AIM

Association for Information and Image Management

1100 Wayne Avenue, Suite 1100
Silver Spring, Maryland 20910
301/587-8202

Centimeter

Inches

MANUFACTURED TO AIM STANDARDS
BY APPLIED IMAGE, INC.

100-1-10-15

**Cover Sheet for a Hanford
Historical Document
Released for Public Availability**

Released 1994

**Prepared for the U.S. Department of Energy
under Contract DE-AC06-76RLO 1830**

**Pacific Northwest Laboratory
Operated for the U.S. Department of Energy
by Battelle Memorial Institute**

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DISCLAIMER

This is a **historical document** that is being released for public availability. This was made from the best available copy. Neither the United States Government nor any agency thereof, nor Battelle Memorial Institute, nor any of their employees, makes any **warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights.** The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DECLASSIFIED

HANFORD TECHNICAL RECORD
GENERAL ELECTRIC

DOCUMENT NO.

HW-61400 RD

SERIES AND COPY NO.

DATE

October 19, 1959

HANFORD ATOMIC PRODUCTS OPERATION - RICHLAND, WASHINGTON

RESTRICTED DATA
THIS DOCUMENT CONTAINS RESTRICTED DATA AS
DEFINED IN THE ATOMIC ENERGY ACT OF 1954.
ITS TRANSMISSION OR THE DISCLOSURE OF ITS
CONTENTS IN ANY MANNER TO AN UNAUTHORIZED
PERSON IS PROHIBITED.

TOPICAL MATERIAL PREPARED FOR AGENDA -

E. J. BLOCK MEETING

JUNE 29, 1959

ISSUED OCT 24 1967

OTHER OFFICIAL CLASSIFIED INFORMATION

THIS MATERIAL CONTAINS INFORMATION AFFECTING
THE NATIONAL DEFENSE OF THE UNITED STATES
WITHIN THE MEANING OF THE ESPIONAGE LAWS,
TITLE 18, U.S.C., SECS. 793 AND 794, THE TRANSMISSION OR REVELATION OF WHICH IN ANY MANNER
TO AN UNAUTHORIZED PERSON IS PROHIBITED BY
LAW.

AUTHOR

Edited by:

R. Nilson

ISSUING FILE

THIS DOCUMENT MUST NOT BE LEFT UNATTENDED WHERE AN UNAUTHORIZED PERSON MAY HAVE ACCESS
TO IT. WHEN NOT IN USE, IT MUST BE KEPT IN AN APPROVED LOCKED RECEPTACLE WITHIN AN APPROVED
GUARDED AREA. WHILE IT IS YOUR RESPONSIBILITY AND UNTIL YOU HAVE OBTAINED A SIGNED RECEIPT FROM
CLASSIFIED FILES, IT IS YOUR RESPONSIBILITY TO KEEP IT IN THE FILES AND TO RETURN IT TO THE
THAT SUBJECT AND FROM AN UNAUTHORIZED PERSON. IT IS ESSENTIAL TO, AND STORAGE OR PLACE
RESIDENCE IS REQUIRED. IT IS NOT TO BE REPRODUCED, IF ADDITIONAL COPIES ARE REQUIRED,
OBTAIN THEM FROM THE RELATED ISSUING OFFICE. PERSONS READING THIS DOCUMENT ARE REQUESTED
TO SIGN IN THE SPACE PROVIDED BELOW.

ROUTE TO:

PAYROLL NO.

LOCATION

FILES ROUTE
DATE

SIGNATURE AND DATE

J. N. Brown

13219

1903d

OCT 22 1959

R. Nilson

13374

6000

DEC 3 1962

~~Feb 26 1965~~

~~13374~~

~~6000~~

~~FEB 12 1965~~

~~2-26-65~~

GENERAL ELECTRIC

RECORD

COPY

DECLASSIFIED

Indefinite Retention

Disposal Date

DECLASSIFIED

HW-61400 RD

D I S T R I B U T I O N

1. RW Benoliel
- 2. JH Brown
- 3-4. AC Callen
5. WR Conley
6. RL Dickeman
7. WD Gilbert
8. PC Jerman
9. LW Lang
- 10-11. R Nilson
12. GJ Rogers
- 13-15. RE Trumble
16. 300 Files
17. Record Center

Classification Cancelled and Changed To

DECLASSIFIED

By Authority of CGPR-2,

WASnyder 1-6-94.

By Jerri Maley 1-10-94.

Verified By J.E. Sautley 1-31-94

RESTRICTED

This document contains restricted data
defined in the Atomic Energy Act of
1954. Its transmittal for the disclosure
of its contents in any manner to an un-
authorized person is prohibited.

This document classified by

R Nilson

This document consists of 58
pages. No 2 of [redacted] copies.
Series [redacted]

TOPICAL MATERIAL, PREPARED FOR AGENDA -
E. J. BLOCK MEETING
JUNE 29, 1959

Edited by R. Nilson

EDITORS NOTE:

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

The material presented herein represents the best knowledge concerning the 2000 Reactor and Metallurgy Program Research and Development effort as of June 29, 1959. Since much of the material involves matters which are flexible in time, it is expected that certain items will become out of date. The principle objective in distributing this information, which required the concerted writing efforts of many persons in Research and Engineering, is to preserve the material in a useable form on which future similar writing efforts can be based. The persons chiefly responsible for the material presented are: W. D. Gilbert, L. W. Lang, R. W. Benoliel, R. E. Trumble, A. C. Callen, R. E. Hall, W. K. Kratzer, N. R. Miller, S. S. Jones, D. H. Curtiss, W. S. Nechodem, G. J. Rogers, A. G. Dunbar, R. O. Gumprecht, T. W. Evans, W. K. Alexander, C. R. Richey, C. W. Allen, W. R. Conley, P. A. Carlson, K. W. Norwood, G. E. Zima, O. E. Adams, and R. Nilson.

DECLASSIFIED

HW-61400 RD

TABLE OF CONTENTS

	<u>Page</u>
I. Development Program Bases	4
II. Category I - FY 60-61	4
A. Category I Program	4
1. Reactor Power Level and Production Capability	5
2. Process Conditions	5
3. Reactor Safety	7
4. Pile Life	7
B. Basic Engineering Factors Limiting Pile Power	7
C. Fuel Program	9
1. Projected Performance of I & E Fuel Elements	9
2. Hot Spot Corrosion Problem	10
D. Engineering	13
1. Bulk Temperature Increases	13
2. Aluminum Process Tube Corrosion	14
3. Zirconium Process Tube Program	16
E. Pile Life	17
F. Nuclear Safety	18
1. Control Criteria	18
2. Water Plant Reliability Criteria	20
3. Reactor Confinement	24
G. Reactor Control Problem	25
1. Start-up Control Problem	25
2. Short-term Equilibrium Control Problems	26
3. Long-term Equilibrium Control Problems	26
III. Category II and NPR - FY 60-61	27
A. Incremental NPR Development Program	27
B. Facilities Available and Required	29
C. Fuel Development	31
1. Engineering, Physics, Fuel Technology Aspects	31
2. Key Metallurgical Considerations and Program Schedules	32
3. Irradiation Program	37
4. Fuel Fabrication Development	38
5. NPR Fuel Fabrication Line Construction	40

DECLASSIFIED

HW-61400 RD

	<u>Page</u>
D. Engineering	40
1. Zr-2 Process Tubes	40
2. Graphite	42
3. Heat Transfer Burn-out Protection for NPR	45
4. Material Development	47
E. Physics	50
1. Lattice Considerations	50
2. Conversion	50
3. Temperature Coefficients	51
4. Migration Area and Reflector Savings	53
5. Spectrum	53
6. Graphite Cooling Tubes	54
7. Control and Instrumentation	54
8. Power Level Stability	55
9. Uncontrolled Slow Flux Distribution Cycling	56
10. Research Program	56
F. Nuclear Safety	57
1. Basic Criteria	57

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DECLASSIFIED

HW 61400 RD

RESEARCH & DEVELOPMENT - 2000 REACTOR AND METALLURGY PROGRAM

I. Development Program Bases

- A. Inter-relation of Category I, Category II, NPR - levels of funding, levels of effort, and major off-site support. (Contributed by R. L. Dickeman separately.)

II. Category I - FY 60-61

A. Category I Program

The Category I program is directed toward an improved technology applicable to existing operations for improved production capability, improved reliability of the production facilities, and better protection of operating personnel and plant environs from radiological hazards. These development programs are consistent with the efforts to obtain increased production capacity at reduced unit costs and to maintain or improve the present reactor safety status.

Following are schedules of projected reactor operating conditions and operating limits which will prevail during FY 1960-1961, assuming that the anticipated improvements, discussed in later sections, are achieved.

DECLASSIFIED

HW-61400 RD

1. Reactor Power Level and Production Capability (From 5 yr. Program Estimate)

<u>Reactor</u>	<u>FY-60 Level</u>	<u>FY-61 Level</u>
B	1.71×10^3 MW	1.82×10^3 MW
C	2.03	2.19
D	1.75	1.81
DR	1.73	1.79
F	1.75	1.75
H	1.75	1.84
KE	3.77	3.83
KW	<u>3.77</u>	<u>3.83</u>
Total	18.26	18.86

Operation of the reactors at the above estimated levels will result in the following production: (N.B. 1960 figures below are from Official Forecast)

	<u>FY-60</u>	<u>FY-61</u>
MWD	4.33×10^6 MWD	5.15×10^6 MWD
Plutonium	3.71×10^6 gm	4.41×10^6 gm

The above increases in production capability reflect reactor operation at increased coolant outlet temperature and flow rate improved production efficiency, and increased reactor conversion efficiency.

2. Process Conditions

- a. Bulk outlet water temperature will be increased to 95°C at a pilot reactor early in FY-1960, and all reactors will be operated with 95°C outlet in FY-1961.
- b. Reactor flow rates will be approximately as follows during the next two years, reflecting minor increases as equipment and process changes are realized.

DECLASSIFIED

HW-61400 RD

<u>Reactor</u>	<u>FY-60 Flow Rate</u>	<u>FY-61 Thousands GPM</u>
B	78	83
C	93	100
D	80	83
DR	79	82
F	80	80
H	80	84
K	172	175

Further increases in coolant flow rates are programmed for the K Reactors in FY-1962, and feasibility studies for coolant flow increases in the older reactors are currently being made.

- c. Assuming achievement of the above flow rates and 95°C bulk outlet temperature, the following values for fuel element specific power are to be anticipated, notwithstanding other limitations:

<u>Reactor</u>	<u>FY-60 Specific Power -</u>	<u>FY-61 KW/Ft.</u>
B	73	78
C	80	84
D	75	78
DR	71	74
F	74	74
H	68	72
K	82	83

- d. Process tube outlet temperatures for control of fuel element corrosion will be maintained as follows:

<u>Reactor</u>	<u>FY-60 Tube Outlet Temperature - °C</u>	<u>FY-61</u>
C	115	125
K	115	115
B, D, DR, F, H	120	120

- e. The graphite will be operated at a maximum temperature of 800°C at the K Reactors and 700°C at all other reactors, utilizing 100% helium as the reactor gas atmosphere.

DECLASSIFIED

HW-61400 RD

- f. The fuel element discharge exposure goals will be maintained at 800 MWD/T for the I & E elements, and at a variable goal up to 800 MWD/T for the solid elements.

3. Reactor Safety

During this production period it is assumed that temporary power level restrictions, which could reduce productive output, will not prevail as an operational limit. Studies of reactor and radiological safety considerations will be continued with high priority to provide a broad background of nuclear safety technology, including the following areas:

- (a) mechanism and consequences of fission product release, (b) vulnerability of the reactor systems to earthquake failure, (c) reliability of the reactor coolant systems, (d) refinement of reactor control criteria, and (e) improved knowledge of physics and engineering fundamentals contributing to nuclear safety technology.

4. Pile Life

It is recognized that continuing contraction of the graphite moderator may be a major factor limiting the productive life of the Hanford reactors. Economic and design studies will be made to devise least costly methods of correcting for excessive graphite distortion to permit graphite lives of at least 25 years.

B. Basic Engineering Factors Limiting Pile Power

Four basic engineering factors can be critical in limiting ultimate pile power. These four factors are tube boiling, sub-cooled burn out, effluent system boiling and graphite damage. The limits and current status for each of these factors are shown in the following table.

BASIC ENGINEERING FACTORS LIMITING
PILE POWER

<u>FACTOR</u>	<u>LIMIT PARAMETER</u>	<u>ULTIMATE CONSERVATIVELY SAFE LIMIT (Present Design)</u>	<u>CURRENT LIMIT</u>	<u>CURRENT STATUS</u>
Tube Boiling Protection	Outlet Tube Temp.(max)	130 C	130 C	90-105 C
Sub-cooled Burnout	Heat Flux	1,500,000 BTU/Ft ² -Hr	None Specified	800,000 to 1,000,000 BTU/Ft ² -Hr
Effluent System Boil- ing Protection	Bulk Outlet Temp.	95 C ^a 105 - 115 C ^b	93.5 C	D-DR 93.5 C Others 93.5 C
Uncertainties in Graphite Contraction	Graphite Temp. (max)	Unknown 850 C permissible	650 C	560 - 590 C

a. Minor or no effluent system modifications.

b. Extensive effluent system modifications.

DECLASSIFIED

HM-61400 RD

██████████

DECLASSIFIED

HW-61400 RD

C. Fuel Program

1. Projected Performance of I & E Fuel Elements

Forecasted performance of the I & E fuel elements may be restricted by fuel fabrication, "hot-spot" corrosion, and uniform corrosion failures, probably in that order. Recent rupture experience indicates that approximately 60% of the recent categorized failures ruptured in the internal coolant passage. Six of these failures exhibited water passages in the bonding layer extending from a penetration in one of the fusion weld areas to the failure point. It is hypothesized from these data that the failures were fabrication defects having weld penetrations caused by unbonded or porous areas adjacent to the weld. Potential solutions currently being evaluated by the Fuels Preparation Department include caustic cap etching to promote better bonding, development of better welding techniques, development of non-destructive testing equipment and tighter weld inspection controls.

Should the "internal" class of fuel failures be reduced significantly by these potential solutions, it is probable that the "side-hot spot" class of fuel failure will limit production increases. Since January of this year approximately 40% of the categorized fuel failures have been of this type.

Elimination of these two principle causes of fuel element failure should permit substantial production increases from increased power and improved efficiency, and unit cost reductions by exposure increases. Operation at power levels corresponding to bulk outlet temperature limits may require slight fuel element geometry modifications. No geometry changes are currently being scheduled since redimensioning

██

DECLASSIFIED

HW-61400 RD

is neither immediately necessary nor economically attractive. For those columns which do require special charging to reduce either fuel or tube corrosion rates, an aluminum spacer has been developed which will mix the coolant to reduce temperature unbalances. This method is currently being utilized at the C and K Reactors and is being tested for possible use at the other production units also.

2. Hot Spot Corrosion Problem

- a. The mechanism postulated for fuel element failures of the "hot-spot" classification implies two independent phenomena, i.e., a rapid corrosive penetration of the aluminum fuel cladding, followed by a penetration through a defect in the aluminum-silicon bonding layer. Conditions of exposure of aligned fuel elements are not sufficiently severe to produce rapid corrosive attack of the intergranular type normally found on hot-spot ruptures; and it has been shown previously that fuel elements purposely misaligned do exhibit the typical flow pattern in the aluminum surface film. It has therefore been assumed that the majority of the hot-spot failures have occurred as a result of fuel element misalignment in the process tube, causing a flow disturbance which in turn creates areas of decreased velocity and localized overheating. At this point it is academic whether or not actual burnout occurs or whether the corrosion rate simply becomes excessive since penetration of the cladding will follow within a relatively short time.

Penetration of the cladding, however, does not of itself imply fuel failure since experience has shown that sound Al-Si bonding layers offer excellent secondary corrosive protection. If the bonding layer contains defects such as cracks, voids and pores, corrosive attack of the exposed uranium occurs and the fuel element "ruptures".

DECLASSIFIED

HW-61400 RD

This postulated mechanism indicates the need for two programs aimed at reducing the "hot-spot" rupture incidence. The first of these is to develop a fuel element and process tube geometry which would prohibit misalignment of the elements to prevent local cladding overheating. The second is to continue to attempt to produce bonding layers which are perfectly sound to provide secondary corrosion protection. The latter is a Fuels Preparation Department Program, the former has been initiated within the Irradiation Processing Department and has been termed the Projection Fuel Program.

b. Projection Fuel Elements

Two concepts have been chosen for initial in-reactor testing of projection fuel elements; the self-supported element for use in ribless or vestigial-ribbed process tubes, and the so called bumper fuel element for use with current process tubes as an interim measure to reduce misalignment.

The self-supported fuel element is a current I & E fuel element having eight - one-inch long supporting projections welded to it (four at each end spaced 90° apart). This concept is attractive since it permits fuel geometry changes without process tube modifications.

The bumper concept uses an element similar except dimensionally to the self-supported element and which is capable of maintaining a minimum annular clearance in the current ribbed process tubing. It was originally conceived to eliminate the fuel element misalignment without replacement of all of the existing process tubes

DECLASSIFIED

HW-61400 RD

with ribless tubes. There is a distinct probability that the concept design will not be successful due to fuel instability problems and since the fuel element is rib-supported, and thus not concentric with the process tube. Nevertheless, it may provide a means of achieving alignment on an immediate schedule to reduce the hot-spot failure incidence.

The proposed development program and the current status of the projection fuel element program are as follows:

- 1) To date irradiation of nine solid fuel columns and thirteen I & E columns of self-supported elements in B Reactor has been completed and neither hot-spot ruptures nor film patterns have been observed. The columns have been recharged for irradiation to 800-900 MWD/T and will continue to be recharged to observe tube effects.
- 2) Ten columns of bumper fuel elements and the necessary controls are currently in-test at KW Reactor. Two of these columns have been discharged and are awaiting visual examination.
- 3) Future tests of both types of projection elements will include high-exposure irradiation of enriched elements to pilot operating conditions up to 100 KW/ft, and demonstration scale loadings of approximately 100 tubes each.

It is anticipated that the preliminary test loading of bumper fuel elements will be discharged in late July or early August. If the results are encouraging, the testing will proceed. If the results are negative or inconclusive, the bumper program will be dropped in favor of the self-supported fuel program. It is anticipated that this decision will be made by the end of CY 59.

DECLASSIFIED

HW-61400 RD

In the event it is deemed uneconomic to replace current process tubes with zirconium tubes, but rather to continue use of aluminum, sufficient data on tube wear are being obtained in the current test at B Reactor to aid in the choice between ribbed and ribless tubes. From the fuel failure and geometry flexibility standpoints it appears that the ribless tube self-supported fuel elements concept will become necessary at Hanford in order that forecasted production increases can be effected, provided the original assumptions of failure mechanisms are correct.

D. Engineering

1. Bulk Temperature Increases

The reactor development aim of increasing the bulk temperature limit is beginning to take form as a specific and detailed program. The DR Reactor has been selected as the lead reactor in this program. The far downcomer is now being modified and instrumented so that the testing program can be started in late July 1959. This specific program calls for a three-phase study of the problem:

- a. An experimental and theoretical study to evaluate more accurately the accuracy of our present limits.
- b. A model study program to study the effects of steady state and transient boiling in the DR effluent system.
- c. Higher temperature operation of the DR Reactor.

Completion of all three phases is tentatively scheduled by August 30 permitting a reevaluation of the bulk temperature limit which may lead to a 95°C bulk limit by August or September 1959. Alternately, the study may lead to a reduced bulk temperature limit depending upon the test and study results.

DECLASSIFIED

HW-61400 RD

The re-evaluation of this bulk temperature limits for all reactors will be completed during FY-1960.

The present program of raising the bulk outlet temperatures with only minor effluent system modifications is self-limiting. The limiting temperature is just under 100°C, perhaps as low as 95 to 96°C. In addition, is the inherent fact that as this limiting bulk temperature is approached the risk of effluent system damage is increased.

By adopting a positive program of effluent system pressurization the bulk limit can be raised to 105-115°C and will probably no longer be a critical power level limiting factor. This increase, however, will require new rear face piping, downcomers, and possibly retention basins.

2. Aluminum Process Tube Corrosion

a. Internal Corrosion

The uniform internal corrosion rates of process tubes with present process water have been determined to be about 3.2 mils/year at 100°C, 9.8 mils/year at 120°C and 30 mils/year at 140°C. The total allowable penetration is 30-40 mils. Therefore, in the absence of annulus eccentricity, appreciably long tube lives can be expected at reactor bulk exit temperatures of 95°C. At a 95°C reactor bulk limit, the maximum tube exit temperatures are above 110°C giving a tube life of about 6.5 years for the maximum temperature tubes.

Actually, eccentricity is a major factor in limiting tube lives and, in the absence of correcting action, tube lives may be more than halved by resulting normal non-uniform annular temperatures. The use of water mixing elements is a fairly well-proven method of

DECLASSIFIED

HW-61400 RD

compensating for annular eccentricity. With such devices, the lives of even the highest temperature tubes should be of the order of 4-5 years at a 95°C reactor bulk temperature. Current plans include the use of mixer elements in all the central zone tubes at C-Reactor beginning probably in August 1959. The utilization of mixer elements at the oldest reactors (B through F) will likely begin in September or October, 1959. In the meantime, small test quantities of the mixer elements are to be charged in C and D Reactors in July to determine better mixer element efficiencies. (Previous studies with mixers aimed at minimizing fuel jacket corrosion rates are not altogether pertinent due to differences in fuel-tube geometry, tube condition, and placement of the mixer device.)

The ultimate reactor bulk temperature feasible with aluminum tubes is not yet well determined. However, with the use of water mixing elements, reactor bulk outlets up to 110°C are not ruled out. At this temperature, a central zone retubing cycle of about 3-4 years would probably be required.

A further extension of permissible temperatures may be obtained by decreasing the pH of process water which should reduce rates somewhat. Work is in progress to determine the technical and economic incentive for such a step.

b. External Galvannic Tube Corrosion

External tube surface corrosion is a quite serious problem at F Reactor and could conceivably become serious at other reactors in the future.

Steps being taken to minimize the costs of external corrosion are:

DECLASSIFIED

HW-61400 RD

- i. Refining methods of predicting internal corrosion rates to insure tubes are removed before they leak due to the internal attack.
- ii. Obtaining better methods of predicting which tubes are externally corroded so they can be replaced before they leak.
- iii. Developing new methods of joining the tubes to nozzles to prevent tube flange failures. When the Van Stone flanges fail, water is forced into the graphite stack with resulting exterior attack on adjacent tubes. Testing of different joints will likely begin at F Reactor during the first quarter of FY-1960 and full scale usage may be started mid-FY-1960.

3. Zirconium Process Tube Program

a. Procurement

Four contracts for a minimum of 75 tubes each have been signed with Tube Reducing Corp., Harvey Aluminum Co., Superior Tube Co., and Mallory-Sharon Metals Corp. The contracts are fixed price contracts with price redetermination features. The contracts are not developed but rather pilot production orders.

The tubes to be provided are ribbed Zr-2 tubes of BDF size. Initial deliveries will begin in November 1959. Contracts are to be complete in 3rd quarter FY-60. Cost \$600,000.

Two fixed price purchase orders have been placed each for 50 smooth bore Zr-2 tubes of C-Reactor size. Delivery will be in late July--early August 1959. Cost \$110,000.

b. Program

The program for demonstrating zirconium tubes consists of two parts: (1) 100 smooth bore tubes with self-supported fuel

DECLASSIFIED

HW-61400 RD

elements to be installed in C-Reactor during September-October 1959, (2) 300 ribbed tubes with I & E fuel elements to be charged in F Reactor late in CY-1959 or early in 1960.

Associated development activities includes development of charging machine, Van Stoning equipment, tube removal equipment, substitute closure for Van Stone (O-rings or equivalent), nozzle modification and dummy patterns. Programs are underway and approximately on schedule.

Economics have been reviewed and integration of associated projects in progress.

E. Pile Life

Contraction of the central zone of all reactors continues at about the same rate of 0.3" to 0.4" per year. However, distortion at BDF reactors, where front face residual expansion exists and where the worst curvature problem exists, continues to improve slowly. Improvement is due mainly to changes in loading pattern and control rod manipulation. Currently there is no 4" metal in our reactor for reasons of graphite curvature. Some 4" metal is being irradiated to deplete inventories.

At the old reactors central zone contraction and resultant tube curvature may ultimately limit reactor life. However, possible mechanical corrective action might readily extend life for additional 10-15 years. With possible reaming of channels in offing in conjunction with zirconium tube replacement, adverse curvature can be remedied. At K Reactors, side expansion is apparently continuing. Immediate effects are not clear. Extensive studies are in progress to evaluate current status, possible corrective action and possible effects to pile life.

DECLASSIFIED

HW-61400 RD

F. Nuclear Safety

The reactor safety program is conducted to assure nuclear, radiological, and operational safety in all phases of the reactor process. Continuous re-evaluation is conducted as changes are made in the process and power levels and exposures increased. The major areas of the continuing studies and development efforts are discussed below.

1. Control Criteria

Relief from power level and exposure restrictions previously imposed by original unrealistic interpretations of the reactor control criteria has been obtained by the adoption of revised interpretations of the criteria. General statements of the control criteria, which have been passed on to the ACRS, and which apply to all Hanford reactors, present or future are as follows:

a. Total Control Criterion

The design and/or operation of the reactor shall be such that in the event of any credible combination of events, including complete and permanent loss of coolant, the total control strength shall be sufficient to insure that no reactivity state will develop which would significantly increase the consequences of the accident.

b. Speed of Control Criterion

The design and/or operation of the reactor shall be such that in the event of a power excursion resulting from any credible combination of events including complete and permanent loss of coolant, the response of the reactor safety control system shall be sufficient to insure that no energy state will develop while the reactor is super-critical during the excursion which would significantly increase the consequences of the accident.

DECLASSIFIED

- 19 -

HW-61400 RD

The presently adopted interpretations of the criteria which have been formulated from comprehensive studies and have been accepted by the ACRS and the Atomic Energy Commission are as follows:

a. Total Control

A prompt critical state is never permitted. Criticality short of prompt is allowed in the reactor during the meltdown process only after a time when it is obvious that emergency procedures to save the reactor from complete meltdown are no longer possible. In the case of the present reactors, with aluminum process tubes, the time at which the reactor complex has reached the melting point of aluminum is considered the time after which a super-critical state can be allowed.

b. Speed of Control

The surface temperature of the fuel element shall not reach the melting point of uranium during the excursion.

The present status of the reactors under the revised control criteria is as follows:

a. Total Control

All reactors are presently operating under the assumption that the HCR systems can be considered as supplementary total control. DR and H Reactors presently require no further supplementary total control strength and thus are not limited by this consideration.

The remaining six reactors (B, D, F, G, KE and KW), require varying amounts of supplementary total control strength during outages and cold startups. These requirements are a function of enrichment level and location, exposure, xenon transients,

20
DECLASSIFIED

HW-61400 RD

and "permanent" pile characteristics such as fuel loading, rod strengths, ball system strengths, etc. The methods of providing supplementary total control vary among the reactors. D, KE, KW and C (very occasionally) require the use of temporary lead-cadmium, poison columns because of the lack of adequate operational supplementary control systems. Fortunately, the need for supplementary total control strength coincides with a need for additional turn-around control, so that no significant production loss effect has been noted. B and F Reactors possess adequate PCCF systems, of which several tubes have been used for supplementary total control use. Essentially no economic effect has been noted at these reactors.

Should the HCR system be declared unreliable for supplementary total control, significant production losses would be incurred at all except possibly H Reactor. Production losses on the order of several percent would be expected at the K piles, and lesser effects in descending order at C and D, B, and F, DR, and possible H.

b. Speed of Control

No reactors are presently limited by this criterion. The power level limits, under the most conservative assumptions regarding exposure and water loss reactivity gain are as follows for the reactors:

KE, KW	> 6000 MW
C, H	> 4000 MW
B,D,DR,F	> 4000 MW

2. Water Plant Reliability Criteria

Adequate cooling of reactor fuel elements is the paramount consideration in nuclear reactor safety. Studies have resulted in a set of criteria

DECLASSIFIED

HW-61400 RD

pertaining to cooling system reliability, and conformance of the existing plants to these criteria is currently being evaluated.

The normal reactor cooling system requirements shall be:

a. Three independent, reliable sources of cooling as follows:

1) Primary Cooling System

The primary cooling system will provide adequate reactor cooling during all phases of operation and shutdown. The design and operation shall be such that the secondary or last ditch backup systems are not called upon to provide the sole source of cooling during any phase of normal operation and would not be expected to be required to function as the sole supply system during the life of the plant.

2) Secondary Cooling System

The secondary cooling system shall be independent of the primary system - (common piping is acceptable) - and capable of providing adequate shutdown cooling indefinitely assuming instantaneous loss of power to the primary system. Serious consideration should be given the operation of the secondary system at partial flow capability at all times and preferably at a flow capability sufficient to satisfy the full requirements in event of primary system failure without relying upon actuation or acceleration of equipment. That flow provided by the secondary system during normal operation may contribute to the primary cooling system requirements.

3) Last Ditch System

The last ditch cooling system must be independent of both the

22
DECLASSIFIED

HW-61400 RD

primary and secondary cooling systems, including piping to the reactor manifolding. It must be capable of providing shutdown flow indefinitely assuming instantaneous loss of power to the primary system and concurrent failure of the secondary system to provide its rated flow.

- b. An independent system is defined as one with power sources and pumping systems such that failure or cause of failure in one cannot induce failure of the alternate systems.
- c. Adequate cooling is defined, as a minimum, as that required to maintain reactor components, including fuel elements, at temperatures below the melting point. In practice, the objective of requiring levels of cooling capable of protecting against all reactor or fuel element damage will be prudent to adopt.
- d. Failure of any single component in a system shall not be capable of rendering that system inadequate. In practice certain piping and valves may be qualified as exceptions.
- e. All engineering, construction, operation and maintenance shall meet or exceed requirements specified by applicable codes and standards.

The conformance of the various reactors to the criteria are outlined as follows:

a. Primary System

1) Old Reactors (B,C,DR,F,H,C)

The primary systems have high integrity and are adequate for all phases of operation.

DECLASSIFIED

HW-61400 RD

2) K-Reactors

The primary systems have high integrity and meet the safety criteria, except for loss of 4.16 KV power during six pump operation at full speed. Under consideration is automatic speed reduction of the primary pumps upon loss of 4.16 KV power. This will assure reliability and adequacy of the system.

b. Secondary System

1) Old Reactors

The old reactor secondary systems meet the criteria for independence and have ample capacity. At C-Reactor, the primary and secondary power supply drive a common shaft and a common pump. This mechanical coupling is considered acceptable. The reliability of the secondary systems is under review. Major problem areas include a) possible failure of the coal-fired boilers to accelerate, b) vulnerability of valve-pits to damage or failure (all systems), and c) vulnerability of above ground steam lines to earthquake or storm damage. Improvement of reliability by providing oil-fired boiler capacity is budgeted for FY-1960.

2) K-Reactors

The reliability of the K-Reactor secondary system is very high, even though compliance with the criteria of independence from the primary system cannot be assured. Project CG-775 and a K-Reactor Improvement project study are in progress to further increase the reliability of the secondary system.

DECLASSIFIED

HW-61400 RD

c. Last Ditch System

1) Old Reactors

The last ditch system (high tanks and export line) at the old reactors meets the criteria for independence. The adequacy and reliability of these systems is now under review, and programs are underway to improve the reliability of the export lines.

2) K-Reactors

The last ditch system meets the criteria for independence. The reliability of the last ditch system is high according to actual tests but there are operating conditions under which the flow capacity of the cross-tie system is inadequate to assure no damage. The K-improvement program will remove this inadequacy and increase the reliability of the last ditch cross-tie system.

3. Reactor Confinement

Programs for the provision of fission product confinement have reached the project stage. The project is proceeding in three distinct phases; a) the installation of fog spray systems in the discharge areas of the reactors to "rain out" particulate matter in the event of irradiated-uranium fires, b) the installation of stack filter systems to further screen out fine particulate matter from the reactor ventilation system, and c) the provision of some means for the removal of halogen gases that otherwise would not be removed from building air released from the stack.

Phase I is well underway and the fog spray systems will be installed at all reactors during outages scheduled for this summer. Design of the Phase II filter buildings is complete. A development contract has been let to Arthur D. Little Company for the development of a suitable halogen removal system for Phase III and testing of filters.

DECLASSIFIED

HW-61400 RD

G. Reactor Control Problem

1. Start-up Control Problem

Conversion to I & E fuel elements has made the startup control problem more severe because of more positive overall temperature coefficients and flux distortion introduced during startup by reactivity-compensating enrichment. Supplementary control systems at most of the older reactors have minimized the effects of the I & E conversion. However, at the K Reactors, startup "losses" (an inverse function of startup efficiency) have nearly doubled. Development of improved reactor instrumentation and supplementary control systems is expected to offset these losses.

- a. Power rate-of-rise instrumentation will permit smoother, faster, yet safer power level raises during startup. Instrumentation should be installed at all reactors during 1959.
- b. "Zone" tube outlet temperature monitoring will permit closer approach to operating limits during startup and more efficient adjustment of the flux distribution, thus permitting higher startup levels. Systems are presently installed at four of the older reactors and will be installed at the other two early in 1960.
- c. Sub-critical monitoring instrumentation will permit faster control rod withdrawal and improved startup efficiency. Instrumentation will be installed at all reactors during the latter part of 1959.
- d. At reactors where control system capacity and flexibility are now limiting, the poison spline supplementary control systems will permit higher startup levels and more efficient startups. D and the K Reactors are presently equipped for poison spline use. It has been found, however, that the original hollow boron carbide-filled splines are not satisfactory, and spline usage has been

DECLASSIFIED

HW-61400-RD

essentially stopped until better splines are obtained. An order has been placed to obtain a supply of solid aluminum-boron splines.

2. Short-Term Equilibrium Control Problems

Both the conversion to I & E loadings and higher equilibrium power levels have tended to make reactor operation more unstable. Limiting problems have appeared only at the larger K piles. At the older reactors "tighter" operating control techniques appear adequate to contain cycling tendencies.

Longitudinal cycling problems became apparent early in K Reactor operation and have been offset by using a control rod configuration which essentially reduces the effective length of the pile.

Appropriate instrumentation is being developed (in-core flux monitoring) which will detect incipient cycling tendencies and permit early counter-acting control efforts. A prototype installation should be operable in July, 1959.

A longitudinal supplementary control system is under development which will permit direct axial cycling control.

The poison spline system installation will permit more precise control over long-term operating reactivity transients, reducing control rod movements which tend to "stir up" cycling tendencies.

3. Long-term Equilibrium Control Problems

During long equilibrium operating periods the buildup of plutonium in the reactor results in a steady long-term increase in reactivity. It has been found that insertion of control rods to control this gain is not fully effective because excessive rod insertion tends to distort the flux distribution, reducing pile flattening and generating distributional

DECLASSIFIED

HW-61400 RD

instabilities. The higher exposures attainable with I & E fuel elements permit longer operating periods and have magnified these problems.

Solutions are as follows:

- a. At the reactors which are equipped with ball valve columns, long term gains are controlled by charging small amounts of poison at short intervals into columns located so as to maintain pile flattening.
- b. Long term gains may also be controlled by reducing the moderator temperature (and reactivity contribution) by increasing the proportion of helium with gas atmosphere. However, at the K piles the gas atmosphere is normally almost all helium and little added control is afforded.
- c. The poison spline systems are expected to provide adequate long term gain control.
- d. Research into the use of nitrogen as a component of the pile atmosphere has been started. Nitrogen is a neutron absorber which has thermal conductivity properties similar to carbon dioxide. By gradually replacing carbon dioxide in the pile atmosphere with nitrogen, long term gain control may be effected. However, use of nitrogen at high power levels may be precluded by the resulting high temperatures of components such as rod tips, shields, seals, etc.

III. Category II and NPR - FY 60 and FY 61

A. Incremental NPR Development Program

The incremental NPR development program is required in addition to the Category I and Category II Reactor and Metallurgy programs to provide needed technological support for the design, construction, and startup of

DECLASSIFIED

HW-61400 RD

the New Production Reactor as scheduled. The need for this program arises from the increased scope of development effort accompanying the authorized construction project. The program is closely inter-related with the design and design test activities conducted under the NPR project, and will provide the technological details necessary for final design and start-up.

The Category II development program is a continuing effort with the objective of developing basic technology and demonstrating technical feasibility. It is not scoped or programmed to develop prototype or pilot plant scale information which is needed for support of a specific reactor concept. The NPR effort is programmed to support these specific areas.

The major areas of development under the NPR program include the following:

- a. Fuel Development to establish design of a fuel prototype and to establish coextrusion technology adequate to support the fuel element fabrication.
- b. Physics Development to determine NPR enrichment requirements, lattice parameters, temperature coefficients, control requirements, and other parameters necessary to evaluate the operational and safety considerations of the overall reactor system complex.
- c. Materials Development leading to the procurement of reactor grade graphite Zircaloy-2 process tubing, and shielding capable of maintaining integrity under the rigorous reactor operational conditions.
- d. Engineering Development to define or develop heat transfer characteristics of the fuel elements, coolant flow requirements, corrosion rates of materials used in the reactor coolant system, effective decontamination methods for removal of fission products and activated corrosion products, and process and nuclear instrumentation.

DECLASSIFIED

HW-61400 RD

- e. Equipment and Design Development to provide components necessary for the NPR design.

The fund requirements for the NPR Incremental Development program may be summarized as follows:

<u>Operating Costs</u>	<u>FY-1960</u>	<u>FY-1961</u>
Development	\$ 2,017,500	\$ 1,710,000
Unique Corrosion Facilities	<u>75,000</u>	<u>-</u>
Total	\$ 2,092,500	\$ 1,710,000

<u>Test Facilities (PAC)</u>	
1706 KER Modifications	\$ 2,400,000
Rupture Test Facility	1,500,000
Heat Transfer Facility	<u>450,000</u>
Total	\$ 4,350,000

GETR Graphite Irradiation	\$ 500,000
---------------------------	------------

B. Facilities Available and Required

Numerous facilities are being utilized to develop technology or to conduct design tests in support of the NPR Project. Those facilities currently in operation include the following:

- 1) Four KER in-pile recirculating coolant loops - used to obtain irradiation data and performance characteristics of fuel element concepts necessary for design of NPR fuel elements. Water quality, corrosion, film build-up, and decontamination data are also obtained to provide design data necessary for NPR systems design. These loops operate at nominal outlet temperatures of 250° - 290°C, and the process channels are not of sufficient size to accommodate full size prototypic NPR fuel elements.
- 2) EIMO Loops - These loops are small scale out-of-pile recirculating loops, operating at temperatures up to 300°C, utilized to obtain

DECLASSIFIED

HW-61400 RD

specific performance data of proposed fuel types. The loops are also utilized to obtain pilot data on water quality, corrosion, and film formation, which is correlated with in-reactor data obtained from the EER loops. Several of the ELMO loops are being utilized to evaluate decontamination procedures, and to obtain corrosion data of candidate decontamination methods on carbon steel, stainless steel, Zircaloy, graphitor, and other materials proposed for use in the NPR coolant systems.

- 3) Fission Product Decontamination Loops - Two small loops have been fabricated to evaluate the decontamination procedures required for the removal of fission products in carbon steel and stainless steel systems. These loops are augmented by two additional out-of-pile loops to evaluate the long term corrosion effects of decontaminants on carbon steel, stainless steel, Zircaloy-2, and other NPR materials.
- 4) ETR Loop - Two fuel irradiation test facilities, the 3 x 3 loop and the 6 x 9 loop, are programmed at the ETR to evaluate the irradiation performance characteristics of small, clad fuel samples. These loops will also be utilized in the development of rupture detection instrumentation, and the development and evaluation of decontamination procedures for stainless steel coolant systems.
- 5) Component Test Loop and Process Tube Rupture Loop - This recirculating loop located in 189-D, designed for operation at 300°C, is being utilized to test performance of specific components designed for NPR use. These include nozzles, joints, packings, closures, instruments, valves, connectors, etc. This loop also serves as a hot coolant source for the Zircaloy-2 process tube rupture loop, which is designed to

DECLASSIFIED

HW-61400 RD

test the venting area requirements in the graphite stack, and to characterize the effects of a process tube rupture.

Two other test loop programs are being designed, and will be placed in operation in FY 1962. These include:

- 1) Full Rupture Loop - This loop will be placed in the PRTR, and will be designed to test the rupture characteristics of full size prototype NPR fuel elements. Data will be obtained to evaluate the mechanism, effects, and consequences of several candidate NPR fuel types.
- 2) KER Modifications - Modifications will be made to two KER loops to permit irradiation testing of full size NPR fuel elements. The hydraulic characteristics of the loop will be designed to simulate typical NPR operating conditions. The loops will also be utilized to obtain corrosion, water quality, film formation, and decontamination data under closely controlled conditions.

C. Fuel Development

1. Engineering, Physics, Fuel Technology Aspects

The interaction of thermal hydraulics, physics, and fuel integrity aspects is complex for the various NPR fuel concepts and operating conditions. In order to properly and expeditiously take into consideration the interacting factors, and IBM-709 reactor system analysis code with built in iterative features has been developed which embodies both physics and thermal hydraulics. Ultimately four or five variations of the basic code will enable the selection of the NPR fuel concept by August 1, 1959 and establishing the final fuel design by January, 1960.

The fuel technology and physics aspects of the NFR are discussed elsewhere. The basic engineering aspects of several prototypic fuel elements are given in the following table.

COMPARISON OF ZIRCALOY-CLAD NFR
PROTOTYPIC FUEL ELEMENTS
(Estimated Maximum Conditions)

<u>Fuel Element Geometry</u>	<u>Cluster</u>	<u>Tube-Rod</u>	<u>Tube-Tube</u>	<u>Tube-Tube-Rod</u>
Dimensions ^a	0.78"	0.79" 1.34" 2.35"	0.52" 1.43" 1.85" 2.46"	0.50" 0.85" 1.48" 1.86" 2.46"
<u>Weight (lbs/ft)</u>	24	25	24-1/2	23-1/2
<u>Engineering Data</u>				
Pressure Drop, Full Column (psi)	105	128	126	154
Clad Surface Temp (°C)	287	306	290	279
Uranium Surface Temp (°C)	376	460	395	348
Max. Uranium Temp (°C)	700	666	525	434
Max. Heat Flux (BTU/hr-A ²)	608,000	870,000	775,000	521,000

a. Dimensions read radially outward.

2. Key Metallurgical Considerations and Program Schedules

Metallic fuels are subject to swelling, an isotopic volume increase, when irradiated at high temperatures. Swelling results from the diffusion of the inert gases formed as fission products to bubbles which have been nucleated in the metal. Temperature appears to be the most important factor, although geometry, cladding thickness, thermal cycling, coolant pressure and the presence of alloying elements also appear to affect the swelling behavior. Although swelling was first observed nearly ten years ago, there has been little systematic study of underlying mechanisms and, except for certain alloys of interest

DECLASSIFIED

HW-61400 RD

in the Naval Reactors Program, there has been little systematic experimental study of the behavior of uranium-base fuels. As a result there is no established theory nor are there sufficient data to permit determination of the degree of swelling to be expected under NPR operating conditions. Irradiation tests performed to date suggest that swelling will constitute a minor dimensional instability problem in extended surface fuel elements provided the exposure level is not too high. Further fuel element tests plus special capsule irradiations are being carried out to provide more definite information on this subject. Temperature monitored (but not controlled) capsules have been and are being irradiated in the MIR and ETR. The capsules contain NaK as a heat transfer medium which will provide no extra restraint and will permit expansion of the fuel samples. Large flux transients in the testing reactors have made it difficult to obtain good data from these tests. An irradiation program has been planned for the Hanford reactors because the flux levels are more steady. Difficulties in the design of a NaK containing capsule for use in a horizontal process tube have delayed the program but at the present time 30 capsules are scheduled for charging by the end of July. Goal exposure for the five process tube charge will be reached at varying times between October 1 and January 1. Examination and determination of dimensional changes will begin by December 1.

Collection and analysis of all available data is in progress with a preliminary estimate of the dimensional changes to be expected under anticipated NPR fuel operating conditions scheduled for preparation by July 15. Efforts to compute the dimensional changes will continue but there is little hope of success because of a) lack of an accepted mechanism and b) lack of data on the mechanical properties of uranium and zirconium.

DECLASSIFIED

HW-61400 RD

The high rate of reaction between uranium and hot water and the volume increase accompanying conversion of uranium to uranium oxide constitute hazards to the NPR in the event of failure of the fuel cladding for any reason. The release of corrosion products and fission products to the coolant stream lead to loop cleanup and decontamination problems while the deformation of the fuel element, as a result of the volume increase due to corrosion, may lead to coolant flow reduction, discharge problems, or, in the worst case, failure of a process tube while it is pressurized. Since it is not economically attractive to avoid all possibility of fuel failure, we must understand the possible fuel failure mechanisms and be prepared to do what is necessary to avoid the possible consequences mentioned above. An extensive program of study of failure mechanisms, of corrosion product release rates and of the rate and nature of the deformation accompanying failure is underway in ex-reactor facilities and is being planned for in-reactor facilities now being readied for the experiments. Studies to date have led us to the following tentative conclusions: 1) suitable metallurgical treatment of the fuel assembly can assure that the failure will be localized, 2) split-type failures will send a strong signal to the rupture monitor assuring easy detection and the corrosion products will be washed downstream rather than accumulate in one spot and block the coolant flow, 3) the dependence of corrosion rate on temperature is not very great, the rate varying by a factor of only 20 over the temperature range 200°C (water) to 485°C (steam). As a consequence, failure behavior depends less on location in the reactor and on the power generation rate than was originally expected. The general conclusion is that safe operation of the reactor in the event of a failure may be assured provided that a reliable, sensitive monitoring system is included in the design and that the reactor is shut down and the failure discharged as soon as possible. Continuing studies in ex-reactor facilities will be carried out to determine the rate of corrosion (and hence fission)

DECLASSIFIED

HW-61400 RD

product release, and the rate and nature of the deformation accompanying failure of the cladding under a wide variety of conditions. In addition, the extent to which failure of one fuel component will interact with adjacent fuel components and with the process tube will be studied.

Besides the dimensional changes that may result from swelling, as has been described above, there are several other potential sources of dimensional instability. Irradiation-induced growth, and the associated phenomena of bumping and warping, may be a source of trouble but careful attention to details of the fabrication process to assure grain size and preferred orientation control should minimize these possibilities. In addition, since the growth rate of uranium goes through a maximum near 250°C and drops to zero near 500°C, there should be less of a problem from this source than in our present reactors.

In two fuel element irradiations recently completed, one at HAP0 and one at Savannah River, a dimensional instability problem has developed which appears not to be related to either swelling or irradiation-induced growth. In the HAP0 irradiation, the fuel rods experienced a diameter increase and internal cracking was noted in the post-irradiation examination. In the duPont irradiation a tubular element underwent an increase in outer diameter and a decrease in inner diameter, the net effect being a considerable increase in volume. Post-irradiation examination has been delayed so that it is not yet known whether cracking has occurred. Both irradiations were conducted at higher power levels than before so the current thinking is that the dimensional changes are somehow related to thermal expansion and stress relaxation by creep. Further irradiation tests in progress and in the planning stages will shed more light on this problem.

DECLASSIFIED

HW-61400 RD

The fuel material will be encased in Zircaloy-2 so some consideration must be given to the pertinent elements of zirconium technology. The mechanical properties, corrosion properties, neutron cross-section, and the weldability are the properties of greatest interest. The Naval Reactors Program has developed a considerable amount of zirconium technology and several useful zirconium alloys, one of which, Zircaloy-2, is currently specified as the fuel cladding material for the NPR. The corrosion resistance of Zircaloy-2 is more than adequate for NPR fuel application so that an incentive exists for reducing costs here, either through reducing the amount of Zircaloy-2 used or else developing an alternate cladding alloy. It may be possible to reduce the dimensional changes by use of a high strength cladding or else to reduce the likelihood of cladding failure by use of a high ductility cladding. A modest alloy evaluation effort is planned along these lines and a close watch on the progress of alloy development efforts off-site will be maintained. As described elsewhere(below), studies of zirconium alloys for braze materials in the closure development effort are in progress.

Technical criteria requirements for the NPR fuel elements are as follows:

- a. The maximum temperature of the uranium in any fuel element shall not exceed the alpha-beta transformation temperature (662°C) at any time.
- b. The average exposure of Zircaloy clad fuel elements shall not exceed 1000 MWD/T for uranium temperatures less than 250°C, and 2000 MWD/T for uranium temperatures greater than 250°C.
- c. Heat transfer and flow calculations should allow for a possible uranium volume increase of 4% per 1000 MWD/T. If the uranium

DECLASSIFIED

HW-61400 RD

temperature does not exceed 500°C, the volume increase may be taken as 0.6% per 1000 MWD/T.

- d. Minimum rod separation in the Zircaloy seven-rod cluster fuel element may be considered to be 0.060 inch. With tubular elements, the criteria is met if the combined water volumes of the annuli surrounding the uranium tube are greater than the volume of the uranium tube.

3. Irradiation Program

a. KER

The KER irradiation program calls for testing modifications of the Zircaloy-2 jacketed cluster and tubular elements to exposures up to 5000 MWD/T, jacket surface temperatures up to 314°C, and heat fluxes to about 0.6×10^6 B/hr ft². Current tests include a natural uranium tube-and-tube element to 300 MWD/T and natural uranium rod-and-tube elements to 2000 MWD/T. A charge of enriched seven-rod cluster elements is being irradiated to 5000 MWD/T. One loop contains a high temperature aluminum corrosion test.

Within the next month natural uranium tube-and-tube elements will be charged to a goal of 2000 MWD/T. Irradiation testing of enriched tube-and-tube elements will start when these become available, probably in August, 1959. An evaluation of the irradiation performance of a modified end closure is planned, using natural uranium cluster elements. Evaluation of the closure on tubular elements will be carried out when the fabrication procedures are developed.

DECLASSIFIED

HW-61400 RD

b. ETR

In addition to the KER irradiation tests, the HAP0 test loops in the ETR will be used to provide further information on the in-reactor behavior of NPR candidate fuel elements. The emphasis must of necessity be on tests involving a small number of samples because of space limitations. One loop is in a very high flux region of the reactor and will be used for screening tests and high heat flux tests while the other is being equipped for failure tests, rupture monitor evaluation and decontamination studies.

The volume of tests to date has been low as one loop has only recently begun operation and the other is not yet ready. Installation of test loops in general has been responsible for the intermittent operation of the reactor recently and this will continue for several more months.

c. FRTR

Under consideration also is a loop especially designed for rupture testing and monitor evaluation. Present plans are for this loop to be installed in the FRTR. It would be designed so that a large number of tests could be conducted as a sort of "pilot plant" for monitor development. It is currently in the project proposal stage.

4. Fuel Fabrication Development

The co-extrusion process for cladding metallic uranium with Zircaloy has been developed by NMI to the point that it can be relied on to produce high quality material consistently, provided very careful control is maintained on all steps of the process. The metallurgical bond thus produced is very strong, in fact in most cases it exceeds the strength of the parent metals.

DECLASSIFIED

HW-61400 RD

[REDACTED]

Cladding thickness is uniform and not subject to penetrations as is the case with a brazed bond such as we now have with the aluminum cladding process. In the case of tubular elements variations of wall thickness are very critical but again careful control of process variables will yield a satisfactory product. There are certainly further refinements to be made which will improve product reliability and reduce costs but we are now confident that a workable cladding process has been demonstrated.

The above comments apply to the cylindrical surface cladding and closure of the ends must be carried out separately. This phase of the program is receiving a large fraction of our effort at present and is the most critical.

Work done by NMI, for both HAPO and duPont, has demonstrated that it is technically feasible to bond end closures to fuel element during the extrusion process. This so-called integral closure results in objectionable percentages of Zr in the finished product with attendant loss of uranium. Such elements have been made and are now being irradiated. Here at HAPO we are working on a welded closure which if successful will minimize the required amount of Zr and should also be more economical to apply. There are a number of variations of the welded closure under development and we are quite confident that within six months we will be able to select the most satisfactory method and proceed with setting up the production facilities.

Other post extrusion steps of the fabrication process under development include straightening, heat treating, surface conditioning, autoclaving, support attachment, and non-destructive testing. Each of these steps

[REDACTED]

DECLASSIFIED

HW-61400 RD

is showing satisfactory progress and it is not expected that they will be critical or will hold up the overall program.

5. NPR Fuel Fabrication Line Construction

Plans for an NPR fuel production plant are in the scoping stage. Very shortly after approval of the Project Proposal now pending, it will be possible to start detail design and go out for bids on Phase I construction. This construction will consist mainly of the building in which the extrusion press and other associated process equipment will be located. The extrusion press to be used and located in the production plant is the one now on order for the pilot plant under Project CGF-810. This change from the original idea of having two presses at HAPO seems prudent in light of the tight money ceiling on the NPR Project.

The second phase of the production plant construction project will include installation of process equipment in the Phase I building.

The Pilot Plant Project mentioned above will provide development facilities for carrying out all steps of the production process on a small scale. It will also be used for training production personnel.

D. Engineering

1) Zr-2 Process Tubes

Three pilot production contracts have been signed with Allegheny Ludlum Steel Corp., Harvey Aluminum Co., and Chase Brass and Copper Co. Each contract is a fixed price contract with price redetermination features for a minimum of 70 tubes. Total maximum cost is $\$3 \times 10^6$. Initial deliveries are due first of the year with completion later in CY-60. Production order placement is scheduled for February 1, 1960 with inquiry being sent out in November 1959. Maximum unit price (not

DECLASSIFIED

HW-61400 RD

including equipment) is \$11,000. After redetermination price may be lowered to \$9000 to \$10,000/tube. The techniques of fabricating process tubes of the required size has been developed. Current efforts are directed toward converting "laboratory" techniques to production and toward cost reduction.

Present knowledge of the physical and mechanical properties of Zr-2, as a function of temperature and mechanical and thermal history, is, in general, adequate to permit specification of in-reactor working conditions corresponding to an acceptable service life (25 years).

The process tubes for the NPR have been designed so that the total principal stresses (includes pressure stresses plus thermal stresses) shall not exceed a creep rupture limit for 20-year life (1% elongation). When this criterion is satisfied, the "Tentative Structural Design Basis for Reactor Pressure Vessels and Directly Associated Components" (Bu Ships ltr. SSN/S51 ((551B) is likewise satisfied. This latter criterion is used in the design of reactor components by other AEC prime contractors. Other points of conservatism that have been adopted in the design include the use of vacuum-annealed properties of Zircaloy-2 rather than cold work properties and the use of properties determined from testing of strip Zircaloy-2 rather than testing a tube.

Areas where present knowledge is not assuredly adequate will be reduced by aggressive investigative programs; service life will be increased by conservative design and careful control of critical environmental condition; and safe operation will be promoted by rigorous monitoring of process tube performance in reactor.

DECLASSIFIED

HW-61400 RD

Areas where investigative programs are in progress or planned are:

- a) Creep properties of Zr-2 under cyclic conditions of stress, temperature and irradiation. Effects of corrosion and abnormal operating conditions on creep. Work hardening (loss of Ductility) under irradiation. Brittle fracture. Stress relaxation.
- b) Zircaloy-2 reactions with gases and vapors such as oxygen nitrogen hydrogen CO₂ and H₂O. Although absorption of these gases are expected to be negligible at normal operating temperatures in the NPR, additional studies will be required in order to permit prediction of long range effects and the establishment of reactor operating limits which do not contain unnecessary conservatism. The potential effects of abnormal operating conditions such as uranium-water reactions adjacent to the zircaloy and excessive impurities in the reactor atmosphere or coolant and decontamination methods will be evaluated.

Technical specifications for the NPR process tubes have been established. These specifications cover dimensional requirements (ID (2.700 - 2.715 inches), OD (3.265 - 3.250 inches), wall thickness (.250 inch minimum), length (57.0 - 57.5 feet), straightness and surface roughness)) with tolerances specified, pressure testing, corrosion testing, chemical composition for sponge and finished tubing, processing requirements (degree of cold work, circumferential butt welding and special requirements for circumferentially butt welded tubes) and shipping instructions.

2) Graphite

Maximum temperature and temperature distribution in graphite stack of NPR are such to preclude any stored energy build-up and require

DECLASSIFIED

HW-61400 RD

a completely inert gas atmosphere thus in essence eliminating the problem of uniform routine graphite oxidation. Graphite oxidation will occur in the event of a ruptured tube, however, consequences of such infrequent oxidations will be minor compared to other effects of ruptured tubes. The stack is being designed to minimize (possibly eliminate) the effects of a tube rupture by use of venting system.

Contraction of the stack appears to be the major problem as regards graphite performance. The stack design is such that the maximum temperature of the load bearing portions of the stack will be maintained at 850°C or less by a side-to-side coolant system using recirculating water.

The R&D program in progress will develop information on high temperature dimensional stability of a number of new candidate graphites. As originally scheduled, choice of raw materials would be made in November 1959, and purchase specifications by February 1960. However, with the current lack of adequate test facilities, this schedule will be difficult to achieve. If the schedule is followed, choice of materials will have to be made on basis of inadequate testing.

Currently, review of all available information substantiates previous theories regarding the effect of crystal size, coke particle, density, fabrication processes on dimensional stability. Present candidates for raw material are two cokes apparently made by same process; one by National Carbon Co. and one by Great Lakes Carbon Co. Fabrication process is straight AGOT process. For purified material, the graphite would first be graphitized at 2800°C and then purified at 2450°C in standard process.

DECLASSIFIED

HW-61400 RD

[REDACTED]

The status of high temperature--high flux facilities is less than adequate. The L-42 position at MTR was unnecessarily lost after two-cycle exposure of a prime candidate graphite on June 8, 1959 to KAPL. There appears no likelihood of any further exposures in this facility. The L-48 position is undergoing the usual debugging shakedowns. Use of this position will always be burdened by operational complexities. Current assembly will be discharged June 29, 1959. The second assembly (charged on that date) will be discharged after a 3 or 4 cycle exposure depending upon performance of temperature control. Data from first assembly are not expected to be completely satisfactory.

In a capsule position in L-4 now, there are currently being exposed four samples. There is no temperature control or monitoring. Samples are scheduled out in August 15, 1959, and results will be available three weeks later. A second exposure will contain more samples. Some qualitative comparison of candidate cokes can be made from these exposures.

Assemblies in E-5 and N-7 of ETR continue to be plagued with failure of reactor to run at full power. It is not expected that data from these exposures will be of any real value in selecting NPR graphite.

A project agreement has been drawn up as a proposed supplement to contract AT(84-3)189, held between GE (AFED) and AEC (San Francisco Operation Office). The supplement is to be negotiated this week (June 22). The contract is for one year with approximately six exposures. The maximum cost is \$500,000 including a fee of \$10,000. The first two exposures will be in center loop and in side core loop.

[REDACTED]

DECLASSIFIED

HW-61400 RD

Each loop has eight positions with four samples at each position. One sample will be CSF, the other three NPR candidates. The sample temperature and flux will be monitored but uncontrolled. Each position will have a different flux level so that several exposures can be obtained from a single operational cycle.

There is a finite probability that there will not be sufficient data available by year's end to permit making a sound choice of graphite. Without the data, we must take a calculated risk or delay the Design and Construction schedule until the data are available.

3) Heat Transfer Burn-out Protection for NPR

The phenomena of heat transfer burn-out occurs when partial or film boiling on the heat transfer surface causes such a reduction in heat transfer coefficients that surface temperatures rapidly increase to the melting point of the heat transfer surface, i.e., the fuel cladding. Heat transfer burn-out can occur in either the non-boiling (sub-cooled burn-out) or the bulk boiling (quality burn-out) regions. The conditions of flow rate, temperature of the local coolant and system pressure are all important in determining the heat flux at which the burn-out occurs. Also, in the boiling region the amount of vapor (quality) is an important variable. There are essentially three questions that must be answered in order that the necessary measures of burn-out protection may be determined. These are: (1) under what steady-state conditions will sub-cooled burn-out occur in NPR fuel elements; (2) under what anomolous conditions (such as the maximum tolerance variations of fuel and process tube dimensions, fuel element dimensional instability, including growth, and the effects of other flow disturbances) will heat transfer burn-out occur under

██████████

DECLASSIFIED

HW-61400 RD

otherwise normal conditions, and (3) under what transient conditions of flow and pressure will burn-out occur.

To answer the first question, an essentially full scale heated mock-up of a single NPR pressure and heat flux burn-out will occur in the simulated heated fuel column. The necessary modification to experimental equipment is underway, and preliminary experiments are scheduled to begin in the fall of 1959. This work will be supplemented by experiments made in short test sections in which local conditions about the fuel charge can be more precisely controlled. These experiments are already underway. It is anticipated that these experiments will reveal that sub-cooled burn-out heat flux values are well in excess of a million Btu/hr/ft² under typical NPR operating conditions, whereas the expected maximum heat flux for NPR fuel will be more like 750,000 Btu/hr-ft² under typical NPR operating conditions.

The short test section equipment described above will also be used to investigate the effects of dimensional variations upon burn-out conditions to answer the second question. The results of these experiments will be analyzed in order to determine the necessary hot-spot or uncertainty factors that will be used to compensate these effects when operating limits designed to avoid heat transfer burn-out (sub-cooled) are established.

To answer the third question involves an extensive program of experimental investigation and in all probability it will be this program that will most strongly affect NPR operating limits. The objective of this program is to determine the necessary margin between operating conditions and true limits such as boiling burn-out that are necessary to account for anomolous or "accident" conditions

██

DECLASSIFIED

HW-61400 RD

of coolant supply. Basically, the program involves the full scale simulation of the conditions (such as the failure of a tube inlet or outlet hydraulic connector, rapid loop depressurization or flow loss, etc.) and determining the point at which burn-out occurs, following the transient heating, cooling, pressure and flow effects all the while. Then with due consideration for the process instrumentation, this information will be used to determine at what points instrument trips would have prevented the burn-out. Project action has already begun that will modify and adopt heat transfer experimental equipment for this transient experimentation. Beneficial use of this equipment (the initial experimentation) is expected in the summer of 1960.

4) Material Development

a) Carbon Steel and Stainless Steel

There is a broad background of technology concerning both carbon and stainless steel usage in high temperature water which is generally adequate. Important areas not defined adequately and which are being studied include:

(1) Decontamination Effect

Corrosion of these materials during the decontamination process and post decontamination effects are being stressed as part of the program to develop decontamination processes. Tests are underway to study uniform, pitting, galvanic, crevice, and stress corrosion effects of repeated decontamination on both carbon and stainless steels.

(2) Stress Corrosion and Cracking

Stress corrosion cracking of primary heat exchanger tubes

DECLASSIFIED

HW-61400 RD

is a problem of great concern. Studies will begin in the first half of FY-60 to better define allowable chloride ion and dissolved oxygen concentrations in the boiling secondary coolant.

b) Water Treatment

(1) Primary System

Tentative primary system coolant qualities have been specified.

- (a) Total solids content ≤ 5 ppm in primary coolant.
- (b) Undissolved solids content ≤ 0.3 ppm in primary coolant.
- (c) For S/S and C/S systems, dissolved O_2 content ≤ 0.07 cc/liter (STP) in primary system.
- (d) For S/S systems chloride ion content ≤ 0.05 ppm.
- (e) Dissolved H_2 content for primary coolant water is specified as a function of temperature and whether or not zirconium alloys are present.

Programs are aimed at establishing the adequacy of the specified quality and to relax requirements wherever possible. The main emphasis is to obtain a water quality which will minimize activity build-up and hence minimize the frequency of decontamination. Much of this work will be accomplished with the KER Loops.

Present plans call for operating KER No. 1 (a carbon steel system) and KER No. 2 (a stainless system) with a water quality (purity, pH, etc.) corresponding to the specified NFR quality. KER Loop No. 4 will be used to study variation of the specified quality. Thus, it is hoped

DECLASSIFIED

HW-61400 RD

to establish the optimum pH, the need, or lack of need, for H₂ addition, required purity levels, etc. This work is required not only to insure the adequacy of the NPR water plant but also as a basis for setting process limits for NPR operation.

(2) Secondary System

It is planned to simulate conditions at each step in the NPR secondary system (the boiler, condensers, condensate return system, and steam system) in ex-reactor tests. With this equipment it is planned to study:

- (a) Phosphate addition requirements to minimize tube fouling.
- (b) Kinetics of oxygen scavenging by hydrogen and sodium sulfate and the need for these scavengers.
- (c) Vapor phase inhibition to minimize steam corrosion.
- (d) Evaluation of various materials which may be used for condenser tubing.

These tests will start in the first half of FY-60 and results will supplement data obtained from more general testing carried out before exact conditions in the secondary system were established.

(3) Miscellaneous System

Other studies are slated at studying problems unique to special cooling systems, e.g., the graphite cooling system. These various systems will be simulated by laboratory equipment in order to isolate problems and determine solutions.

DECLASSIFIED

HW-61400 RD

E. Physics

1. Lattice Considerations

The physics of the NPR differs from that of the present Hanford Reactors in several important aspects. First is the fact that the NPR is designed below the cross over point. Thus, the lattice cannot support a chain reaction with natural uranium; the enrichment required is in the neighborhood of 0.9% U²³⁵, depending on the final choice of fuel element.

The lattice spacing of the reactor is 8" x 9", as compared to 7-1/2" x 7-1/2" and 8-3/8" x 8-3/8" for the present reactors. The graphite to uranium atom ratio is 36-46, compared to 67 for the present I & E loaded K reactors. It is thus evident that the NPR is quite undermoderated as far as the graphite moderator alone is concerned.

However, additional moderation is furnished by the large volumes of coolant water present in the NPR. Again, compared to the K Reactors with a water to uranium atom ratio of 0.31 the cold NPR ratio is 0.5 - 0.85. The fact that the NPR is undermoderated results in a) a fail-safe lattice on water loss, b) a large epithermal component in the neutron spectrum, c) a good product conversion ratio in spite of other effects such as excessive water which lower the conversion efficiency, and d) a high age to diffusion length ratio.

2. Conversion

The NPR reactor is optimized on a power plus production basis. Thus, it does not reflect the ultimate of a production-only graphite-moderated reactor. Assuming no production in control rods, the atom/atom conversion ratio of the NPR will be higher than for the older reactors but lower than for the K Reactors. Furthermore, design has

DECLASSIFIED

HW-61400 RD

been optimized on total Pu, not Pu²³⁹. Due in part to the inherent "harder" neutron spectrum of the new reactor, the concentration of the higher plutonium isotopes will be greater than in our present piles. Typical values for production yields of three prototypical NPR elements are shown in the table below. It is seen that on a yearly basis, the NPR out-produces the K Reactors, due to the higher specific power even though the intrinsic conversion ratio is less.

CONVERSION AND
TYPICAL PRODUCTION FIGURES OF NPR CONCEPTS

<u>Parameter</u>	<u>20#/ft Cluster</u>	<u>24#/ft Cluster</u>	<u>26#/ft Cluster</u>	<u>KIIIN</u>
Gm Pu/MWD (initial)	0.813	0.889	0.882	0.894
Gm Pu/Ton (2000 MWD/T)	1520	1620	1680	712 ^a
KG Pu/Yr (P = 4000 MW) TOE = 80%	888	940	975	698 ^b
% 240 (2000 MWD/T)	12.	13.	12.	7 ^c

^aValue of 800 MWD/T exposure

^bTime operating efficiency = 79%. Figure is for CY 58.

^cValue at 800 MWD/T exposure

3. Temperature Coefficients

A third aspect is the high temperature and pressure feature of the NPR and the attendant effects. In addition to the customary moderator and metal temperature coefficients, the NPR will experience a loss in reactivity as the coolant heats to equilibrium temperature and density. The loss of water from the density change is 17 percent.

DECLASSIFIED

HW-61400 RD

Since the reactor is already undermoderated, the water is worth less as a poison than as a moderator, thus, the reactor lattice multiplication factor decreases.

In general, the various temperature reactivity coefficients behave as follows. The metal temperature is negative in all cases, and the magnitude increases with fuel specific weight due to the increased amount of U^{238} in the lattice with fuel weight. Typical average values are shown in the next table. The graphite coefficient is positive and is a function of fuel specific weight and geometry. In general, the coefficient is larger, the smaller the element -- a consequence of both larger coolant annuli (more effective neutron blocking) and smaller thermal utilization with the smaller weight elements. Furthermore, the ripe reactor coefficient is larger than the green reactor coefficient due to the plutonium buildup. Typical ranges of values for the coefficients are shown in the table.

The integral coolant density effect is negative for all cases considered; however, the differential coefficient may be positive especially for the light fuel elements. Again as is the case of the graphite coefficient, the coolant coefficient magnitude is larger in the ripe reactor. The magnitude of the coefficient increases with fuel weight which is a consequence of the increased degree of undermoderation with the larger elements and the attendant steepness of the reactivity vs moderator to uranium ratio curve in this region. The typical range of coefficients to be expected are shown in the table.

DECLASSIFIED

HW-61400 RD

NPR TEMPERATURE COEFFICIENTS (AVERAGE)

<u>Reactor</u>	<u>Fuel</u>	<u>Metal Coeff.</u>	<u>Graphite Coeff.</u>	<u>Coolant Coeff.</u>
NPR	20 lb/ft	$-2.5 \times 10^{-5}/^{\circ}\text{C}$	$+2.6 - 7.8 \times 10^{-5}/^{\circ}\text{C}$	$-3.7 - 10 \times 10^{-5}/^{\circ}\text{C}$
NPR	22 lb/ft	$-3.0 \times 10^{-5}/^{\circ}\text{C}$	$+2.4 - 7.2 \times 10^{-5}/^{\circ}\text{C}$	$-4.5 - 11 \times 10^{-5}/^{\circ}\text{C}$
NPR	24 lb/ft	$-3.5 \times 10^{-5}/^{\circ}\text{C}$	$+1.8 - 6.2 \times 10^{-5}/^{\circ}\text{C}$	$-6 - 12 \times 10^{-5}/^{\circ}\text{C}$
NPR	26 lb/ft	$-3.9 \times 10^{-5}/^{\circ}\text{C}$	$+1.5 - 5.4 \times 10^{-5}/^{\circ}\text{C}$	$-6.4 - 12 \times 10^{-5}/^{\circ}\text{C}$
K	Solid	$-2.3 \times 10^{-5}/^{\circ}\text{C}$	$+3.75 \times 10^{-5}/^{\circ}\text{C}$	0
K	KII	$-2.4 \times 10^{-5}/^{\circ}\text{C}$	$+3.12 \times 10^{-5}/^{\circ}\text{C}$	0

4. Migration Area and Reflector Savings

The high temperature and pressure features dictate adequate venting spaces in the graphite. The total graphite void fraction is on the order of 15 percent which includes vent spaces, rod and ball channels, gas passages, etc., etc. Thus, the neutron streaming corrections in the NPR are large and the age and diffusion area accordingly large. Since the reflector graphite is solid a very good reflector savings is achieved in the NPR due largely to the large slowing down and diffusion areas. Compared to the K- Reactors the reflector savings is 1-1/2 to 2 times as much.

5. Spectrum

The fact that the graphite and coolant temperatures are higher in the NPR than in the present reactors, and the fact that the reactor is undermoderated produces a neutron spectrum which differs in two respects from the normal Hanford reactor spectrum. First, the average energy of the Maxwellian is higher; second, the epithermal component is stronger. Both these differences, plus the existence of larger fuel elements which will "harden" the spectrum even more by selective absorption make the NPR less thermal

5
DECLASSIFIED

HW-61400 RD

than our other reactors. The first effect of the "higher temperature spectrum" is of course greater Pu^{239} burnout. Additional effects which will be difficult to calculate are the effects of resonance absorption (just above thermal) on the temperature coefficients, long term effects, etc.

6. Graphite Cooling Tubes

The necessity of graphite cooling tubes, complicates the physics of the reactor as non-essential (as far as physics is concerned) poison and moderator materials are added to the reactor. The effect of these materials on the over-all performance, however, is small.

7. Control and Instrumentation

A fourth difference in the new reactor is the fact that the design of the control rods, (of which there are 87) and the ball 3-X system (of which there are 108 channels) comes closest of any previous Hanford designs of being adequate for a flexible reactor. Fuel loadings composed of elements weighing in a range of 20 to 26 pounds per linear foot can be accommodated in the reactor with varying degrees of adequate control and safety. Of course, it is not possible for a reactor with a fixed lattice to accept radically different fuel elements - all optimum; however, in the NPR a surprisingly wide range of elements will be operable (with different degrees of optimization). The strength of the control system is also functionally dependent on the fuel element weight and shape through the age to diffusion area ratio.

The instrumentation, safety and control systems of the NPR will incorporate all the best and latest developments from the present

██████████

DECLASSIFIED

HW-61400 RD

reactors. In addition, set-back and some degree of automatic control will be provided.

8. Power Level Stability

a. Negative Prompt Reactivity - Temperature Coefficient

It is conceivable that the characteristics of the reactor and a recirculating heat dissipation system could be such that self-sustained divergent power oscillation could be set-up. However, this is not considered to be a serious consideration if the prompt reactivity change with a water temperature increase is always negative. The power level should either tend to be self-dampening or easily controlled by rod movement. If the prompt power coefficient is negative through all normal operating conditions the reactor operation with constant inlet temperature should be free from rapid total power level and flux distribution "drifting". With a strongly negative temperature coefficient of reactivity excessive rod movement could be necessary to prevent over compensation of the power level to adjust the system temperatures if the inlet temperature fluctuated appreciably. Such a consideration will weigh in consideration of whether or not the heat exchanger system must be capable of maintaining the inlet temperature to the reactor rigidly free of fluctuation. However, if the prompt water temperature coefficient is zero to mildly negative reasonable variations in inlet coolant temperature should not adversely affect operation. In fact, through proper balance of inlet temperature variation with power load and demand with the prompt reactivity-water temperature coefficient the power level could automatically adjust to the load demand with negligible recourse to rod motion.

██

54
DECLASSIFIED

HW-61400 RD

b. Positive Prompt Reactivity - Temperature Coefficient

With a positive prompt coefficient the power level would tend to diverge following any disturbance. This would be intolerable.

One grey zone is with a negative metal coefficient and a slightly smaller positive water temperature effect. With certain discrete combinations of coefficients and time delays between power level and temperatures a rapid undamped power oscillation could occur. However, a slightly positive water temperature coefficient would be permitted if it were strongly over-shadowed by the negative uranium temperature coefficient.

9. Uncontrolled Slow Flux Distribution Cycling

The traverse motion of control rods is not well suited to the prevention of cycling in the longitudinal direction. This cycling is the result of several factors: a) iodine¹³⁵ decay to xenon¹³⁵ which decays and is burned out, (tendency follows specific power), b) positive temperature coefficient (cycling increases with coefficient and power level), c) length to migration area ratio of the reactor, and d) relative fraction of flux outside control rod pattern.

It is believed that either a control movement parallel to the process tubes or a means of fine control in the axial fringes should be provisioned for in the NPR design to offset any longitudinal cycling tendencies.

10. Research Program

An intensive laboratory program is and has been underway to attempt to satisfy the physics needs. Cold, clean measurements of the

DECLASSIFIED

HW-61400 RD

lattice parameters of solid, cored, I & E and cluster fuel elements have enabled the devising of theoretical models which now allow quite accurate calculations.

Similar measurements are now commencing for either NPR candidate element types such as tube-rod and concentric tube. Exponential and PCTR mockups of the final NPR lattice will be used to peg the enrichment level of the NPR slug.

TTR neutron temperature experiments and neutron-water differential scattering cross section experiments are commencing to yield basic data vital to the temperature coefficient problem. Doppler coefficient experiments are programmed to extend our knowledge in the metal-temperature coefficient field. Mocked-up reduced coolant density experiments will be performed to get at part of the coolant coefficient. It is hoped that sufficient basic knowledge on the separate components of the temperature-dependence problem will enable an accurate assessment of the integral temperature effect. Probably no other physics problem will receive so much attention for the NPR.

Other areas of physics, also important are being studied. Reactor control, nuclear safety, conversion ratio, cold, clean parameters, neutron spectra, flux distributions, xenon transients, power and heat generations, criticality in handling E-metal, and instrumentation etc., etc.

F. Nuclear Safety

1. Basic Criteria

a. The NPR must satisfy the generalized control criteria (see IIF).

DECLASSIFIED

HW-61400 RD

Satisfactory conformance will be considerably easier however than in the present reactors due mainly to the fact that all fuel concepts and sizes under consideration make the reactor so under-moderated that it is fail-safe upon loss of water. Thus it is conceivable that there will be no speed of control or total control problems associated with water loss. A new type of accident is possible and this is the cold-water accident. Here, a large reactivity gain can be experienced which would require the insertion of safety control elements. However, this accident is far from as demanding as the water loss accident in the existing reactors due to the ample warning time to scram and the fact that no meltdown disaster can occur with cooling water present.

- b. The NPR must also satisfy the water plant reliability criteria. The fact that the NPR is a recirculating reactor complicates considerably the reactor and heat exchanger heat dissipation systems. Burns and Roe are currently designing the entire NPR heat dissipation system to conform to the criteria.

R. Nilson

R. Nilson, Supervisor
Reactor Physics Unit
Research & Engineering Section
IRRADIATION PROCESSING DEPARTMENT

vb

**DATE
FILMED**

6/23/94

END