November 30 through December 2.

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 November, 1964. 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.

M. C. Leverett
Manager
Research and Engineering

MC Leverett:LCC:vb

DATE

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8/4/94

END