**DECLASSIFIED
WITH DELETIONS**

HW--80559-8-Del.

DE93 003968

ts of

A.

89479

Classification Cancelled (Change to:
DECLASSIFIED *w/ deletions*)

**N-REACTOR DEPARTMENT
MONTHLY REPORT - AUGUST, 1964**

By Authority of

CB-PR-2
DS Lewis 9/15/92
DG Greiner 10/5/92
FM Cick 10-6-92

R L Dickeman
R. L. Dickeman
General Manager

CLASSIFICATION REVIEW FOR
DECLASSIFICATION NOT LEFT
UNCHANGED

By...

Date

REC Division of Classification

Work performed under Contract No. AT(45-1) - 1350 between
the Atomic Energy Commission and General Electric Company.

GROUP I

Excluded from Automatic Downgrading and Declassification.

**DECLASSIFIED
WITH DELETIONS**

MASTER

-1-

EP

~~SECRET~~ **DECLASSIFIED**

HW-80559 8 DEL

DISTRIBUTION

Copy Number

1
2
3
4
5-10
11-12

13
14
15
16
17
18
19
20
21-23
24
25

W. E. Johnson
R. L. Dickeman
A. B. Greninger
H. M. Parker
J. E. Travis, RL00-AEC
E. P. Baranowski
Director - Div. of
Production, Washington,
D. C.
F. E. Crever
Savannah River Operations
M. C. Leverett
L. M. Loeb
W. M. Mathis
J. S. McMahon
J. Milne
W. A. Shanks
Extra Copies
300 Files
Records Center

ROUTE

Name	Payroll Number	Location	Files Route Date	Signature and Date
<i>J. E. Travis</i>	<i>RL00-AEC</i>	<i>703</i>		

TABLE OF CONTENTS

Operating Summary	4
Summary	6
Reactor Plant Operation	7
Fuel Manufacture and Technology	21
Plant Modifications	25
Conversion	25
Hydraulics and Process Limits	26-45
Technical Activities	46
Administrative Summary	55
Index	61

DECLASSIFIED

~~SECRET~~
DECLASSIFIED

HW-80559 8

DEL

OPERATING SUMMARY

Input production Pu (KMWD)	13.2
Input production T (KMWD)	0.
Maximum power level attained (MW)	2,300
Average power level while operating (MW)	1,560
Time operated efficiency (%)	27.4
Plant utilization (%)	28.7
Number of Shutdowns	
Scheduled	2
Unscheduled	4
Reactor Process Data	
Power at which computed (MW)	2,300
Primary coolant inlet temperature (°F)	360
Primary coolant outlet temperature (°F)	462
Maximum graphite temperature (°F)	800
Heat Loss	
Graphite cooling (MW)	31
Shield cooling (MW)	1.8
Rod cooling (MW)	2.1
Steam temperature (°F)	324
Steam pressure (PSIG)	80
Dump condenser flow (gpm)	375,000
Inlet temperature (°F)	68
Outlet temperature (°F)	160
Nuclear Steam Generating Rate	
Maximum (1000's lb/hr)	7,500
Average while operating at (1000's lb/hr)	4,490
Water Collected from Gas Atmosphere (gal/day)	Negligible
Demineralized Water Produced (gal/hr)	85,524
Helium Losses (1000's cu. ft.)	505
Fuel Oil Usage (bbls.)	25,690
Flattening Efficiency (%)	75-80

Fuel Conditions

Heat flux

Maximum (BTU/hr/ft. ²)	370,000
------------------------------------	---------

Average (BTU/hr/ft. ²)	265,000
------------------------------------	---------

Clad temperature

Maximum (°F)	480
--------------	-----

Average (°F)	455
--------------	-----

Uranium temperature

Maximum (°F)	650
--------------	-----

Average (°F)	590
--------------	-----

Fuel Failures

0

Fuel Balance	<u>.094</u>	<u>1.25</u>	<u>Co-Prod.</u>	<u>Other</u>	<u>Total</u>
Fuel charged (Tons)	6.23	18.77	0	0	25.0
Fuel discharged (Tons)	25.10	0	0	0	25.10
Net change (Tons)	-18.87	18.77	0	0	-0.10
Total in reactor at month end (Tons)	331.39	36.43	0	0	367.82

Fuel Element Output

0.947%	0
--------	---

1.25% (driver)	331
----------------	-----

1.25% (spike)	0
---------------	---

1.25% + 0.947% (spike)	<u>786</u>
------------------------	------------

Total assemblies	1,117
------------------	-------

Total tons	23.3
------------	------

% of forecast	67
---------------	----

Uranium Utilization

70%

DECLASSIFIED

DECLASSIFIED

HW-80559 8 **SECRET**

SUMMARY

The reactor was operated at 1600, 2000 and 2300 MW, and was shut down just short of the planned power ascension step of 2400 MW for adjustments in the secondary system.

Findings at the 50 percent power level show that both critical and supporting systems are functioning well, and that there are no design problems apparent that would prevent the reactor from operating safely at the design power level of 4000 MW. In the two planned scrams from 50 percent power, automatic post-scram actions responded as programmed. The boiler responded normally from its standby steaming rate of 250,000 lb/hr.

The total clean excess reactivity has been re-estimated at 36-37 mk, slightly higher than the previous figure. This may allow some change in the loading pattern for flattening.

Revised estimates of the nuclear characteristics of the reactor have been made. The effective cross section of Xenon appears lower than expected, and this will raise the reactivity at full power. A computer program has been completed for calculation of steady state temperature distribution in the graphite. It will be used to predict temperatures at higher powers.

A stress rupture test on a section of Zircaloy tubing provided further evidence that radiation in itself does not cause deterioration of the mechanical properties of Zircaloy.

An extended outage of nineteen days allowed for scheduled maintenance and repair and for charging 68 columns of spike fuel. The maintenance items completed during the outage included correction of pressure surges in the primary loop, replacement of the primary pump shaft seal with a different type, rebanding of the tubes in the dump condensers, changing pressure trip points in the confinement system, and installation of ground detectors in the 125V DC system.

DECLASSIFIED

HW-80559 8

DEL

REACTOR PLANT OPERATION

OPERATING PATTERN

N Reactor operated for 8-1/2 days at levels of 40%, 50%, and 57% of design power. From August 5 until August 24, the plant was shut down for loading additional "spike" enrichment fuel columns and for planned maintenance. Two planned scram tests were made from 50% power to ascertain plant responses, and there were four unscheduled shutdowns.

On August 2, a sizeable leak was discovered at a welded fitting on the 2" diameter check valve bypass line on the No. 5 primary loop. The reactor was shut down, the primary loop depressurized, and the substandard weld repaired. A second leak in the primary system, a cracked elbow in a 1/2" diameter drain line from the No. 1 primary loop, was repaired. This failure was believed to be due to vibration in the long run of unsupported, field-run drain piping. Additional support was provided to the line.

Power Ascension

On August 4, power was increased to 50%, a step in the planned power ascension program. Operation at both 40% and 50% was stable and well controlled. There was no tendency for neutron flux cycling, even with relatively unflattened flux distribution. All of the primary and secondary control instrumentation was operated in automatic control. The twelve dump condensers in service were only partially loaded.

Scram Test at 50 Percent Power

On August 5, the first scram test at 50% power was initiated. All systems responded reasonably well, but extrapolation of the results to a scram at full-power conditions indicated that the existing pressurizer level control response would result in significant over-pressurization of the primary loop. This would occur as the result of over-correction of pressurizer level after water in the primary loop had contracted.

There was also some secondary system upset as the result of instability following the scram. Heat dissipation was not impaired and steam generators remained flooded, but feedwater pump suction pressure was lost momentarily when the surge tank lost pressure control. The boiler responded normally to the increased post-scram demand from its standby steaming rate, 250,000 lb./hr. All programmed automatic post-scram actions responded correctly.

Ball safety system channel rows were actuated on either side of rod column No. 5, indicating two slow rods in that column. (Later investigation disclosed that the partial ball drop was actually due to a malfunctioning relay in the ball-backup-to-rods circuit.) On the

DECLASSIFIED

DECLASSIFIED

HW-80559 8

DEL

partial ball drop, ball hopper 71 failed to open because of a stuck hopper gate.

To correct the conditions noted above the following actions were taken:

- a. The pressurizer level control fill valves were recalibrated to improve their response and the level control transmitter was replaced with one of a wider range, thus reducing the magnitude of error signal per foot of level change.
- b. The surge tank was modified by installing internal baffles at both condensate and steam entrances and a damping orifice in the condensate line, and by increasing the size of the steam pressurizing line from 6" to 12".
- c. The mechanism in ball hopper 71 was replaced with a new tested mechanism.

On August 25, the scram test was repeated from 50% power to evaluate the system response after modifications just described. There was material improvement in the stability of the secondary system and the over-pressure in the primary system was significantly reduced. All other responses were normal; there were no slow control rods indicated, and no ball channel hoppers were actuated.

Vibration of Air Line

On August 28 the actuating air line to the pressure reducing valve which controls the diversion system backpressure failed because of vibration. The consequent loss of diversion capability required a manual plant shutdown.

Power Ascension

A programmed power increase beyond 50% power was made on August 29. Because of difficulty with excess feed water rates to steam generators 1A and 5B, the power increase was halted at 2300 MW to permit balancing of process conditions. During this period, there were indications that more coolant could be used to improve performance of the turbogenerator surface condenser. When the No. 4 river pump was started, there resulted a low frequency condition on B bus, which in turn initiated a low flow trip on the circulating raw water system as No. 2 and No. 4 pumps tripped out. The plant is down while the problem of the excess feed water is being investigated. It is believed that the steam level water level measuring instrumentation may be falsely signalling too low a water level. If steam generator levels are too high, water carry over can occur; this would explain the observed behavior.

DECLASSIFIED

Water Quality

All equipment in the water quality sampling and analysis systems has been activated. The pH, conductivity, and sodium equipment is operational although a high level of maintenance is still required. Oxygen, chloride, and hydrogen monitors are in service but performance does not yet satisfy requirements. During run-in of the equipment, manual techniques are being used to control coolant quality.

Operations Staffing and Certification

Twenty-seven of the initial 48 operator candidates and all 10 of the supervisory participants have now passed both written tests for certification. They have thus qualified to continue with the oral examination and with the demonstration phase of certification training.

PROCESS PHYSICS

Equilibrium Conditions

The reactor was approaching equilibrium at 1900 MW on August 5:

Reactivity in Rods	=	4.3 mk (projects to 3.4 mk at equilibrium)
Flattening Efficiency	=	70%
Rod System Worth	=	67 + 4 mk
Spike Enrichment Inventory	=	64 columns, loaded in the third lattice unit
Reactivity Contribution of Spike Enrichment	=	2.2 mk
Inferred Reactivity held in Rods without Enrichment	=	1.2 mk

The predicted reactivity at 1900 MW was about 3.5 mk based on data from the zero power physics tests and on the calculated effects of buildup of xenon and exposure. The difference between the predicted and observed reactivity (+4.7 mk) is mostly attributable to two factors (a) an overestimate of the worth of xenon (2.3 mk), and (b) an overestimate of the initial reactivity losses due to exposure buildup - the samarium-neptunium transient (approximately 4 mk). The predicted flattening efficiency with 64 columns of spike fuel and 5 mk invested in control rods was 65 to 70 percent.

DECLASSIFIED

CLASSIFIED

HW-80559 8

Reactivity and Flattening Efficiency

Sixty-eight additional columns of spike enrichment were loaded into the third and fourth lattice units, increasing the cold reactivity by 3.4 mk. The total reactivity contribution of 132 spike columns is 5.6 mk. The predicted contribution of 132 spike columns in the third and fourth lattice units was 5.2 mk, and the predicted flattening efficiency was 78.2 percent.

Figure 1 shows how closely the calculated reactivity effect of spike loading agrees with the observed data.

Parameters of Reactor Refinement

Refined estimates of some of the nuclear characteristics of the reactor have been made. The poison worth of xenon has been re-evaluated on the basis of observations made on the reactor. The effective cross section of the xenon appears now to be lower than had been initially expected and reactivity at full power will be about 2 mk more than had been estimated.

Weighting factors for computing the effect of graphite temperature on reactivity have been derived, and coolant and graphite temperature coefficients have been developed for estimated the effects of changes in these temperatures on control rod worth. All these refinements are now in use for the prediction of reactivity and for other purposes.

Graphite Temperature Studies

Work is now complete on modification and refinement of a computer program for calculation of steady state temperature distribution in the graphite. Refinements include recent data on effects of radiation on thermal conductivity of the graphite, differential thermal expansion effects, contact coefficients, film heat transfer coefficients and emissivity of the Zircaloy.

Reasonable agreement between computed results and in-pile measurements has been achieved. However, all measurements have thus far been made during periods under non-stable xenon conditions. Future measurements will be made after the xenon buildup.

Heat Generation in Irradiated Fuel

Neutron and fission product power decay curves have been generated for the 1.25 and 1.95 weight percent U^{235} co-product demonstration fuel during a loss of water accident in the reactor. The power decay for both of these fuel types is essentially the same. Since these studies were for demonstration loadings the neutron power decay was assumed to

DECLASSIFIED

DECLASSIFIED

HW-80559 8

DEL

X = Observed Data

Reluctivity Effect (mk)

Number of Spike Columns

FIGURE 1

Calculated Effect of Spike Columns in 3rd and 4th Lattice

K-E
10 X 10 TO THE CM. 358-14
KEUFFEL & ESSER CO. MADE IN U.S.A.

[REDACTED]
DECLASSIFIED

be that exhibited by the tube-in-tube fuel. However, the fission product power decay was calculated for the specific co-product fuel. At equilibrium conditions the fission product, U^{239} , and Np^{239} beta particles and gamma rays were found to contribute about 5.7% of the total reactor power.

FLEX-3 Normalization

Two changes in the FLEX 3 computer code, (a) decreasing the effective cross section of Xe^{135} used, and (b) treating the Zr-2 resonance as a resonance have greatly improved the agreement between observed and calculated N-Reactor nuclear parameters. A correction for thermal expansion of the fuel as power increases has also been incorporated.

MECHANICAL AND THERMAL EQUIPMENT PERFORMANCE

Dummy Spacers

The foremost upstream dummy spacer was removed from each fuel column to alleviate previously noted rattling of the spacer against the tube nozzle wall. This change was recommended as the result of ex-reactor tests in the development laboratory which indicated that the rattling on the foremost dummy spacer was caused by impingement of the cooling water stream on the side of the spacer. No damage to the nozzles was evident, although the upstream support on most of the spacers was well peened from the water action.

Rupture Monitor Differential Pressure Control Valve

A failure of the right side control valve was experienced, similar to that observed in July. The valve was replaced temporarily with a spool piece-orifice assembly to allow resumption of plant operation. The damaged valve was repaired and installed as a replacement for the left side control valve. Valve life has been extended by installing a hardened stainless steel insert sleeve over the region where erosion had cut through the valve body.

Rupture Monitor Spectrometer

All spectrometers of the rupture monitor system have been recalibrated and are operating satisfactorily.

Grayloc Coupling Leakage

Several Grayloc fittings (out of over 8000 of this type) have leaked noticeably. Most of the leakage has been confined to the diversion valve

couplings on the right side of the reactor. Minor leaks were repaired by cleaning the flange surfaces and reburnishing the Grayloc seats; more serious leaks were repaired either by building up the faulty seat with weld metal and refacing, or by seal welding the joint.

Dry Film Lubricants

An assistance-to-Hanford contract has been established with the General Electric Company Advanced Technology Laboratory to provide suitable dry film lubricants for use in the high temperature primary loop environment.

Pressurizer Level Control

Pressure surges have been experienced in the primary loop after a reactor scram and also during relief valve tests. The pressure surges are caused by the piston-like action of the rising water level in the surge tank as it compresses the steam mass above it. This mass first increases because of flashing during the initial drop in pressurizer level. When the level rises rapidly the pressure increases faster than the sprays can condense steam and compensate for it. Modifications that have been considered to eliminate the surges include (1) connecting the level control circuit to the wide-range transmitters, and (2) adding rate action to the existing proportional reset unit. These methods of control were successfully demonstrated in a previous analog study. The control circuit has already been transferred to the wide-range level transmitters as mentioned previously, but the rate action will not be added until further tests are completed.

Low Level Trip

The scram circuit was modified to provide a ten-second time delay on the surge tank low level trip. No other scram trips were affected. This change was made to prevent scrams from occurring as the result of momentary secondary system upsets where return to normal is accomplished within ten seconds. Reactor shutdown is not required unless the surge tank level stays below the low-level scram point for long enough to indicate loss of heat dissipation capability, a period of time significantly longer than ten seconds.

Rod Drive Hydraulic Piping

The suction line between the hydraulic reservoirs and pumping units has an inverted loop branch line leading to the low-volume pump. Air or nitrogen gas inherently in the hydraulic system collects in the loop to form an air lock and causes the low-volume pumps to lose their prime. When the low-volume pumps are air-locked, there is no backup pump for the high-

DECLASSIFIED

DECLASSIFIED

HW-80559 8

DEL

volume pump, and recharge time for the scram accumulators is increased. Continued running of an unprimed pump causes knocking and results in exceptional wear. Rerouting of the short branch line to each of four low-volume pumps will eliminate the air lock.

Optimization of Steam Generator Level Control Systems

Efforts to optimize these systems have resulted in greatly improved performance. Further improvement should be attained with thorough recalibration and checking. This work will be started on the level control system for steam generator 1A.

Steam Generator 4A

Retubing work was completed on the 4A heat exchanger on schedule. Hydro testing was performed up to 2500 psi, at which point it was interrupted by the walkout of the plumbers. For code certification, the hydro test must go to 2740 psi. Repetition of the test will therefore be necessary.

Stem Failure in Primary Coolant Supply Valve

Metallographic examination of the failed 17-4 PH stem has produced evidence of stress corrosion cracking. All other 18, 24, and 26 inch valve stems supplied by the same vendor have been ultrasonically inspected for flaws. Final results are not yet available. Two replacement stems fabricated from Inconel X750 have been ordered. One of these will replace the broken stem and the other will be used to replace another operating stem so that further metallurgical examinations can be made on it.

Primary Pump Shaft Seal Confinement

Modifications were completed to provide a felt wiper-type seal instead of the original water seal. Removal of the breakdown bushing on the turbine end of the extension shaft will greatly alleviate future problems in shaft alignment.

Injection Water Valves

The response time on these valves to open from a closed position was found to be too long. This was caused by the output pressure of the electro-pneumatic converter being less than 3 psig. The signal limiter in the control circuit was readjusted to give a 5 psig output. This change keeps a constant loading pressure on the loading diaphragm of the valves.

Secondary System High Pressure Valves

The hydraulic systems of all 16 high pressure valves admitting steam to the dump condensers have been flushed. Modifications have been started to provide positive locking of pressure switch settings controlling the hydraulic system. The servo-control valves have been scheduled for factory reconditioning and three have already been repaired and tested by the vendor.

Response Time of V-4 Emergency Cooling System Valves

The V-4 valves are gas-operated high-pressure cone valves, connected by piping to an emergency dump tank. Their function is to dump the pressure of the primary coolant system in an emergency. The originally specified opening time for these valves was 6 seconds. The actual maximum time was recently observed to be approximately 25 seconds. An analysis made to determine if the longer time is tolerable indicated that an opening time of up to 25 seconds would not be critical for any of the accidents which require the use of the emergency cooling system. It is concluded, therefore, that the present valve opening times are acceptable for full power operation.

CRWS Pump Discharge Valve Program

An engineering study of the long-range program for upgrading the circulating raw water system (CRWS) to meet all technical criteria was completed. The use of an extremely fast-acting, spring-loaded check valve in the discharge line of each CRWS pump has been recommended to eliminate the reverse flow which occurs upon loss of pumping power. Of the two alternate arrangements suggested, one calls for the complete removal of the existing 66" butterfly valve and installation of a special fast-acting 48" check valve between the existing 66" flanging. The other arrangement calls for installation of a standard fast-acting 48" check valve in the 72" piping downstream of the existing 66" butterfly valve, and leaving the butterfly valve in place for greater redundancy. This would increase the cost of installation. Both arrangements would use a modified flow nozzle to accommodate the 48" check valve in either the 66" or 72" piping.

Eroded Elbow Below V17-1 Valve

A leaking elbow was found below the V17-1 valve that controls back-pressure to the process tube diversion header. It was radiographed and found to be eroded through approximately half the wall thickness. A water leg was installed below the valve to function as an energy absorber.

At the same time the valve was also disassembled and inspected. The

DECLASSIFIED [REDACTED]

~~DECLASSIFIED~~

HW-80559 8

DECL

seat showed no signs of wear other than a few insignificant pits and the piping below the valve appeared to be free of cavitation damage.

Steam Line to High Pressure Valves

Installation was completed of the auxiliary steam line to provide control power for the high pressure valves admitting steam to the dump condenser, except for the insulation.

Dump Condensers

All 16 dump condensers were modified and repaired. Tubes were rebanded with a braided copper strap in place of the previously used steel bands, which had fretted certain tubes excessively. Tubes with serious defects were plugged by welding. Of 103 tubes plugged, 17 were in unit 11. No plugs were required in units 12, 15 or 16.

Sump Pumps

In all cells but No. 4, modifications were completed to the level probes in the sump tanks, to the float switches in the sumps, and to the lubrication system of the sump pump bearings.

Fog Spray Trip

Concern that spurious confinement system trips from ventilation unbalance would result in unnecessary wetting of reactor components has led to a separation of the fog spray trip from the ventilation closure trip. The former is to be set at 10" (water column) instead of 2", and the latter retained at 2". After the ventilation system is shut off at 2" it is estimated that the new 10" fog spray trip pressure will be reached in about 7 minutes in 109-N and about 1.6 minutes in 105-N. These are the times required for the temperature to rise about 10 F, and produce a 10" pressure rise in a sealed building.

Ventilation Pressure Control

Problems have been experienced with instability of the 105-N ventilation pressure control systems. Stable pressure control was achieved in Zone I by utilizing only one of the three exhaust dampers for control and leaving the other two wide open. It was also necessary to dampen the pressure transmitter control signal. The transmitter range and the controller scale were changed to a range of + 0.1 to -0.3" (water column) since normal Zone I operating pressure is now required to be -0.05 to -0.1" in order to minimize loss of helium. Similar modifications appear desirable for Zone II and III when time permits.

DECLASSIFIED

Confinement Valves

During tests of the confinement system, the steam vent butterfly valves failed to close. Investigation revealed that deformation in the Kel-F seats in the three-way control valves caused the poppet assemblies to stick. These seats were replaced with aluminum material, and the valves are now operating properly. As a preventive measure, similar changes will be made in valve seats for the ventilation confinement valves.

Fog Spray Diesel Starting Signals

Two independent starting signals are being added to the present single starting signal to the high-lift diesel-operated fog spray pumps. The present signal is transmitted when the water level in the accumulator tank falls to the 9'-0" level. The new signals will be transmitted by an additional level switch located at the 8'-6" level and by a pressure switch located in the accumulator tank. This modification will provide increased assurance that the fog spray diesel pumps will start when required.

Fog Spray Control

A number of other modifications were completed, including (1) installing a test line for routine testing of control valve action as well as for evaluating the potential for nozzle plugging; (2) closing and inactivating certain exhaust dampers over the process unit so that exhaust air would pass through an active fog spray network; (3) adjusting the 109-N vacuum relief valve to prevent negative pressure in excess of -0.5 psig; and (4) modifying circuits to eliminate the fog spray automatic shutoff trip.

Irradiated Fuel Element Handling

The discharge conveyor, which had been removed for repair of chain links, was reinstalled. During the subsequent discharge of 68 tubes from the reactor, the entire underwater metal handling equipment was utilized; however, several difficulties with bearings and clutches were experienced and most of the fuel elements had to be stored temporarily in baskets rather than in the standard canisters. The last nine tubes discharged were successfully routed through the segregation equipment and into canisters.

Filter Cooling Tests

Cooling tests have been conducted on stack exhaust filters to demonstrate the effectiveness of the water spray cooling system in

DECLASSIFIED

removing heat of particulate fission product decay from the stack filter system. These tests indicates satisfactory on-off control of the filter cooling arrangement. After more than ten cycles, the appearance of the filters was good but failure of DOP (dioctyl phthalate) testing equipment prevented a quantitative evaluation. In the meantime, testing of the halogen filters has been started.

ELECTRICAL EQUIPMENT PERFORMANCE

Reactor Nuclear Flux Monitor

The interim proportional counter detector in channel 1 was replaced with the operational fission counter detector, as designed. All twelve high range primary channels are now in service, following the installation of two high range ion chambers in position 5A on top of the unit. Ten of the twelve galvanometer channels are in service. The two out-of-service channels are awaiting return of repaired chambers from the vendor; delivery is expected by the middle of September.

125V DC Ground Detectors

Four temporary ground detectors have been built and installed in the control room to monitor the key 125V DC power supply to various control systems in both 105 and 109 Buildings. These new ground detectors will detect grounds in the order of megohms, while previous ground detectors would detect grounds only up to 5,000 ohms, which is less than that required for maximum reliability.

Boiling Point Calibration

A calibration and trip check of all four channels of boiling point instrumentation was made during the August outage. There had been no shift in calibration. The check was made at two temperatures for each of the two pressures.

Other Maintenance

Other maintenance items of note were:

- a. Replacement of ten resistance temperature detectors on process tube discharge connectors at the reactor rear face.
- b. Correction of false indications from dump condenser level instrumentation, caused by leaks in the instrument system.
- c. Correction of problems in the pile gas chromatograph caused by leakage in a gas sample line and by failure of a valve.

Status of Production Tests

PT No.	HW No.	Title and Date	Status		
			Approved	In Progress	Completed
1		(Cancelled)			
2	80369	Routine Graphite Irradiations, 1/3/64	X		
3	81478	Routing Monitoring of Graphite Oxidation in N-Reactor, 3/26/64	X	X	
4		(Rough Draft - Not Issued)			
5		(Rough Draft - Not Issued)			
6	81339	The Use of Isotope Producing Rods in N-Reactor, 3/17/64	X		
7	81029	NPR Shield Evaluation; Shield Plug Measurement During Start- up Test N-2, 3/2/64	X	X	
8	81327	Co-Producer Demonstration Test (1.25), 5/4/64	X		
9	81609	Low Goal Irradiation of Fuel Elements With Varying Amounts of White Oxide on Surface, 3/30/64	X	X	
10	82136	N-Reactor Corrosion Monitoring, 5/25/64	X		
11	82074	105-N Fog Spray Control Valves, 4/30/64	X	X	
12	82241	Copper Base Alloy Corrosion Program, 6/4/64	X	X	
13		(Rough Draft - Not Issued)			
14	82385	Irradiation of Thorium-Uranium Crud Monitor Elements in N- Reactor, 5/21/64	X		

DECLASSIFIED

Status of Production Tests - cont'd

PT No.	HW No.	Title and Date	Status		
			Approved	In Progress	Completed
15	82361	N-Reactor Scram from Loss of BPA- 13.8 KV Bus 6/12/64	X		X
16	82468	Horizontal Rod Scram from 400 MW, N-Reactor 5/26/64	X		X
17	82619	Horizontal Rod Scram No. 1, Power Ascension Program, N-3 Test, 6/8/64	X		X
18	82659	Primary Loop Pressure Relief Valves (RV-2-1 and 2) Relief and Reseating Pressure Test, 6/11/64	X	X	(one RV-2 valve tested)
19	82800	Surge Tank Inert Gas Evaluation, 6/12/64	X		
20	82933	Hydraulic Actuators for RWSV-805 Valves, 6/23/64	X	X	
21	83200	Optimization of Secondary Coolant Supply Operational Modes, 7/8/64	X	X	
22		(Cancelled)			
23	83389	Horizontal Rod Scram No. 2, Power Ascension Program, N-3 Test, 7/22/64	X	X	

OUTAGES

<u>Date</u>	<u>Type</u>	<u>Duration</u>	<u>Cause</u>
8/2/64	Manual Shutdown	27.0 hrs.	Leaking weld on primary piping.
8/5/64	Scram (Production Test)	447.8 hrs.	Scheduled charging, maintenance and construction tie-ins.
8/25/64	Scram (Production Test)	5.8 hrs.	PT-NR-23 data.
8/26/64	Manual Scram	2.1 hrs.	Ground in 109 Building 125V DC control power.
8/28/64	Manual Shutdown	10.1 hrs.	Loss of diversion capability due to control air line failure.
8/30/64	Scram	45.6* hrs.	Low flow trip on raw water system due to B bus low frequency condition.

*Outage continued through month end.

DECLASSIFIED

DECLASSIFIED

FUEL MANUFACTURE AND TECHNOLOGY

PRODUCTION STATISTICS

	<u>Current Month</u>	<u>CYTD</u>
1. <u>Input (extrusions)</u>		
Outer Billets - 0.95%	32	527
1.25% Driver	0	85
1.25% Spike	43	378
Inner Billets - 0.95%	<u>160</u>	<u>756</u>
Total Extrusions	235	1746
Total Tons	39.1	320.0
% of Forecast	91	90
2. <u>Output (Finished Production)</u>		
Assemblies - 0.95%	0	8252
1.25% Driver	331	655
1.25% + 0.947% Spike	<u>786</u>	<u>2141</u>
Total Assemblies	1117	11,048
Total Tons	23.3	214.6
% of Forecast	67	88
3. <u>Uranium Receipts</u>	<u>Current Month</u>	
Total Billets	255	
Total Tons	40	
4. <u>Uranium Utilization</u>	70%	71%
5. <u>Shop Yield</u>		
Outers	<u>82.3%</u>	<u>82.8%</u>
Inners	<u>67.5%</u>	<u>81.6%</u>
Total	80.0%	82.4%

FUEL MANUFACTURE AND TECHNOLOGY

PRODUCTION SUMMARY

Input production was 92 percent of forecast, the underrun being partly attributed to an increase in extrusion time allotted for engineering development on the co-producer program, and to bonding difficulties with inner extrusions temporarily reducing the extrusion rate.

Finished production consisted entirely of 1.25 percent spike enrichment and co-producer fuel elements, and was 67 percent of forecast. There was a temporary holdup of in-process material to permit extensive dimensional measurements prior to reactor charging. These measurements are part of a program to gather information on the dimensional changes occurring in fuel elements during irradiation.

Because permission has not yet been received for reactor charging of the 1.25 percent enriched co-product fuel elements, they will be returned to the 300 Area for visual reinspection of the end closures under magnification. This is being done because of a very small possibility that end closure defects could have passed undetected into finished inventory.

PROCESS DEVELOPMENT

Process Flow

Several changes in process flow were put into effect to improve productivity without decreasing product quality. The fuel elements are being vacuum blasted after heat treating and prior to the step-cut operation. This produces more uniformly clean fuel elements because it avoids the oily film remaining from the lathe coolant. It also results in less clamp-pad slippage in the lathe because the fuel surface is not as smooth as in the previous process. An added minor change is elimination of the drying operation after end testing.

Inner Tube Support Improvement

Process tests have shown that if inner tube supports are vacuum annealed at 675°C prior to welding, they can be sized after autoclaving, without cracking. A manufacturing process has been developed for annealing these supports in the extrusion preheat pit furnaces. Reject copper billet cans will provide the vacuum chamber. Adoption of this technique will permit discontinuation of post-autoclave annealing of all inner tubes, thereby reducing manufacturing costs and permitting an improved process flow through the shop.

DECLASSIFIED

Heavy Braze Rings

A test was initiated to determine the effect of a heavier outer tube braze ring in improving the reject rate for end welding of beveled cap closures. In previous tests to compare the braze of beveled caps with that of flat caps, the reject rate for beveled caps was significantly higher. The reject rate for either cap was higher for end welding than for any other part of the brazing process.

This test was performed in two parts. In the first part, 176 fuel elements were brazed. The heavier ring produced a significant improvement in the weld reject rate. There was also a notable decrease in uranium contamination in the welds.

In the second part of the test, 396 outer fuel element ends were brazed. Again, significant improvement was registered. The results warrant further testing.

Braze Spill Over

Spilling of molten braze material onto the cladding surfaces during end cap brazing continues to be a small but significant cause of rejects. Minor braze spill over is removed from the inner diameter of outer fuel elements by grinding. These pieces are then processed through the remainder of the operation as normal production and closely examined after autoclaving. No white oxide has been observed, and the reworked fuel elements have qualified as acceptable production.

Improvements in Clad Thickness Uniformity

A test has been completed in which 13 tipped mandrel extrusion billets, seven cupping extrusion billets, and ten normal production control billets were coextruded to outer enriched fuel element size. The coextruded tubes were cut to length and after copper stripping, two tubes from each co-extrusion were end faced and clad-and-bond tested. All tubes were found acceptable at the peel test and free of clad-to-core bonding defects. Both tipped mandrel and cupping billets displayed a statistically significant superiority over the control billets in uniformity of clad thickness on the inside diameter.

Cost Estimates on 13-Inch Castings

A cost study is underway at Fernald to determine the capital cost that would be incurred if the standard production furnaces were to be converted to make 13-inch castings. This cost estimate is necessary to determine the feasibility of eventual conversion to 13-inch tipped mandrel extrusions.

Inspection and Testing

A review is being conducted of the final certification step performed on the fuel elements to assure that no rejects or duplicate numbers are in the outgoing product.

The x-ray fluorescence machine has been successfully operated for seven shifts to determine uranium contamination levels in fuel end closures. There is evidence of a lack of correlation between x-ray fluorescence results and autoradiographic-alpha-counting results. A sample of material that had been rejected by the auto radiographic and alpha-counting technique passed the x-ray fluorescence machine. Sixty per cent of the material fails the 20,000 ppm x-ray fluorescence standard. A sample selected from current production prior to autoradiography indicated that 80 per cent of present production would fail at a 20,000 ppm x-ray fluorescence test. A special production evaluation is now underway in the physical constants test reactor using neutron activation techniques to determine the degree of bias present in the alloy used in setting the x-ray fluorescence standards.

Repeated marring of the outer clad was detected by the monotube loading crew on one occasion. Investigation disclosed that the marring had occurred in a de-shoeing operation that was conducted on ten boxes of fuel elements. Fortunately the material was not marred beyond the 0.002-inch limit of the specification, and was accepted as of reactor quality.

Oxide Burning

The backlog of uranium turnings and soft fines at the oxide burner was reduced to normal level despite the equipment being shut down for one week for replacement of the air filter and repair to the burner barrel. All 1.25 percent turnings, and all inventories of natural, 1.25 and 1.45 percent enriched uranium turnings held by Hanford Laboratories have been converted to oxide. It is anticipated that inventories of 0.95 percent enriched turnings currently held by the Laboratories can be burned in September. This will allow the oxide burning operation to return to a one-shift schedule.

FUEL FAILURE INVESTIGATIONS

Rupture Investigation - Fuel Element 3048-16

All evidence indicates conclusively that the failure of this element in July was caused by the manufacture of a defective end cap from material containing the extrusion end defect, with part of this defect remaining in the cap. During welding, the crack apparently caused a small weld blowout or pinhole, thereby providing a corrosion path to the braze. Subsequent corrosion through the braze probably took place during residence in-reactor.

DECLASSIFIED

To verify this theory four defective caps were machined from selected end defect pieces. The crack-like defects in these were larger radially in the ruptured element. These caps were used to make end closures on fuel elements. Nothing unusual was apparent upon brazing. All four produced easily visible blowouts upon welding and when rewelding was attempted on two of them, blowouts were again observed. The defects extended beyond the finished weld. Holes were produced in the weld near about half of the cracks. These fuels were then given a 72-hour autoclave cycle at 400°C and 1500 psig steam. One of the caps failed, two did not. The fourth was on the opposite end of the one that failed, so behavior there is uncertain. There was a cap-core unbond on the one that failed. There was no indication of white oxide on any of the defective end caps. Caps containing smaller cracks are now being produced. They will also be used on end closures to determine their behavior.

PLANT MODIFICATION

Included here are larger long range projects such as waste management, 4A re-tubing, ERW storage tank modifications - principally changes for which AEC approval must be obtained, and excluding changes of lesser importance carried out without such prior approval.

Metal Examination Facility

Project schedules for design and construction work have been prepared and submitted to the RLOO-AEC for approval. Design work is in progress on the structural revisions necessary to the examination basin and on the special handling equipment for installation in the rear face area of the reactor.

CONVERSION

Surge Tank Stability Analysis

A computer program is being developed in support of studies to determine the operating characteristics and the stability of the three-vessel surge tank system for Phase II operation.

UNCLASSIFIED

~~SECRET~~
DECLASSIFIED

HW-80559 8

DEL

CURRENT STATUS OF N-REACTOR THERMAL
HYDRAULICS AND PROCESS LIMITS*

ABSTRACT

Experimental studies of the N-Reactor fuel column have defined its thermal and hydraulic behavior under a variety of steady-state operating conditions. These studies show:

- 1) Larger deviations of flow and pressure would be required to initiate instability or film boiling conditions in the limiting power tube than indicated by previous studies⁽¹⁾,
- 2) Process tube operating limits that prevent bulk boiling at the exit of a tube outlet connector will preclude occurrence of film boiling in the fuel column under normal or trip conditions,
- 3) Plugging upstream from the fuel column in tubes generating up to the maximum of 5000 KW cannot initiate boiling instability when the system is operated with rear riser pressures greater than 1200 psig,
- 4) The subcooling (boiling suppression) available at the exit of the outlet connector of a process tube is a satisfactory index of the tube operating margins with respect to the flow reductions required to initiate film boiling or flow instability, irrespective of the tube power.

* Abridged from HW-83655, August 14, 1964. (Confidential)

(1) See bibliography on page 45 .

DECLASSIFIED

DECLASSIFIED

HW-80559 8 DEL

The experimental data reported in HW-80692⁽¹⁾ and the interpretations of those data in this report may be used directly to establish process limits and to evaluate behavior of the fuel column under abnormal conditions. Respective benefits are 1) optimization of production through establishment of realistic margins of safety, and 2) identification of the consequences of postulated accidents affecting adequacy of cooling, without recourse to empirical correlations that may yield either unduly conservative (or unfavorably inaccurate) results.

INTERPRETATION OF THERMAL HYDRAULICS DATA

Process Channel Profiles

The primary cooling system is designed so that the pressure differential available to force coolant through each of the 1004 process channels is the same under normal conditions. For construction of characteristic curves a process channel may be broken up into two components as shown in Figure 2: Section A consists of all parts of the channel from the inlet header to the upstream end of the fuel column. Section B consists of all components of the process channel between the upstream end of the fuel column and the outlet header. The hydraulic resistance of A is a function only of flowrate and inlet temperature; that of B is a function of a fuel geometry, inlet temperature, flowrate, heat generation in the fuel column, the regime of flow, and system pressures.

Supply and Demand Curves

The supply curve (Fig. 3) represents the pressure difference available to force flow through the fuel column and outlet piping. The demand curve represents the pressure difference required to force flow through the fuel column and outlet piping. This curve is based on constant tube power, reactor inlet temperature and outlet header pressure. The point of intersection of the two curves determines the flowrate at which the process channel would operate under the chosen conditions.

The normal supply curve corresponds to a total reactor flowrate under full design power conditions. The demand curve consists of two segments. The portion A-B is the measured characteristic under liquid flow conditions; the portion B-C-D-E-F is the boiling (two-phase flow) part of the curve. If there is upstream throttling, and the tube power is constant, the temperature rise along the fuel column would increase with each reduction of flow. At point B the temperature at the outlet of the center channel of the fuel column would reach saturation and boiling would commence in that channel. The curve B-C-F represents the locus of all stable operating points under boiling conditions for constant inlet temperature, tube power and outlet riser pressures shown.

DECLASSIFIED

UNCLASSIFIED

HW-80559 8 DEL

Schematic Representation of Process Channel and Pressure Profiles

Instability

Instability is the condition under which continuous reductions in flow would result in a movement along the demand curve from A toward C, then to a point slightly below the point of tangency, thus causing a rapid and discontinuous reduction of flow and a shift in the operating point to point E.

There are a number of conditions that could cause unstable flow: inlet throttling at low flow or low pressure, decrease in pressure, increase in power, plugging of an annulus or a leak in a fuel column. Minute reductions in flow or minute increases in demand can result in gross inadequacy in cooling if the power remains constant during the transient. This inadequacy would be avoided by having low flow trips set to scram the reactor before instability could occur or immediately on onset.

Boiling Heat Transfer Effects

As flow is reduced along the demand curve of Figure 3 bulk boiling eventually will commence in the fuel column. The character of boiling at a point on the surface of the fuel will progress from saturated nucleate boiling, through stages of unstable film boiling and finally to stable film boiling, under which conditions the local film heat transfer coefficient would undergo radical changes. These changes are schematically depicted in Fig. 4.

The ultimate condition of stable film boiling is characterized by steam blanketing of the fuel surface. This insulating layer of steam causes a drastic reduction in local film coefficient and results in a temperature rise of the surface that is mainly dependent upon the existing surface heat flux.

The severity of consequences of any cooling system incident involving film boiling is evaluated in terms of the specific conditions involved and the fuel temperature increases that are calculated to occur under these conditions. For this reason use of the term "boiling burnout" is avoided in this report, and film boiling is used to describe situations under which local heat transfer coefficients deteriorate due to steam blanketing conditions.

Application of Thermal Hydraulics Data to Reactor Conditions

Experimental data now available enable construction of supply and demand curves under any chosen set of reactor operating situations. In studies of the behavior of an individual process tube under abnormal conditions, families of such curves are constructed, considering reactor inlet temperature constant and separately varying either tube power or outlet pressure.

DECLASSIFIED

Supply and Demand Curves for Normal and Throttled Condition Upstream from Fuel Column. $F_0 = 82,000$ lb/hr, Inlet Temperature 360°F , Outlet Header Pressure 1296 psig, Tube Power (limiting) 5000 KW.

FIGURE 3

DECLASSIFIED

Schematic Variation of Heat Transfer Coefficient
Under Boiling Conditions - Forced Convection.

FIGURE 4

DECLASSIFIED

DECLASSIFIED

HW-80559 8

For studies of an individual process tube the assumption is made that the inlet temperature, total reactor flow and outlet (rear riser) pressure is constant despite any change of flowrate in that tube. Changes in reactor operating parameters (flow, pressure and power) or in upstream resistance that could cause bulk boiling, instability or film boiling can readily be evaluated by inspection of curves such as are shown in Figure 3. Thus, operating margins available under a given set of reactor conditions can be determined. The margins required (and those used for establishing safety circuit trip settings) are determined by the criteria adopted for protection of the fuel against thermal damage.

COMPARISON OF EXPERIMENTAL RESULTS WITH EARLIER CALCULATIONS

Estimated Performance of the Primary Cooling System

Pre-nuclear test data and measurements obtained during low-level nuclear testing of the reactor not only have shown that the flow capabilities of the primary cooling system have exceeded design expectations but also have provided the information needed to accurately estimate the performance of the cooling system under full reactor power conditions. Thus, it is now possible to reduce conservatism previously required for evaluation of operating margins with respect to boiling. This conservatism in choice of operating parameters was employed both in analytical studies published in the Hazards Report⁽²⁾ and in the experimental work reported recently⁽¹⁾. The relative effects of choice of cooling system parameters on operating margins of the maximum (limiting) tube are demonstrated in Table 1. This table compares the parameters used for the earlier experimental and analytical studies with those currently estimated from operating experience.

Comparison of available subcooling at the exit of the outlet connector (bottom line of Table 1) will show that the currently estimated parameters combine to yield much more favorable operating margins below boiling temperatures than do those used in previously reported studies. As will be discussed further, the outlet subcooling is the best index for evaluation of boiling operating margins. Available subcooling is determined by reactor flow, inlet temperature and rear riser pressure. The required subcooling is determined both by the thermal and hydraulic characteristics of the fuel column and outlet fittings and by the safety criteria adopted.

Comparison of Experimental with Analytical Results

Most of the data presented in the Thermal Hydraulics Test Report⁽¹⁾ may be used to establish process limits and to update previous studies. Figure 5 illustrates the basic differences between the predicted behavior of the limiting

TABLE 1
EFFECT OF CHOICE OF OPERATING PARAMETERS ON
OPERATING MARGINS

	<u>Experimental Studies⁽¹⁾</u>	<u>Hazards Report⁽²⁾</u>	<u>Current Estimates</u>
Central Tube Flowrate lb/hr	80,000	80,000	82,000 ^(a) to 86,000 ^(b)
Reactor Inlet Temperature °F	369	369	342 ^(c) to 360 ^(a)
Limiting Tube Power-kw ^(d)	5,000	5,000	5,000
Normal Rear Riser Pressure, psig		1,310	1,421
Minimum Rear Riser Pressure, psig (trip level)	1,200	1,265	1,296
Outlet Temperature of Limiting Tube, °F	552	552	518 ^(b,c) to 541 ^(e)
Connector Outlet Subcooling at Low Pressure Trip, °F	17	23	38 ^(e) to 61 ^(b,c)

(a) Pump speed 3,600 rpm, fouled steam generators.
(b) Based on best estimated pump performance, no orificing.
(c) Based on best estimated steam generator performance.
(d) Design "hot channel". Heat transferred to coolant is 4,900 kw.
(e) Flow of 82,000 lb/hr, 360°F inlet.

power tube and that determined by experiment, under equivalent reactor operating conditions. These differences are discussed as follows:

- (1) The analysis used for construction of Figure 5-21 of HW-76930 Vol. 1 did not factor in a 15 per cent imbalance between enthalpy rise in the center channel of the tube-in-tube fuel column and the bulk enthalpy rise. The significance of this with respect to operational requirements is that greater amounts of tube outlet subcooling (boiling point suppression) than initially expected are required to prevent boiling in the center channel of the fuel on approach-to-trip conditions.
- (2) The relative flatness of the experimental curves for boiling demonstrates that there will be far less potential for instability than initially expected.

DECLASSIFIED

Comparison of Hydraulic Characteristics in HW-76930 with those in HW-80692 at same outlet pressures. Tube power 5,000 kw, $F_0 = 80,000$ lb/hr; Rear Riser Pressure 1,200 psig; Inlet Temperature 369°F.

FIGURE 5

DECLASSIFIED

HW-80559 8

DEL

Hydraulic Characteristics of the 5,000 kw Limiting Tube Under Currently Expected Operating Conditions, Based on Data in H-80692.

FIGURE 6

DECL

- (3) The flow reduction due to film boiling (inner annulus) for the 5,000 kw limiting tube is less than shown in HW-76930 Vol. 1. Process limits established to prevent boiling in the fuel channels (or at the extreme, boiling at the exit of the outlet connector) will preclude occurrence of film boiling for the majority of cases.
- (4) The range of expected capacity of the cooling system (shown by the upper two supply curves in Figure 5) will provide greater operating margins (82,000 vs. 80,000 lb/hr) with respect to flow instability than indicated in previous reports.

Thermal-Hydraulic Characteristics of Limiting Tube

Figure 6 was plotted for low pressure, high inlet temperature and low flow conditions in order to enable direct comparison between experimental and analytical results. To allow a more realistic appraisal of operating margins available for the 5,000 kw tube, Figure 6 was prepared from the experimental data, using as a basis the maximum expected reactor inlet temperature of 360°F and a minimum (normalizing) tube flow of 82,000 lb/hr (see Table 1).

The curves of Figure 6 illustrate the interdependency that exists between reactor cooling system capabilities and fuel column characteristics. Any change in cooling system parameters which widens margins with respect to boiling in the process channel also increases the degree of subcooling available at the exit of the outlet connectors. Based upon conservative conditions and the data given in Figure 6, a comparison can be made in Table 2 with operating margins given in the Hazards Summary Report.

TABLE 2
COMPARISON OF OPERATING MARGINS FOR 5,000 kw TUBE

	Hazards Report (2) F=80,000 lb/hr T _{in} = 369°F	Current Data F=82,000 lb/hr T _{in} = 360°F	Nominal Trip Margin*
Instability:			
Flow Reduction (a) (f)	14% (d) (e)	24% (e)	10%
Outlet Pressure Decrease (b)	15%	>37%	9%
Tube Power Increase (c) (f)	13%	16%	10%
Film Boiling in Inner Annulus:			
Flow Reduction (d) (f)	14%	44% (a)	10%
Flow Reduction (e) (f)	14%	24%	10%

DECLASSIFIED

HW-80559 8

- (a) at low pressure trip level
- (b) at normal flow
- (c) at low pressure trip and normal flow
- (d) due to plugging upstream from the fuel column
- (e) due to system flow decrease (pump loss or pump speed decrease)
- (f) constant rear riser pressure

* The nominal trip margins given are those utilized to prevent boiling in any part of the process channel at the trip condition. This explains why they are so small in relation to those required to prevent flow instability or film boiling.

The data of Table 2 show that both instability and film boiling margins are more favorable than previously estimated. Observations that can be made from both Table 2 and Figure 6 are as follows:

- (1) Instability margins will be greater than previously estimated.
- (2) With flow trips set at 90 per cent of normal flow a reactor scram would occur before bulk boiling could take place due to flow reduction in any part of a maximum power tube under normal rear riser conditions.
- (3) At low rear riser pressure trip conditions, a low flow trip setting of 79 per cent or more of normal flow would prevent film boiling in the center channel for upstream plugging events.
- (4) At low rear riser pressure trip conditions, a reduction in reactor flow of greater than 20 per cent would be required to initiate instability in the limiting power tube.

It is important to note that all operating margins given are based on a constant tube power (hot channel) and thus do not account for the rapid decrease in heat generation that would occur after initiation of a scram. For this reason, the actual margins available will be larger than those shown for all situations leading to a scram trip.

Bulk Boiling

As employed in this report, the term bulk of boiling describes a situation in which the average coolant stream is at saturated liquid or quality conditions; e.g., any addition of heat results in net steam generation. With respect to N-Reactor applications, these corrosion and hydraulic effects of bulk boiling have been evaluated to the following extent:

- (1) Corrosion Rates - Corrosive effects of boiling have been minimized by the use of Zircaloy-2 as process tube and fuel cladding material, by rigid water quality control, and by design limitations on cladding surface temperatures. At present there is no indication that significant amounts of boiling will cause any surface corrosion problems.

DECLASSIFIED

DECLASSIFIED

HW-80559 8

- (2) Hydraulic Effects - The significance of boiling with respect to over-all behavior of the fuel column is shown in both Figures 5 and 6. Bulk boiling in individual subchannels of the fuel column (outer annulus, inner annulus and center channel) would also cause a shift in flow distribution among the subchannels. Figure 6 shows that there would be a significant margin between the tube flow at which bulk boiling would commence at the outlet of the center channel and that at which film boiling would occur in the same channel.

Film Boiling Margins

The hydraulic demand curves for the limiting tube in Figure 6 show that the reductions in process tube flowrate that would be required to initiate film boiling would be about 21 per cent and 44 per cent for the center channel and inner annulus, respectively. Experimental results have shown that even a greater flow reduction would be required to cause a film boiling in the outer annulus⁽¹⁾.

The curves of Figure 7 are valuable for use in establishing operating limits. In addition, they demonstrate a very significant principle with respect to film boiling margins. This principle is as follows: Within the scope of applicability of Figure 7, film boiling cannot occur in any part of the tube-in-tube fuel column unless bulk boiling occurs at the exit of the outlet connector. The principle is valid for tube powers up to and including 5,200 kw and for outlet riser pressures equal to or greater than 1,200 psig.

The principle stated in the foregoing paragraphs has many ramifications, the most significant of which are listed below:

- (1) Process tube operating limits established to prevent bulk boiling at the exit of the outlet connector will preclude film boiling in the fuel column^(a).
- (2) For any given amount of subcooling available at the exit of the outlet connector, the flow reduction required to produce film boiling is independent of tube power^(a).
- (3) The "burnout ratio" is meaningless as an index of safety margin with respect to film boiling, with the present N-Reactor fuel geometry and operating conditions.

The most appropriate expression of margins for film boiling appears to lie in the change of the basic thermodynamic or hydraulic parameters that would be required to initiate film boiling. According to Figure 7, the tube flow and initial outlet subcooling are the most significant of these parameters.

(a) except for fuel subchannel plugging situations.

Ratio of Tube Flowrate at Film Boiling Conditions to Normal Tube Flow vs Outlet Connector Subcooling at Normal Flow, for Tube Flow Decrease Resulting from Plugging Upstream from the Fuel Column.

FIGURE 7

DECLASSIFIED

Significance of Film Boiling

In application to nuclear hazards studies, the significance of occurrence of film boiling in any given situation is judged in terms of the severity of the rise in fuel temperature that results from deterioration of the heat transfer coefficient. Where such conservatism can be tolerated, an assumption is made that the fuel surface involved in film boiling is completely insulated and the heat generation rate is constant, so that the total heat generated in a cylindrical fuel element must be transferred from only one surface. Without this conservatism, a film boiling heat transfer coefficient is utilized⁽⁶⁾ and the scram power decay transient is taken into account.

The principal fuel column channels of interest with respect to film boiling are the inner annulus and center flow passage, because the fuel surface heat flux is the highest for these channels, and the flow reductions required to initiate film boiling are the smallest. An additional consideration is the reported finding^(1,2) that plugging of the center channel sufficiently great to cause film boiling in that channel would reduce the over-all tube flowrate by less than four per cent, an amount barely detectable by the flow monitor system. Recent calculations have shown that continuous film boiling in the center channel of a limiting tube would have no nuclear safety implications. A temporary film boiling condition in the center channel would not result in significant damage to the fuel or cladding.

DERIVATION OF PROCESS LIMITS FROM THERMAL HYDRAULICS DATA

Cooling System Control

When the reactor is started up from a cold shutdown state, the minimum pressure maintained at a given reactor outlet temperature is that required to provide ample boiling point suppression in the limiting process tube to prevent boiling in the fuel column during an approach to low flow trip. Reactor power is limited to a specific low level during the period within which primary cooling system flowrates and pressures are brought up to equilibrium operating levels.

As soon as the primary cooling system has reached full flow and pressure conditions, the reactor power is gradually increased to operating level and safety circuit instrumentation is re-set to pre-established equilibrium trip settings. Under these conditions,

- (1) The primary coolant flowrate is maintained at a constant magnitude by control of pump speed,
- (2) The rear riser pressure is maintained at a constant level by the pressurizer, and
- (3) Reactor inlet temperature is controlled by the steam pressure in the secondary heat dissipation system.

Thus, three of the major operating parameters are held to essentially constant, pre-established levels.

Process Limits for the Core Cooling System

The criterion presently in use as a basis for establishment of process limits is that of avoidance of bulk boiling (net steam generation) in any process channel under normal operating conditions or during rod scrams. This limitation applies not only to the exit of the process channel but to the individual coolant passages of the fuel column as well.

Current Process Limits

The fundamental principle for the current process limits is that of operating with sufficient boiling suppression under normal conditions to prevent occurrence of bulk boiling anywhere at each of the limiting (trip) conditions. It has been demonstrated that the critical location with respect to boiling potential is the outlet of the center channel of the fuel column. Thus, an allowance is made for the difference between the coolant temperature at that location and the measured bulk coolant outlet temperature of a process channel. This allowance is applied to determine both trip limits for tube outlet temperatures and the maximum tube outlet temperatures allowed during normal operating conditions. One of the methods by which these limits are determined is that of outlet temperature trip limits.

In any situation resulting in a rise of tube outlet temperature, the trip limits for outlet temperature are arranged so that a trip or alarm will occur before boiling begins at the outlet of the center channel of the fuel column. Thus, the measured maximum coolant enthalpy difference between the tube inlet and outlet must be limited to some fraction of the enthalpy difference between inlet and saturation at the center channel outlet. Thus,

$$(h_o - h_i)_{\max} = \frac{1}{K} (h_s - h_i) \dots \dots \dots (1)$$

in which: h_i = enthalpy at tube inlet, Btu/lb
 h_o = enthalpy at tube outlet (mixed conditions)
 h_s = saturation enthalpy corresponding to the pressure at the downstream end of the fuel column.
 K = the ratio of the enthalpy difference in the center fuel channel to that in the mixed tube, adjusted for instrument error and variations in fuel manufacturing tolerances. Values estimated for K range between 1.15 and 1.25; the maximum value is currently in use^(c).

In application of Equation 1, the pressure at the downstream end of the fuel column can be related to rear riser pressure through knowledge of the pressure drop between these two points at a given flowrate (at the maximum about 20 psi, taken at low flow trip conditions). In turn, the rear riser pressure

DECLASSIFIED

chosen for the formulation of trip limits is normal pressure less a factor to account for momentary transients (100 to 150 psi) and is the low pressure trip point. A pressure margin of 150 psi currently is being used; the minimum will be determined by operating experience.

The enthalpies in Equation 1 are readily expressed in terms of coolant temperatures through use of steam tables. Thus, curves can be constructed such as those shown in Figure 8, in which outlet temperature trip limits can be determined for any given inlet temperature and rear riser pressure. (These curves are currently being used for interim process limits).

Future Process Limits

It has been demonstrated in Figure 7 that the subcooling available at the outlet of a process channel at normal flowrate is the principal factor that determines the operating margin with respect to film boiling.^(a) Figure 9 illustrates that the same factor is applicable for determining instability margins.^(a) Both figures were determined by use of process channel supply and demand curves developed from the experimental studies and are applicable over the ranges of reactor power, flowrate and rear riser conditions expected during the production phase of reactor operation.

The basic feature of both Figures 7 and 9 is that limiting values of flow monitor trip settings can be evaluated directly in terms of the outlet subcooling required to prevent flow instability or film boiling under approach-to-trip conditions. Conversely with a given flow monitor trip setting, it is possible to determine the outlet subcooling required at normal flowrate to prevent film boiling or instability at low flow trip conditions. For the maximum (limiting) power tube, the principal factors that affect available subcooling are the inlet temperature, normal tube flowrate and rear riser pressure. Thus, limiting magnitudes of these parameters can be determined through use of the curves and application of a heat balance between power generation and coolant temperature rise.^(b) In a similar fashion, by assuming a given low flow trip setting, normal flowrate and other cooling system parameters, a limiting magnitude of tube power can be established.

(a) under conditions shown.

(b) It is estimated that 98 per cent of the tube power (heat generated in a lattice unit) is transferred to the coolant.

Maximum Tube Outlet Temperature Limits to Avoid Center Channel Bulk Boiling.

DECLASSIFIED

FIGURE 8.

DECLASSIFIED

FIGURE 9

BIBLIOGRAPHY

1. HW-80692, "Thermal and Hydraulic Characteristics of an N-Reactor Fuel Column", J. K. Anderson, et al, June 25, 1964 (CONFIDENTIAL).
2. HW-79630, Vol. 1, "N-Reactor Hazards Summary Report: Volume 1, Summary and Safety Analysis", N. R. Miller and R. E. Trumble, Ed., April 15, 1963 (SECRET).
3. HW-79295 RD, "Ammonium Hydroxide for pH Control in Pressurized Water Systems...", T. F. Demmitt, V. R. Fitzpatrick and A. P. Larrick, December 12, 1963 (CONFIDENTIAL).
4. HW-77925, "Design Test No. 1160: N-Reactor Pressure Loop and Plugging Tests", M. Pociluyko, July 25, 1963 (UNCLASSIFIED).
5. HW-77470, "Burnout Heat Fluxes for NPR Tube-in-Tube Fuel Elements", J. K. Anderson, et al, June 12, 1964 (CONFIDENTIAL).
6. HW-81424, "Analysis of the N-Reactor Hypothetical Second Pressurization Incident", D. D. Stepnewski, et al, March 23, 1964 (CONFIDENTIAL).
7. HW-79240, "NPR Outlet Riser Pressure Requirements (Non-Bulk Tube Boiling Criteria)", J. Muroaka, March 20, 1964 (CONFIDENTIAL).
8. HAN-88602, "Operating Safety Limits; N-Reactor", Production Division, Health and Safety Division, RLOO-AEC, June 10, 1964 (CONFIDENTIAL).
9. HW-76500, "Process Standards: NPR", Staff Research and Engineering Operation, N-Reactor Department, Current Issue (SECRET).

DECLASSIFIED

TECHNICAL ACTIVITIES

NUCLEAR SAFETY

Fuels Criticality Studies

Changes in exposure levels and the use of higher enrichments for spike and co-product fuel result in increased irradiated fuel storage requirements. Storage problems for enrichments up to 2.98 weight percent U^{235} are being studied at this time. Storage configurations under examination are: 1) the current canister in the existing storage configuration, 2) canisters of essentially current design but incorporating cadmium sheet in their walls to allow increased weight limitations and more compact canister storage, and 3) identical to item 2) above but having individual fuel compartments formed by cadmium and aluminum sheets.

The critical mass data for the proposed fuel designs and enrichments is being generated with the GAMTEC computer code currently in use by Hanford Laboratories. An effort is also being made to generate conservative critical mass data with the FLEX-3 computer code.

Waste Management

There has been no evidence of any significant discharge of radioactive material to the environs from the N-Reactor during its period of operation.

Gaseous Wastes

Air exhausted from the 200-foot vent stack has been free of fission product or any significant activation product contamination as determined by routine monitoring of the stack effluent. These measurements are backed up by periodic searches for ground contamination above background in areas downwind from the stack and by periodic air sampling in the vicinity of N-Reactor.

Liquid Wastes

A review of sampling locations for liquid wastes is underway to insure that adequate sampling locations are available to provide for adequate routine sampling of all potentially contaminated streams which could enter the river. Proposals for additional sampling taps will be prepared as required.

Solid Wastes

As of July 1, an estimated 500 cubic feet of solid radioactive waste (principally trash and waste paper) with an activity level of less than 10 curies per cubic foot had been produced at N-Reactor and transferred to and buried at the 100-D area burial site.

NEW EQUIPMENT

Front Face Nozzle Dynamic Gas Seal

A program has been underway for several months to evaluate adequacy of the front face dynamic gas seal and to explore alternate seals. Fabrication of a test station to be used in environmental testing of the various seal packing materials is now approximately 70 percent complete.

Special Reactor Tools

A special tool has been designed to remove the stem liners from the rupture monitor toggle valves. The special pusher-puller machine for inserting and removing process tubes and for removing ruptured slugs has been fabricated, and the prototype process tube cutter completed. Fabrication was initiated on a prototype connector spreader device to aid in removing and installing nozzles.

Underwater Metal Handling Capabilities

The capabilities of the underwater metal handling equipment were examined to determine what modifications could be made to accommodate a wider range of fuel element lengths. It appears possible to handle somewhat longer fuel elements and also a greater variety of lengths. A specially programmed discharge of fuel elements by type and by length will be required.

NEW PRODUCTS

Melting Time Calculations: Co-Producer Fuels

Melting time calculations for both the 1.25% and 1.95% enriched test load co-producer fuel geometries have been completed except for the power decay curve. The assumptions made were identical to those used in the tube-in-tube study reported in HW-79791 (NPR Fuel Melting Times Following Loss of Coolant, December 14, 1963). The power decay curve used takes

DECLASSIFIED

credit for reactivity effects associated with complete loss of water from the active zone two seconds following the assumed inlet header failure (one second following the start of the power decay). The calculated power decay, based upon a 40 mk shutdown margin, corresponds to a scram from equilibrium full power operation. A power decay curve based upon scram from the hot turnaround point, also with credit taken for loss of water, was completed, but found to be less severe than the 40 mk equilibrium case. This is due to the reduced fission product inventory present at turnaround. A degree of conservatism is incorporated in these calculations, since the power distribution between the driver and graphite was held at the steady state full power value throughout the power decay.

Figures 10 and 11 present target and driver melting times for both test loads as a function of tube power. These tube power levels are based on the assumption of a uniformly loaded tube (only one fuel type in any given tube) which probably will not be the case for the 1.95 percent test load. The 1.95 percent test fuel operates at considerably higher specific powers than adjacent columns of tube-in-tube fuel and will therefore probably be specially loaded. Nevertheless, the 1.95 percent test load closely resembles the maximum weight balanced fuel design based on the low melting point alloy target currently under consideration. For this reason the plots of the figures are useful to evaluate melting times for a full reactor load of fuel with low melting point targets.

Melting times at the average central zone power and hot tube power are listed below:

<u>Tube Power</u>	<u>Target Melting Times (Sec.)</u>		<u>Driver Melting Times (Sec.)</u>	
	<u>1.25%</u>	<u>1.95%</u>	<u>1.25%</u>	<u>1.95%</u>
4400 kw	143	197	No melting	
5000 kw	121	167	285	No melting

Co-Product Physics Studies

Attention has been directed this month to generating physics and thermal hydraulic data via the FLEX-3 computer code for the full reactor co-product load. The relations among driver enrichment, exposure, fuel weight per foot, conversion ratio, and fuel temperature have been calculated for the case of 5 kg T/yr, at 4000 MW reactor power. Additional calculations, at 4800 MW, will be done. These results will then form the basic technical input for the economic optimization studies which are required to support the choice of the fuel weight, enrichment, etc.

DECLASSIFIED

DECLASSIFIED

HW-80559 8 CS

K&E 10 X 10 TO THE 1/2 INCH 358-11
KEUFFEL & ESSER CO. MADE IN U.S.A.

FIGURE 2

Driver Turnaround and Melting Times at 1/25 and 1/95 Attached to Producer Test Loads

DECLASSIFIED

1.95 Percent Driver Tube

Five special coextrusions have been made with 0.95 percent enriched uranium. They are dimensionally prototypical of the 1.95 percent enriched uranium driver fuels scheduled for the co-product production at the end of the year. Highly worked uranium sleeves were used on the inner surface with the 7-1/2 inch diameter extrusion line, but without Zircaloy-2 sub-assemblies.

Evaluation of metallurgical samples indicates that:

1. Complete bonding was obtained between the claddings and the uranium, as determined by the peel test.
2. Bonding was obtained at the uranium-uranium interface and grain growth did occur across this interface.
3. Grain sizes across the coextrusion were normal.

The resulting fuel elements have been processed through end recessing and no particular problems in quality have been detected.

Driver Fuels, 1.95 Percent

Modifications in equipment are necessary to fabricate Model No. 2 driver fuel elements of 1.95 percent enrichment. Tooling design for the pilot run has been completed. Most of the pilot run tooling will also be used for the production run.

Driver-Target Components

Arrangements have been made for all of the components needed to fabricate the 1.95 percent enriched uranium driver target fuels later in the year. It was not possible to obtain a satisfactory price or delivery date on the inner cladding component for the driver fuel; so it was extruded onsite from Zircaloy-2 billets. It was found that improved material utilization on Zircaloy parts was obtained by extruding them essentially to size on the inner diameter. The range of inner diameters was controlled to within 0.003 inch, with an average range of 0.001 inch. Finishing will consist of honing the inner diameter and machining the outer.

Target Fuels

Five lithium-aluminum coextrusions dimensionally prototypical of the target fuels to be made for the 1.95 percent co-product test were co-extruded. The extrusion billet design was identical to that planned for the production run of target fuels. The 4-inch liner was used in the 2750 ton press and the extrusion preheat temperatures were varied from

~~SECRET~~
DECLASSIFIED
WITH DELETIONS

HW-80559 8

675°F to 800°F.

The overall results of the extrusions were good. Aquadag was used as a coating on all billet cores to prevent bonding with the aluminum. No bonding occurred on any of the extrusions as measured by peel tests. The outer diameter of the coextrusions averaged 0.669 inch (0.001 inch under design) and the maximum range of the diameters was 0.004 inch. Metallurgical samples are being measured to determine clad thickness and clad thickness variation.

To assure that the aquadag coating used to prevent bonding will not interfere with reactor operations or with separations activities, tests have been made of the reactivity of the graphite, its chemistry, and any impact on hydrogen content of the coextruded lithium-aluminum.

A nondestructive ultrasonic test has been developed, by Hanford Laboratories, to assure that no spot bonding exists between the aluminum and the lithium-aluminum, and a successful prototype has been installed in the pilot plant testing tank.

Production of Po-210

Production of Pu with 12% Pu-240

Alternate schemes for operating N-Reactor were devised to produce plutonium containing 12% Pu-240 in quantities to meet the AEC's estimated requirements. If only limited quantities of weapon material are needed, it would be possible to produce about 1130 kg of Pu with 12% Pu-240 prior to the January 1967 transition to co-product operation. This would fulfill the AEC's estimated requirement through mid-1968 for high-240 material as well as producing 450 kg of weapon grade metal.

If there is a strong demand for weapon grade metal during the period 1965 to 1967, the transition to equilibrium operation could be altered and the final goal exposure reduced to 1700 MWD/T to produce 1050 kg of weapon grade metal and 600 kg of Pu containing about 12% Pu-240.

Both schemes utilize fuel segregation by location in reactor and separation of inner and outer tubes.

DECLASSIFIED
WITH DELETIONS
52-

DECLASSIFIED

NEW TECHNOLOGIC DEVELOPMENTS

KSE-5 Tests

The mixed charge of KSE-5 elements (four elements of British alloy composition interspersed with elements of standard uranium alloy composition) was discharged from KER Loop August 19 at goal exposure of 2500 MWD/T. The water channel-temperature monitors remained steady throughout the irradiation, indicating that no serious unbalance of water flow developed due to excess swelling. Examination of this fuel is now underway.

Zircaloy Process Tubes

The first stress-rupture test on irradiated tubing was completed. A section of irradiated KER tube failed in 490 hours at 300 C at a stress level of 61,000 psi. Unirradiated tubing at this stress and temperature would have failed in 20 hours. A stress level of 58,000 psi would have caused failure in 490 hours at 300 C for unirradiated tubing. While only one test has been performed, this result, combined with other tests, (e.g., burst test, crack propagation), continues to indicate that irradiation per se does not cause deterioration of the mechanical properties of Zircaloy.

A key test in determining fracture characteristics of the process tubes was completed. An N-Reactor tube section with a one-inch slot milled 80 percent through the wall was burst at $\sim 13,000$ psi internal pressure. The resultant crack did not leak at 7000 psi internal pressure at room temperature. When the temperature was raised to 300 C the crack was leaktight at 2360 psi internal pressure. Each end of the milled slot was then extended $1/4$ inch and the pressure cycled ten times between 1000 psi and 8500 psi at room temperature. The crack was thereby extended to the end of the milled slot, and was about $1-1/2$ inches long. (To cycle the pressure an internal "mouse" is used to prevent leakage.) The $1-1/2$ inch crack leaked badly at a pressure of 1200 psi. The pressure cycling was continued for 22 more cycles and the crack was extended into the heavy wall beyond the end of the slot. The maximum pressure was lowered to 7000 psi for an additional 91 cycles, then to 6500 for an additional 56 cycles. At this point the crack was 1.81 inches long. The tube section was then cycled for 65 more cycles at 6500 psi at which time failure occurred. At failure the crack was 2.25 inches long. There are a number of significant points about this test. First, a fatigue crack of any significant length can be expected to leak at operating pressure at room temperature. Second, a crack, even though completely through the wall, was over two inches long before failure occurred at room temperature at four times operating pressure. Third, even though cycling was performed at room temperature, at pressures up to five times operating pressure, and starting with an inch long crack, over 240 cycles were required to obtain failure.

Graphite Test Facilities

Construction was completed on the 9-tube graphite mockup. The mockup is located in the airlock of 105-N building. Initial distorting of the stack will begin after some small scale tests can be performed on full-size bars to provide crude limits of distortion at which keys break.

DECLASSIFIED

ADMINISTRATIVE SUMMARY

AUGUST, 1964

EMPLOYMENT

Section	Permanent		Temporary		Total		Change	
	E	NE	E	NE	E	NE	E	NE
General	1	1			1	1	-0-	-0-
Finance	15	13			15	13	+1	+5
N-Reactor Fuels	46	98	6	-0-	52	98	+1	-0-
N-Reactor Plant	68	219	3	-0-	71	219	+3	(6)
N-Reactor Project	69	22	-0-	-0-	69	22	-0-	-0-
Research and Engineering	46	9	2	-0-	48	9	+1	-(1)
Employee Relations	<u>5</u>	<u>2</u>	-0-	-0-	<u>5</u>	<u>2</u>	<u>+1</u>	<u>+1</u>
	250	364	11	-0-	261	364	7	(1)
Total employees						<u>625</u>		
Net change								<u>+6</u>

SAFETY

Days without a disabling injury 640

Hours worked without a disabling injury 2,073,230

DECLASSIFIED

VISITORS

Name	Company	Contact(s)	Date(s)	Purpose
R.A. Graham	Wah Chang Albany, Oregon	R. R. Studer	7/30/64	End cap failure.
J. Grammer	Rem. Inc., Albany, Oregon	R. R. Studer	7/30/64	End cap failure.
B. R. Pepka	Renton Coil Spring Renton, Washington	R. R. Studer	8/7/64	T-section support.
J. L. Klein	Nuclear Metals, Inc. Concord, Massachusetts	L. M. Loeb	8/6-7/64	Review N-fuel failure.
M. J. StClair	Advanced Alloys Palo Alto, California	L. H. Steves	8/20/64	Discuss zircaloy-2 support fabrication as a potential source.
W. G. McCarthy	Pacific Steel Products Seattle, Washington	N. W. McKay	7/27/64	Welding machine component requirements.
S. Dunning	Sheldon Dunning, Inc. Seattle, Washington	W. W. Arnold	7/28/64	Paints for chem exhaust stack.
J. Kenneth Salisbury	Consulting Engineer Atherton, California		8/13-14/64	Consult on heat dissipation.
R. Shurstrin W. Pitman	American Machine and Foundry		8/4-11/64	Assist in troubleshooting of irradiated metal handling equipment.
T. M. Snyder	General Electric Co. San Jose, California	M.C. Leverett and Staff	8/18-21/64	Consulted on physics problems.
N.V. Ross	Ajax Magnethermic Corp., Warren, Ohio	Milton Lewis A.E. Guay	8/10-12/64	Consulted on the fuel element failure.

UNCLASSIFIED

HW-80559 8

VISITS

<u>Name</u>	<u>Company</u>	<u>Contact(s)</u>	<u>Date(s)</u>	<u>Purpose</u>
J.W. Riches	General Electric Co. Palo Alto, California	J. F. Young	8/6-7/64	Discuss desalination.
J.W. Riches	A.E.C., Washington, D. C.	W. A. Williams	8/10-12/64	Discuss desalination.
C.F. Poor	Tidewater Oil Co. Martinez, Calif.	Phil Pyle	8/19-20/64	Discuss computing system facility, ISI-609 Computer.
J. D. McCullough	Wescon Show Los Angeles, Calif	Vendor contacts	8/24-28/64	Attend Wescon Show and to contact vendors.
W. J. Dowis	APD, Palo Alto, California		8/6-8/64	Discussions on desalination program.
W. J. Dowis	AEC, Washington, D. C.		8/10-12/64	To meet with the AEC-Depart- ment of Interior Task Force planning a research and development program for de- salting of sea water with nuclear energy.

UNCLASSIFIED

SIGNIFICANT REPORTS

UNCLASSIFIED

HW-80559 8

Number	Title	Author	Date
HW-83722 Conf.	Appraisal of Monitor Performance During July 7, 1964 Fuel Element Rupture	D. W. Leiby	August 21, 1964
HW-83655 Conf.	Current Status: N-Reactor Thermal Hydraulics and Process Limits.	R. H. Shoemaker	August 14, 1964
HW-83783 Conf.	Escapement of Fuel #3048-16	W. R. Krehbiel	August 12, 1964
HW-83244 Conf.	Essential Material Cost Analysis, N-Reactor Fuels Operation	H. P. Kraemer	July 13, 1964
HW-83529 Conf.	Fabrication History - Fuel #3048-16	D. W. Darsow	August 4, 1964
HW-81845 E Uncl.	Failure Analysis of 109 Instrumentation System, Part 5, Steam Generator Level Control	A. A. Maupin, Jr.	July 23, 1964
HW-83520 Secret	N-Reactor Co-Product Demonstration, Requirements for Aluminum-Lithium Alloy	L. M. Loeb	August 4, 1964
HW-83227 Uncl.	N-Reactor Fuel Spacer Test Summary Report	M. Pociluyko	August 6, 1964
HW-83681 Conf.	N Reactor Status Report - 50% Power	R. E. Trumble, et al	August 21, 1964
HW-83794 Part 1 Uncl.	100-N Equipment Experience Report 0-20 Percent Design Power Level	J. W. Helton. Ed.,	August 27, 1964
HW-83769 Uncl.	109-N Dump Condenser Modifications	J. W. Helton	August 28, 1964
HW-83315REV Secret	Official Production Forecast, N-Reactor Fuels Operations	L. M. Loeb	August 19, 1964
HW-83571 Uncl.	Oxygen Control in the N-Reactor Graphite Cooling System	W. K. Kratzer	August 5, 1964
HW-83440 Conf.	Review of the Program to Assure the Compatibility of the N-Reactor Gas Atmosphere, Graphite Moderator, and Zircaloy Process Tubes	D. H. Curtiss	June 9, 1964
HW-83821 Secret	Shop Yield, July, 1964	D. W. Darsow	August 24, 1964

UNCLASSIFIED

SIGNIFICANT REPORTS - continued

<u>Number</u>	<u>Title</u>	<u>Author</u>	<u>Date</u>
HW-83213REV Secret	Specification for Aluminum-Lithium Alloy Billets	R. R. Studer	August, 1964
HW-83575 Uncl.	Strength Test of Autoclave Rupture Disc Exit Lines - 333 Building	J. E. Hansen	August 10, 1964
HW-81973 Uncl.	Testing and Use of a New Type of Flareless Tubing Fitting	C. W. Ward	June 29, 1964

INVENTIONS OR DISCOVERIES

All persons 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 August, 1964, except as noted below. Such persons further advise that for the period therein covered by this report, notebook records, if any, kept in the course of their work have been examined for possible inventions or discoveries.

None

UNCLASSIFIED

HW-80559 8

[REDACTED]

SUGGESTION PLAN PARTICIPATION

	<u>AUGUST</u>	<u>CY 1964 TO DATE</u>
Number of Eligible Employees	364	364
Number of Suggestions Received	50	383
Number of Suggestions Acted Upon	47	387
Number of Suggestions Adopted	25	156
Net Annual Savings	\$ 229.	\$127,690
Amount of Awards	\$ 280	\$ 6,770
Per cent of Awards to Savings	122	5.3
Average Amount of Awards	\$ 9.16	\$ 43.40

SUGGESTIONS ADOPTED/1000 EMPLOYEES/YEAR

1964

UNCLASSIFIED

INDEX

Boiling point calibration	17
British alloy tests	53
Braze rings	22
Certification of operators	9
Circulating raw water valves - Back flow in	14
Clad thickness uniformity	23
Confinement system	15-16
Co-product fuel components	51
Co-product fuel melting calculations	47
Co-product physics	48
Discharged fuel element examination facility	25
Dump condenser tube banding	15
Emergency cooling system valves	14
Erosion in pipe elbow	14
Flattening efficiency	10
FLEX-3 normalization	11
Fog spray control	15
Fog spray diesel starting signals	16
Fuel element discharge equipment	16
Fuel element dummy spacers	11
Fuel element inspection and testing	24
Fuel element rupture investigation	24
Fuel element segregation equipment	47
Fuel element storage basin - criticality studies	46
Fuel element supports	22

Graphite distortion tests	54
Graphite temperatures	10
Grayloc couplings	11
Ground detectors	17
High pressure valve controls	14
Horizontal rod drive hydraulic piping	17
Hydraulics and process limits	26-45
Injection water valves	13
KSE-5 tests	53
Nuclear flux monitor	17
Outages	20
Oxide burning	24
Polonium 210	52
Power ascension	7-8
Power decay curves	10
Pressurizer level control	12
Pressurizer level control fill valves	8
Primary pump shaft seal confinement	13
Process physics	9
Production tests	18-19
Reactivity from spike enrichment	9, 10a
Rupture monitor differential pressure control valve	11
Scram test	7
Seals for front face nozzles	47
Stack exhaust filters	16

Steam generator level control systems	13
Steam generator retubing	13
Sump pump modifications	15
Surge tank baffles	8
Tools	47
Valve stem failure	13
Vibration of air line	8
Waste management	46
Water quality	9
Xenon worth	9
Zircaloy stress rupture tests	53

END

**DATE
FILMED**

1 / 26 / 93