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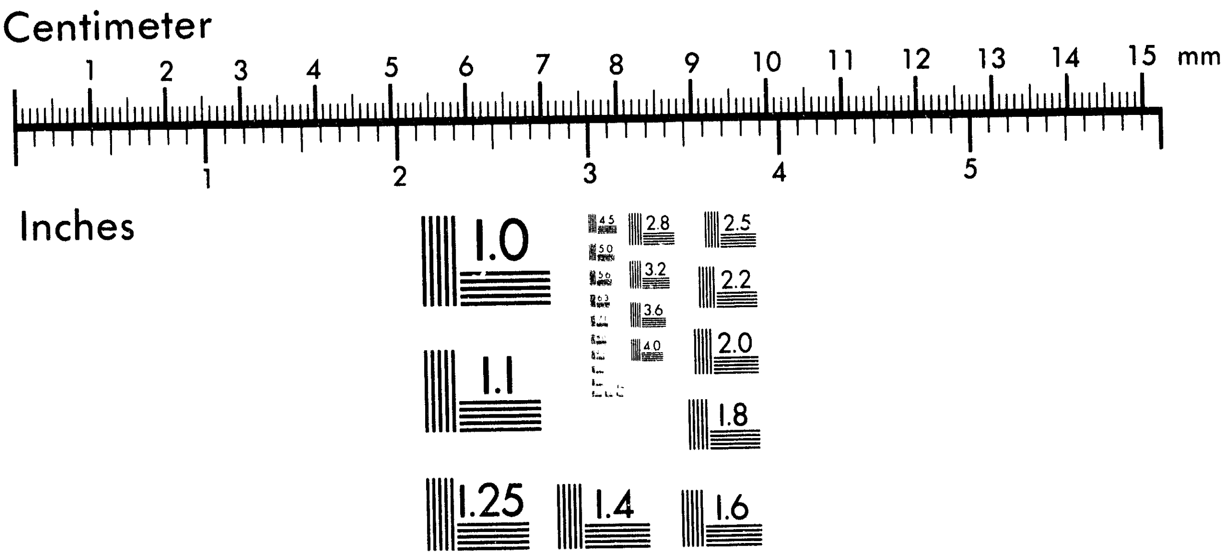

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GENERAL ELECTRIC

HANFORD ATOMIC PRODUCTS OPERATION - RICHLAND, WASHINGTON

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RL-NRD-150 3C

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TITLE

MONTHLY RECORD REPORT RESEARCH AND ENGINEERING N-REACTOR DEPARTMENT

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M. C. Leverett

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RL-NRD-150 3C

This document consists of 19 pages. [redacted]

MONTHLY RECORD REPORT

RESEARCH AND ENGINEERING OPERATION

N-REACTOR DEPARTMENT

March 31, 1965

M. C. Leverett

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TECHNICAL ACTIVITIES

Simulated Driver Target Test (PT-NR-36)

In addition to the standard and British (400 Fe-800 Al) alloys scheduled for use in the 1.95 enriched coproducer test, three special test alloys were made up. These were comprised of the following compositions: 150 Fe-350 Si, 250 Fe-250 Si, and 250 Al. Much of this material cracked at the quench step after heat treating in the ingot form. Some of the higher alloy material was successfully handled by heat-treating in the billet form and these billets are in transit to HAPO.

To help solve the problems of alloy billet manufacture, two actions have been taken. First, directly cast billets of the crack-prone alloys have been ordered. This action is based on the unexpected good coextrusion behavior of sample direct-cast material received last month. This will supplement the trial of wrought material heat treated in billet rather than ingot form. Second, a modest order of British composition has been made to assure through early trial that the desired process route, i.e., massive billet - no subassembly, will be effective for this alloy.

Target Melting Studies

A fourth irradiated target element was heated to gain insight into the mechanism by which irradiated coproduct targets might fail in a temperature excursion. The target chosen for this test had been irradiated to 1.5 gas volume ratio (GVR). It was heated in a quartz tube so that it could be visually observed during the heating cycle. During heat-up, the target moved from relief of internal stresses and, as a result, contact between the target and the control thermocouple was lost at 500 C. Visually, the temperature of the target, when it failed, was estimated to be 700 C. A small hole was seen to open in the cladding and molten target material spewed out and impinged against the quartz tube due probably to hydrostatic pressure exerted by tritium and helium in the alloy.

The temperature of cladding failure in this target element was lower than encountered with the other three irradiated targets that were fail tested. Calculations indicate that at this failure temperature there is insufficient internal gas pressure from 1.5 GVR to fail the cladding. It is thought that the absence of the iron tube (removed to see the target) that has been used as the induction heating coupler caused sufficient inductive stirring in the molten core metal to promote the rate of corrosion penetration. Justification for this explanation comes from the fact that five unirradiated targets, which were used in similar mockup tests to adjust power regulation to attain the correct heating rate, failed between 700 and 900 C. This was an entirely unexpected result because earlier direct heating tests showed that unirradiated targets remain intact upon heating to 1100 C and are more prone to fail during cooling.

[redacted]

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[redacted]

Coproduct Experimental Physics Program

Certain results of PCTR and exponential pile tests are now available; these results, and the status of remaining tests follow.

PCTR:

k_∞, wet, with targets	$= 1.117 \pm .006$
k_∞, dry, with targets	$= 1.027 \pm .003$
Poison spline worth (spline in each x-coolant tube) $\frac{\Delta k_\infty}{k_\infty}$	= 61 mk
k_∞, wet, no targets	- tentatively planned for mid-April
Target poison worth comparisons	- planned for early April
Control rod worth comparisons	- " " " "

EXPONENTIAL PILE:

Exponential diffusion length	$= 41.6 \pm 6$ cm
b_m^2, wet, with targets	$= 197.4 \pm 1.6\,\mu b$
b_m^2, dry, with targets	$= 22 \pm 1.7\,\mu b$
b_m^2, wet, with targets and control rods	$= 12.2 \pm 2.9\,\mu b$
b_m^2, wet, with targets, $\sim$4.6 v/o water in voids	$= 232.9\,\mu b$
b_m^2, wet, with targets, $\sim$9.2 v/o water in voids	$= 272.6\,\mu b$
b_m^2, wet, with targets, $\sim$13.8 v/o water in voids	$= 286.2\,\mu b$
b_m^2, wet, with targets, $\sim$18.4 v/o water in voids	$= 290.1\,\mu b$

Neither of the experiments exactly duplicate the N-Reactor lattice; however, conventional calculational techniques which use the test geometries, are used to predict and confirm the observed data.

[redacted]

[redacted]

Technique	k_{∞}, wet	k_{∞}, dry
Experiment	1.117	1.027
FLEX-3	1.116	1.040
4-group diffusion (HFN)	1.090	not reported

Critical Mass Experiments - 1.95 Material

Loading for these experiments began on March 25. The critical mass of 1.95 percent enriched drivers, and of the assembled coproduct fuel, will be measured. Target date for completion of these experiments is April 30, 1965.

Storage Basin Cubicles

Basic physics calculations have been completed, using 4-group diffusion theory, which indicate that the proposed cubicle storage system will be safe and will satisfy the requirements for fuel storage, both with the present fuel load, and with metallic coproduct fuel enriched to 1.95 per cent U-235. In addition, preliminary estimates indicate that storage of uranium oxide fuel up to enrichment levels of about 3.5 percent will be feasible. Detailed calculation of the parameters of an oxide fuel element at 3.5 percent U-235 are being pursued.

Attenuation measurements through test slabs at the C-Reactor test facility have been completed. Calculations will be normalized to the observed, unattenuated spectrum, and the accuracy of the theoretical attenuation calculation verified.

Graphite Program

Considerable effort has been spent recently in reviewing available data and current programs which have been used to predict N-Reactor life expectancy and to outline future programs, and the basic graphite data and reactor life predictions have been presented to RLOO-AEC. A visit to Savannah River Plant was scheduled, with RLOO-AEC and Washington-AEC representatives attending also, to further evaluate the technical and administrative feasibility of using space in a high flux region of a Savannah River reactor to perform needed irradiation experiments.

Coolant Systems Inspections

Several components in the primary coolant system were visually examined for pitting, film buildup and any observable condition which might be deleterious to long term operation of the carbon steel systems. All of the components examined exhibited a good magnetite film on all surfaces exposed to the primary coolant. Slight pitting was noted in the return lines from steam generator 1A and 1B but it is not of a magnitude to cause concern.

[redacted]

The secondary system is also, in general, in excellent condition. Most surfaces have a light magnetite film, with a light red oxide film overlay. The steam generators have a nominal amount of crevice corrosion between the tubes and the tube sheets, and between the tubes and their supports. Therefore, the primary and secondary systems do not, at this time, present any evidence of corrosion which should cause concern as to their future reliability.

Steam Generator Decontamination

Decontamination of Cell 2 steam generators by the alkaline permanganate-sulfamic acid process reduced radiation levels of stainless components by roughly a factor of four to five. Decontamination effectiveness was less than expected because of the loss of chemicals by leakage into other portions of the primary system.

Primary Cooling System Supply Characteristics

Supply pressure differential versus flow characteristics for the active zone and overall reactor core have been developed. These are based on recently developed hydraulic resistance equations for ex-reactor primary loop piping. Characteristics computed for dual purpose reactor operation with both five and six loops in operation are given in Figures 1 and 2. These assume a 3600 rpm pump speed and 4000 MW reactor power. The first of these figures gives supply delta-P for the active zone versus flow rate and the second, minimum reactor inlet temperatures required to produce the steam pressures corresponding to the two modes of reactor operation. The temperatures of Figure 2 were utilized in pressure drop calculations for development of Figure 1. (A similar set of characteristics is being developed for 120 per cent power operation case.)

Hydraulic Characteristics of Connectors and Fuel

An analysis of data from laboratory experiments has been completed to define the liquid flow hydraulic characteristics of the connectors and process tube (HW-77925, "Design Test 1160, N-Reactor Pressure Drop and Plugging Tests," M. Pociluyko, 1/25/63). Results of the analysis have found wide use in cooling system flow studies and it is expected that the final report will further enhance our ability to predict flows and pressure drops under various modes of reactor operation. In-Reactor pressure drop measurements across the fuel columns are being obtained to extend our knowledge of hydraulic behavior of the reactor core.

Thermocouple Train Measurements (PT-NR-33)

Despite difficulties in obtaining a sufficiently accurate and reliable readout device, and despite limited reactor operating time since the

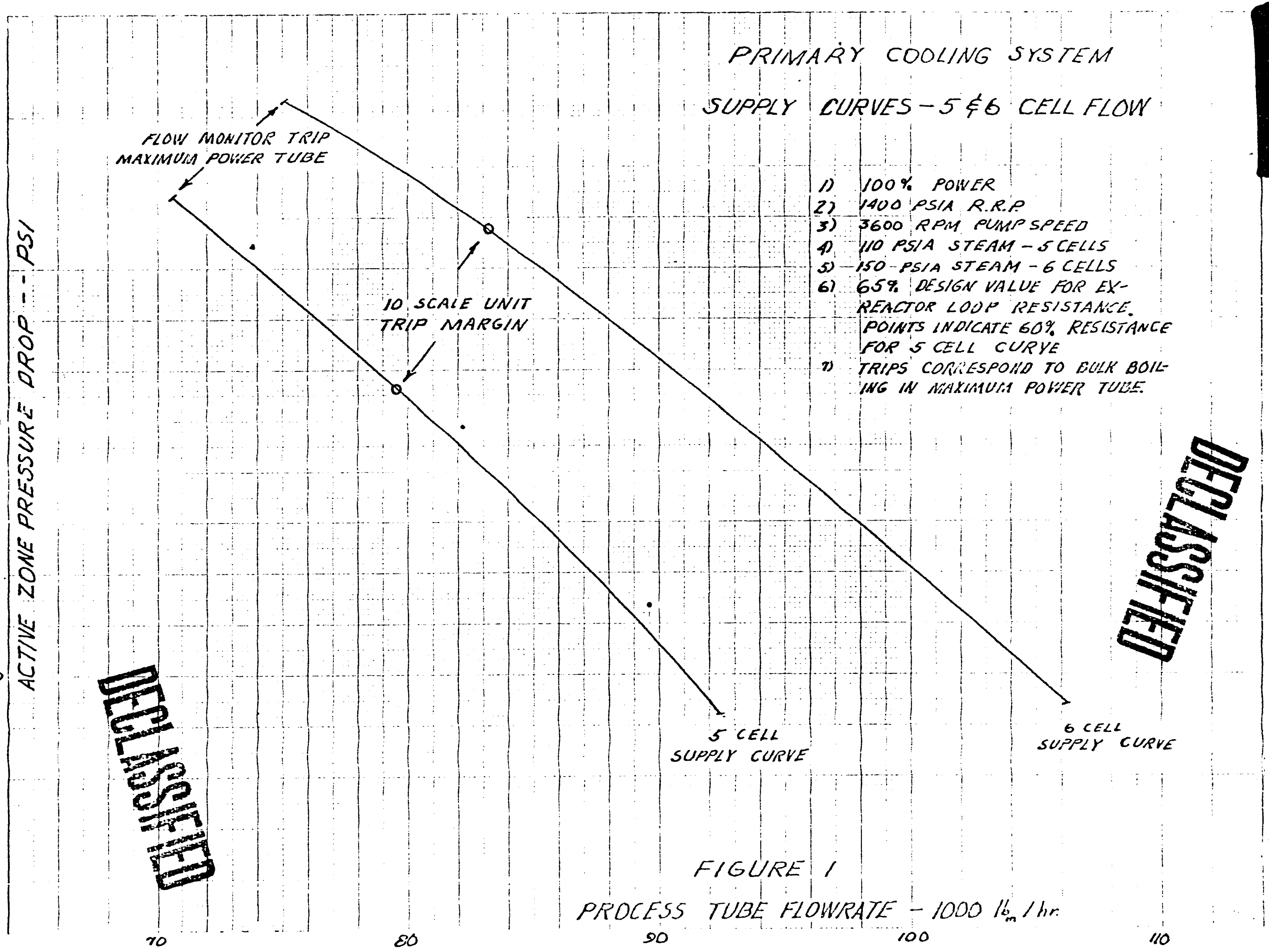

FIGURE 1

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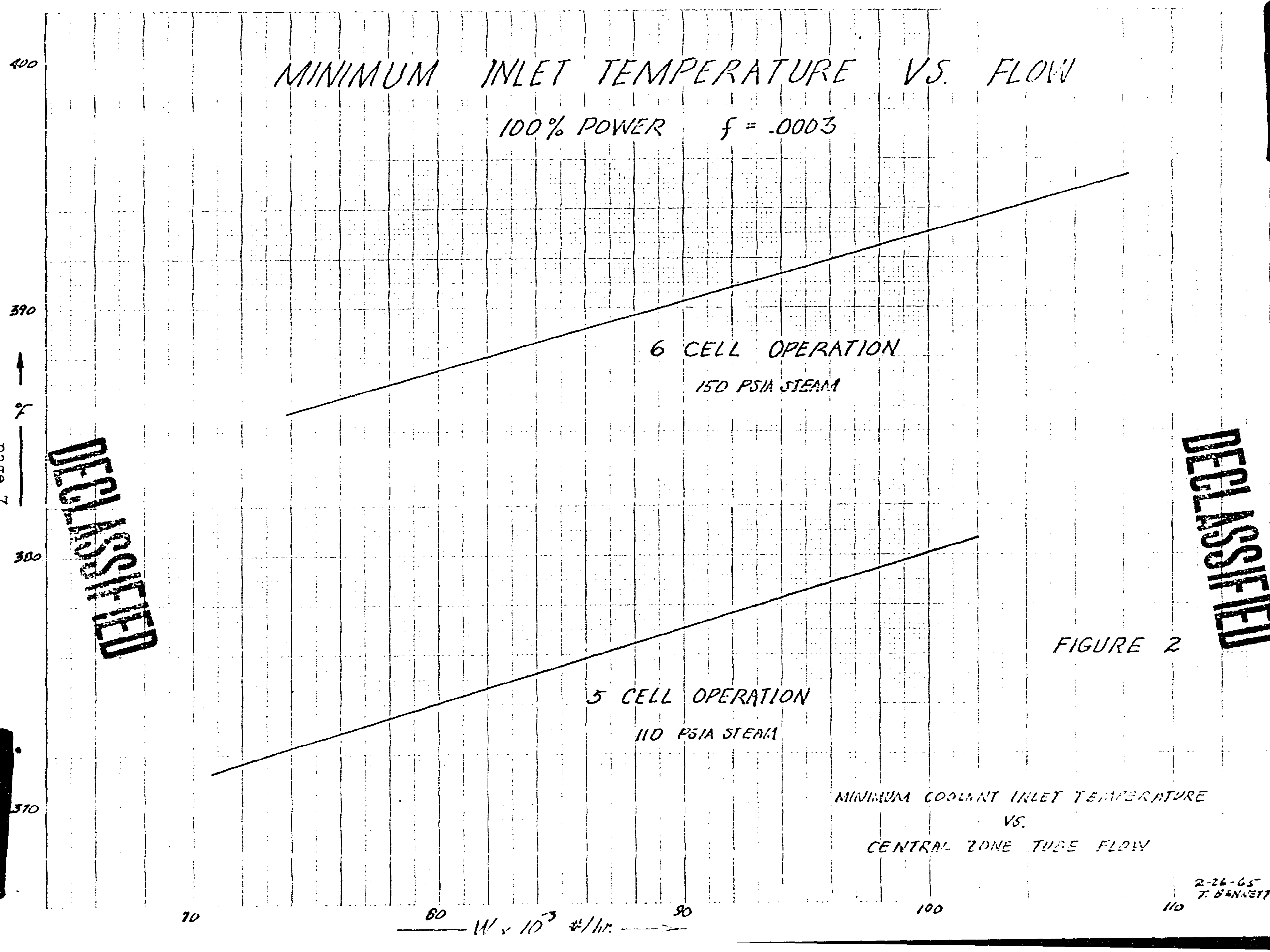

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installation of the four tube-in-tube thermocouple assemblies currently operational, considerable progress has been made toward achieving the objectives of this production test.

Five sets of data were recorded from February 1 to 26 and have now been reduced. The reduced data, without interpretation, appear in the following table. The data shows that all three subchannels are producing water of higher temperature than results from mixing the three channels together. The favored explanation for this anomaly is that the thermocouples in the outer annular subchannel are located at the bottom of the annulus. Since the fuel element axis is slightly below the process tube axis, this portion of the outer annulus produces hotter water than does the outer annulus generally. The thermocouples will be rotated to a new azimuth to test this explanation.

It is worth noting that none of the imbalances is near the 25% figure used as a technical basis for establishing process limits. A comparison between FLEX program predictions and PT-NR-33 observations to date is presented in the table below.

ENTHALPY IMBALANCE(1) COMPARISON
FLEX PROGRAM VS EXPERIMENTAL RESULTS

Channel	Average FLEX Results (Percent)	Experimental Results(3) (Percent) Train No. 2	3	4	5
Center	14	12.4	13.6	13.8	11.0
Inner Annulus	2	6.0	9.1	9.7	8.0
Outer Annulus(2)	6	3.2	4.4	8.1	13.7

(1) Enthalpy imbalance is defined as: (subchannel bulk enthalpy rise, inlet to outlet) ÷ (process tube bulk enthalpy rise, inlet to outlet) minus one.

(2) Bottom of outer annulus, both for FLEX results and experimental results.

(3) Average of five sets of data for each process tube.

Fringe Zone Orificing

An orificing scheme has been devised together with the necessary investigation of process tube thermal hydraulic limits to support a program of orificing 310 fringe zone tubes. This would increase orificing efficiency by 5%.

Key features of the program include the following:

a. Establishment of four reactor flow zones: This will include the unorificed central zone plus three zones of flow orificing where flow will be reduced by 11 to 21 percent.

b. Establishing suitable high-trip protection for the orificed tubes; tubes with orifice will operate at lower flow and, hence, further from the high-flow trip. Further, the orifice makes the flow monitor less sensitive to leaks in the connector on inlet nozzles downstream of the orifice plates.

c. Identifying fringe zone tubes which should not be orificed; the tube charge with the 1.95% material will be located in the left fringe of the reactor. These tubes will not be orificed, otherwise, they would probably limit reactor power. Also, tubes exhibiting abnormally low flow, real or because of instrument error, will not be orificed.

d. Establishing a means for administering zone temperature monitoring protection for orificed tubes.

While the current efforts should have a significant benefit, further orificing gains would likely be possible. Specifically, if nozzle inserts were used for orificing flows, the problem of high-trip flow protection would disappear.

Controlled charging of short-length fuel elements would improve orificing. Charges of standard length with short elements have increased hydraulic resistance. Such charges should not be placed in central zone tubes. Rather, they should be put in fringe-zone tubes where the additional flow reduction is actually desired.

Gamma Energy (Rupture) Monitor

Problems have been encountered in calibrating the gamma energy monitor spectrometer discriminator dials. An interaction exists between the upper and lower discriminator circuits of the signal channels. The discriminators appear to track each other. Although the two circuits are completely independent, they are mounted on a common component board. The problem is probably due to "pickup" between the two signal channel discriminators, and magnetic shielding may be required if it is deemed necessary to eliminate this effect.

The spectrometers can be adjusted as desired, in spite of the interaction, as long as the discriminator settings are not readjusted without the use of calibrating equipment. This effect does not hinder the operation of the system, it only adds to the complexity and inconvenience of adjusting the system. Additional calibration and checking equipment have been defined by Research and Engineering as being necessary for efficient bench checking and spectrometer maintenance. This equipment is on order.

The power supply transfer circuits for the gamma monitor detector tubes and count rate circuits have also been modified at Research and Engineering's request to insure proper operation. These circuits provide redundancy between the power supplies and have failed to operate on occasion. Failure of these circuits causes confusion in reactor operation and the burnout of gamma monitor detector tubes.

Pre-Irradiation Enrichment Analyzer

Development tests are continuing at BNW. A prototypic model of the tester has been completed and is currently undergoing laboratory tests.

Process Tube Monitoring

Granger Associates of Palo Alto, California, who were awarded the contract for the radiation resistant television system, started work about February 15, with delivery expected within 150 days. A newly designed solid state modular plug-in control unit will be used.

The drive unit, depth gauge, probe, control console, recorder, and inner diameter probe have all been checked out together as a system and are operable.

The push-rod has been installed in the special magazine, and the system is ready to check the operation of the ID probe through a full-length process tube.

Zircaloy Process Tube Evaluation - KW Reactor

Research and Engineering has been assisting IPD in the evaluation of a Zr-2 process tube problem. During a recent outage at KW Reactor, two suspect water leaker tubes were pressure tested. Tube 3065, fabricated by one vendor, was installed in August 1962. It was found to have a cracked front Van Stone flance. Tube 3075, fabricated by a second vendor, was installed in August 1963. It was found to leak.

Metallographic samples have shown a hydride layer two to five mils thick on the inside surface of both tubes in the region of the downstream dummy pattern.

Significance to N-Reactor is difficult to assess since the cause of the K Reactor situation is as yet unknown. However, the autoclaved surface, the hotter water (sufficient to oxidize Zr), the past experience with KER facilities all tend to predict no such hydriding with N-Reactor process tubes.

The Testing Methods Laboratory, BNW, has resumed development work on the ultrasonic hydride probe to detect 300 ppm at 3 rpm rotation speed. Good progress is being made on this probe system.

Boiler Base Load Reduction

Investigations into the steam availability following reactor scram have been initiated to define the reactor after-heat-decay curves based on actual plant operating data. Scram data have been analyzed, and data from scrams from 4000 Mw, 3600 Mw, and 2800 Mw agree quite well within the initial five to ten minutes; however, as expected, these curves are significantly higher than the calculated nuclear heat decay data provided by Reactor Physics. The difference is obviously the latent heat stored in the moderator fuel and coolant system. Therefore, previous calculations on boiler availability appear to be conservative. This program will continue to support the boiler base load reduction program and will be utilized to justify a re-evaluation of the current boiler operating restrictions outlined in PCA-25-65.

Nuclear Safety

Standards and Process Change Authorizations

New process standard A-105, Reactor Startup, was issued. A total of eleven PCA's were issued since the last report; 37 PCA's are in force.

Environmental Effects

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

Hazards Review for 1.95 Coproducer Test Load

A hazards review covering the irradiation of a 3x6 lattice block of coproducer material utilizing drivers fabricated from uranium enriched to 1.95% U-235 has been issued.

Conversion to Phase II Operation

Analog System Analysis

The initial analog model for use in the analog simulation of the N-Reactor, including the power generating facility, has been completed. Analog runs were made which included one- and two-turbine trips at equilibrium power level for six loop operation under varying operating conditions of different controller settings and time lags in control systems. The initial efforts were directed toward providing proper guidance and direction for the major phases of the study.

Data from five-loop, single turbine trip off and reactor full power condition were assembled in visual display and presentation format for a conversion status review on March 9.

Analog system changeover is underway to increase flexibility and expansion of reactor model and conversion to Phase II model. Expansion is to include reactor pressurizer characteristics, export turbine controls, deaerator tank, and increased surge tank capacities and characteristics.

Secondary Steam System Pressure Profiles

The analysis of the possible pressure distribution within the secondary steam system under various combinations of steam source and steam loading unbalance conditions has been completed. Results show that the worst case condition is that in which the plant is operating at full power with the No. 5 steam generator cell out of service and all the steam going to the dump condensers equally loaded. Preferential loading of the dump condensers, with those on header section having excess steam flow availability being loaded to 120% of their rated load, tends to alleviate the situation. Additions to the control system are being considered to provide automatic steam load distribution when either one or both turbine trip-offs require unbalanced load acquisition by the dump condensers.

Advance Studies

Graphite Heat Generation for Oxide Fuels

At Research and Engineering's request, Design Analysis has utilized the results of a newly made heat generation estimate to estimate the increase in graphite and process tube temperatures with full load of a candidate tube-in-tube oxide fuel having seven pounds uranium per foot of fuel columns. The results are shown below.

[redacted]

[redacted]

COMPARISON OF CALCULATED GRAPHITE AND PROCESS TUBE TEMPERATURES - METALLIC URANIUM AND UO_2 FUEL (TUBE-IN-TUBE GEOMETRY) - 4000 MW

	Metallic °F	Oxide °F	Increase %
Average Graphite Temperature	973	1180	21
Maximum Graphite Temperature*	1357	1700	25
Process Tube Temperature	611	680	11

*Measured temperatures at 4000 MW are somewhat lower than calculated. Maximum temperatures in uncooled filler blocks, cooled filler blocks and tube blocks are 1130 F, 842 F, and 896 F, respectively.

Pu-238 Physics Studies

These studies are 95% complete. Cases which have been developed are:

1. Flux-trap cases with metallic fuel, ranging in fuel weights from four to eight pounds per foot.
2. Flux-trap cases with oxide fuel, ranging in fuel weights from three to six pounds per foot.
3. A flux-trap case, using the present fuel geometry for driver, fuel enriched to ~1.2% U-235 (essentially equivalent to the present spike fuel).
4. A flux-trap case, using a coproduct (17 lb/ft) fuel as the driver, obtaining the reactivity to support the target columns by reducing the lithium-aluminum content of the coproduct fuel targets.
5. A true coproduct case, Pu (normal) plus Pu-238, constructed by replacing the coproduct lithium-aluminum targets with neptunium oxide.

All flux-trap cases use a driver-to-target column ratio of three to one. For purposes of computing Np-237 burnout and Pu-238 burnout, the following assumptions are made:

1. Fifty kg of Np-237 are available for use at the beginning of the production campaign.
2. Neptunium-237 becomes available continuously thereafter at a rate of 50 kg/yr. In reality, reactor operation will be a stepwise approximation to this assumption.

Results for each case will be presented as: a) percent Np-237 burnout versus irradiation time, b) Pu-238 buildup versus irradiation time, and c) Pu-238 purity versus irradiation time.

[redacted]

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 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 Amounds 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		

PT No.	Document No.	Title, Author, Date	STATUS Approved	In Progress	Com-pleted	Final Report
13	RL-NRD-299	Coproducer Demonstration Test (1.95) EG Pierick, 3/26/65				
14	HW-82385	Irradiation of Thorium-Uranium Crud Monitor Elements in N-Reactor WK Kratzer, 5/21/64		X		
18	HW-82659	Primary Loop Pressure Relief Valves (RV-2-1 and 2) Relief and Reseating Pressure Test, KL Berrett, 6/11/64	X			HW-83175 (partially complet
20	HW-82933	Hydraulic Actuators for RWSV-805 Valves, CL Goss & NR Miller, 6/23/64			X	
21	HW-83200	Optimization of Secondary Coolant Supply Operational Modes FJ Mollerus, Jr., 7/8/64			X	
21 SUP	HW-83216	Surge Tank Presure Test, FJ Mollerus, Jr. 7/8/64			X	
21 SUP	HW-83442	184-N Turbine Generator Tests FJ Mollerus, Jr., 6/27/64			X	
21 SUP	HW-83857	Pressurizer Level Control Test EE Leitz, 8/28/64 *(Final report included in test document)				HW-83857*
26	HW-84094	Calibration of an Operational Rod Withdrawal Sequence RA Chitwood, 9/25/64			X	
27	HW-84238	Steam Generator Moisture Carryover Detection LD Smith, 9/24/64	X			
30	HW-84232	Evaluation of Ammonia for Controlling Oxygen in the Graphite Cooling System, WK Kratzer, 10/5/64		X		

PT No.	Document No.	Title, Author Date	STATUS Approved	In Progress	Com-pleted	Final Report
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		
34	RL-NRD-26	Horizontal Rod Scram No. 5, Power Ascension Program, N-3 DL Renberger, 10/26/64			X	RL-NRD-172
35		(rough draft - not issued)				
36	RL-NRD-55	Simulated Driver-Target Element AC Callen, 11/6/64	X			
37	RL-NRD-98	Horizontal Rod Scram No. 6, Power Ascension Program, N-3 DL Renberger			X	RL-NRD-259
38	RL-NRD-193	Evaluation of N-Reactor Monitor Modifications for Desensitizing Response of Gamma-Energy Monitor to Activate Corrosion Coupons DO Allred, 1/20/65	X			
39	RL-NRD-204	Operation of "E" Panel Transfer Switch - 184-N JR Bolliger, PC Althoff, 1/23/65	X			

- - - - - - - - - - - - - -

PERSONNEL CHANGES

Additions

Joan Workman, Secretary, returned from leave of absence March 1, to Thermal Hydraulics Subsection

M. O. Clement, Engineer, transferred to Advanced Technology effective March 1.

C. A. Mansius, Engineer, transferred to Advanced Technology effective March 1.

Ray D. Benson, Technician, transferred to Process Analysis and Evaluation March 15.

Removals

R. V. Poe, Engineer, transferred to GE-APED, San Jose on March 12.

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 January 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.

JWRiches
Acting for
Manager
Research and Engineering

MC Leverett:JWR:vb

SIGNIFICANT REPORTS

Report Number	Class.	Title	Author(s)	Date
RL-NRD-121	Uncl	Summary Hazards Report, Use of S-1 and SCRUP Casks for Transporting N-Reactor Fuel	JW Vanderbeek, RE Trumble	12/11/64
RL-NRD-121 SUP1	Uncl	Supplement 1 (same title as above)	JW Vanderbeek, RE Trumble	3/5/65
RL-NRD-149 SUPA	Conf	PT-NR-28, PT1, Process Tube Orificing with V-11 Butterfly Valves	JL Benson	1/27/65
RL-NRD-172	Conf	Evaluation Report - PT-NR-34, Horizontal Rod Scram No. 5, Power Ascension Program, N-3	DL Renberger	1/12/65
RL-NRD-239	Uncl	Hazards Aspects of the Dual-Purpose Operation of N-Reactor Plant	RE Trumble, et al	3/1/65
RL-NRD-249	Uncl	N-Reactor Graphite Moderator: Predicted Behavior and Development Program	DH Curtiss	3/12/65
RL-NRD-265	Secret	Transition Methods for Conversion of N-Reactor to Coproduct Operation	EG Pierick, RD Shimer	3/9/65
RL-NRD-266 RD	Secret	N-Reactor Department Research and Development Budget for FY 1967 and Revision of Budget for FY 1966	Staff, N-Reactor Department	3/15/65
RL-NRD-275	Uncl	N-Reactor Axial Traverses	RA Chitwood	3/15/65
RL-NRD-285	Conf	Life Expectancy of N-Reactor	RL Dickeman	3/19/65
RL-NRD-289	Conf	N-Reactor Active Zone Supply and Demand Curves: 5 and 6 Cell Operation	TJ Bennett	3/25/65
HW-83887 REV	Secret	Hazards Review - N-Reactor 1.95% Coproducer Fuel Element Test	JL Benson, LL Grumme	3/15/65
RLSA-12A	Uncl	The Effect of Anisotropic-Contraction on Reactor Moderator Integrity	DH Curtiss	3/16/65

VISITORS

Name	Company	Contact(s)	Date	Purpose
Thoma Snyder	APED, General Electric, San Jose, California	M.C. Leverett	3/11-12/65	Physics Consultations

VISITS

Name	Company	Contact(s)	Date	Purpose
Milt Lewis	AIChE, Idaho Falls, Idaho	D.C. Hampson	3/10/65	Address on N-Reactor
" "	AIChE, Salt Lake City, Utah	E. Butler	3/11/65	" " "
B.S. Kosut	ASTM Northwest Districit Council Meeting, Portland, Oregon		3/17-18/65	Attend meeting as Sec/Treas
R.E. Hall	Burns & Roe, Hempstead, L.I., NY	I. Gabel	3/29-30/65	Conversion
" "	Philadelphia Electric Co.	W. Wilsey	3/31/65	Re: Peach Bottom Reactor Fire
J.W. Vanderbeek	Savannah River Plant, Augusta, Ga.	C.M. Patterson	3/31 to 4/2/65	Waste Management
L.C. Clossey	Classification Symposium AEC Headquarters, Washington, D.C.	C.L. Marshall	3/16-18/65	Attend classification symposium
M.C. Leverett	Savannah River Plant, Augusta, Ga.	A.A. Johnson	3/24-25/65	Technical discussions
D.H. Curtiss	" " " "			
M.C. Leverett	General Electric, New York Office	F.K. McCune	2/26/65	THC business
" "	Engineering Headquarters, New York	O.J. DuTemple	2/26/65	Committee work