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HANFORD ATOMIC PRODUCTS OPERATION - RICHLAND, WASHINGTON

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MONTHLY RECORD REPORT
RESEARCH AND ENGINEERING OPERATION
N-REACTOR DEPARTMENT

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MONTHLY RECORD REPORT
RESEARCH AND ENGINEERING OPERATION
N-REACTOR DEPARTMENT

May 31, 1965

M.C. Leverett

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PLANT STATUS

Physics Status

On May 24, 1965 at 8:00 a.m. the power level was 3680 mw, average fuel exposure was 1146 MWD/T, reactivity held in the rods was 8.37 mk (xenon equilibrium had been obtained), the hottest graphite temperature was 1030 F, and the flattening efficiency was 79%.

Stack Measurements

A document, RL-NRD-331, covering measurements of net change of dimension of the reactor stack was issued. The maximum change in dimension is near the center of the top surface of the stack where it is found that a net contraction of about .12 inch has occurred since the zero measurement was taken on February 23, 1964. This amount of contraction is approximately what had been predicted.

Process Tube Outlet Temperature Detector Failures

Progressive failures of the resistance temperature elements utilized in the all-tube temperature monitor has required their removal from the normal readout circuits. Currently, approximately 5% (51) of the process tube RTD's must be read manually in the instrumentation room rather than from temperature map printouts. The failure is a result of a vibration at the sensing wire through the micaprotective shield such that the RTD grounds internally to the RTD body. This ground places an excessive load on the power supply for the bridge readout circuitry and disturbs the accuracy of the remaining measurements. As a result, when a failure occurs the element is disconnected from the normal readout circuits and set up for manual readout. Replacement RTD's in sufficient quantities are not yet available for repair of those out of service. Manual readout on an interim basis is a safe and acceptable mode of operation.

Fuel Element Rupture - May 15

On May 15, a fuel element rupture occurred in process tube 0643. The first indication of the rupture was shown by the tripping of the gross gamma monitor. The activity of the gross gamma monitor was approaching such a magnitude that the background noise caused it to enter the tripping threshold. A check of the gamma energy monitor showed a signal building up on the No. 1 spectrometer. A marked-up copy of the strip-chart recorder for this particular turret of the gamma energy monitor is shown in the chart on the next page. The individual spikes representing the gamma energy monitor signal in tube 0643 are accentuated with dots at their maximum value points which are connected by the dashed line. This dashed line shows the exponential buildup of fission product activity as the rupture proceeded.

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There were indications of increases of fission product activity in the primary loop water at this time. After the reactor had been shut down, examinations of the downstream dummies in process tube 0643 showed more than average contamination. It is clear that the operation of the gamma energy monitor system has significantly improved following its modifications under PT-NR-38.

Examination of the ruptured fuel element was initiated in the Radiometallurgy Laboratory. This work showed that the outer fuel tube had two small perforations on its inner surface near but not at the location of a locking clip. The perforations are small and their cause has not been determined. The section of the fuel element containing the perforations has been removed from the rest of the fuel element and detailed metallurgical examination is proceeding. The failed fuel element was located in the third position from the downstream end of the fuel column. Exposure, power density and temperature were therefore low as compared with reactor average.

Earthquake

At 8:28 a.m. on April 29 a mild earthquake shock was experienced. The reactor seismoscope system received one trip, which is insufficient to cause scrambling of the reactor. The seismoscopes are set to trip at MM 4 (on the Modified Mercalli scale) and from this information and other information, it is inferred that the intensity of the earthquake at the Hanford site was about 3.5 to 4.0 on the Modified Mercalli scale. No damage or interference with plant operation was experienced although swinging light fixtures and other minor manifestations were evident to employees at N-Area.

TECHNICAL ACTIVITIES

1.25 Enriched Coproduct Test

The 36 columns of fuel in this test have reached ~800 MWD/T average exposure with no evidence of malperformance. Six central and two peripheral columns will be discharged at the next outage and replaced with standard tube-in-tube fuel. Flux distribution, conversion ratios, and fuel evaluation including swelling will be obtained from the discharged fuel. Since equipment is not available at Hanford to determine tritium content, the RLOO-AEC is making arrangements with SRP for the determination of the tritium content of these targets.

Fabrication and testing of the fuel separator and target stripper equipment have been completed. Numerous equipment dry-runs for equipment operability have been made. On two occasions the equipment has successfully separated the target from the fuel and stripped the zirconium can from unirradiated targets. Installation of the separator and stripping equipment in the storage basin, in-place testing of equipment, and operator training still remain to be done.

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STRIP-CHART RECORDER FOR TURRET OF
GAMMA ENERGY MONITOR SIGNALLING
FUEL ELEMENT RUPTURE

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Monitor Thermocouple Trains - 1.25 Coproduct Test

Thermocouple trains were installed in four process tubes containing 1.25 enriched coproduct test columns during the April outage. Measurements from these trains have indicated a slight shift in heat balance from the inner to outer annulus. However, this shift (on the order of one-half percent on the average) has not been sufficiently great to suggest the existence of any abnormalities due to swelling. Rather, the consistency of data indicate that abnormalities (such as buckling of inner cladding of the drivers) would be readily detectable. Since the fuel assemblies are now reaching exposures at which swelling could occur, monitoring frequencies have been increased. Simplified methods of reducing data have also been developed so that rapid interpretations can be made by shift engineers if the need arises.

1.95 Coproduct Test

The fuel assemblies for the 1.95 block test are in shipping boxes ready for delivery to the reactor site. Publication of the 1.95 production test is pending until the calculated safe handling mass in the discharge basin has been verified by critical mass experiments.

The critical approach experiments on 1.95 driver fuel pieces have been completed. Results indicate a minimum critical mass of 1000 pounds and a critical mass per unit area of 210 lb/ft³. Handling and processing of 1.95 drivers have been performed utilizing a calculated minimum critical mass of 770 pounds, or 77% of the experimental value. It is necessary to set up another approach-to-critical experiment using complete target fuel assemblies. Preparations for this experiment are under way, and data will be available for analysis early in June.

Target Development - Coproduct Program

Special metal target developmental studies are centered around a "buried poison" concept. This consists of a Zr-2 clad lithium-aluminum rod identical to that in the 1.95 coproduct design except that it has a 1/16 inch to 1/8 inch full length axial hole filled with neutron absorbing powder or rod. In this form, the poisonous material would be expected to cause no more than a 10% production loss. In case of loss of coolant and expulsion of LiAl from the Zr-2 jacket, the poison would be dispersed throughout the inside of the cladding to create a highly neutron absorbing device.

Melting tests have been performed on two irradiated targets, one filled with powdered boron and the other with powdered boron carbide. Comparison of 305 pile measurements before and after melting showed that the melted targets were between 92% and 96% as black as the unmelted ones -- even though 90% of the LiAl had been expelled from the boron-filled target and

70% from the boron carbide one. Two more drilled targets will be tested in a similar manner; one with gadolinium oxide and one with copper-cadmium alloy.

A lithium-aluminate ceramic target element is also being readied for charging with the 1.95 coproduct test. No fabrication, performance or hazards problems are anticipated with this material. The separation scheme is still being developed, however, and research on tritium extraction from ceramic has been accelerated. A number of approaches are being examined. These include 1) heating through several phase changes, 2) fusing in borax, and 3) electrolysis and chemical reduction in fused salts.

Physics Studies - Coproduct Fuel Design

Calculated physics parameters have been assembled for a 21 lb/ft tube and rod coproduct element, where the fuel charge is shortened to 32.8 feet, thus increasing the rod system strength sufficiently to satisfy total control requirements (not considering flooding or target melting). Advanced Technology is performing an economic analysis of these data.

Transient calculations are being performed for supercell designs at three driver fuel weights (22, 24 and 26 lb/ft) and three target-to-driver ratios (1/6, 1/7 and 1/8) to determine a limiting case from a reactivity control standpoint. These calculations, however, will again ignore the effects of flooding and target melting. When these calculations are complete, a case equivalent to that existing for the tube-and-rod design will be in hand. Calculations to define a solution to the target melting problem on this single supercell design will then be performed.

Long-Term Reactivity Changes with Fuel Exposure

Monitoring of the long-term reactivity transient has continued during the month; the data confirm last month's extrapolations. Reactivity losses are occurring at slightly more than 0.1 mk per day at full power load. The reactivity will decrease to about 5 mk in control rods at about 1350 MWD/T and to zero at about 1700 MWD/T. Somewhere between these points, the reactor will become inadequately controlled, demanding a power cut or the insertion of more spike enrichment. Since fuel exposure accumulates at ~11 MWD/T - day, and the average is now about 1050 MWD/T, the exposures indicated cover a range of 27 to 60 full power days from now.

Process Physics Support

A physics training session for Plant Engineering personnel has been developed to improve their knowledge related to the physics behavior of the reactor. This material will also be used for presentation to cleared WPPSS personnel.

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Simulated Driver-Target Test, PT-NR-36

Pilot fabrication of three alloy compositions was completed. Review of the data and the autoradiographs indicated that the 150 Fe, 350 Si alloy fuel and the 250 Fe, 250 Si, 250 Al alloy fuel which were fabricated from direct-cast billets had marked clad variation patterns in the form of axial streaking. This variation was barely within current tester limits. As the original billets were selected from a group which had generally poor as-cast quality also, the fuel will not be accepted for in-reactor testing.

The fuel from the 350 Fe, 350 Si, 800 Al alloy have excellent clad thickness uniformity and are generally acceptable for reactor use. This material was cast as large ingots and extruded to billet size prior to heat treatment. Four charges of this material are being prepared for the next charge-discharge outage.

Zircaloy Tubes

The recent observation at K reactors of hydriding of the downstream end of certain Zr-2 process tubes suggests that similar reactions may possibly occur in the N-Reactor graphite cooling system. Both the K reactor process tubes and the N-Reactor graphite cooling system tubes were manufactured by the same vendor using essentially the same fabrication process and were installed in an unautoclaved condition. Coolant temperature of the graphite cooling system is only a few degrees less than the K reactor system. A significant difference in water quality exists however; the graphite cooling system has a pH=10 with LiOH or NH₃ additive whereas the K system has a pH≤7.0. The H₂ gas content of the graphite cooling system is much higher than at the K reactors. In view of the similarities and lack of explanation for K reactor hydriding, plans are under way to remove two of the N-Reactor graphite tubes for examination.

No evidence of such hydriding has been found in the examinations which have been made of the process pressure tubes removed from the KER loops. These tubes have had a service environment equivalent to N-Reactor process tubes. Hydriding similar to that found in the K reactors is therefore not anticipated for N-Reactor process tubes.

Use of Nitrogen as a Reactor Gas Atmosphere

The addition of nitrogen to the gas atmosphere is being evaluated as a technique for increasing graphite temperatures to decrease the rate of graphite contraction.

The hazards associated with the use of nitrogen in the gas atmosphere have been evaluated. It appears that the desired graphite temperatures could be obtained by operating with a gas atmosphere consisting partially of nitrogen. It is

possible to schedule the nitrogen addition so as to preclude any nuclear safety problems and minimize any potential for detrimental effects to reactor components. This study indicates that the reactor nuclear safety can be maintained with graphite temperature in excess of design temperatures but some restrictions in other plant operating parameters would be required. Increases in graphite temperatures above those anticipated with a gas composition consisting of 50% nitrogen and 50% helium would also require further analysis of the potential influences on the Zr-2-graphite compatibility relationships. Work is continuing to more exactly define the various interdependent relationships.

Systems Analysis

Analog simulations are under way to determine the effect of primary pump coastdown, cell valve failure, post-scrum main steam header setpoint, and number of injection pumps in-service on primary volumetric contraction.

Typical results are shown in the attached figure for Phase I operating conditions and four injection pumps on-line. The power levels giving the equivalent primary coolant volumetric contraction are plotted as functions of inoperative steam generator cell bypass valves at two different primary pump coastdown flows. As shown on the 33% curve, which is close to the actual coastdown, and for one inoperative bypass valve, the power level giving the equivalent contraction is about 3625 megawatts. The total equivalent pressurizer level change is 14 feet.

In an actual reactor scram from 3650 megawatts on May 14, the pressurizer level dropped from 20 to 6 feet, and the total flow went from 290,000 lb/min to 100,000 lb/min (34%); this shows good agreement between the model and the reactor plant.

Data obtained in this way is very useful in establishing proper limits of operation under non-standard as well as standard conditions.

Upgrading Central Data Logger

An improved means for the rapid computation of reactor operating parameters would enhance reactor safety and operating efficiency. Efforts are therefore under way to upgrade the Central Data Logging Facility to provide a more flexible service output. A study is in progress to determine the changes and funds which would be necessary to accomplish these objectives. The following additions or modifications are being analyzed.

1. Modify tape reader to handle eight channel paper tape
2. Add a card reader

POWER LEVEL GIVING EQUIVALENT PRIMARY COOLANT
VOLUMETRIC CONTRACTION VS NUMBER OF INOPERATIVE
STEAM GENERATOR BYPASS VALVES

Phase I Operating Conditions
Four Injection Pumps

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3. Read in directly into the computer the riser inlet temperatures, the process tube outlet temperature, the percent delta-P.
4. Add a 4096 39-bit word memory unit
5. Install two typewriters, one for a tube power map output, and the other for miscellaneous output.
6. Make it possible to calculate any selected process tube power and display the results on a digital display.

Thermocouple Train Measurements - PT-NR-33

During the April outage the four thermocouple trains installed in the reactor were rotated 180 degrees to locate the outer annulus thermocouples at the top of the annulus. Data from thermocouples in this position are mutually consistent with all others obtained previously, from the standpoint of energy distribution in the three subchannels of the fuel column and around the outer annulus. The table summarizes the data obtained thus far in production test PT-NR-33.

Enthalpy Imbalance Data from PT NR-33

<u>Fuel Subchannel</u>	<u>Ratio: Subchannel Enthalpy Rise to Bulk tube Enthalpy Rise</u>
Center	1.10 to 1.15
Inner Annulus	1.06 to 1.11
Outer Annulus: Bottom	1.02 to 1.15
Top	0.716 to 0.791

Examination of these data show that;

1. Enthalpy imbalances appear to be independent of tube power,
2. No discernible shift has occurred in imbalances since the first measurements were taken (February 1), indicating that swelling has not yet created significant changes in fuel dimensions.
3. At a reactor power level of 92%, a 60 F difference was indicated between top and bottom of the outer annulus at the downstream end of the fuel column.

Eccentricity effects appear to be largely responsible for the temperature differences in the outer annulus. If any changes in fuel design are made to produce a better balance in temperatures among the three subchannels of the fuel, such modifications must include measures to reduce the degree of eccentricity that can exist in the outer annulus. Otherwise, mixers would be required. An addendum to the production test is being prepared to evaluate the potential magnitude of outer annulus imbalance that could occur in the reactor as a result of eccentricity.

The development by Thermal Hydraulics (BNW) of devices to measure flow splits in the fuel column is complete. Orders have been placed for a sufficient number of thermocouples to make an additional 16 trains, four of which will be used to monitor performance of the 1.95 test load.

Environmental Effects

No significant release of radioactive effluent occurred during this report period.

Nuclear Safety Audit

A nuclear safety audit of the 333, 306, 303J and 3715 buildings was performed April 16. No violations of nuclear safety specifications were observed.

Operating Safety Limits

The Operating Safety Limits were revised to accommodate Phase II operation and to reflect knowledge gained from plant operating experience. A draft has been routed to RLOO-AEC for their review.

Advanced Technology Studies

Polonium-210

Utilizing the 68 spare channels and 100 process channels in N-Reactor, 1050 grams of polonium could be produced per year with a capital cost for reactor modification of 1.7 million dollars. Of the 1050 grams of polonium produced, 500 grams could be made in the 68 channels and 550 grams in the 100 process channels.

Previous calculations of polonium production in the 68 fringe tubes indicated a yearly yield of 750 grams of product. The lower yield results from using a lower cross section for bismuth which corresponds to the average neutron energy in the fringe tube section.

Capital costs for reactor modification include installation of the following:
1) filtered water supply and raw water backup headers from 109 once-through

18 inch supply header to 105 front face; 2) thin wall zirconium tubes, nozzles and connectors for the 68 fringe tubes; 3) drain piping for all tubes; 4) instrument and electrical equipment; 5) charge-discharge equipment; and 6) plant engineering.

Neptunium-237

The production of Np^{237} from recycled uranium and the subsequent conversion to Pu^{238} were studied further.

The attached figure shows the unit cost for producing neptunium from varying blend enrichments from the Cascades. If the average conversion cost ($\sim \$40$ per gram) of $\text{Np}^{237} \rightarrow \text{Pu}^{238}$ is added the result is the average unit cost for Pu^{238} over a 15-year campaign period.

A second figure shows the kilograms of Pu^{238} and Np^{237} over the campaign period. The value marked at a blend of 1.7% with the note, $e = 1660$ ppm, is the value calculated for recycle using power reactor fuels with U^{236} content of 1660 ppm for blend enrichment.

Plutonium-238

A third figure shows similar results for Pu^{238} . The production of Pu^{238} from an annual feed of 50 kg of Np^{237} has been summarized in the Pu^{238} report, RL-REA-2029 (prepared by Engineering Staffs of CPD, IPD, NRD), describing the conversion process and the production results. For comparability with Savannah River studies the calculations were repeated using 135 days cooling time. It was found that production of 460 kg of Pu^{238} in N-Reactor from a fixed Np^{237} input compared favorably with the SRP production of 450 kilograms using similar ground rules.

Average Unit Cost for Np^{237}
15 Year Campaign

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Np²³⁷ Production by Recycle

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Pu²³⁸ Production

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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 Thorium-Uranium Crud Monitor Elements in N WK Kratzer, 5/21/65		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 Completed	
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		
21	HW-83200	Optimization of Secondary Coolant Supply Operational Modes, FJ Mollerus, Jr., 7/8/64			X	
21 SUP	HW-83261 SUP	Surge Tank Pressure Test, FJ Mollerus, Jr., 7/8/64			X	
21 SUP	HW-83442 SUP	184-N Turbine Generator Tests FJ Mollerus, Jr., 6/27/64			X	
27	HW-84238	Steam Generator Moisture Carryover Detection LD Smith, 9/24/64			X	
27 SUP	HW-84238 SUP 1	Supplemental Testing Procedure			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	

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PT No.	Document No.	Title, Author, Date	Approved	In Progress	Com- pleted	Final Report
30	HW-84232	Evaluation of Ammonia for Controlling Oxygen in the Graphite Cooling System, WK Kratzer, 10/5/64			X	
30	RL-NRD-196 SUP	Supplement A, 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	HW-84401 ADD 1 ADD1	Reactor Enthalpy Imbalance Measurements and Inlet Riser Strap-on RTD Calibrations, DO Allred, 11/2/64		X		
33	HW-84401 ADD 2 ADD2	Addendum 2: Eccentricity Experiments RH Shoemaker, 5/18/65		X		
35		(rough draft - not issued)				
36	RL-NRD-55	Simulated Driver-Target Element, AC Callen, 11/6/64	X			
38	RL-NRD-193	Evaluation of N-Reactor Rupture Monitor Modifications for Desensitizing the Response of the Gamma Energy Monitor to Activated Corrosion Products, DO Allred, 1/20/65		X		
38	RL-NRD-198 ADD ADD 1	Addendum 1 to PT-NR-38 DO Allred, 5/7/65		X		
39	RL-NRD-204	Operation of "E" Panel Transfer Switch - 184-N JR Bolliger, PC Althoff, 1/23/65			X	
40		(rough draft - not issued)				

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PT No.	Document No.	Title, Author, Date	STATUS			
			Approved	In Progress	Completed	Final Report
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				
44	RL-NRD-372	Evaluation of Dump Condenser Condensate Outlet Valve CONDV-203 Control Characteristics, LD Smith, 5/13/65				X
45	RL-NRD-390	Increase in Primary Coolant System Pressurizer Pressure, RE Hall, EA Grimm, 5/27/65				

PERSONNEL CHANGESAdditions

G. H. Bateman, Technician, transferred to Thermal Hydraulics effective May 3.

Removals

C. A. Mansius, Engineer, transferred to Project Section effective May 17.

J. W. Vanderbeek, Specialist-Radiological Engineering, terminated on May 14 to accept employment with Walston and Company, Investors.

SECURITY VIOLATIONS

Georgia Stoffels, open safe, May 11, 1965.

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>
HW-82162 REV 1	S	Hazards Review, N-Reactor 1.25% Coproducer Fuel Element Test	NR Miller and WS Nechodom	3/31/65
HW-84401 ADD 2	U	PT-NR-33, In-Reactor Enthalpy Imbalance Measurements; Addendum 2: Eccentricity Experiments	RH Shoemaker	5/18/65
RL-NRD-193 ADD 1	U	PT-NR-38, Addendum 1, Evaluation of N-Reactor Rupture Monitor Modifications for Desensitizing the Response of the Gamma Energy Monitor to Activated Corrosion Products	DO Allred	5/20/65
RL-NRD-274	C	N-Reactor Plant Response to a Scram	EE Leitz	3/15/65
RL-NRD-293	S	Product Grading at N-Reactor	EE Leitz	4/16/65
RL-NRD-331	C	Pile Motion	VJ DeJong	4/19/65
RL-NRD-343 RD	C	Draft - Operating Safety Limits, N-Reactor (Phase I and II Operation)	Staff, Process Eval. & Control	4/27/65
RL-NRD-344	S	N-Reactor Department Semiannual Report of Research and Development	Staff of Research and Engineering	5/19/65
RL-NRD-356	U	Trip Report of Foreign Travel, April 1 to April 15, 1965	DH Curtiss	5/10/65
RL-NRD-372	U	PT-NR-44, Evaluation of Dump Condenser Condensate Outlet Valve CONDV-203, Control Characteristics	LD Smith	5/13/65
RL-NRD-374	C	Charts Used in Presentations before ACRS on N-Reactor Experience and Phase II Operation, Richland, Washington, May 7-8, 1965	Edited by RE Trumble	5/14/65
RL-NRD-387	S	Report of Investigation of the Second Fuel Element Rupture in N-Reactor (Tube 0643, May 15, 1965) Daily Report No. 6, May 24, 1965.	AE Guay	5/25/65

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VISITORS

<u>Name</u>	<u>Company and Location</u>	<u>Contact(s)</u>	<u>Date(s)</u>	<u>Remarks</u>
LF Coffin, Jr. WE Tragert DW White	Research Laboratory, GE Schenectady, New York	Milton Lewis	May 3-4, '65	R&D Program Review: Technical Audit
Prof. FN Rhines	Metallurgy Eng'g Department U. of Florida, Gainesville,	TW Evans	May 6-7, '65	Consult on intermediate temperature brittleness in metal with R&E staff
Hanford Subcommittee of ACRS	Advisory Committee on Reactor Safeguards, AEC	RL Dickeman	May 7-8, '65	Safety presentations on N-Reactor

VISITS

Milton Lewis	AAUW, Ellensburg, Washington		May 11, '65	Speech to AAUW
RE Hall MC Leverett WS Nechodom	ACRS - AEC Headquarters, Washington, D.C.		May 13-14, 1965	Safety presentations on N-Reactor
WK Kratzer	General Dynamic Corp. Electric Boat Division Groton, Connecticut	Tom D'Muhola	May 18-20 1965	Obtain info on decon- tamination processes and chemicals. Observe chemical waste disposal facility.
DH Curtiss	Research Laboratory, GE Schenectady, New York	WE Tragert	May 18, '65	Attend Symposium on Zirconium Technology
RE Trumble	APED, General Electric, San Jose, California	JC Russ	May 17-21, 1965	Attend meetings to review KRB Reactor
DW Leiby	APED, General Electric, San Jose, California	JC Russ	May 10-13, 1965	Nuclear safety functional review - In-Core Flux Monitoring
KO Creek	Granger Associates, Palo Alto, California	D Meadows	May 21, '65	Review of radiation resistant TV camera

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