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GENERAL ELECTRIC

HANFORD ATOMIC PRODUCTS OPERATION - RICHLAND, WASHINGTON

XX

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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

M. C. Leverett

January 31, 1966

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PLANT STATUS

The reactor was manually scrammed from 3870 Mw on January 4 due to a rupture suspect. The failure was subsequently identified as a fretting type on the inner surface of a single tube test fuel element. The evidence indicates that a buggy spring support vibrating against the inner spring of the fuel caused the clad perforation. The remaining column of single tube fuel was discharged.

The reactor operated at 3730 Mw until January 7, when the loss of instrument air supply caused the reactor to scram. Upon reinstatement of the instrument air supply the reactor was started up on January 8 and achieved 4000 Mw operation on January 10.

After the startup on January 8, the plant operated at essentially full power until January 13. At this time a scram occurred due to loss of header pressure on the steam supply to the drive turbines. The scram was found to have been due to energizing an improperly wired control panel which was a part of the tie-in of the sixth cell work.

While the reactor was down, heat exchanger 3B was hydrostatically tested and repaired. A leak in the auxiliary plant boiler was also repaired and an inspection of the boiler carried out. The reactor was started up on January 19. Operation at essentially rated power continued until January 24, when a scram was caused by an individual who was tracing wiring connections. A graphite cooling system dump resulted which in turn caused automatic scram of the reactor.

Since a number of scrams have been caused by the actions of individuals engaged in tie-in, maintenance, or investigative work on the reactor control systems, new and tighter procedures covering such work are being prepared. Until these procedures are in effect, maintenance, tie-in, or investigative work on critical portions of the reactor system will not be permitted while the reactor is in operation.

Startup of the reactor was initiated on January 27 and routine operation at rated power through the end of the report period is planned.

Reactor Physics

On January 24 at 8:00 a.m. the power level was 4000 Mw; the reactivity held in the rods was 8.45 mk; the flattening efficiency was 83 percent and the hottest graphite was 1168 F.

Loading changes during the early part of January were performed to improve the reactor flattening efficiency. Four columns of highly exposed Mark I fuel were replaced by one column of natural tube-in-tube fuel, one column of lithium aluminate target material; and two columns of green Mark I fuel. Two columns of single tube test fuel were replaced with Mark I fuel.

A total of 98 columns were discharged during the outage of January 24. Seventy-four columns of standard fuel were discharged and replaced with standard fuel. Twenty monitor columns (also standard fuel) were also discharged and replaced with standard fuel. The coproduct target irradiation (PT-NR-57) and one column of standard fuel were discharged and replaced with natural uranium fuel in order to suppress local high flux zones in the reactor. The remaining two columns of 1.25 percent coproduct fuel were discharged and replaced with standard fuel.

TECHNICAL SUPPORT OF CURRENT OPERATIONS

Fuel Designations

New designations for the principal N fuel elements have been adopted as follows:

- Mark I The present N-standard, plutonium-only fuel element, enrichment 0.947 percent.
- Mark I-A Mark I with inner member enrichment 1.25 percent; that is, the present N-spike element
- Mark I-B Mark I, enrichment natural; that is, the present N-poison element
- Mark II The coproduct demonstration element; 2.1 percent enriched driver, lithium aluminate target of standard Li⁶ content
- Mark II-A The coproduct demonstration spike element. Same as Mark II except for lower Li⁶ content of target.
- Mark III The higher performance plutonium-only fuel element which will be developed and proposed for use after the coproduct demonstration.

The coded alpha-numeric designations for the various parts of the several types of fuel elements now in existence will be continued for record keeping and specification recording purposes.

Indicated Failure Investigations

Two additional failure indications, 16 and 17, have occurred since the December report. A list of all the failure indications with their assigned causes is given in Table I. Table II provides pertinent operating data for the latest failures.

Indicated Failure 17 (Prototype Mark II Driver Fuel)

Failure was indicated in column 1553 on January 4. Two visual examinations of all assemblies failed to confirm a failure, but one was detected by fission gas sniffing. Visual examination failed to disclose a failure point in the suspect assembly, but after disassembly it was seen that one buggy spring support on the flow splitter appears to have fretted through the inner cladding of the driver. There is no blister on the cladding adjacent to the fretted volume of cladding.

Indicated Failure 16 (Mark I Fuel)

Failure was indicated in column 2961 on December 25. Visual examination showed a small blister on an outer. The blister is immediately adjacent to a weld tab on a suit case handle support. A wire-like object was seen under the support during examination of the element.

Indicated Failure 15 (Mark I Fuel)

The failed outer from this column showed a blister under a suit case handle support and was thought to have been caused by fretting. A wire-like object was observed under the support during examination of the element.

Indicated Failure 14 (Mark I Fuel)

Careful examination of the failed high-exposure inner in the N-basin failed to disclose the cause of failure. End cap shear and clad thinning have been essentially eliminated as probable failure mechanisms. There is a strong similarity to failure indication 7.

TABLE I
CAUSES OF INDICATED FUEL FAILURES

<u>Failure Indication</u>	<u>Tube No.</u>	<u>Date</u>	<u>Fuel Type</u>	<u>Description and Cause</u>
1	3361	7- 7-64	Mark I Inner	Closure; defective end cap
2	0643	5-15-64	Mark I Outer	Fretting; unknown fretted inner clad
3	2960	6- 6-65	Mark I Outer	Fretting; Cr plate chip fretted inner clad
4	1258	6-15-65	Proto Mark II	Corrosion; hypothesized to be F ion accelerated under locking clip
5	0756	7-23-65	Mark I	Unconfirmed
6	2747	9- 1-65	Mark I	Unconfirmed
7	2864	9- 3-65	Mark I Inner	End associated; cause unassigned
8	0353	9-10-65	Mark I	Unconfirmed
9	2557	9-26-65	Mark I Outer	End associated; cause unassigned
10	2361	10-17-65	Proto Mark II	Unconfirmed
11	1864	10-25-65	Mark I Outer	Fretting; Zr-2 chip off suit case handle
12	2844	1- 1-65	Mark I	Unconfirmed
13	1562	11-12-65	Proto Mark II	Unconfirmed
14	0664	12-12-65	Mark I Inner	End associated; cause unassigned
15	1147	12-16-65	Mark I Outer	Fretting; as yet unidentified wire under suit case handle
16	2961	12-25-65	Mark I Outer	Fretting; as yet unidentified wire under suit case handle
17	1553	1- 4-66	Proto Mark II	Fretting; buggy spring on flow splitter fretted inner clad
18	2173	1-28-66	Mark I	Unconfirmed

Indicated Failures 10 and 13 (Prototype Mark II Driver Fuel)

Re-examination of the autoradiographs of the ends of the fuel elements from these columns by N-Fuels Quality Control disclosed that three elements had uranium contamination in their end welds and should have been rejected. However, there is at present no clearly identified rupture site or site of uranium penetration in significant quantity.

Table I shows that of 17 failure indications, one was due to a faulty end cap; six were due to one or another form of fretting corrosion; one was due to accelerated corrosion under a locking clip (possibly due to fluoride residues); three were end-associated; and seven are as yet unconfirmed.

TABLE IISUMMARY OF FUEL ELEMENT FAILURE INDICATIONS
DURING JANUARY

<u>Failure Indication</u>	<u>Tube No.</u>	<u>Date</u>	<u>Tube Power</u>	<u>Exposure Tube Element</u> (MWD/T) (MWD/T)	<u>Fuel Type</u>	<u>Failure Type</u>	<u>Prod. Hours Loss</u>	<u>Hours Lost</u>
17	1553	1- 4-66	4355	1272	--	Proto. Mark II Driver	Fretting by flow splitter	3830 23
18	2173	1-28-66	2886	1950	--	Mark I	Uncon- firmed	2920 17.5

Achievement of Design Power Level

The reactor reached design power on January 10; the power level was cut 50 Mw later that day, due to a somewhat unstable power distribution and has since been increased to 3970 Mw. Adequate margins existed at this time to permit an increase to 4000 Mw, but the power was restricted slightly because of a modest cycling tendency.

Prior to this operating period, the reactor power had been restricted to 3800 - 3875 Mw, or 3 to 5 percent below design level, for a number of months. An examination of the operating data indicates a number of reasons why this was so:

- 1) A number of operating periods have been so short that maximum flattening of the reactor has not been attained; after reaching equilibrium xenon concentrations, little time was available for moving the rods into a better flattening configuration before the next outage.
- 2) Power has been restricted by reaching limits on a few isolated columns. Closer inspection reveals that:
 - a) The most limiting columns were at nearly goal exposure, while surrounded with low exposure columns. High exposure columns tend to operate one to two percent higher in power than surrounding green fuel.
 - b) The most limiting column charges contained short fuel pieces, which resulted in a higher pressure drop due to the larger number of supports and junctions, a low flow and a high outlet temperature.
 - c) High exposure fuel charges exhibit high pressure drops and low flow, due to fuel swelling, crud buildup, or other causes not understood.
- 3) Removal of fuel from process tube 1648 (to provide an empty channel) has resulted in overmoderation of the surrounding lattices and tube power peaking.
- 4) Some tube outlet temperature RTD's have drifted in calibration, or have become non-linear. Several cases where such erroneous indications have restricted power have been found.

To resolve these problems sufficiently to operate at full power, several loading changes were made:

- Process tube 1649, which was not only adjacent to empty channel 1648, but also contained a high exposure charge which was partially made up of short pieces, was recharged with natural uranium (0.712% U-235).

- Process tube 1162, which contained a high exposure charge with short pieces, and which was immediately adjacent to a similar channel, was loaded with a prototype coproduct target test, which acts as a poison column to control local power peaking.
- Process tubes 1561 and 1662 contained high exposure charges with some short pieces, exhibited unusually low flows, and the outlet temperatures were at least 5 to 7°F higher than surrounding tubes. These charges were replaced with green, 18-piece, Mark I charges.

Graphite Heat Generation Rate

The fraction of the total fission energy deposited in the graphite has been recalculated from first principles for both the Mark I and Mark II fuel designs. Results indicate that 4.46 percent of the total heat is generated in the graphite with a Mark I fuel load and 4.74 percent with a Mark II fuel load; an increase of six percent.

The increase in the graphite heat load on changing to coproduct fuel is substantially less in these calculations than in preliminary estimates. The major difference is found to be that the total gamma source in the coproduct case is lower due to the substitution of n, α reactions in lithium for n, γ reactions in U-238. This reduction was ignored in earlier approximations. Thus, while these new calculations confirm that the reduction in fuel mass substantially reduces the fraction of gammas absorbed in the fuel (by more than 20 percent), it was erroneous to assume that the unabsorbed fraction was distributed to the rest of the lattice.

It is to be noted that a 15+ percent increase would be encountered if the target material underwent n, γ reactions, and that about a ten percent increase will be encountered in Mark II-A fuel due to the reduced lithium concentration.

N-Ingot Chemistry (NIC)

A major addition to the Reactor Information File (RIF) includes the N-Ingot Chemistry parameters listed as follows:

Ingot Analyses - Specification

Ingot Analyses - Information

Ingot Number	Silicon	Waiver No.	Oxygen	Thorium	Calcium
Carbon	Hydrogen	Date	Tin	Molybdenum	Chlorine
Nitrogen	Aluminum		Billet or Core H ₂	Lead	Date
Boron	Beryllium		Phosphorus	Bismuth	
Chromium	Cadmium		Dysprosium	Zinc	
Iron	Copper		Europium	Vanadium	
Magnesium	Zirconium		Gadolinium	Niobium	
Manganese	Isotopic		Samarium	Titanium	
Nickel	Phosphorus		Hafnium	Strontium	
			Sulphur	Sodium	

Closed Circuit Television System

The radiation resistant TV system has been received from the vendor. The system was tested with several 35 mm lenses and a test pattern and operates very well. The camera was tried with the depth gauge probe with good results. The metallic parts of the probe were blackened, and additional small 60 volt lamps, and a larger mirror, were installed to improve the size and quality of the picture of the inside of the process tube. Lines as small as 0.002-inch, or 0.003-inch wide, separated by >0.006-inch, can be seen on the wall of the process tube.

Polaroid pictures of oscilloscope traces of the waveforms of the various circuits in the system were made to be used as reference material for future maintenance of the closed loop TV system. A thermister was installed inside the camera head near the Vidicon tube to read out camera head temperature. Tests to date have shown no serious temperature problems in that the camera can run for extended periods of time with no cooling air being supplied.

The monitoring system has been assembled in conjunction with the graphite mockup assembly; and internal inspection of a full sized process tube utilizing the closed loop TV circuit probe has been started. This process tube had been utilized as one of the tubes during the development of the charging machine, and, therefore, has considerable abrasion and wear due to those testing operations. This tube has been installed in the graphite mockup with standard front and rear nozzles rolled into place. The process tube had been cleaned utilizing the motor-driven scrubber and then air dried. The TV probe was driven through the process tube and the internal surface conditions of the process tube were observed.

The high resolution of the process tube monitoring system provides sufficient detail to determine the characteristics of the off-normal conditions of the inner surface of the process tube. The ability to look at the surface both obliquely and in a normal manner, with both front and rear lighting, provides a broad range of viewing and lighting combinations, which can show up any discontinuities to their maximum contrast. Since the dial gage and mirror assembly mechanism is operated from a variable speed motor drive, it is possible to stop the mirror at any position between 45° , when it is fully in place, and the fully retracted position. This provides a variation in viewing angle, such that information can be obtained on shape of defects in the process tube wall.

Redesign of the Rupture Monitor Sample Chamber

A false rupture signal caused the unnecessary shutdown of the reactor in November 1965. The cause of the rupture alarm was **activated** particulates, probably fission products from a former rupture, which became trapped in the sample chamber. A program was immediately initiated to prevent recurrence of this type of incident. As a result, a new sample chamber has been designed and tested. The new chamber is a helical coil of tubing which provides a sample volume of about 41 percent of the original chamber. The principal advantage of the helical chamber is that the sample velocity through its 1/4-inch tubing coils is about 30 times faster than through the present cylindrical chambers and, of course, it contains no traps for solids. The new chamber, therefore, should be continuously self-flushing. The test results show that, with minor modifications to the sample turrets, the new chambers are completely satisfactory for the operation of the rupture monitor system.

Effects of Subcooled Nucleate Boiling on Hydraulic Stability

A brief study was made of pressure drop data obtained in BNW experiments with the original cosine heated model of a tube-tube fuel column, with the object of determining whether subcooled nucleate boiling would affect the hydraulic characteristics of a single coolant subchannel. The results as shown in Figure 1 indicated no significant effects, since the parameter involving pressure would increase in direct proportion to pressure drop if such effects were present. Thus it is evident that limited subcooled nucleate boiling would have no adverse effects on subchannel flow stability and thus could be tolerated during normal operation. The coolant pressures and temperatures for the case studied matched reasonably well with those expected during coproduct operation.

Figure 1

Effects of Subcooled Nucleate Boiling In
Center Channel of Cosine Heated Test Section

Conditions at the outlet of the subchannel: pressures, 1430 to 1448 psia; temperatures, 527 to 586 F; constant power at approximately 2400 kw (for half column length); flow reduced in steps to create desired outlet conditions.

The only area in which there may be some question as to effects of subcooled boiling is the relative significance of equivalent diameters (the subchannel investigated had an equivalent diameter of 0.440 inches as compared with 0.198 inches for the inner annulus of the coproduct Mark II fuel). This question cannot be resolved until the cosine-heated model of the coproduct fuel is available (September). However, since calculations have shown that subcooled nucleate boiling would not occur for any tube powers expected during 4000 Mw operation of the reactor, the information will be available in ample time.

Valve Modification

Modification of the emergency supply and dump valves by replacement of the valve springs with ones of stronger design has provided improved valve opening times and increased valve reliability.

Safety Circuit

The recommendations for modifying the reactor safety circuit such that a reactor scram will be initiated if a graphite coolant flow loss occurs and persists for a significant interval has been re-evaluated. Because the "rate-of-rise" safety circuit trip is now required in service during operation, this modification is not necessary.

Connectors

Ex-reactor examination and testing of graphite cooling system connectors did not reveal any reason for concern about component integrity. A series of tests showed that a greater deflection can be tolerated than that which was previously calculated. These data further confirm that the allowable stress of 18,000 psi in the connectors was not exceeded during the graphite cooling system dump on May 8, 1965.

Cell Isolation Valves

The cell isolation valves have been placed in the open position such that they can be closed from the control room but will not close automatically on a "dump" signal.

NUCLEAR SAFETY AND ENVIRONMENTAL CONTROL

A PERT chart has been prepared describing the Five Year Nuclear Safety Mission. The work planned falls under the headings:

- 1) Fuel temperature transients under accident conditions
- 2) Quantity, nature and chemical form of fission product release from ruptured or molten fuel
- 3) Retention of noble gases
- 4) Local meteorology
- 5) Ground disposal of radioactive effluents
- 6) Particulates and aerosols from burning solutions

Reactor Process Limits Program

A digital computer program for calculating reactor process limits has been placed into service for routine reactor limit application. Further emphasis in this program will be to incorporate a tube power limit calculation and a flow monitor high trip limit calculation and limit comparison. This program has been utilized throughout the month during reactor operation. The primary difficulty remaining is the turnaround time involved for the transmission of data to and from the computing center.

Because of classification restrictions, teledata transmission cannot be used. Messenger transmission of the data is therefore being used.

Thermal Flywheel Evaluation

Use of steam generated from reactor afterheat to drive the plant turbine generator set and the primary loop drive turbines would provide increased plant safety following a failure of the plant service boiler. However, before plans and procedures can be devised which would utilize this energy source and before this capability can be depended upon, quantitative measurements of steam generation and consumption are required to confirm calculation. A production test, PT-NR-56, "N-Reactor Thermal Flywheel Evaluation Test," has been prepared to obtain these measurements.

Environmental Effects

No significant release of radionuclides from N-Reactor to the environs occurred during the past report period.

CONVERSION

Delays in the completion of the Washington Public Power Supply System (WPPSS) electrical generation facility and the delivery schedule for the 500/230 kv transformer at Vantage has resulted in the need for rescope of the Conversion Testing Program. The test program is being modified to eliminate much of the testing previously planned for 110 psia steam pressures in favor of performance of similar tests at 150 psia. Elimination of the lower pressure testing will eliminate the need for several tests which were repeated at the higher steam pressure and should shorten significantly the test program.

COPRODUCT PROGRAM

Program Coordination

The following key dates for the coproduct schedule program have been established:

Joint Program review with the AEC	April 1966
Start of fuel fabrication	May 1, 1966
Start of N-Reactor loading	September 1966
First load discharged at 1700 MWD/T - Second load fully charged	July 1, 1967
Two load demonstration complete	January 1968

The requirements for the critical action schedule are about 90 percent established. The requirements that have not yet been established are those involving fuel storage in the N-Reactor basin, fuel shipment to Purex, and the number of campaigns to be run by Isochem on coproduct fuel.

Capital funding for the coproduct program is to be authorized by the Commission in three parts:

- 1) Project CGN-184, Phase I - Fuel-Target Fabrication Equipment. This proposal was submitted to the Commission on December 27, 1965. The directive to proceed was received Jan. 17, 1966. Orders are being placed for the necessary equipment.
- 2) Project CGN-184, Phase II - Reactor Oriented Modifications for Coproduct Demonstration. This proposal is to be submitted by February 1, 1966.
- 3) Chemical Processing Oriented Modifications for Coproduct by Isochem are scheduled for submission in the near future.

Some of the key actions in the R&D programs described in RL-NRD-629, "Coproduct Schedule," have been rescheduled, but the new completion dates are consistent with the over-all program goals. The new R&D schedule will be consolidated with the critical action schedule.

The 20-tube R&D test (PT-NR-66) is receiving strong emphasis at this point due to its very tight schedule for getting fuel material procurement and fabrication, fuel handling, reactor charging and in-reactor operating data. At present, all facets of the program are on schedule, in line with plans to charge the first five columns in the reactor by March 15, and the remaining 15 columns between April 1 and April 15. During January procurement was started on the target and driver component materials. Lithium aluminate was ordered for the targets for the 20-tube R&D test. Uranium billets were also ordered. The first column (2458) of prototypical lithium aluminate targets was discharged from the reactor for shipment to Savannah River for extraction process development. The targets are scheduled to arrive at SRL on February 7.

Ceramic Target Extraction Studies

Preliminary pilot plant scale studies are reported by the Savannah River Laboratories. Their tests showed that moisture is still evolved from the molten technical grade anhydrous borax after eight hours of heating, leading to focusing attention on 1) a possible excessive isotopic dilution due to the presence of moisture in the flux, and 2) a possible loss of tritium, retained as residual moisture in the molten flux at the completion

of the extraction cycle. Dehydration studies using deuterated water as a tracer are being conducted at BNW to provide information necessary for solving these problems. Simultaneously, sources of drier and purer borax are being sought. In addition, the possible use of other fluxing agents which would permit the extractions to be performed at lower temperatures and with a greater degree of freedom from potential isotopic dilution will be investigated. Potential fluxing agents being considered include metals (single or in combinations) such as, Al, Li, Pb, Sn and mixtures of anhydrous borax with anhydrous boric oxide or sodium carbonate.

PT-NR-57, Ceramic Target Irradiations

A second column of ceramic target elements was charged early in January. These elements were essentially the same as the first column charged except that Zr-2 cladding of nominal 95 mil thickness was used. The outer diameter of the Zr-2 cladding and the inner diameter of the Zr-2 shroud tube were increased accordingly. Free standing Zr-2 cladding was specified for this column after the initial three columns of elements experienced partial cladding collapse during autoclaving.

Twelve of the sixteen elements were made with sintered pellets of γ -lithium aluminate, two were made with the same material prepared by direct encapsulation in aluminum by high energy impaction and two were made with vibrationally compacted, sintered lithium aluminate powder.

The first column of prototype ceramic targets was discharged, from column 2458, during the January 14 outage. The average exposure was calculated to be 2.8 Gas Volume Ratio (GVR) (24.5 effective full power days). This is somewhat lower than the 3.5 goal planned for the coproduct element, but the alternate appeared to be waiting until mid-February or later and this would not permit meeting a commitment to have irradiated targets at Savannah River early in February.

The elements have been examined visually in the N-basin during separation from the shroud tubes and appear to be undamaged by irradiation.

The irradiated targets will be used in tritium extraction tests at BNW and Savannah River.

Coproduct Target Supports

It has been planned that hard sized buggy spring supports with conventional fixing clips would be used to position coproduct targets inside the driver components. Because the annulus is significantly smaller than the conventional tube-tube annulus (0.19-inch versus 0.25-inch), redesign of the buggy spring, fixing clip, and associated cooling has been required. The main change is use of thinner strip (0.020-inch) in order to accommodate the conventional design in the thinner annulus. In addition, widening of the supports to 0.315-inch has been required to compensate hydraulically for leaving off the center supports on the target. Work has been in progress to develop and test new tooling for fabrication of the components and for the sizing step. Mechanical evaluation of the buggy spring flow and vibration testing, and drop testing are in progress.

PT-NR-8, Coproduct Test (1.25% Enriched Drivers)

One of the last two columns of this test remaining in the reactor was discharged during the January 14 outage and weighed and examined. The average exposure of the fuel in this channel (1156) was 2800 MWD/T. Volume increases up to 5.7 percent were measured and several elements were observed to have buckled inner cladding. Elements from channel 1060, discharged last month at 2400 MWD/T, had volume increases up to 4.0 percent and rough appearing inner surfaces. However, the scratch marks formed when the target elements were pushed out appeared to be generally straight, indicating an insignificant amount of buckling even though the inner cladding looked very rough. In this last column, however, the target support scratches had a wavy appearance, indicating that the degree of buckling was no longer insignificant.

The remaining column of this production test was then discharged (column 1259), because the potential gain in information on fuel behavior at swelling levels above 5.7 percent was not judged to be worth the risk of failure estimated for this remaining column.

The water channel thermocouples had not yet indicated an enthalpy unbalance. This is taken as substantiation of the visual estimate that the degree of buckling was not yet very severe.

Data for the peak element in each of the PT-NR-8 columns is tabulated here for comparison. The maximum exposure shown for column 1156 is the highest for any metallic fuel we know either here or elsewhere.

<u>Column</u>	<u>Exposure</u>		$100 \frac{\Delta V}{V}$
	<u>Avg.</u>	<u>Max.</u>	
1258	1100	1450	0.78
0958	1700	2500	2.64
1060	2200	3300	4.03*
1156	2800	4200	5.68*

*Buckling of inner cladding observed

Behavior of the 1.25 percent drivers which were used in PT-NR-8 is regarded as highly satisfactory.

PT-NR-13, Coproduct Test (1.96% Enriched Drivers)

The fifteen column 1.96 percent coproduct test block reached an average exposure of 1150 MWD/T on January 13. Goal exposure of about 1700 MWD/T for the 1.96 test block is scheduled for achievement in April. By the next outage, the fuel will have reached sufficient exposure to discharge a column and compare the volume changes between the standard uranium alloy and the low swelling, 350 ppm Fe + 800 ppm Al, uranium alloy.

The six highest power columns in this test are operating at the following conditions:

<u>Process tube No.</u>	<u>Power (kw)</u>	<u>Max. U Temperature (°C)*</u>
1344	4510	568
1444	4420	561
1544	4563	573
1644	4596	576
1744	4475	566
1844	4518	569

* Calculated by FLEX-3, assuming a cosine power distribution

Conversion Ratio

Available records now indicate that the previously calculated exposure for the 1.25 percent coproducer test may be erroneous, due to a misunderstanding of the flow monitor calibration for those channels. A correction to the fuel exposure of +100 MWD/T or less may result from further detailed investigation; this correction would reduce the indicated experimental conversion ratio by something less than ten percent, so that the corrected data would be more in line with calculated results.

Possible Chattering of Coproduct Target

Fretting due to differential oscillation between simulated target and driver has been seen in several SDT (Simulated Driver-Target) assemblies, and is tentatively identified as the cause of failure 17. Since this finding implied a possible deficiency in the target support design, out-of-pile tests have been done on hydraulically-induced target vibration. It appears that the degree of post-assembly expansion of the buggy spring support, by which it is deformed so as to press firmly against the inner surface of the driver, is critical. If sufficient deformation is achieved, no vibration occurs even in spite of enlargement of the driver-target annulus by the greater thermal expansion of the former. The possibility that such a mechanism may occur has directed attention to the adequacy of the coproduct supports as discussed below.

BNW has completed vibration testing with a model of the coproduct Mark II target with a 0.296-inch support width. With target supports "sized" in a steel ring having the inside diameter of a cold driver element, no vibration was observed through a lucite window in the test section at flow velocities of approximately 47 feet per second. This experiment was repeated with the target supports inside rings with inside diameters representative of dimensions of a hot driver (which would undergo a thermal expansion of 15 mils). Again no vibration was detected. Measurements taken after removal from the "cold" test section indicated that the sized supports would spring back an amount between 10 and 15 mils. Thus it is indicated that the spring forces available through the 2.5 to 7.5 mil compression in the "hot" case were adequate to prevent relative motion of the target with respect to the driver due to pure hydrodynamic forces. Follow-up experiments were made with support heights reduced to make a slight (less than 2 mils) interference fit. In that case a slight fluttering motion was observed, having a small irregular amplitude and a frequency of less than 10 cps. It thus appears that if the supports are properly sized, no difficulties should be experienced with flow-induced vibration.

Enthalpy Imbalance Studies

Additional calculations were made for Mark II fuel to determine the effects of a six percent enthalpy rise imbalance on operating margins for the conditions reported in the hazards review. As determined from conservatively calculated boiling pressure drop characteristics corresponding to a tube power of 4700 kw, the film boiling ratio (trip flow divided by film boiling flow) for a balanced condition was 1.94 as compared with the 1.49 ratio corresponding to the six percent unbalance. Thus with no credit taken for interchannel mixing at fuel element junctions for either case, the margin (or ratio) to film boiling in the inner annulus would be reduced by about 25 percent by an imbalance of 6 percent between enthalpy rise in the inner annulus and mixed coolant in the process tube. No significant difference was noted in margins to flow instability. Thus it is concluded that the 6 percent imbalance (the maximum expected) can be accommodated satisfactorily.

Work is progressing on determining the effects of manufacturing tolerance variations on the performance of the Mark II fuel, and in addition a two-dimensional mathematical model has been set up to determine the effects of end cap peaking on uranium and cladding temperatures.

OTHER ADVANCED PROGRAMS

Cobalt-60 Program

Work has been started on the design and development of supports for the irradiation of cobalt strips in the graphite cooling tubes and the cobalt target material has been ordered. The following requirements are recognized:

- 1) low rate of tube wear per charge-discharge pass
- 2) low vibration amplitude during irradiation
- 3) quick and simple support removal by remote means
- 4) low cost

Design of suitable low cost, neutron transparent dummy elements is a part of this effort.

For the short term, irradiation of one cobalt target (under non-prototype conditions) is planned by the physicists to obtain data on activation rates in N-Reactor. Current plans are to end load a column of dummies, with the target in the center, with a coil spring to prevent vibration and tube damage.

Po-210 Production Study

There is a continued interest in Po-210 as a heat source for space applications. A recent study has indicated that Po-210 production can be enhanced by peaking the flux in the 68 spare channels from 6.74×10^{13} n/cm²-sec to 9×10^{13} n/cm²-sec. This flux peak is created by increasing the tube power in the last ring of tubes opposite the spare channels. By use of this technique, polonium production in the 68 spare channels could be increased from 725 grams per year to 935 grams per year at discharge. N-Reactor's capabilities and cost determination for polonium production are in progress; documentation will be completed during February.

Irradiation of 5 kg of Np-237

The concurrence, by the Commission, in the proposed Pu-238 production demonstration has initiated the full development of the program. The current status of the program shows the following progress:

- Neptunium concentration started on January 12. Calcination started January 17 and the first shipment to BNW (for casting and fabrication) was made on January 19.
- BNW has fabricated two dummy elements and has tested these for clad wrinkle. Element No. 1, which did not contain a graphite plug, collapsed in the autoclave under 1300 psi at 300 C. Element No. 21 which contained a graphite plug, gave very good results under the same conditions.
- The aluminum inner clad tubing has been drawn to final dimensions by BNW.
- The fabrication of the Zr-2 tubing by N-Fuels has been completed.
- The end caps are being procured off site and will be on hand by February 15. These dates are compatible with the scheduled charging of one column of targets by March 17.
- Transducers required for flow monitoring have been ordered.

TARGET SCHEDULE FOR NEPTUNIUM IRRADIATION

1. Conversion of 1870 grams of Np-237 and Pu-238

Convert neptunium nitrate to neptunium oxide	January 17
Start fabrication of targets	January 24
Charge one column (16 targets)	March 21
Discharge targets	August
Cool (20 days)	
Separate Pu-238	August
Ship Neptunium nitrate to SRP	October
Storage, Pu-238	

2. Conversion of 2670 grams of Np-237

Convert neptunium nitrate to neptunium oxide	March 14
Start fabrication of targets	March 21
Charge two columns (32 targets)	May 16
Discharge targets	October
Cool (20 days)	
Separate Pu-238	October
Ship neptunium nitrate to SRP	December
Convert Pu-238 to oxide and ship to Mound Laboratory	December

IN-PLANT TESTS

The status of production tests is as follows:

PT No.	Document No.	Title	STATUS			
			Approved	In Progress	Completed	Final Report
2	HW-80369	Routine Graphite Sample Irradiations		X		
3	HW-81478	Routine Monitoring of Graphite Oxidation in N-Reactor		X		
4	RL-NRD-218	Evaluation of Monitor Column Fuel Elements in N-Reactor		X		
6	HW-81339	Isotope Producing Rods in N		X		
8	HW-81327	Coproducer Demonstration Test (1.25)				X
9	HW-81609	Low Goal Irradiation of Fuel Elements with Varying Amounts of White Oxide on Surface				X
10	HW-82136	N-Reactor Corrosion Monitoring	X			
12	HW-82241	Copper Alloys in the N-Reactor Primary System		X		
13	RL-NRD-299	Coproducer Demonstration Test (1.96)		X		
14	HW-82385	Irradiation of Th-U Crud Monitor Elements in N-Reactor				X
14 SUP	HW-82385 SUP A	Supplement to PT-14, Irradiation of 8.47" Long Fuel Element Assemblies				X
18	HW-82659	Primary Loop Pressure Relief Valves (RV-2-1 and 2) Relief and Reseating Pressure Test				X
		No further testing will be done				HW-83175

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PT No.	Document No.	Title	STATUS			
			Approved	In Progress	Completed	Final Report
20	HW-82933	Hydraulic Actuators for RWSV-805 Valves			X	
24	HW-84095	Plant Service Boiler Base Load Reduction Test Program		X		
27	HW-84238	Steam Generator Moisture Carryover Detection			X	
27 SUP	HW-84238 SUP 1	Supplement 1 to PT-27			X	
30	HW-84232	Eval. of Ammonia for Controlling Oxygen in the Graphite Cooling System			X	
32	HW-84367	Exposure of Corrosion Coupons in N-Reactor Process Tubes		X		
33	HW-84401	In-Reactor Enthalpy Imbalance Measurements		X		
33 ADD	HW-84401 ADD 2	Addendum 2 to PT-33, Eccentricity Experiments		X		
36	RL-NRD-55	Simulated Driver-Target Elements			X	
41	RL-NRD-297	Eval. of Direct Addition of Ammonia to Graphite Cooling System		X		
43	RL-NRD-379	Testing of Converted Reactor Plant			X	
46	RL-NRD-427	Primary Drive Turbine Speed Reduction Tests			X	
47	RL-NRD-426	Horizontal Rod Coolant Channel Purging			X	RL-NRD-745
48	RL-NRD-466	Corrosion Test Samples in End Caps of Graphite Cooling System		X		

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PT No.	Document No.	Title	STATUS			
			Approved	In Progress	Completed	Final Report
49	RL-NRD-492	100-N Area A Bus Power Supply Staged Transient Loading			X	
52	RL-NRD-511	Fast Indexing Drive Modification to Rupture Monitor Turrets			X	
55	RL-NRD-581	Eval. of Flow Nozzle Tap Relocation in the Steam Generator Steam and Feedwater Lines	X			
57	RL-NRD-657	Coproducer Target Element Evaluation		X		
57 SUP	RL-NRD-657 SUP	Supplement to PT-57			X	
57 SUP	RL-NRD-657 SUP B	Supplement B to PT-57		X		
61	RL-NRD-662	Eval. of Uncompensated Level Control for 109-N Steam Generators	X			
62	RL-NRD-668	Eval. of Single Pass Filtered Water Cooling	X			
63	RL-NRD-695	On-Reactor Tests for New Resistance Temperature Detectors	X			

PERSONNEL CHANGESAdditions

R. D. Shimer, Senior Engineer, transferred to Advanced Technology from Finance on January 15.

Removals

Georgia Stoffels, Secretary, transferred to Finance January 3.

M. O. Clement, Engineer, Advanced Technology, leave of absence January 4.

A. G. Dunbar, Senior Engineer, CISA, terminate for transfer to BNW January 3.

D. L. Moore, Chemical Technologist, C&M, terminate January 5.

Internal

J. R. Bolliger, Senior Engineer, transferred to Advanced Technology January 1.

Sharon Edwards, Secretary, transferred to opening with Manager, PECO, on January 17.

Joan Workman, Secretary, transferred to Manager-Reactor Physics, January 24.

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 1966. 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.

McLeverett

Manager
Research and Engineering

MC Leverett:LCC:vb

SIGNIFICANT REPORTS

<u>Report Number and Class.</u>		<u>Title</u>	<u>Author</u>	<u>Date</u>
RL-NRD-515	U	Evaluation Report, PT-NR-43, Horizontal Rod Scram	AJ Ebens	August 31, 1965
RL-NRD-581	U	Evaluation of Flow Nozzle Tap Reloca- tion in the Steam Generator Steam and Feedwater Lines	RA Rohrbacher DL Renberger	October 11, 1965
RL-NRD-599	S	Fuel Element Designs for Economic Production in Pu-Only Modes of Operation	AC Callen	October 15, 1965
RL-NRD-662	U	Evaluation of Uncompensated Level Control for 10 -N Steam Generators, PT-NR-61	JK Flickinger	November 19, 1965
RL-NRD-668	U	PT-NR-62, Evaluation of Single Pass Filtered Water Cooling	WK Kratzer	December 1, 1965
RL-NRD-681	U	Final Report, PT-NR-60, Minimum Startup Steam Consumption Rate Evaluation	AJ Ebens	December 14, 1965
RL-NRD-694	U	Summary of Discussions of Zr Pressure Tube Technology at a Joint Meeting of AECL-UKAEA and Hanford representatives	DH Curtiss	December 15, 1965
RL-NRD-685	S	N-Reactor Department Semiannual Report of Research and Development	R&E Staff, NRD	December 15, 1965
RL-NRD-704	S	Defense Plutonium Mission Plan	MC Leverett	January 4, 1966
RL-NRD-708	U	Circulating Raw Water Flow Distribution	WM Sefton	December 29, 1965
RL-NRD-710	S	Report to the FEDC Working Committee for the Period September 1965 to January 1966	M Lewis	January 5, 1966

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SIGNIFICANT REPORTS (Cont'd)

<u>Report Number and Class.</u>		<u>Title</u>	<u>Author</u>	<u>Date</u>
RL-NRD-719	S	Technical Requirements and Criteria for the Coproduct Target Element	TW Evans	January 7, 1966
RL-NRD-720	U	A Proposed Redesign of the Rupture Monitor Sample Chambers	DO Allred	January 28, 1966
RL-NRD-721	S	Minutes of Meetings to Discuss Extraction Test Studies of Ceramic Coproduct Targets	R Cooperstein	January 3, 1966
RL-NRD-722 1	C	Combined Status Report - Fuel Element Failures in N-Reactor	AE Guay	January 14, 1966
RL-NRD-730	U	TRUE-K, A Reactivity Code Based on Tube Exposures for Calculating Long Term Gains	JA Marks	January 20, 1966
RL-NRD-731	C	Conversion Ratio Error	JA Marks	January 18, 1966
RL-NRD-737	S	Fuel Reference Design and Pertinent Operating Data for Coproduct Demonstration Load	EG Pierick	January 14, 1966
RL-NRD-740	U	Simulation Techniques Applied to the System Engineering of a Large, Dual Purpose Nuclear Reactor Plant	DW Leiby	January 28, 1966
RL-NRD-745	U	Evaluation Report, PT-NR-47, Horizontal Control Rod Coolant Channel Purging	PK Telford	January 18, 1966
RL-NRD-748	S	Coproduct Program Technical Criteria	MC Leverett	January 17, 1966
RL-NRD-752	C	Heat Generation and Spatial Deposition for NPR .947 Fuel and 2.1 Coproducer	JA Marks	January 26, 1966
RL-NRD-755	S	N-Reactor, Np ²³⁷ - Pu ²³⁸ Program	RL Dickeman	January 10, 1966
RL-NRD-757	S	N-Reactor Product Flexibility	EG Pierick	January 25, 1966

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SIGNIFICANT REPORTS (Cont'd)

<u>Report Number and Class.</u>		<u>Title</u>	<u>Author</u>	<u>Date</u>
RL-NRD-763 1	S	Coproduct Program Schedule Report	JW Riches	January 24, 1966
RL-NRD-763 2	S	Coproduct Program Schedule Report	JW Riches	January 31, 1966
RL-NRD-764	S	Trip Report, Coproduct Extraction Discussions with Savannah River Laboratory, January 14, 1966	R Cooperstein M Lewis	January 19, 1966
RL-GEN-33	S	Richland Integrated Process Development Program, Defense Plutonium Mission	MC Leverett, et al	January 17, 1966
RL-NRD-173 PT 2 REV	S	Alternate Fuel Cycles for Power- Only Operation of N-Reactor	Staff, NRD	September 30, 1965

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VISITS

<u>Name</u>	<u>Company</u>	<u>Contact</u>	<u>Date</u>	<u>Purpose</u>
WM Scott	GE Research & Development Center, Schenectady, N.Y.	DA Vermilyea	Jan. 3-7	Corrosion conference
RE Trumble	GE Atomic Power Equipment Dept., San Jose, California	JC Russ	Jan. 3-7	KRB Design Safety Review meeting
M Lewis	Savannah River Plant Augusta, Georgia	TC Evans	Jan 10-13	FEDC Working Committee meeting
M Lewis R Cooperstein	" " "	CH Ice	Jan. 14 Jan. 11-14	Coproduct extraction study " " "
LL Grumme	E.I. duPont Washington, D.C.	JE Suich	Jan. 10-12	Conference on computer code HAMMER
MC Leverett	ASA Sectional Committee Washington, D.C.		Jan. 12	Meeting of N6 committee
" "	AEC Headquarters Washington, D. C.	CK Beck FP Baranowski	Jan 13-14	Discuss reactor production
MM Hendrickson	ANL, Commonwealth Edison-Dresden Reactor Morris, Illinois	GT Longergan	Jan. 11	Radiological effluent control and protection
" "	AEC-Contractor Health Protection Conference, Chicago, Illinois		Jan. 12-14	Attend conference
" "	Consolidated Edison, Indian-Point Reactor, Peakskille, New York	James Tamburri	Jan. 18	Effluent control and radiological protection

VISITORS

DW White JE Burke	GE Research & Development Center, Schenectady, N.Y.	M Lewis	Jan 24-25	Review chemistry problems with respect to coproduct extraction
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