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MONTHLY RECORD REPORT
RESEARCH AND ENGINEERING
N-REACTOR DEPARTMENT

7

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AUTHOR

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MONTHLY RECORD REPORT
RESEARCH AND ENGINEERING OPERATION
N-REACTOR DEPARTMENT

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December 31, 1964

By Authority of CG-PR-2,
DS Lewis, 4-21-94.

By Jessie Maley 5-10-94 M. C. Leverett
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PLANT STATUSPower Ascension Program

The reactor power was elevated to 4000 megawatts reaching that level at 10:45 p.m. December 9. The final ascension from 3600 megawatts to 4000 megawatts took place without incident. The following table summarizes plant operating conditions at 4000 megawatts with five pumps running and sixteen dump condensers on the line.

Plant Operating Conditions at 4000 MW

<u>Parameter</u>	<u>Expected</u>	<u>Observed</u>
Average Reactor Inlet Temperature, °F	354	353
Average Reactor Outlet Temperature, °F	497	494
Max. Reactor Tube Outlet Temperature, °F	525	526*
Primary Coolant Flow, lb/hr	85.8 x 10 ⁶	85.5 x 10 ⁶
Horizontal Rod System Worth, mk	67 ± 4	Same
Maximum Tube Power, kw	5000	4930**
Maximum Power Density, kw/ft	230	215
Maximum Fuel Cladding Temperature, °F	600	575
Excess Reactivity Held in Control Rods, mk	8	8.5 ± 1.5
Flattening Efficiency	0.8	0.81
Pressurizer Pressure, psig	1425	1420
Maximum Steam Pressure, psia	80	80

* Maximum allowed by Process Standards is 530°F

** Includes estimated 75 mw actually removed via graphite cooling system.

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This performance was almost exactly as had been predicted prior to the final ascension from 3600 to 4000 megawatts. Operation was well within the limits imposed by the Process Standards and the Operating Limits. Operation of the reactor was stable although a slow drift of power from front to rear was noted.

The reactor was shut down on December 15 following breakage of a 17-4 FH valve stem in cell No. 1. The valve stem was removed and the plant returned to service.

At the time of this report the N-Reactor was operating at full design power. The physics assessment for equilibrium operation was essentially the same as reported in the above table, except for a slight (0.5 mk) increase in operating reactivity which resulted from increased fuel exposure. Maximum axial peaking indicated by in-core monitors was 1.51 peak/average.

The next engineering goal is elevation of the steam pressure from its present level of 80 psia to 110 psia as required for phase II operation. Programs to produce this result by June 1 are under development.

Dump Condensers

On December 16 dump condenser No. 1 was entered and inspected and the tube bundle was found to be in good condition.

Plant Service Boiler Tests

The minimum base load condition of the plant service boiler was reduced from 180,000 pounds per hour to 120,000 pounds per hour, and as many as four burners were authorized to be placed in a standby status on December 29, as authorized by the second step of production test NR-24.

Steam Generator Level Control

Tests of the steam generator level controls have been performed at each power level plateau since 2400 megawatts to develop empirical relationships necessary for control of the liquid-steam interface near the desired control point. Based upon these data, necessary control system adjustments have been successfully made.

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TECHNICAL ACTIVITIES

Evaluation of KER Fuel Elements

A previous report disclosed that the UT-10B nondestructive test for cap to core bonding in the KER Metal Examination Facility had found a fuel element with a defective end cap. Visual examination of the fuel element in Radiometallurgy showed a dent on the outer weld and metallographic examination showed a pair of cracks in the braze layer adjacent to the dent. The dent and the cracks in the braze were shown to be superficial, and after grinding 100 mils deep, all disappeared. Both have been attributed to damage during handling after discharge.

Primary System Pressure Limit

A review of the basis for the process tube strain limit was made in an attempt to determine the level to which the primary system pressure can be raised. On the basis of data from conventional tensile samples irradiated in the G-7 loop of the ETR under the Division of Reactor Development Surveillance Program, it was concluded that a strain limit of 0.01 rather than the current 0.0055 can be used with negligible decrease in conservatism. Additional data on fracture studies of crack propagation, bending, and variable loading rates, indicate that even the limit of 0.01 is very conservative.

Oxygen Control in the Graphite Cooling System

Production Test NR-30 authorized the evaluation of ammonia for control of oxygen in the graphite cooling system for a period of seven equilibrium operating days. The seven days were completed on December 14; five of the days at full reactor power. The test demonstrated dramatically that oxygen could be maintained at the desired level of ~7 ppb if ammonia is used for pH control.

Typical coolant conditions before, during and after the change to ammonia are given below.

	Before Change to NH ₄ OH	During NH ₄ OH Test	24 Hours After Termination of NH ₄ OH Test
pH Control Material	LiOH	NH ₄ OH	LiOH
pH	9.8	9.8	9.8
NH ₃ conc; ppm	.02 - .05	8 - 15	0.1
N ₂ H ₄ conc; ppb	<3	110	<3
O ₂ conc; ppb	30,000-40,000*	<3	>26,000
H ₂ O ₂ conc; ppb	100,000-150,000	<25	150,000
H ₂ conc; cc/liter	>50*	>50*	>50*

* Estimated. Concentration greater than saturation in samples taken.

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HW-83892, "Copper Base Alloys in the Graphite System" has been issued in anticipation of the permanent use of ammonium hydroxide for pH and oxygen control in this system. A copper corrosion test program to determine the effects of continuous operation with ammonium hydroxide is being prepared. Emphasis will be placed on determining whether the ammoniated water will damage the 1280 copper sensing lines that are part of this system.

Crud Probe Elements

The crud probe elements installed in the N-Reactor primary coolant on October 16 have been functioning without difficulty. The elements showed no significant increase in fuel crudding either after the crud burst on November 24 or after the unexplained period of low primary coolant pH between December 10 and 12.

Interpretation of N-1 Startup Spectral Indices Experiments

Reaction rates of Cu, Au, Eu, Lu, Pu, U²³⁸ and U²³⁵ radioactivities were determined in N-1 startup tests and have been corrected for radioactive decay and other minor experimental variation. Interpretation of the corrected results, reported in HW-84085, has begun. Unavoidable and unmeasured control rod perturbation of the neutron spectrum along the test fuel pieces has been calculated to be only a few tenths of a percent and can be neglected in the cold experiments.

The application of the concept of only two spectral indices, neutron temperature and Westcott's r (fast to slow flux ratio) has been shown in transport calculations to be inadequate to describe the detailed neutron spectral shifts in the fuel annuli. Emphasis has therefore been shifted to attempts to fit the observed thermal reaction rate distributions with transport calculations. To date the relative magnitudes of the Cu⁶⁴ activity data have been fit quite well in both annuli. The precise distribution in each annulus is not well predicted because of an inadequacy of the cell model used in the THERMOS code. Work to improve the cell model is under way.

Process Tube Monitoring

Step one of the technical negotiation for the television system was completed and three companies were asked to submit price proposals. Although higher than expected the responses to bid proposals differed by only about ten per cent.

The radiation resistant television camera lens was received from Lerma Engineering. The process tube water plugs were delivered.

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Fabrication was started on the drive unit by J.A. Jones. The Hanford Laboratories Electrical Shop has started assembly of the depth gauge probe and control console.

Detailed design of the vacuum line for the rear of the process tube, and detailed design of a probe tray and calibration block for the front of the process tube are completed and ready for fabrication.

Fuel Enrichment Analyzer

A device is being developed for determining the enrichment of fuel elements as they are charged into the fuel magazines and analyzing the magazines for enrichment content before they are charged into the reactor.

The general specifications for such an enrichment analyzer have been written, and will be used as a basis for a development program to be conducted in conjunction with Testing Methods of Hanford Laboratories (at a cost of \$2500). The components for the analyzer will mostly be standard, commercially available equipment.

Fuel Exposure Analyzer

A feasibility study on separating irradiated fuel elements by plutonium content is also being made.

Rupture Monitor System

In late November the N-Reactor was shut down because of large signals occurring in both the gross gamma monitor and the gamma energy monitor for a particular process tube. These rupture indications were later shown to be the result of corrosion products trapped in the sample chamber.

Modifications of the rupture monitor system are being evaluated in an effort to desensitize it to corrosion products. It has been determined that the gamma energy monitor's signal channel energy levels must be raised beyond the photopeaks of corrosion products. A multichannel analyzer spectrum will be taken of the coolant to determine the optimum setting of the signal channels.

N-Reactor Control Systems Paper

A technical paper, RL-NRD-105, "Control Aspects of a Large, Dual-Purpose Nuclear Reactor Plant," has been prepared by D. W. Leiby. It will be presented at the Joint Automatic Control Conference in Troy, New York in June 1965. All the necessary General Electric and Atomic Energy Commission clearances have been received.

Thermocouple Train Progress

Thermocouples for the first train were charged into the N-Reactor November 18. The assembly was found to leak slightly in the vicinity of the nozzle cap, and was eventually discharged and decontaminated. Examination revealed an incomplete braze joint at one of the twelve penetrations where the metal sheathed couples entered the primary coolant. Although the assembly had been vigorously tested before installation, the leak was not detected because high temperature brazing flux temporarily plugged the hole. This type of leak will be prevented in the future by thorough washing to remove the flux and close examination following the brazing operation. The thermocouple program is designed to give important empirical information for the precise establishment of standards and controls.

Thermal Hydraulic Experiments

Experiments are under way in the Hanford Laboratories with the modified cosine heated test section representing the tube-in-tube fuel loadings. Steady-state two-phase pressure drop data have been obtained at rear riser pressures of 1200 and 1300 psi, as well as equivalent tube power values ranging from average to hot channel conditions and exit steam qualities from zero to 30 per cent by weight.

Observations made during the 1200 and 1300 psi runs indicate that inter-channel mixing may have a significantly beneficial effect upon flow reductions required to initiate film boiling in the fuel. Thus, it seems certain that the data obtained with the previous model are conservative (by an indefinite amount) with respect to predicted flow reductions required to initiate film boiling.

Conversion to Phase II

The secondary steam system model for pressure and flow distribution analysis has been revised and updated to include information received during the November visit to Burns & Roe. Computer runs are being made to predict the expected pressure differences and flow distributions which will exist under the possible inlet and outlet steam flow conditions.

A mathematical model of the modifications to the heat dissipation plant due to conversion is being developed in conjunction with personnel of the Systems Research Operation of Hanford Laboratories.

The analog computer simulation of the N-Reactor plant will be modified to include the effects of the alterations and additions due to conversion.

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PCTR-k_∞ Experiments on the 1.95% Enriched Coproduct Cell

The k_∞ experiments on the wet and dry N-Reactor coproduct lattice have been completed and are presently being analyzed. Standard K-Reactor splines were evaluated in these experiments also. Preliminary review of the experimental data indicates that the spline worth is down approximately 20 per cent from that found for the 0.95 U²³⁵ fuel loading.

Coproduct Exponential Experiments

A delay in receipt of the coproduct fuel resulted in a delay in starting of the exponential experiments. The aluminum process tubes employed in earlier 0.95 fuel exponential experiments have been readied for these experiments. Fuel delivery was initiated December 29.

Coproduct 1.95% Block Test

The distribution of power in the test block will contain a peak near its interface with the rest of the reactor. The relative height of this peak can be depressed by introducing a buffering layer of natural uranium columns between the block test and the rest of the reactor. The average power in the block can be depressed by locating it in a relatively low flux part of the reactor. The test is feasible if conducted in this way.

Current plans therefore call for the charging of 21 tubes of 1.95 material into a 3 x 7 block located in 3 columns near one side of the reactor. This will require replacement of one or two columns of spike and two or one columns of standard fuel. A buffering layer of natural uranium may be added in column 4 or 5 if tube power measurements show it to be necessary, or, alternatively, shallow insertion of rod tips from that side of the reactor may be used to control the power in the test tubes. It is believed this procedure will give satisfactory test results without limiting the power of the reactor as a whole.

Theoretical investigation of this problem and all other N-Reactor studies to date have been based upon diffusion models of the space-energy distribution of neutrons. The inadequacy of this approximation is emphasized in the black target of the coproduct fuel element. In an attempt to improve the theoretical model of N-Reactor the transport IBM code THERMOS is being used to obtain average thermal absorption and fission cross sections for homogenized multizoned reactor calculations. To date a ten per cent reduction has been found for the average thermal values for the coproduct cell. Values for the 0.95 and spike cells are currently being calculated.

Target Integrity Tests

Two more individual target heating tests have been completed, bringing the total to nine. In the first seven tests, unirradiated lithium-aluminum alloy target material canned in aluminum and jacketed in Zircaloy was heated in a helium atmosphere to 1100 C in approximately five minutes, held at 1100 C for times up to one hour and either slowly or rapidly cooled back to room temperature. In the eighth and ninth tests, the atmosphere was steam rather than helium. A thermocouple malfunction caused the ninth test to be terminated after reaching 900 C.

Four of the nine targets survived with no damage other than a pronounced swelling at the base of the Zircaloy end caps adjacent to the internal aluminum end caps. That swelling may have been caused by hydrostatic pressure of the lithium-aluminum target material as it melted and was superheated. The cladding that incurred preferential strain was that which had been annealed during the welding of the end caps.

The five targets that failed did so either immediately upon reaching 1100 C or during cooling when the temperature reached about 850 C and an exothermic reaction was noted. However, the targets that failed did not release fluid Li-Al alloy. Rather, the targets seemed to be consumed by a reaction of molten aluminum and solid zirconium to form Al_3Zr . Cross sectioning the failed targets showed that the Al_3Zr product formed wherever molten aluminum contacted zirconium.

It is tentatively concluded that the autoclaved oxide layer on the interior of the Zircaloy jacket prevents direct contact of the molten aluminum with the zirconium. If so, then the formation of Al_3Zr would be limited by the rate of diffusion of Al through ZrO_2 . At the beta to alpha transformation of the Zircaloy (at 850 C) during cooling, ZrO_2 and Al_3Zr may develop gross defects from differential expansion relative to Zr-2. The exothermic reaction may be simply the formation of Al_3Zr from the elements.

Spectrographic analysis of a target that failed showed that lithium was not "lost" from the system. The material that appears to be unreacted Li-Al alloy showed a slight (though not significant) loss of lithium and the Al_3Zr showed no loss of lithium. However, a thin metallic-appearing film on the Vycor tube (that contained the target during heating) was rich in lithium and tends to indicate that the Li-Al matrix suffered some lithium depletion. A complete material balance cannot be made until chemical analyses are finished.

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The target element that was heated to 900 C in the ninth test was re-heated several times in the course of investigating the cause of the thermocouple failure. When this repeatedly reheated target reached 1100 C, cooling water was intentionally introduced.

This last test seems to indicate that the heat of formation of the oxides of the metals present tends to prolong a thermal excursion if any failed targets come in contact with water. However, the extrapolation of this experience to that which might be encountered with irradiated targets is premature.

The heating of the first irradiated target was initiated the week beginning December 14.

Simulated Driver-Target Test

As reported previously, initial trial fabrication of uranium alloy drivers with non-heat-treated billets was unsuccessful and billet cracking during heat treatment prevented early trial of heat-treated alloys. Extrusion of heat-treated British alloy in the test geometry has now been demonstrated. In addition some short billets of the other alloys were heat-treated locally and have been successfully coextruded. With these additional tests complete, it appears that heat treatment is effective and can be done successfully on most alloys. Some changes in quenching medium or procedure may be needed however. Extrusion pressures were not as high as anticipated and the billet will be re-designed to obtain a higher reduction ratio in order to improve clad-core bonding.

Coproduct Conversion Study

A study of methods to convert from a plutonium-only operation to a coproduct operation has been made. Two methods of accomplishing the conversion are incremental replacement of the present production fuel with coproducer fuel over about a six month period, or a block discharge of present production fuel with a block charge of coproducer fuel. The block transition allows for some desirable but not mandatory physics tests not possible with incremental transition method. The incremental transition appears to be the most desirable from the operating and cost viewpoint since it would not require early increases in plant capacity and billet orders.

Hazards Support - Coproduct

Loss-of-coolant temperature transients were calculated for a variety of fuel weights for use in parametric studies. These calculations confirm that the lighter weight drivers melt or "peak out" in longer periods of time than do the heavy ones.

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Nuclear Safety

Storage Basin

Criticality calculations for the fuel storage basin capacity were completed by Reactor Physics and are being documented.

Phase II Hazards Review

The comment issue of the hazards review for phase II operation was issued on December 18 and circulated to the AEC for further distribution to BPA, WPPSS and Burns & Roe. Comments are due by January 4.

Safety Audits

Nuclear safety audits of both the 333 and 306 buildings were conducted on two separate occasions. On November 30 a sampling in each building for adherence to 1.95% Nuclear Safety Specifications was performed. On December 15 a complete audit of the entire facility was made. No nuclear safety specification violations were observed in either audit.

Safety Specifications

A number of changes were made in the N-Fuels Nuclear Safety Specifications to promote a safe and yet efficient operation. The areas affected were the billet preparation; billet extrusion; welding, testing and inspection; and the quality control laboratory. Most of these changes concerned 1.95% material with the exception of the quality control area specifications which were completely rewritten.

Cask Safety Limits

Criticality safety of the SCRUP and Stan-Ray casks was examined, and the pertinent limits established.

Personnel Exposure Status

There were no reported abnormal employee exposures during December. For the year 1964, a tabulation of employee exposure puts the maximum individual exposure at approximately 2.5 rem and the average employee's exposure at approximately 0.5 rem (the department exposure limit is 5.0 rem/year for an individual).

Standards and PCA's

There were no revised or new standards issued since the last report period.

A total of 31 Process Change Authorizations (PCA's) were issued since the last report period. Forty-eight PCA's were in force as of this date.

Alternate Fuel Cycles for Power Only Operation

The $\text{PuO}_2\text{-UO}_2$ fuel data necessary for economic evaluation has been completed. The problems of neutron spectrum calculations have been resolved. Calculations of the physics have also been completed.

Resolution of the problems of neutron spectrum calculation was achieved by changing the TEMPEST calculational model from the "heavy moderator" equations (Wigner-Wilkins model). Use of the heavy moderator equation does not properly allow for scattering of neutrons into the lower end of the thermal energy range (0 - 0.68 ev). The light moderator equations are felt to yield more valid results for fuels with relatively high Pu concentrations such as those studied.

The data obtained are summarized in the following table.

Nominal (Initial) Enrichment (Wt % PuO_2 in $\text{PuO}_2\text{-UO}_2$)	True Enrichment (Initial) (Wt % Fissile Atoms in Total Metal)	Attainable Goal Exposure (MWD/T of Metal) (Cosine Flux)
0.6	0.737	1,720
1.2	1.220	10,140
2.0	1.868	19,500
3.0	2.674	29,800
4.0	3.482	39,100
6.0	5.096	57,500

In general, the differences between the $\text{PuO}_2\text{-UO}_2$ fuel and the UO_2 fuel (previously reported) are small in contradiction to earlier analyses. The following observations can be made from these data.

- for a given initial weight percent fissile content, the $\text{PuO}_2\text{-UO}_2$ fuel gives a higher goal exposure (x 40% higher at 1.5 wt % and 10% at 3.5 wt %).
- the net burnout of input fissile material, for a given goal exposure, is approximately 10% less for the $\text{PuO}_2\text{-UO}_2$ cases than for the UO_2 only cases - apparently indicating a higher conversion of fertile material in the $\text{PuO}_2\text{-UO}_2$ cases.

Work is now under way to calculate the effects of temperature changes on reactivity. More data will probably be required for these cases than for the UO_2 or ThO_2 cases, since the effects on $\text{PuO}_2\text{-UO}_2$ systems are not as well behaved as for the former fuels. Further analysis of the data in hand is also required to provide tube power data and other important information.

Po²¹⁰ Production in a Rod-in-Tube Configuration

Studies of the use of a Bi₂O₃ rod as a flow splitter in place of an aluminum rod show relatively little advantage. About 185 grams of Po²¹⁰ would be produced annually. Reduction in spike enrichment requirement and increase in Pu production would yield a credit of up to \$150,000 per year when debited with the additional costs using the Bi₂O₃ rods.

IN-PLANT TESTS

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

PT No.	Document No.		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				
7	HW-81029	NPR Shield Evaluation; Shield Plug Measurement During 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		

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PT No.	Document No.	Title, Author, Date	Status			
			Approved	In Progress	Com- pleted	Final Report
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)				
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		
20	HW-82933	Hydraulic Actuators for RWSV-805 Valves, C.L. Goss and 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

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PT No.	Document No.	Title, Author, Date	Status			
			Approved	In Progress	Com- pleted	Final Report
26	HW-84094	Calibration of an Operational Rod Withdrawal Sequence, R.A. Chitwood, 9/25/64	X			
27		(not issued)				
28		(rough draft - not issued)				
30	HW-84232	Evaluation of Ammonia for Controlling Oxygen in the Graphite Cooling System, W.K. Kratzer, 10/5/64	X	X		
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
35		(rough draft - not issued)				
36	RL-NRD-55	Simulated Driver-Target Element, A.C. Callen, 11/6/64	X			

PERSONNEL CHANGESAdditions

A. G. Dunbar transferred to the Controls, Instruments and Systems Analysis Subsection from APED, San Jose effective December 1.

D. W. Constable transferred to the Advanced Technology Subsection from Contract and Accounting Operation effective December 15.

SIGNIFICANT REPORTS

<u>Report Number</u>	<u>Class.</u>	<u>Title</u>	<u>Author(s)</u>	<u>Date</u>
RL-NRD-65	Secret	Projected Costs for Fuel Ruptures	D.H. Curtiss	December 28, 1964
RL-NRD-103	Confid.	Half-Rods for N-Reactor	W.S. Nechodom	December 2, 1964
RL-NRD-104	Uncl.	Interim Test Report No. 2 - Plant Service Boiler Base Load Reduction Program (PT-NR-24)	D.L. Renberger	November 24, 1964
RL-NRD-105	Uncl.	Control Aspects of a Large, Dual-Purpose Nuclear Reactor Plant	D.W. Leiby	December 10, 1964
RL-NRD-106 RD	Confid.	N-Reactor Phase II Operation Hazards Summary Report (Comment Issue)	R.E. Trumble, N.R. Miller J.R. Bolliger	December 18, 1964
RL-NRD-107	Uncl.	Interim Test Report No. 3 - Plant Service Boiler Base Load Reduction Program (PT-NR-24)	L.D. Smith	December 16, 1964
RL-NRD-119	Secret	N-Reactor Department Semiannual Report of Research and Development	Research and Engineering Staff	December 15, 1964
RL-NRD-124	Uncl.	Some Possible Small Digital Computers for Application to N-Reactor Process Tube Power Calculations	C.F. Poor	December 14, 1964
RL-NRD-127	Uncl.	Preliminary Specifications for a N-Reactor Fuel Element Pre-Irradiation Enrichment Analyzer	D.O. Allred	December 16, 1964
RL-NRD-135	Uncl.	Evaluation and Remedial Considerations of N-Reactor Rupture Monitor System False Indications	D.O. Allred	December 22, 1964
RL-NRD-147	Uncl.	Nondestructive Analysis of Pu ²⁴⁰ Content of Post-Irradiated N-Reactor Fuel Elements	D.O. Allred	December 31, 1964

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VISITORS

<u>Name</u>	<u>Company</u>	<u>Contact(s)</u>	<u>Date(s)</u>	<u>Purpose</u>
Doug. Jessup	Scientific Data Systems, Bellevue, Washington	C. F. Poor	12/7/64	Discuss capabilities of their small computers
Ralph Blair Jeffrey Gorman George Wyss	Bettis Atomic Power Lab. Naval Reactors Div., AEC RLOO-AEC	R. H. Shoemaker	12/15/64	Discuss primary pump seal experience
G. J. Haysom	Computer Control Co., Inc. Kent, Washington	C. F. Poor	12/17/64	Discuss capabilities of their small computers.

VISITS

J. W. Riches	AIF Conference, San Francisco		11/30/64	Attend meetings
J. L. Benson	ANS-AIF Conference, San Francisco		through	
M. C. Leverett	" " " "		12/3/64	
A. E. Guay	E. I. duPont	R. T. Huntton	12/7/64	Discuss irradiation
J. E. Hammond	Savannah River Laboratory		through	performance of alloyed
			12/9/64	fuel elements
R. E. Hall	Burns & Roe, Inc.	I. Gabel	12/7/64	Discuss conversion
	Hempstead, Long Island, N.Y.		through	
			12/10/64	

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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 December 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.

Manager
Research and Engineering

MC Leverett:LCC:vb

DATE

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8/15/94

END