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NUCLEAR SUPERHEAT PROJECT SEVENTH QUARTERLY
PROGRESS REPORT, JANUARY-MARCH 1961

By
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September 1961
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Atomic Power Equipment Department
General Electric Company
San Jose, California

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NUCLEAR SUPERHEAT PROJECT
SEVENTH QUARTERLY PROGRESS REPORT

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by

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Atomic Power Equipment Department

GENERAL ELECTRIC COMPANY

San Jose, California

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1.0 INTRODUCTION AND SUMMARY

1.1 Introduction

This is the seventh of a series of quarterly reports which will cover the progress and results from the conceptual design, economic evaluations and research and development work performed by the General Electric Company as part of the Nuclear Superheat Project under Contract AT(04-3)-189, Project Agreement No. 13. The following list of progress reports and topical reports have been published as a result of this work.

GEAP-3290, First Quarterly Progress Report, July - September, 1959

GEAP-3319, Superheat Process Tube Heat Transfer Tests

GEAP-3371, Second Quarterly Progress Report, October - December, 1959

GEAP-3387, Fabrication, Irradiation and Evaluation of Superheat Fuel Elements

GEAP-3468, Third Quarterly Progress Report, January - March, 1960

GEAP-3538, Fourth Quarterly Progress Report, April - June, 1960

GEAP-3563, Interim Report on Steam Dryer Development

GEAP-3564, Results of Air-Water Steam-Water Tests on Primary Steam Separators - October, 1960

GEAP-3581, Fifth Quarterly Progress Report, July - September

GEAP-3589, Economic Study for 300 MW(e)
Separate Superheat Reactor

GEAP-3590, Economic Study of the Mixed Spectrum Superheater

GEAP-3591, Manufacture of the Adhesive Bonded
AEC Superheat Critical Fuel

GEAP-3633, Economic Study for 300 MW(e)
Once-Thru Superheat Reactor

GEAP-3686, Sixth Quarterly Progress Report,
October - December, 1960

Section I of the First Quarterly Report (GEAP-3290) presented a description of the Nuclear Superheat Project including objectives, approach to the problem and expected results by individual task. The following tabulation of task titles is listed for easy reference.

TASK A - Conceptual Design and Program Evaluation

TASK B - Fuel Technology

TASK C - Materials Development

TASK D - Experimental Physics

TASK E - Coolant Chemistry

TASK F - Heat Transfer

TASK G - Mechanical Development

TASK H - SADE

TASK J - Mixed Spectrum Superheat Study

1.2 Summary

The following section provides a brief summary of significant results by task.

1.2.1 TASK A - Conceptual Design and Program Evaluation

An analog computer analysis of the Separate Superheat Reactor is being carried out. Phases of this study have been completed with regard to relief valve requirements, by-pass valve requirements, and the automatic control system. Work is continuing on selection of an emergency back-up cooling system and on preparation of a complete control-safety document.

A study has been initiated on a computer program for a high quality, once-through boiler. This code when developed will permit more systematic evaluation and optimization of this concept.

Preliminary calculations have been done for a typical superheat element which contains varying concentrations of erbium oxide, a burnable poison, mixed with the fuel. It has been tentatively concluded that greater reactivity control and increased exposure time may be realized at certain concentrations of fuel and erbium oxide. Additional work is in progress on such problems as economic effects, effect upon control rod worth, etc.

1.2.2 TASK B - Fuel Technology

During January, the fuel element designated SH-4B was irradiated in VBWR for a period of 15 days. Following the irradiation, the element was examined at the Radioactive Materials Laboratory.

A comparison of the pellet process with vibratory compaction showed no real economic incentive for fabricating fuel using the powder process.

Development work was continued on zirconium process tubes and on insulated stainless steel process tubes.

Evaluation was completed of the two experimental runs concerning dynamic erosion of UO_2 . Analysis of the results of these two sets of experiments are discussed.

The continued data analysis of the Trail Cable Capsule Experiments has resulted in the correlation of the observed UO_2 growth with the clad restraining force.

A high temperature pressure vessel has been installed and thermal cycling tests have begun on superheat fuel elements.

1.2.3 TASK C - Materials Development

A Cal-Tester tensile machine has been procured and modified for in-cell elevated tensile tests on coupon-type specimens. Tensile specimens have been prepared from samples procured from vendors.

Tubing has been ordered to be used for obtaining information necessary to write material specifications for nuclear superheat development.

1.2.4 TASK D - Experimental Physics

Experimental gross reactivity measurements have been obtained for uniformly spaced arrays of annular, slightly enriched, oxidized fuel. Correlation of these results with a mathematical model shows some discrepancies which should be resolved by further experiments. At present, there is some uncertainty over extrapolating from this critical data to a large superheat reactor.

1.2.5 TASK E - Coolant Chemistry

Tests of steam corrosion of type 304 stainless steel were made in the corrosion loop at San Jose. The amount of corrosion and the metal loss to the system were calculated. Further work has been started on the evaluation of metallurgical changes caused by long exposure with a 50°F ΔT across the cladding.

1.2.6 TASK F - Heat Transfer

GEAP 3703, a topical report on Once-through boiling heat transfer, is essentially complete and will be issued shortly.

Manufacture and inspection of the superheat test vessel auxiliary steam drum was completed. Final installation was scheduled for April, 1961.

1.2.7 TASK G - Mechanical Development

The multi-arc (radial vane) steam separator showed the greatest promise of the types tested. Development of this separator is continuing and air-water tests of an adjustable two-vane separator have been completed.

Due to unsatisfactory test results, it is recommended that no additional work be performed on an integral dryer for a superheat fuel assembly, since satisfactory large dryers are available for use in lieu of integral dryers.

Development has been initiated on a device which will seal the joint between a superheat fuel rod and associated steam piping. A seal actuated by a low melting alloy appears to be the most feasible, and further studies of this seal are now being done.

Tests have been conducted on E-SADE process tube seals and also on the problem of thread galling under reactor operating conditions.

1.2.8 TASK H - SADE and E-SADE

1.2.8.1 SADE

Fuel element SH-4B was tested in the SADE loop from January 15 to January 29, when it had to be removed due to VBWR shutdown. The element was taken to the RML for inspection to determine its exact condition. SH-4B was visually examined, photographed, dimensioned, and gamma scanned at the VAL, RML facilities.

The delayed neutron leak detection system has been designed.

Extensive design studies have been made to determine how the NUSU superheat element could be irradiated in the SADE test loop. Preliminary analysis indicates that the element must be enclosed in a 1.90 I.D. flow tube.

1.2.8.2 E-SADE

Work is continuing on E-SADE design and installation. A detailed schedule has been prepared for installation of the external system in order that interruption of VBWR operations will be a minimum. The Piping and Instrumentation drawing is now being finalized. At present, the analytical portion of the Loop Safeguards analysis is approximately 50 per cent complete.

1.2.9 TASK J - MIXED SPECTRUM SUPERHEAT REACTOR DESIGN

Work on the Mixed Spectrum Superheater consisted of: (a) Completing minor points on the reference 300 MW plant design for the steam cooled reactor study, and (b) Further evaluation of flooding effects.

2.0 TASK A - CONCEPTUAL DESIGN

2.1. Analog Computer Study for Separate Superheat Reactor

Three phases of the Separate Superheat Reactor analog study have been completed and are briefly reported in this quarterly. These three phases are:

- (a) Relief Valve Requirements.
- (b) By-Pass Valve Requirements.
- (c) Automatic Control System.

Control and analog computer studies are continuing in these areas:

- (a) Selection of back-up emergency cooling method to provide cooling upon failure of by-pass valves.
- (b) Contacting vendors to determine availability of control components to meet separate superheat reactor requirements.
- (c) Preparation of a complete plant control and safety system design, review of the system performance, and issuance of a complete control-safety document.

2.1.1 Relief Valve Requirements

An analog computer study has been made to determine the Relief Valve requirements for the Boiling Water-Separate Superheat reactor combination. The recommended requirements are as follows:

- (1) Separate Superheat Reactor Relief Valve Requirements:
 - (a) Capacity - 100% of rated steam flow
 - (b) Location - Exit steam line
 - (c) Valve lifting pressure - 150 psi above operating pressure
- (2) BW Reactor Relief Valve Requirements:
 - (a) Capacity - 140% of rated steam flow
 - (b) Location - BWR reactor vessel
 - (c) Valve lifting pressure - 200 psi above operating pressure

2.1.2 Criteria For Selection

The criteria for selection of relief valve requirements is the following accidents: With both reactors operating at rated power, the turbine stop valve closes. All by-pass valves, emergency cooling methods, and reactor safety systems fail to operate.

The relief valves should limit the resulting nuclear excursion and limit the pressure rise to less than 300 psi.

The selected relief valve characteristics meet the above requirements and also result in an adequate operating pressure range.

It should be noted that the relief valves limit the nuclear power and pressure excursion but they do not prevent SH fuel cladding or BW fuel melting.

2.1.3 Method of Study

The method of studying this problem was to use the analog computer to study the effects of the more important variables. As the relief valve requirements are set by the boiling water reactor design, the recommended relief valve requirements will change as the BWR design changes. The recommended relief valve requirements are based on these BWR characteristics:

- (1) BWR void reactivity = 2.80 \$
- (2) Fuel element time constant = 13 seconds
- (3) Initial BWR pressure rise = 18 psi/sec
- (4) Fuel Doppler Coefficient = 1 \$ per 1000°F
- (5) SHR moderator void reactivity = \$1.00

2.1.4 Computer Results for Selected Relief Valve Capacity

Figure 2.1 illustrates the effect of 100% relief capacity located on the SSHR exit and 140% capacity on the BWR vessel. The following sequence of events occur.

- (1) The stop valve closes at time zero and turbine steam flow ceases.
- (2) The pressure in the BWR and the SSH reactor start to climb.
- (3) The rising pressure produces steam void collapse in both the BWR and the SSHR and the fission power of both reactors begins rising.
- (4) At time of 4 seconds, the SHR exit steam line pressure has risen 150 psi and the SHR vessel pressure has risen 100 psi. The SHR relief valves start to open.

At this time, the SHR fission power is up to 280% and the BWR fission power is up to 300% of rated power.

- (5) At time of 9 seconds, the BWR pressure is 200 psi over operating pressure and the BWR relief valves open.
- (6) The excursion ends a few seconds later and the maximum pressure rise is 220 psi.

At the peak of the excursion, the BWR power has increased by 320%, and the SHR power by 350%.

During the transient, the clad temperature rises 700°F. For a hot channel clad factor of 2, the maximum clad temperature rise would be 1400°F, to a total temperature of 2600°F.

2.1.5 By-Pass Valve Requirements

The following discussion is a summary of the By-Pass Valve

Control Study. The criteria for acceptable By-Pass Valve performance is as follows:

- (1) When turbine control valve closes due to speed governor acting, the by-pass valve system must:
 - (a) Prevent high flux scram
 - (b) Prevent high pressure scram
 - (c) Maintain superheat clad temperature below 1350°F. The transient clad temperature limit is set at 1350°F in order to permit detection of by-pass valve failure and to permit activation of back-up emergency cooling to keep clad temperature below 1500°F.
- (2) The by-pass valve must meet the requirements of (1) when the turbine is at any load level and for any degree of control valve closure.
- (3) When the turbine stop valve closes, the bypass valve must provide emergency cooling to maintain clad temperature below 1350°F.

2.1.6 By-Pass Valve Characteristics

A by-pass valve system to meet the above requirements can be met by a system of the following characteristics:

- (1) Capacity - 120% of full steam flow
- (2) Speed - Full stroke in 0.5 seconds.
- (3) Controller action - The speed governor opens the bypass valve by approximately the same amount as it closes the control valve. The pressure regulator then acts on the by-pass valve to maintain system pressure.

The emergency speed governor closes the stop valves and control valves, and simultaneously opens the by-pass valve.

2.1.7 Automatic Control System

The automatic control system study came to the conclusion that automatic temperature control is very feasible from a control viewpoint. The unknown factor is the life of a superheat fuel element under load when subjected to constantly changing clad temperature conditions.

The automatic control system objective is:

- (1) Hold steam temperature to within $\pm 25\%$ for any load change at a rate of 3% per minute.
- (2) Return steam temperature to within $\pm 50^{\circ}\text{F}$ within one minute after a $\pm 10\%$ step load change.

The automatic temperature control system has these characteristics:

- (1) Controlled variable - outlet steam temperatures
- (2) Control Means - Control rod position
- (3) Controller Parameters
 - (a) Temperature proportional gain - $10\phi/25^{\circ}\text{F}$
 - (b) Temperature integral gain $2\phi/25^{\circ}\text{F-sec}$
- (4) Temperature sensor time constant - 1 second
- (5) Control rod reactivity rate - $2\phi/\text{sec}$
- (6) Control rod position control time constant 1.0 sec.

2.2 Once Through Superheater Optimization Code

The development of this code is considered necessary to permit evaluation of the numerous variables and the complex iterations required to optimize the performance and power costs of a once-through superheater.

The following input data is provided for the program:

1. Exit steam quality from the annulus of hot element hot channel.

2. Exit steam quality from hole of hot element hot channel transition element.
3. Steam quality at end of first low quality boiler pass.
4. All the fuel dimensions for both fuels, except for the center hole on the annular element and the length of the core which are omitted.
5. The only flow specified is the mass flow rate in the hole of an average transition element in a hot channel.
6. Other data such as conductivities, pressures, convergence error, peaking factors, etc. are also given. The flux distribution in the form of a chopped and/or skewed cosine curve is specified but not the magnitude of the heat rate.

Items 1 through 5 are the major input items. The program adjusts heat rate, inner hole size, flow rates, and core length by systematic iterations using a special forced convergence routine until one of the transition element clads is operating at 1250°F and the first pass boiler rod fuel is up to peak performance. The center line and internal maximum temperatures are checked for both fuels at overpower transient conditions. If the temperature is excessive the fuel size is adjusted to give a maximum temperature of 4500°F. If the fuel sizes specified in the input produce cool fuel temperatures, no changes are made in the sizes. The program, while doing all the above, must balance the pressure losses in the hole and annulus of both the hot and average transition elements as well as balancing the pressure loss across the average and hot elements themselves.

Once the core is sized, the program continues with the calculation of the fuel cycle costs and adjusted capital costs. By making a systematic parametric study of the input variables, the optimum economic reactor configuration could be determined.

2.3 Separate Superheat Reactor Design Studies

2.3.1 Introduction

A study has been initiated to determine the advantages and limitations of using natural erbium oxide, intimately mixed with the uranium oxide fuel, as a burnable poison.

The most significant effect that the addition of burnable poison in the fuel brings about is a reduction in the slope of the reactivity versus burnup curve. Thus, at any given time, fuel containing burnable poison will always have a lower excess reactivity than corresponding fuel without it. This leads to several very important considerations, one of which is the achievement of higher fuel exposures for a given control rod strength. Conversely, a reduction in control rod strength for a given exposure can be accomplished, thus reducing the severity of accidents associated with control rod movement. Since flooding reactivity effects are directly proportional to excess reactivity, these also will be reduced in a burnable poison containing fuel.

If the slope of the reactivity versus time or burnup curve can be made to approach zero (flat), then power shifting with burnup can perhaps be held to a minimum and an "optimum" power shape maintained throughout life. This is particularly desirable from the standpoint of shaping axial power.

In general, it appears that the use of erbium as a burnable poison can help to make a safer reactor for any given fuel exposure. If the economic advantages of longer exposure and better power shape outweigh the costs associated with using erbium, then a safer reactor with a very desirable economic advantage will result.

2.3.2 Summary and Conclusions

Preliminary zero-dimensional burnup calculations have been carried out for a typical superheat element for several concentrations of erbium oxide intimately mixed with the uranium oxide fuel. The U-235 enrichment was 2.6% and the water to fuel volume ratio was 2.6. These results were then used in radial one-dimensional calculations to determine the equilibrium conditions for a core with fuel containing erbium and a core without burnable poison.

The tentative conclusions that appear at this early stage of the analysis are:

- (a) The concentration limits for the use of erbium in the fuel for a physically realizable core will be determined by the reactivity swings associated with burnup and temperature changes that can be tolerated from the standpoint of providing adequate shutdown margin, with control rods, in the cold flooded condition.
- (b) The concentration limits for erbium for an economic advantage will depend on the price structure of erbium and the degree to which the additional fabrication costs and economic penalty due to erbium residue (unburned) at end of core life are balanced by the increased exposure and/or changes in specific power. (For certain considerations, the radial power shape can be worse for erbia containing fuel than that without, as is mentioned in conclusion d.)
- (c) The reactivity changes associated with burnup and temperature are less severe than predicted in earlier estimates and a major share of reactivity control can be accomplished with erbia in the fuel.

- (d) The analysis of spatial burnup and power peaking effects for cores containing burnable poisons is considerably increased in complexity over conventional cores. This is due to the gross changes in the slope of the reactivity versus burnup curve as burnup takes place. Thus, the problem of finding an optimum refueling procedure becomes more difficult. For example, in the conventional multi-batch outside-in fuel shuffling procedure, the power shape can actually worsen as the equilibrium core is reached for certain values of U-235 enrichment and erbium concentrations. However, it is expected that higher values of enrichments and erbium concentrations may yield as good a power shape at these higher exposures as the conventional fuel has at 12,000 - 15,000 MWD/T.

2.3.3. Discussion of Results

2.3.3.1 Zero-Dimensional Burnup Calculations

A first step in the preliminary scoping phase of an analysis program aimed at determining a technically feasible and economically attractive method of achieving high fuel exposures with burnable poisons has been completed. Zero-dimensional burnup calculations have been carried out for several concentrations of erbium oxide intimately mixed with the uranium oxide fuel. The results are shown in Tables 2.1 through 2.3 for a typical superheat element having a U-235 enrichment of 2.6% and a water to fuel volume ratio of 2.6. None of the spatial and other non-uniform burnup effects have been accounted for in these tables:

TABLE 2.1

Initial Core Reactivity Held Down By Erbium-167

<u>Initial Erbium Concentration</u> <u>(atom Er-167/atom U)</u>	Δk_{∞} at 0.0 MWD/T <u>($\Delta k_{\infty} - k_{\infty}$ (No Er) - k_{∞} (Er))</u>
0.0	0.00
0.00175	0.13
0.0025	0.18
0.0035	0.24

Note: 1. Moderator temperature = 285°C

TABLE 2.2

Reactivity Change With Moderator Temperature Vs.

Erbium-167 Concentration For The Initial Core

<u>Initial Erbium Concentration</u> <u>(atom Er-167/atom U)</u>	<u>%$\Delta k/k$ (hot to cold)*</u>
0.0	+1.2
0.00175	+3.5
0.0025	+4.6
0.0035	+6.0

$$* \% \Delta k/k = \frac{k(20^{\circ}\text{C}) - k(285^{\circ}\text{C})}{k(20^{\circ}\text{C})} \times 100$$

TABLE 2.3

Positive Reactivity Swing Vs. MWD/T Exposure

<u>Erbium Concentration</u> <u>(atom Er-167/atom U)</u>	<u>%$\Delta k/k$*</u>	<u>MWD/T At Which k</u> <u>Goes Through A Maximum</u>
0.0	---	None
0.00175	---	None
0.0025	0.73	2,000
0.0035	1.91	3,500

$$* \% \Delta k/k = \frac{k(\text{maximum}) - k(\text{initial core})}{k(\text{initial core})} \times 100$$

From the work done to date, it appears that 15 to 20% Δk in the initial core can be held down with erbium.

It appears that both the temperature reactivity effect (Table 2.2) and the positive reactivity swing with life (Table 2.3) are less serious than indicated by earlier estimates, and a major share of reactivity control in the superheater can be accomplished with erbia in the fuel.

2.3.3.2 Radial One-Dimensional "Spatial" Effects

The above zero-dimensional ("point") burnup results were used in radial one-dimensional calculations to determine the equilibrium conditions for a core without burnable poison and a core with Erbium-167 intimately mixed with the uranium-dioxide as a burnable poison. A three-batch, outside-in, fuel movement sequence, discharging fuel at 12,000 MWD/T, was assumed for both cores. Control rod patterns and movements were neglected in this study. The results are shown in the following tables.

TABLE 2.4

End of Equilibrium Cycle Exposures

Equilibrium	% Enrichment	Average Burnup, At End Of Cycle, Per Core Region In MWD/T			Core Average At End of Cycle MWD/T
		Central	Intermediate	Outer	
Without Erbium	2.3	12,000	7,400	3,100	7,500
Er-167 Concentration = 0.0025 atom/atom U	2.9	12,000	6,700	2,600	7,100

TABLE 2.5

Power Shape at End of Cycle

Equilibrium	P max	$\bar{P}$ ave in reg			P ave in core
	$\bar{P}$ core ave	Central	Intermediate	Outer	
Without Erbium	1.181	1.151	1.076		0.773
Er-167 Concentra- tion = 0.0025 atom/ atom U	1.465	1.315	1.032		0.653

From the tables, it is seen that an enrichment penalty of 0.6% is incurred for this particular erbium concentration (which has not been determined to be optimum).

The equilibrium core without erbium at these exposures has a flatter power shape than the core containing erbium, since the former has the more optimum ratio of k_{∞} outside region to k_{∞} inside region, from the standpoint of outside-in fuel movement desirability. However, at greater discharge exposures (perhaps approaching 20,000 MWD/T), this will probably be reversed and the erbium core at equilibrium may have the better inherent power shape for multi-batched outside-in fuel shuffling.

3.0 TASK B - FUEL TECHNOLOGY

3.1 Irradiation Tests in SADE

An irradiation experiment was performed in VBWR during the January run. The fuel element was designated SH-4B, containing 4.5% U-235 isotopic enrichment. The bayonet, annular type fuel element was 1.25" outside and .75" inside diameter clad with 0.028" tubing weld drawn 304 type. The initial pellet to jacket diametrical clearance was $0.005 \pm .002$ inch, the UO₂ pellets were sintered to 94% minimum theoretical density. It was the first fuel element with 28 mil clad irradiated under superheat conditions.

The total irradiation lasted 15 days with maximum exit steam temperature of 740°F and estimated clad temperature of 910°F. These conditions were maintained for a total of 3 days. During the remainder of the time the reactor operated at lower levels. For short periods during reactor transient tests the maximum temperature of the exit steam was recorded at 750°F with a corresponding clad temperature of 930°F. Maximum heat flux was 200,000 Btu/hr-ft².

There was a total of 19 power cycles of the VBWR during the irradiation of SH-4B (i.e., more than 30% variation in power level).

The inlet and outlet steam in the SADE loop was monitored during the irradiation period to study fission product escape from the fuel element. No indication of fission gas release was detected in the coolant, although due to power level changes the sampling was not sufficient to establish a base trend of loop activity conditions.

Fuel element SH-4B was non-destructively examined at the Radioactive Materials Laboratory. The surface scale appeared normal and adherent on

all fuel element surfaces. Scale color variation was observed, in line with the temperature variation throughout the fuel length. The flow booster tube was not removed, because it was tightly held in the inner annulus. (During pre-irradiation assembly the flow tube was tightly fitted into the annulus.)

Maximum dimensional changes included .007" diametrical decrease at the hot (lower) portion of the outside diameter, and approximately .019" bow in one of the four measured directions. Bow in the other three directions was less than 0.003".

Following modification of the VBWR core, the fuel element will be reinserted and irradiated at 350,000 Btu/hr-ft² for one month.

3.2 Fuel Fabrication Development

A comparison of the pellet process with vibratory compaction for superheat fuel of present reference design (annular pellets, similar to AEC Superheat Critical Fuel) shows no real economic incentive for going to the powder process. The advantages which can be shown for the powder process in small diameter, rod-type fuel are substantially reduced in the superheat fuel design. Heavier pellets (large diameters) allow marked reductions in handling costs; i.e., less pellets to press, grind, inspect, and load. This savings, coupled with the premium price required for vibratory compaction type powder, essentially cancels the cost advantage which the powder process might offer. Figure 3.1 shows the fuel fabrication cost for powder annular fuel elements as affected by the premium paid for arc-fused powder. Also plotted are fabricating costs of pellets of the sizes being considered for present design concepts.

A clad sizing process has been developed for obtaining clad-pellet contact

Figure 3.1

ANNULAR FUEL ELEMENT ASSEMBLY

SWAGED ANNULAR FUEL

SWAGED ANNULAR FUEL

Figure 3.2

with thin wall (.016 inches) tubes. Pellet loading requires that the outer cladding tube have an internal dimension approximately .005 inches greater than the outside diameters of the pellets. This gap must be removed prior to reactor insertion or the high pressure will collapse the tube and cause sharp wrinkles on the outer surface. A low energy swaging process has been developed which allows the reduction of the outer clad with very little damage to the pellet structure. Figure 3.2 (top view) shows an annular fuel element in the final stages of assembly. The middle view shows a fuel element that has been swaged by conventional techniques and then split open for examination. The lower view is of a similar fuel element that has been swaged by the low energy method.

A double spring plenum support, Figure 3.3, has been designed and tested in the high temperature autoclave. Round wire Inconel "X" springs were used and contact surfaces ground to achieve maximum support of the plenum wall. The element was first tested by subjecting it to a pressure of 1,300 psi at room temperature and a second cycle of 1,300 psi at 1400°F. No detrimental effects were observed in the plenum area after either cycle. A wrinkle had formed in the outer cladding over the fuel area after the high temperature cycle. The element was then cycled six times from 300°F to 1400°F at 1,000 psi. During this period, the plenum failed as shown in Figure 3.4. This was considered an extreme test of a plenum support since in the reactor the maximum plenum temperature should be several hundred degrees lower. Additional tests will be conducted to determine temperature limitations on the above design and to evaluate alternate designs.

A program was initiated to determine the best method of fabricating spacers for annular fuel elements. Two concepts are currently being considered.

SECTION SHOWING PLENUM SUPPORT

Figure 3.3

Plenum Collapse
Figure 3.4

The first is a spiral fin spacer formed as an integral part of the cladding. Both chem milling and a special tube drawing process are being investigated as means of manufacturing this tubing. The second concept employs spacers which are welded to the cladding. Several resistance welding processes which allow a high degree of control are being considered. These processes have been used for accurate-control welding, and are expected to be suitable for spacer attachment to thin clad fuel elements.

3.3 Process Tube Development

3.3.1 Zirconium Process Tube - CL-1 Loop Tests

Metallographic examination was completed of the uninsulated zirconium process tube which had been exposed for 1,000 hours to dynamic superheat conditions. Fretting marks were discovered at the points where the stainless steel loop pipe spacers made contact with the zirconium tube. The deepest penetration was .008" and examination of this area at high magnification showed no oxide film near the bottom of the groove, indicating that the fretting occurred during the entire test. There was no appreciable difference in the oxide film thickness on the inside and outside of the tube. No hydrogen redistribution was observed at any point of the process tube tested, indicating a low temperature gradient from steam to water across the tube wall. In the final design, a stainless inner sleeve will reduce the temperature gradient across the tube walls even further. Hydride content was uniform along the length of the tube (30 ppm) except near the fretting marks where it was 80 ppm. Fretting or wear of the metal was accompanied by a localized increase in hydrogen pickup. Weight loss was 1.8 mg/dm²/day.

Another zirconium tube with a .010" wall stainless steel thermal liner was inserted in the loop to gain information on corrosion phenomena and insulating properties of the configuration. See Figure 3.5. This test ran for only 10 minutes at maximum steam flow (500 lbs/hr at 110 ft/sec) before the test had to be shut down due to a loss of steam flow. The test was repeated three times and had to be discontinued each time because of high pressure which developed across the flow path of the process tube under test. Since the possibility of a leak at the specimen loop seal had been eliminated, it was deduced that the liner was collapsing elastically and creating pressure conditions which caused the steam flow stoppage. During the next test, a .028" wall thermal liner was inserted in place of the .010" wall liner. This test has functioned satisfactorily for 750 hours, which indicates that the thinner walled liner was distorting either by pressure differential or vibration during the run. A "ribbed" .010" wall liner is being fabricated and will be placed in the loop at the conclusion of the present test.

3.3.2 Insulated Stainless Steel Process Tube

At-temperature (650°F) collapse tests on the annular thin walled stainless steel configuration are continuing. A prototype with .030" of flame-sprayed oxide on the inner tube withstood a room temperature differential pressure of 250 psi and was cycled twice at temperature with a maximum differential pressure of 50 psi without deforming. Because the at-temperature strength of the flame sprayed oxide configuration is still not demonstrated, alternate concepts are being investigated simultaneously. These include:

DETAIL - DYNAMIC SUPERHEAT TEST ON INSULATED PROCESS TUBE (.010" S.S. LINER)

Figure 3.5

1. Stiffening rings vibratory compacted in the annulus.
2. Stiffening rings brazed or spot-welded to the inner tube before assembly and oxide compaction.
3. Spring-wound spiral stiffeners brazed or spot-welded to the inner tube before assembly and oxide compaction.

Calculations based on thermal conductivity of compacted Al_2O_3 (reported in the last quarterly) indicate that the configuration is an excellent insulator. Heat losses to the boiling water would be about 3% as compared to the 40% heat losses with a solid stainless steel process tube.

3.4 Dynamic Erosion Studies of UO_2

Evaluation of the two experimental runs was completed, and the results were compiled in a topical report, GEAP 3698, "Erosion Studies of Powder Compacted Uranium Dioxide Under Dynamic Steam Flow," by C. N. Spalaris, F. A. Camprelli and M. Siegler.

The summary and conclusions from this work on dynamic erosion are included below. One more erosion run is presently planned to examine the erosion behavior of arc-fused powder-compacted fuel elements containing larger particles than those previously tested.

3.4.1 Summary of Work Reported in GEAP-3698 "Erosion Studies of Powder Compacted Uranium Dioxide Under Dynamic Steam Flow"

Erosion tests of purposely defected UO_2 , powder-containing fuel elements show that under certain conditions UO_2 powder can escape through defected clad into the coolant stream. It is too early to assign a cause for the loss of powder in some samples with large clad defect, whereas, in other cases under similar conditions the

powder was adequately contained within the purposely defected fuel specimens. In all cases studied, oxidation of UO_2 to higher oxides was identified both by metallographic and X-ray methods. Specifically, U_4O_9 as well as U_3O_8 were found in all defected specimens after the loop exposures of 780 and 946 operating hours. Because the steam (when pure) is a reducing rather than an oxidizing medium, it was concluded that the small amounts of dissolved oxygen added to simulate reactor steam chemistry were chiefly responsible for the observed oxidation of UO_2 . The presence of the oxides identified after the runs would be sufficient to reduce the UO_2 bulk thermal conductivity such that, if the oxidation were to take place in actual fuel elements, thermal performance of the fuel elements would be impaired. It was demonstrated that the escaping UO_2 can be collected in suitable traps and thus prevent contamination of the rest of the system. The specially designed holder was effective enough as to contain all of the escaping powder in a predetermined area. The UO_2 escaping from the specimens was oxidized to U_3O_8 as was determined by the X-ray analysis. The extent of oxidation was about 5%.

The arc fused UO_2 powder presently available is contaminated with nitrogen to concentrations of approximately 2000 ppm. The slight weight gain experienced with some samples was attributed to oxidation of the dioxide and that of the nitride. In the same tests, pellets suffered no weight loss but rather a slight gain which can be explained because of oxidation to U_3O_8 determined by X-ray and metallographic analysis. Although the available data is very limited, an extrapolation of the results on hand to actual fuel elements in a

reactor environment show that excessive down stream contamination is possible in cases where large splits occur in the cladding. On the other hand, fuel clad defects which are known to have occurred in stainless fuel elements are comparable to those studied in these tests. Under these conditions downstream contamination would be insignificant, especially if traps are provided in the fuel bundles or other parts of the piping. However, small defects may cause the UO_2 to oxidize, thus permitting the escape of the fission gasses in the coolant stream at higher rates than when the UO_2 is unoxidized. The greater fission gas escape rates may in time force an unscheduled reactor shutdown and thus limit the useful fuel lifetime. On the other hand, the higher fission gas release rates will permit an early detection of the clad failure and facilitate its subsequent identification and removal from the core, thus preventing excessive UO_2 erosion from the defect.

3.4.2 Conclusions of Erosion Results Reported in GEAP-3698

The experimental results obtained point to the following conclusions.

1. Uranium dioxide powder escaped through purposely defected stainless clad fuel elements during out of reactor simulated superheated steam dynamic flow conditions. Three out of six swaged specimens, containing arc fused powder in the unsintered condition, exhibited powder loss in the coolant stream.
2. In all specimens tested, oxidation of uranium dioxide was observed in varying amounts, the end products identified by X-ray and metallographic techniques as U_4O_9 and U_3O_8 . Because

the thermodynamics of reaction involving pure steam do not favor oxidation of UO_2 , it was concluded that the observed reactions were due to dissolved oxygen in the coolant.

3. Powder escaping was effectively collected by a suitable holder located in the test section of the loop. Five per cent of the powder collected in the holder was oxidized to U_3O_8 .
4. The degree of oxidation observed in all fuel specimens (both runs) is sufficient to cause substantial changes in the physical properties of UO_2 . Thus, when coolant contacts the UO_2 through a clad defect, eventual loss of heat producing capabilities of the fuel elements must be expected.
5. Leaching-out of aluminum, chromium and boron impurities was observed, as a result of coolant entry into the fuel. This points out that a certain degree of caution must be exercised in adding burnable poisons, such as boron, to UO_2 .

3.5 Trail Cable Capsule Experiments

3.5.1 Data Analysis

A continuation of the data analysis effort reported in the previous quarterly has resulted in a correlation of the experimental capsule outside diameter growth measurements that relate the observed UO_2

growth to the clad restraining force present during the irradiation. (See Figure 3.6) The correlation indicates a nearly inverse linear relationship between the UO_2 growth and the clad restraining force. It may be significant that of the data points falling appreciably below the correlation curve, all represent capsules with swaged cladding. It was recognized at the outset of the program that the swaged capsules might act differently under irradiation due to the fracturing of the pellets as a result of swaging.

The procedure for determining the "estimated UO_2 growth" plotted in Figure 3.6 was identical to that described in the last quarterly report. The values of the maximum clad restraining force were determined using a cladding stress equal to the yield stress of the cladding material in the relation $p = \frac{\sigma_t}{\gamma}$. The projection of an elastic modulus line up to the yield stress on the strain coordinate was taken as the amount of elastic springback following irradiation and used to estimate the UO_2 growth during irradiation. The calculations of the UO_2 growth and clad restraining force for each capsule are summarized in Table 3.1.

3.5.2 Significance of Results to Superheat Fuel Operation

An analysis has been made to predict the number of cycles to failure for superheat fuel element cladding subjected to repeated reversed cycles of plastic strain of the same magnitude as indicated by the above correlation of experimental results for unswaged capsules.

Two clad thicknesses were considered in the analysis - .016" and .028" and

FIGURE 3.6
 UO_2 DIAMETER INCREASE AT 350,000 BTU/hr-ft² VS.
 CLAD RESTRAINING FORCE

TABLE 3.1

ESTIMATED UO_2 GROWTH BASED ON OBSERVED CLAD GROWTH

<u>Capsule Number</u>	<u>Clad Material</u>	<u>Clad Thickness (in.)</u>	<u>Heat Flux (Btu/hr-ft²)</u>	<u>Observed Average Growth (mils)</u>	<u>Correct. to 350,000 Heat Flux (mils)</u>	<u>Correct. for Measured Initial Gap (mils)</u>	<u>Correct. for Springback (mils)</u>	<u>Correct. for Gap Change in Heating (mils)</u>	<u>Final Estimated UO_2 Growth (mils)</u>
TR-1	3003 AL	.049	160,000	6.06	/ 2.4	/ 4	0.75	/ 1.5	14.71
TR-2	3003 AL	.049	309,000	8.91	/ 0.5	/ 4	0.75	1.5	15.66
TR-3	3003 AL	.049	585,000	4.03	- 3.1	/ 9.0	0.75	1.5	12.88
TR-4	3003 AL	.049	350,000	6.0	---	0	0.75	1.5	8.25
TR-5	3003 AL	.028	350,000	4.2	---	/ 7.0	0.75	1.5	13.45
TR-6	3003 AL	.016	220,000	2.05	1.7	/ 10.0	0.75	1.5	16.00
TR-13	3003 AL	.049	426,000	2.06	1.0	/ 2.0	0.75	1.5	5.25
TR-15	304 S.S.	.028	>350,000	2.63	- 0 to .8	/ 1.5	1.75	0.8	5.88 to 6.68
TR-16	304 S.S.	.028	397,000	1.75	- 0.7	/ 1.0	1.75	0.8	4.60
TR-17	304 S.S.	.016	397,000	4.62	- 0.7	/ 0.0	1.75	0.8	6.47
TR-19	304 S.S.	.028	326,000	3.2	/ 0.2	~ / 3.9	1.75	0.8	9.9
TR-20	304 S.S.	.016	334,000	10.6	/ 0.2	0	1.75	0.8	13.41
TR-21	304 S.S.	.028	344,000	2.7	----	~ / 4.8	1.75	0.8	10.00

TABLE 3.1A
OBSERVED CLAD GROWTH

<u>Capsule Number</u>	<u>Clad Material</u>	<u>Clad Thickness (in.)</u>	<u>Heat Flux (Btu/hr-ft²)</u>	<u>Observed Average Growth (mils)</u>
TR-22	304 S.S.	.016	358,000	.0037
TR-23	304 S.S.	.028	---	---
TR-24	304 S.S.	.016	---	---
TR-25	3003 Al	.049	378,000	.0055
TR-26	3003 Al	.049	304,000	.0022
TR-27	3003 Al	.049	369,000	.0015
TR-28	3003 Al	.028	348,000	.0055
TR-29	3003 Al	.028	347,000	.0041
TR-30	304 S.S.	.049	300,500	.0000
TR-31	304 S.S.	.028	367,500	.0020
TR-32	304 S.S.	.016	360,500	.0018
TR-33	3003 Al	.028	329,000	.0002
TR-34	304 S.S.	.049	332,000	.0008
TR-35	304 S.S.	.049	344,000	.0000
TR-36	304 S.S.	.028	385,500	.0037
TR-37	304 S.S.	.028	376,000	.0002
TR-38	304 S.S.	.016	374,000	.0041
TR-39	304 S.S.	.016	343,000	.0022

* Correlation to UO₂ growth has not yet been calculated.

corrections were made to the data to account for the increase in the average UO_2 thermal expansion coefficient at superheat temperatures and for the retarding effect of the 1000 psia external pressure in the superheat environment. The usual Coffin lifetime relation was employed with the results shown in the Table below:

TABLE 3.2

PREDICTED CYCLES BEFORE CLAD FAILURE

<u>Clad Thickness</u> <u>Inches</u>	<u>Total Strain</u> <u>Range</u>	<u>Reversed Plastic</u> <u>Strain Range</u>	<u>Predicted Cycles</u> <u>Before Failures</u>
0.028	0.0203	0.0084	1835
0.016	0.0224	0.0101	1270

The predicted lifetimes are greater than the total number of full power reactor cycles (approximately 500) expected during the fuel life. Cycling tests of fuel elements in the E-SADE superheat facility are planned to confirm the above analysis.

3.5.3 Fuel Pellet Gas Release

As a result of the severe pellet gas release observed in capsule TR-3, the capsule fabrication procedure was modified to include a heating step to 800°F followed by six cycles of evacuation and backfilling with helium at temperature; then cooling to room temperature and seal welding without any further exposure to the atmosphere. The capsules design was also modified to provide a plenum at each end. Analysis of the filler gas in capsule TR-30 following irradiation indicates a factor of 6.8 reduction in the quantity of gas released. The contaminating gas was less than 25% by volume of the residual gas (helium) compared to 80% in capsule TR-3. At this concentration,

the internal pressure developed within the capsule during irradiation could not have been sufficient to cause clad growth. Further analysis is planned to determine whether the growth in the last fifteen capsules was significantly lower than those irradiated previously.

3.5.4 Experimental Results

During the last quarter fifteen new capsules were fabricated, irradiated, and growth measurements were completed. All of the capsules incorporated a gas plenum at each end as shown in Figure 3.7 and each received the pellet gas release reduction procedure prior to the final seal welding. All of the cladding employed in the capsules was fully annealed prior to use. The fabrication parameters incorporated in these capsules are listed in Table 3.1 and were selected to provide additional data in the higher range of clad strength for the correlation in Figure 3.6.

3.5.5 Future Experimental Work

Two annular capsules remain to be built and irradiated. These contain vibratory compacted, arc-fused UO_2 powder followed by swaging. One attempt has been made at filling these two capsules using aluminum cladding, but the resulting density of less than 75% of theoretical was considered unsatisfactory. One explanation for the low density is the possibility that the sharp UO_2 particles "hang up" on the very soft aluminum and prevent compaction. A second attempt will be made using stainless steel cladding.

In an effort to determine whether or not the observed clad growths are a unique characteristic of annular geometry fuel, a series of irradiations are planned of six rod type fuel capsules.

FUEL ELEMENT ASSEMBLY - SECTION
SCALE: FULL SIZE

FIGURE 3.7

A capsule holder has been built to permit irradiation of the 0.5" diameter, solid pellet filled capsules at 400,000 Btu/hr-ft² surface heat flux. (See Figure 3.8.) Three capsules will be irradiated at one time. The rods will all be stainless steel clad with various initial gaps and clad thickness. Fabrication of the capsules is nearly complete and irradiation will be performed during the coming quarter.

3.6 High Temperature Pressure Testing

A pressure vessel capable of operating at 5000 psi at 1000°F has been installed. The pressure vessel is equipped with an internal furnace and thereby requires a dry gas pressurizing media. The furnace has a usable diameter of $2\frac{1}{4}$ inches and a uniform temperature zone of 18 inches.

The following thermal cycling tests have been conducted under a 1000 psi argon pressure. The test specimen experiences a 1400°F to 300°F cycle every twelve hours. The specimen is at 1400°F for approximately four hours, and at 300°F for approximately two hours. The remaining time is spent coming up to temperature and cooling down from 1400°F.

A swaged .016 inch clad superheat fuel element has been cycled six times.

A free-standing .028 inch clad superheat fuel element has been cycled $6\frac{1}{2}$ times.

Difficulties have been encountered with short circuits in the furnace electrical leads entering the pressure vessel cover. The short circuits were attributed to insulation breakdown with time at temperature. The insulation has been replaced with a new type which has worked satisfactorily to date.

FIGURE 3.8
TRAIL CABLE FACILITY IRRADIATION CAPSULE HOLDER
FOR THREE 0.5" DIAMETER RODS.

4.0 TASK C - MATERIALS DEVELOPMENT

4.1 Materials Evaluation, Tensile Tests

A Cal-Tester tensile machine has been procured and modified for in-cell elevated temperature tensile tests on coupon-type specimens of dimensions shown in Figure 4.1. A die is available for cutting these specimens from sheet material. The tensile test machine is shown in Figure 4.2. The furnace and heaters on the specimen grips were completed after the photograph was taken. In Figure 4.3 is shown the specimen grip configuration. The photograph was made before the heaters were installed on the grips. These heaters compensate for heat loss through the pull rods from the tensile tester. The testing machine has been checked out and found to perform properly for in-cell tensile tests at room temperature. The machine is now being prepared for tests at elevated temperature.

Tensile specimens have been prepared from samples of stainless steel tubing procured through normal vendor channels. The following tensile specimens have been selected for irradiation in the SH-V instrument tube:

- A. Type 316, Transverse to tube axis: 5 specimens
- B. Type 316L, Transverse to tube axis: 5 specimens
- C. Type 347, Transverse to tube axis: 5 specimens
- D. Type 304, Transverse to tube axis: 10 specimens
- E. Type 304, Parallel to tube axis: 5 specimens

These specimens will be irradiated with the necessary neutron flux monitor wires needed to determine total exposure and subsequently tested at elevated temperature.

FIGURE 4.1 COUPON TENSILE TEST SPECIMEN

FIGURE 4.2 TENSILE TEST MACHINE

FIGURE 4.3 SPECIMEN GRIP CONFIGURATION

4.2 Strain Cycle Tests - In Reactor Test Equipment

The combined design calculations (thermal, stress, sizing, etc.) are approximately 90% complete. Detailed design of the bench machine is approximately 75% complete and detailed design of the capsules is approximately 50% complete. Concept design for the instrumentation lead-out system has been completed and submitted to all individuals concerned for comments. The instrumentation philosophy was decided upon, and a workable system laid out schematically. The bellows bids have been received and the bellows concept design of the low bidder meets with the specifications.

4.3 Cladding Evaluation

The purpose of this task is to obtain sufficient information concerning the material properties of austenitic stainless steel tubing in order that a realistic material specification for nuclear superheat tubing may be written.

Non-destructive testing methods, such as eddy current and ultrasonic, are available to inspect tubing to within 10% of the wall defect level at production capacities.

A program is being initiated to determine what level of tubing integrity is acceptable from a mechanical property viewpoint. The acceptable defect level will be investigated by conducting a series of 1050°F to 1350°F burst tests on natural and artificially defected superheat tubing. When the acceptable defect level has been firmly established, eddy current and ultrasonic test methods will be selected for inspection to this level at production capacities.

Tubing of the following geometries and AISI types has been ordered to be used for obtaining information necessary to write a material specification and to build superheat fuel element prototypes:

AISI Stainless Steel Types

304L, 316L	1.250" O.D. x .028"
	1.250" O.D. x .016"
	.750" O.D. x .028"
	.750" O.D. x .016"
	.625" O.D. x .028"
304	.750" O.D. x .028"
	.750" O.D. x .016"
316	.750" O.D. x .035"
347	.750" O.D. x .028"
	.750" O.D. x .016"

5.0 TASK D - EXPERIMENTAL PHYSICS

5.1 Superheat Critical Experiment Analysis

The superheat critical program at APED has been directed toward obtaining the basic physics information over a wide range of variables which is necessary to predict with confidence the nuclear behavior of superheat cores utilizing slightly enriched annular oxide fuel. The first portion of the superheat critical program has been devoted to the study of small uniformly spaced arrays of superheat fuel without controls present. The measurements included were:

1. Critical size
2. $\left(\frac{\partial \rho}{\partial H}\right)$ vs. H
3. Temperature coefficient
4. Void coefficient
5. Flux distributions

The fuel element consists of an annular ring of UO₂, 3.4 per cent enriched, and clad internally and externally by 28 mil SS-304 tubing. The clad fuel is surrounded by a 25 mil SS-304 process tube which bounds the external steam coolant passage. The UO₂ density is approximately 93 percent of theoretical including the effect of chips and air gaps between the pellets. The active fuel length is 48 inches.

Selective flooding of the inner and outer steam coolant passages with water gives rise to four flooding configurations, and insertion of an aluminum sleeve in the outer coolant passage gives rise to two additional configurations. The aluminum sleeve is introduced to increase heat transfer between the fuel and the moderator water when the outer coolant passage is not flooded and it is desired to work with a core having a significant excess reactivity. The maximum credible accident postulates failure of all the scram mechanisms and continued water addition until the core is fully reflected. The safety criterion being applied in that there be no fuel damage (clad melting) in the maximum credible accident.

The experimental results were interpreted in the following manner: First the experimental values were compared to the predictions of a calculational model which has demonstrated reasonable accuracy in predicting the nuclear behavior of boiling water lattices; then changes or renormalizations were evaluated that would be required in the model to obtain reasonable agreement for the superheat lattices. It should be mentioned here that this normalization process is frequently necessary when a significant change in lattice parameters is made and that it is often true that the model that gives the best reactivity agreement may contain significant compensating errors in the individual parameters.

The standard model used in this study was the synthesis of several individual models. It involved the following steps: The "L Factor," which is the ratio of the heterogeneous to homogeneous resonance integral for U-238, was calculated for an equivalent rod; the equivalent rod was defined as one having the same effective surface to mass ratio. The "L Factor" and the smeared atomic concentrations were then used as input to the MUFT-IV code⁽¹⁾. MUFT calculates the fast and epithermal group parameters. Independent calculations were then made of the resonance escape probability for U-238, using Hellstrand's surface to mass ratio dependence of the effective resonance integral⁽²⁾. The fast fission effect was calculated using Roderick's model. These parameters were then used to adjust the MUFT output. The adjusted MUFT parameters were

(1) Bohl, H. J., Gelbard, E. M., Ryan, G. H., "MUFT-4 - Fast Neutron Spectrum Code for the IBM-704," WAPD-TM-72, (July, 1957).

(2) Hellstrand, E., "Measurements of the Effective Resonance Integral in Uranium Metal and Oxide in Different Geometries," Journal of Applied Physics 28, (December, 1957).

(3) Roderick, H., "A Simple Approximate Method for Calculating the Fast Fission Factor," Unpublished memorandum, GE-APED (April, 1956).

then corrected for the Behren's⁽⁴⁾ enhanced leakage effect of voids and were used for groups one and two of a three-group model.

The thermal parameters were obtained by using the Wilkins' heavy gas model to determine the thermal spectrum weighting of the microscopic cross sections and by using the P-3 geometric flux weighting for the thermal flux depression in the fuel. A Behren's correction was also applied to the thermal group parameters. The presence of structural materials, controls, or other discontinuities was next taken into account by a two dimensional diffusion flux weighting calculation. Finally either a one or two dimensional diffusion calculation for the core and reflector was performed to yield the gross reactivity and flux distributions.

Several just critical arrays of uniformly spaced superheat elements have been determined. Three element pitches were included; 1.800, 1.900 and 2.000 inches. These pitches correspond to the water to fuel ratios 1.858, 2.324, and 2.816 for the voided element. In addition to the various flooding configurations, additional measurements at an even lower water to fuel ratio were obtained by inserting aluminum displacers in the moderator region of the 1.800 inch pitch array lowering the water to fuel ratio to 1.471.

The reactivity was calculated for each "just critical" array by the model previously described and compared to the experimental value of unity. Table 5.1 summarizes all the fully reflected critical arrays determined. It gives the element pitch and fuel configuration of each array. The critical size is given

⁽⁴⁾Behren, D. J., "The Effect of Holes in Reacting Materials on the Passage of Neutrons," Proc. Physical Society (London), A62, 607, (1949).

T A B L E 5.1

AEC SUPERHEAT CRITICALS - CRITICALITY COMPARISONS

EXPERIMENT VS. THEORY - FULLY REFLECTED CORES 20°C

Element Pitch (inches)	Configuration†† Inside-Outside	Actual W/F	Measured Critical Size No. of Elements	Array	Equivalent Cylindrical Size (cm)	T°C	Observed k_{eff} Fully Reflected	Calculated k_{eff} For This Array	k_{∞}	P_{28}	f	ϵ	Migration Area (cm ²)	Behren's Δk
1.8000 ↓	V - V	1.858	171	13 x 13 / 2	33.01	21.0	1.000829	1.022	1.283	0.863	0.767	1.034	56.5	-0.020
	V - Al*	1.858	172	13 x 13 / 3	33.09	21.4	1.000418	1.023	1.283	0.863	0.767	1.034	56.5	-0.020
	V - Al*	2.178	132	10 x 13 / 2	28.79	18.9	1.000215	1.020	1.275	0.866	0.766	1.029	45.6	-0.012
	F - V	2.178	131	10 x 13 / 1	28.68	19.0	1.000246	1.019	1.275	0.866	0.766	1.029	45.6	-0.012
	F - F	2.713	118	9 x 13 / 1	26.99	19.2	0.999934	1.037	1.244	0.891	0.730	1.025	33.4	0
	V - F	2.392	144	12 x 12	30.36	19.5	0.999709	1.038	1.253	0.893	0.729	1.029	40.0	-0.008
	V - Al† (Al† Displacers)	1.471	246	15 x 16 / 6	39.44	19.4	1.000399	1.044	1.232	0.832	0.763	1.040	47.1	-0.020
	↓	1.471	252	14 x 18	39.50	19.5	0.999620	1.044	1.232	0.832	0.763	1.040	47.1	-0.020
	↓	1.471	253	14 x 18 / 1	39.59	19.6	1.000472	1.045	1.232	0.832	0.763	1.040	47.1	-0.020
	V - Al*	2.324	155	11 x 14 / 1	32.87	18.6	0.999762	1.031	1.277	0.889	0.747	1.030	51.6	-0.019
1.9000 ↓	↓	2.324	156	11 x 14 / 2	32.98	18.7	1.000620	1.032	1.277	0.889	0.747	1.030	51.6	-0.019
	F - V	2.645	120	10 x 12	29.12	16.5	1.000915	1.028	1.268	0.889	0.746	1.025	43.0	-0.011
	F - F	3.180	120	10 x 12	29.12	18.1	0.999946	1.042	1.225	0.906	0.708	1.023	32.8	0
	V - F	2.859	147	12 x 12 / 3	32.29	16.8	0.999997	1.044	1.232	0.910	0.706	1.026	38.4	-0.007
2.0000 ↓	V - Al*	2.816	144	12 x 12	33.66	47.3	1.000044	1.042	1.269	0.913	0.722	1.026	48.2	-0.017
	F - F	3.672	135	11 x 12 / 3	32.56	18.7	1.000847	1.050	1.205	0.922	0.682	1.021	32.4	0
	↓	3.672	140	10 x 14	32.58	18.8	0.999985	1.050	1.205	0.922	0.682	1.021	32.4	0

* Al not included in calculation

† Al included in calculation

†† V = Void, F = Flooded, Al = Aluminum (present in coolant passages)

in terms of the total number of elements in the core loading, the actual rectangular array present, and an equivalent cylindrical size. In all cases, the fully reflected multiplication was very near unity. The calculated multiplication is given for comparison as well as the individual parameters entering the calculation. It can be seen that a significant over-prediction of reactivity was made using the standard model.

The reactivity discrepancy has been plotted as a function of element pitch (Figure 5.1). It can be seen that the discrepancies group into two categories, one for cases where the outer coolant annulus is void, and another for case where it is flooded. In addition, it can be seen that the discrepancies increase at higher element pitches. This means that inclusion of large water gaps between fuel surfaces is not accurately treated by the calculational model.

The possibility of a neutron poison being present in a significant concentration in the stainless steel clad or process tube has been ruled out by spectrographic analyses of samples from the three heats of stainless-304 that were used in the core fabrication. These analyses indicated that the concentration of neutron poisons (rare earths, boron, and others) was less than one part per million (boron equivalent) in the steel. In addition, the uranium enrichment of 7 of the 11 blends used in fabricating the core fuel was checked by mass spectrographic analyses and the average enrichment found to be 3.43 ± 0.03 weight percent U-235 compared to the specified 3.40 ± 0.07 weight percent U-235. This rules out low enrichment as a possible explanation of the reactivity over-predictions.

Four areas of uncertainty in the calculational model that are being considered are:

1. Spectral shifts	3. Thermal disadvantage factor
2. Core leakage	4. Resonance escape

REACTIVITY DISCREPANCIES USING STANDARD MODEL

FIG.5.1

Spectral shifts would be expected in this lattice due to the large element size and the large water gap between fuel surfaces. When the annular passage is flooded, there are over five mean free paths of water between fuel surfaces on the diagonal. This size is comparable to the water gaps between fuel bundles in a BWR where it has been found that using completely unhardened cross sections in the gap yields good agreement with experiment. To estimate the effect of spectral shifts, water greater than one mean free path from the fuel surface was assumed to see an unhardened spectrum. This increases water absorption and hence lowers reactivity. Moreover, it lowers reactivity more in those cases that have the greatest reactivity discrepancy. The reactivity discrepancies, with a spectral shift taken into account, have again been plotted as a function of element pitch. (Figure 5.2) It can be seen that a significant improvement is obtained although the discrepancies are still largest when the annulus is flooded at high element pitches.

The basis for the uncertainty in core leakage is the apparent contradiction between age measurements in pure water which presently indicate an age of about 27 cm^2 , and the fact that an age of 30.5 cm^2 seems to more adequately predict the leakage from small slightly enriched cores⁽⁵⁾. In this regard, the standard model using MUFT is normalized to the value of 27 cm^2 for the age in water. By assuming that the effective age is 30.5 cm^2 rather than 27 cm^2 , a significant decrease in the calculated reactivity, due to the enhanced leakage, is obtained. The reactivity discrepancy including the enhanced leakage as well as the spectral shift correction has been plotted as a function

(5) Deutsch, R. W., "Method for Analyzing Low-Enrichment Light-Water Cores," GNEC-133, (October, 1960).

REACTIVITY ERROR WITH SPECTRAL SHIFT INCLUDED

FIG 5.2

of element pitch. (Figure 5.3) Although there is still some scatter, the general agreement between experiment and theory is now quite good. It should be emphasized here that these modifications of the basic model have not been demonstrated to be the correct answer to the problem, and that further experimental work will be required to fully resolve the source of the discrepancies. It may well be that either resonance escape or thermal disadvantage factor calculations hold the key to the explanation.

The temperature and void coefficient results also indicate that the standard model is not accurately treating the water gaps between the fuel elements. In all cases, a decrease in moderator density was less negative a reactivity effect than the model predicted. Figure 5.4 shows the temperature coefficient for a "just critical" array of elements having 1.800 inch pitch in the void-void configuration versus temperature. Graphs 5.5 - 5.9 show the temperature coefficient for other fuel configurations. Flooding the annulus or increasing the element pitch caused an increase in the discrepancies. Any effect that increases neutron capture in the moderator would tend to bring experiment and theory into agreement. The previously considered spectral shift is such an effect. It has been estimated that this effect will explain half of the temperature coefficient discrepancies.

The void coefficients in the moderator again demonstrated that a decrease in moderator density was less negative of a reactivity effect than the standard model predicted. The moderator void coefficient has been plotted as a function of element pitch for those cases where the annulus is void (Figure 5.10) and for those cases where the annulus is flooded (Figure 5.11). Here again it has been estimated that the spectral shift effect would explain half of the discrepancies.

REACTIVITY ERROR WITH SPECTRAL SHIFT AND ENHANCED LEAKAGE

FIG.5.3

TEMPERATURE COEFFICIENT-SUPERHEAT CRITICAL-1.8000 INCH PITCH V-V

FIG. 5.4

TEMPERATURE COEFFICIENT-SUPERHEAT CRITICAL-1.8000 INCH PITCH F-F

FIG.5.5

TEMPERATURE COEFFICIENT-SUPERHEAT CRITICAL-1.9000 INCH PITCH V-V

FIG. 5.6

TEMPERATURE COEFFICIENT-SUPERHEAT CRITICAL-1.9000 INCH PITCH-F

FIG. 5.7

-09-

TEMPERATURE COEFFICIENT - SUPERHEAT CRITICAL 2.000 INCH PITCH F-F

FIG. 5.8

TEMPERATURE COEFFICIENT - SUPERHEAT CRITICAL - 1.8000 INCH PITCH V-AL-AL

FIG. 5.9

MODERATOR VOID COEFFICIENT V.S. ELEMENT PITCH

FIG. 5.10

MODERATOR VOID COEFFICIENT V.S. ELEMENT PITCH

FIG. 5.II

MODERATOR VOID COEFFICIENT V.S. ELEMENT PITCH

FIG. 5.12

Voiding in the flooded internal coolant passage is, however, well predicted by the standard model. The void coefficient in the flooded internal passage has been plotted as a function of element pitch (Figure 5.12). The solid lines represent the predictions of the standard model. It can be seen that the agreement between experiment and theory is excellent. This is not surprising since it was observed that there was no change in the reactivity discrepancy when the inner passage was flooded.

5.2 Summary of the Results

1. A consistent set of experimental gross reactivity measurements have been obtained for uniformly spaced arrays of annular slightly enriched oxidized fuel.
2. Correlation of these results with a calculational model normalized to boiling water lattices indicates some problems and suggests the necessity of renormalizing the model.
3. In order to find the true cause of the discrepancies, further experimental work will be required.
4. At present, it appears that some uncertainty would be involved in extrapolating from the uniform uncontrolled critical data to a large separate superheat reactor with water gaps, less leakage, controls, and other structural materials present.

Experiments which are being planned to reduce the uncertainties in the required physics information are:

1. Reactivity measurements for a clumped array with water gaps.
2. Controls measurements in the clumped array.
3. Void and temperature coefficients in the clumped array.

4. Flux distribution in the clumped array including intracell measurements to confirm the thermal disadvantage factor and to estimate the spectral shift present.
5. Conversion ratio (resonance escape) as a function of fuel location in the clumped array.

It is felt that these additional experiments will cover the areas of uncertainty and therefore provide a sufficiently wide base for establishing a normalized calculational system applicable to large superheat reactors.

6.0 TASK E- COOLANT CHEMISTRY

6.1 Out-of-Pile Superheated Steam Corrosion Test Loop

6.1.1 Preparation of Specimens

The heat transfer specimens consisted of three Type 304 welded cold drawn annealed and pickled tubes to ASTM Spec. A-249. They are $34\frac{1}{4}$ " long x $9/16$ " O.D. x .500 I.D. and were degreased and then pickled in a 20% HNO_3 bath operating at 130°F for $\frac{1}{2}$ hour. No other surface treatment was performed. The specimens were weighed on a 1000 gm. Mettler balance prior to insertion of the internal heaters.

The coupons exposed in the superheated steam corrosion holder were sheared from $1/16$ " thick sheet and all edges were machined. The coupon dimensions are $3" \times 3/4" \times 1/16"$. They were degreased and cleaned by same procedures as the tubes.

6.1.2 Method of Test

Dynamic tests were carried out in the CL-1 corrosion loop. The three tubes operated in series to superheat steam from 546°F .

Table 6.1 contains the loop operating parameters.

TABLE 6.1

Steam flow	524 lb./hr
Percent moisture in steam	1 w/o
Steam temperatures	
Inlet	545°F
Exit of first heater	653°F
Exit of second heater	844°F
Exit of third heater	1050°F
Steam inlet pressure	1000 psig
Superheater exit pressure	900 psig

Chemistry:	O ₂ (time weight average)	19.0 ppm
	H ₂ (time weight average)	2.6 ppm
	O ₂ max.	23 ppm
	O ₂ min.	14 ppm
	H ₂ max.	4 ppm
	H ₂ min.	2 ppm
Heat flux		180,000 Btu/hr-ft ²
Velocities		
	Inlet	110 ft./sec.
	Outlet first heater	145 ft./sec.
	Outlet second heater	190 ft./sec.
	Outlet third heater	220 ft./sec.

6.1.3 Evaluation of Results

The tubes and test coupons were weighed after removal from the loop. Descaling was performed using a solution of 20% NaOH, 3% KMnO₄, operating at 210°F. Nylon brushes and nylon wool were used to remove the rotted scale. The usual 10-20% HNO₃ second step was not used, since base metal appeared to be attacked on a test piece. Control pieces gave less than 2 mg/dm² weight loss of the base metal for the times (1½ hrs.) used.

6.1.4 Results

Table 6.2 contains the results of the two tests performed under dynamic heat transfer conditions. The exposure parameters were effectively the same for the two runs with the exception of gas control.

TABLE 6.2

1000 HRS. - O₂ AND H₂ NOT CONTROLLED

Calculated Metal Temperature OF	Total Corrosion mg/dm ²	mils	Metal-to-System Loss mg/dm ²
800 - 900	28	0.01	7
900 - 1100	191	0.10	23
1100 - 1300	256	0.13	3

950 HRS. - O₂ AND H₂ CONTROLLED

800 - 900	12	0.006	10
900 - 1100	143	0.07	9
1100 - 1300	452	0.23	120

In the calculations of metal loss to environment it was assumed that 72% of the scale was metal. This assumption was borne out by the corrosion coupons in the 20 ft./sec., 1050°F steam, where scale loss could be expected to be minimal.

The results of the coupon exposure in the 20 ft./sec., 1050°F steam at the exit of the superheater are as follows:

Exposure Time hrs.	Descaled wt. Loss mg/dm ²	Scale Retained mg/dm ²
450	188	250
1000	227	302
2465	276	371

These data are being submitted for statistical analysis and determination of the rate law.

Work has been started on the evaluation of possible metallurgical changes during long time exposure under an approximate 50°F Δ T across the cladding. The specimen with a temperature range of

1100°F - 1300°F shows increasing decarburization with temperature and an increasing thickness of the structure layer associated with it. Attempts are being made to analyze this layer and the scale.

6.1.5 Conclusions

1. The coupon data at 1050°F indicate the corrosion at that temperature is following the expected pattern of either a parabolic or log-law with a continually decreasing rate with time.
2. The single time point data from the heater with a metal temperature from 1100°F - 1300°F do not lend themselves to interpretation until the 2465-hour data have been obtained.
3. The data from the heaters with a metal temperature up to 1100°F, when combined with the 1050°F coupon data indicate good performance up to 1050°F.

6.2 E-SADE Loop Evaluations

Samples were taken of the E-SADE steam inlet and exit for radioactive gas analysis. Two samples were taken but the loop was not operated for a long enough period to establish a reference level of radioactivity for sound superheat fuel elements. It was determined, however, that the superheat fuel element was not contributing any measurable fission product activity to the SADE loop system.

7.0 TASK F - HEAT TRANSFER

7.1 "Once-Through" Boiling Heat Transfer

A topical report on this phase of the task is essentially completed. Some further drafting was required due to graphical presentation of additional data on departure from nucleate boiling.

One of the recommended film boiling correlations was presented in the last quarterly report, and should be corrected by addition of brackets as follows:

The right member of the equation on page 165 should read,

$$\left[\frac{DG}{\mu} \cdot \frac{1-x}{x} \right]^{0.853}$$

The summary to the forthcoming topical report, GEAP-3703, "Heat Transfer Coefficients with Annular Flow During Once-Through Boiling of Water to 100% Quality at 800, 1100, and 1400 psi," is included below as advanced information.

SUMMARY OF GEAP-3703 - "HEAT TRANSFER COEFFICIENTS WITH ANNULAR FLOW DURING ONCE-THROUGH BOILING OF WATER TO 100% QUALITY at 800, 1100 AND 1400 psi"

"Once-Through" boiling of water to 100 percent steam quality was successfully accomplished in a pre-boiler followed by 0.120 inch and 0.060 inch annular flow test sections, with electrically heated tubes 0.625 inch O.D. and 40 inches long. Sufficient data was obtained for preliminary design of "Once-Through" boilers in the regions of departure from nucleate boiling (DNB); transition to film boiling; and film boiling, at system pressures of 800, 1100, and 1400 psi. Mass velocity was varied from 0.75 to 1.9×10^6 lb/hr-ft² and heat flux from 190,000 to 700,000 Btu/hr-ft². Steam

qualities in the annular heater were in the range of 15-100 percent.

Results of DNB, transition and film boiling tests can be summarized as follows:

1. At medium and high steam qualities, operation could be sustained beyond so-called "burnout" design lines based on DNB data, because film boiling coefficients were high enough to maintain the heated wall temperature at acceptable values. For the flow conditions tested, high performance heat flux values of 300,000 to 400,000 Btu/hr-ft² can be used with heater temperature below 1000°F.
2. At low enthalpies, heat flux at DNB decreases sharply with increase of steam quality; reaches a minimum in the region of 50-60% steam quality at all three pressures investigated; and finally, increases in the higher steam quality regions.
3. The transition zone is characterized by sharply declining heat transfer coefficients and temperature oscillations. Reduced temperature oscillations are associated with increased pressure and points of reduced heat flux at DNB.
4. Film boiling heat transfer coefficients ranged from 800 to 2000 Btu/hr-ft² °F. The film boiling coefficient increases with flow rate. It approaches the Colburn equation value at 100 percent steam, decreases slightly as the steam quality is reduced, passes through a minimum, and increases as the water weight fraction exceeds 0.45 to 0.65.
5. Correlation of the film boiling data was achieved within $\pm$ 20 percent using a Colburn type equation modified by steam quality and steam void groups.

7.2 Heat Transfer to Superheated Steam

Manufacture and inspection of the Superheat Test Vessel auxiliary steam drum was completed.

Most of the piping and auxiliary equipment has been received and manufacturing is essentially complete.

Bids were awarded for installation of the equipment into the existing facilities. Final installation was scheduled for April, 1961.

8.0 TASK G - MECHANICAL DEVELOPMENT

8.1 Primary Separator Development

8.1.1 Introduction

Of several basic types of separators previously tested and reported in GEAP 3564, the multi-arc (radial vane) type showed the greatest promise of further development. The development of this type of separator for use inside a boiling water reactor is now continuing.

Basic criteria to be satisfied are:

1. Moisture carryover should be compatible with limitations imposed by the steam drying devices. It is indicated that the separator exit quality should not exceed 6%.
2. Vapor carryunder should not affect plant performance significantly. A carryunder of 0.2% or less by weight of vapor would be satisfactory.
3. The size per unit capacity should be as low as possible. Separators should be contained in the reactor vessel and be capable of separating at least the design mixture flow capacity of the reactor without significantly penalizing the reactor size for the internal separation process.

The approach to the development of the radial vane separator is summarized in Table 8.1.

8.1.2 Summary and Conclusions

Air-water tests of an adjustable two-vane radial separator assembly have been completed. Generally the results showed:

1. Less than 0.2% equivalent steam quality at the point of discharge from the vane ends for all vane arrangements and mixture flows

TABLE 8.1

Classification	Component	Test Medium	What can be Obtained	What Can't be Obtained
Type F	Modified 2-vane radial separator assembly	Ambient air-water mixture	<ol style="list-style-type: none"> 1. Flow configuration 2. Carryunder along vane exit 3. Effect on 1 & 2 of: <ol style="list-style-type: none"> a. Volume flow & air void fraction b. Vane radius c. Inlet opening width, length and height d. Vane spacing 	<ol style="list-style-type: none"> 1. Carryover between vanes 2. Effect of vane discharge to shroud on carryover and carryunder.
	2-vane radial separator incorp. best design features established from test above	1000 psig saturated steam-water mixture	<ol style="list-style-type: none"> 1. Minimum vane spacing to satisfy carryover limitations at maximum capacities for various inlet qualities. 	<ol style="list-style-type: none"> 1. Effect of vane discharge to shroud on carryover and carryunder.
Type G	Full circle radial vane separator	Ambient air-water mixture	<ol style="list-style-type: none"> 1. Effect on flow configuration & discharge on shroud of <ol style="list-style-type: none"> a. Water level b. Change in nozzle height c. Change in volume flow & air void fraction 2. Carryunder 	<ol style="list-style-type: none"> 1. Carryover 2. Effect of vane sizes and spacing 3. Effect of increased over-all size
		1000 psig saturated steam-water mixture	<ol style="list-style-type: none"> 1. Carryover 2. Carryunder 3. Capacity and inlet quality effects on 1 and 2 above 	<ol style="list-style-type: none"> 1. Visual observation 2. Effect of changes in sizes and spacing

during tests.

2. Of the two radii vanes, four and six inch, the former was observed to have the better size/capacity characteristics.
3. Flow rates were tested which were several times higher than any obtained in previous tests of the radial separator. However, these new tests do not indicate carryover or the effect of discharge from the vanes into the water circulating against the shroud. The principal value of the air-water tests is in guiding the efforts of the rest of the program.

Limited capacities of the test loops preclude obtaining data for full size separator assemblies.

Conclusions are:

1. Generally, the capacity increases with increased inlet void fraction (or equivalent steam quality) in air-water.
2. Reducing vane radius and vane spacing tend to reduce nozzle area and increase pressure drop.
3. Increased inlet void fraction reduces pressure drop for the same volumetric flow rate.
4. Nozzle dimensions have a significant effect on the flow configuration.
5. Water level above the lower end of the nozzle appears to disturb the flow pattern.

.1.3 Future Plans

1. A two-vane separator assembly with adjustable space between vanes is being designed and will be constructed for determining vane

steam discharge area effect on carryover in steam-water loop tests.

2. A full-circle radial-vane separator with integral shroud will be tested to show vane discharge effect on carryover and carryunder in steam-water loop tests. The effect of nozzle height on separator operating characteristics will be investigated with this unit.
3. Only gross effects of liquid velocity and steam volumetric flow can be obtained in the steam-water tests.

8.1.4 Discussion

Two basic type assemblies are being used to show the operating characteristics of radial vane-type steam separators: two vanes mounted on a plenum and a full circle of radial vanes mounted on a plenum and surrounded by a shroud. A summary of the tests to be performed with these assemblies and what information can and cannot be obtained from the tests is shown in Table 8.1.

The tests listed in the table will give considerable insight on work that should be performed in the future. Some consideration for future work should include determination of steam velocity effect on the moving liquid film and the associated acceleration field, model tests for determining size extrapolation limits, full size assemblies for tests in air-water on high pressure steam, water mixtures, improvement of the separation process by revisions or additions to the test units and improvement of nozzle designs.

8.2 Two-vane Separator Assembly

Shown in Figures 8.1 and 8.2 are the basic construction features of the two-vane separator for air-water tests. The assembly has been designed to

permit variation of vane radius, vane spacing and inlet nozzle dimensions. By observing the affects of the geometry on the flow configurations during tests, the design of the unit to be tested in steam-water can be established. For use internally in a boiling-water reactor it is important to reduce separator size to the minimum feasible for the operating criteria regarding carryover (6% by weight moisture, maximum) and carryunder (arbitrarily 0.2% by weight vapor, maximum). Test results shown in Figures 8.3 and 8.4 indicate the effect of equivalent steam quality and volume flow of mixture on flow configuration for changes in the geometry of the two-vane assembly. In classifying flow configuration ratings, rising of water droplets between vanes, gross spashing between vanes, water spraying from the mixture to the face of the opposite vane, breaking away of the mixture from the nozzle exit, and fogging between vanes, were considered undesirable. Lack of any of these characteristics was considered good; not more than one was considered fair and acceptable, and two or more considered poor and unacceptable. Flow configurations are shown in the photographs on Figure 8.5 where equivalent vane steam quality (X_S) and volume flow (Q_m) are:

W_w lb/hr	W_a lb/hr	X_S %	Q_m ft ³ /hr	Classification
400,000	1500	13.8	26,200	Poor
375,000	765	7.7	16,000	Fair
280,000	1250	16	21,000	Poor
350,000	720	7.7	15,000	Good

Observations indicated that increased inlet void fraction should improve separating characteristics for a given volume flow and that smaller radius vanes with narrower nozzles provide a more stable liquid film on the vane.

198E168
RADIAL VANES AIR-WATER TEST LOOP
PART 1 OF 2
1-79E168

PL. ISSUED

VIEW 'B'

PART NO.	DIM A
24	2.5
25	3
26	3

NOTE:
PARTS 24, 25, 26, 27, 28, 29, 30, 31 ARE TO BE WELDED TO VANES, PYS. 12, 13, 14, 24, 18, RESPECTIVELY WITH APPROX. 1/8" FILLET WELD.

ELEV. VIEW (TYPICAL)

Fig. 8.1

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(G1)
(G2)
(G3)

Fig. 8.2

FLOW CONFIGURATION OBSERVATIONS FOR VOLUME FLOW & EQUIVALENT STEAM QUALITY
AIR-WATER TEST LOOP

FIG. 8.3

FLOW CONFIGURATION OBSERVATIONS FOR VOLUME FLOW, & EQUIVALENT STEAM QUALITY

FIG.8.4

$W_w = 400,000 \text{ LB/HR}$
 $W_a = 1500 \text{ LB/HR}$

$W_w = 375,000$
 $W_a = 765 \text{ LB/HR}$

$W_w = 280,000 \text{ LB/HR}$
 $W_a = 1250 \text{ LB/HR}$

$W_w = 350,000 \text{ LB/HR}$
 $W_a = 720 \text{ LB/HR}$

Fig. 8.5

FLOW CONFIGURATION
 2-VANE RADIAL SEPARATOR A-W TESTS
 VANE RADIUS = 6", Q_L SPACING = 3",
 INLET NOZZLE = 1/2" x 11/16" x 18"

The limit of acceptable air-water flow rates for the 6-inch radius vanes appears to be about 7,000 ft³/hr per inch of plenum peripheral length at about 12% equivalent steam quality and 5,000 ft³/hr per inch at less than 9% equivalent qualities. Corresponding to this, the 4-inch radius vanes appear to be limited to about 8,000 ft³/hr per inch at 12% equivalent quality and about 6,500 ft³/hr per inch at qualities lower than 9%. Although use of the smaller vanes results in higher pressure drop, due to smaller required nozzle width, the size is of such importance that it is considered of much more value to continue tests with the smaller unit. For plenum peripheral applications the smaller vanes should give capacity per unit length of periphery that is approximately 130% of the 6-inch vane capacity at qualities of 9% or lower. Future tests will determine separation characteristics with steam-water mixtures.

8.3 Full Circle Radial Vane Separator

As shown in Table 8.1, a full circle radial vane separator surrounded by a shroud is planned for tests to determine how liquid discharge from the vanes onto the shroud effects carryover and carryunder. Figure 8.6 shows the design features of a full circle radial vane separator which is presently under construction. In addition to the effect of the vane discharge onto the shroud and the vortex on the shroud, the assembly is designed to show the effect of changes in nozzle height on separating capacity.

This assembly will first be tested in the air-water facility to show flow configuration, then in saturated steam water at 1000 psig.

8.4 STEAM DRYER DEVELOPMENT

8.4.1 Introduction

Previous development work was performed on a single stage dryer designed as an integral part of a superheat fuel assembly. Results were not considered satisfactory for superficial steam velocities in excess of 5 fps through the dryer when inlet moisture exceeded 1.0%. A subsequent design utilizing two stages of steam separation, the dryer of which is shown in a previous quarterly report, was constructed and tested in air-water with satisfactory results. Steam-water tests, however, proved unsatisfactory, possibly because of mechanical defects produced during drilling for pipe taps in the shell. No additional work has been performed on the two-stage dryer.

8.4.2 Summary and Conclusions

No significant information was obtained during steam-water tests of the two-stage dryer assembly.

Mechanical defects may have been responsible for the poor operating characteristics obtained during tests.

8.4.3 Recommendations

It is recommended that no additional work be performed on a superheat fuel assembly integral dryer. Large dryers exhibiting satisfactory drying characteristics are available for application inside a reactor vessel and can be used in lieu of the integral dryers with assurance regarding steam drying functions and at considerably lower total cost.

8.5 Seal Development

8.5.1 Introduction

Development of a device which will satisfactorily seal the joint between a superheat fuel rod and associated steam piping has been initiated. Studies of several types of seal configurating led to the selection of a temperature actuated seal for further study. This seal relays on the expansion of a low melting alloy as it changes phase from solid to liquid. The design of prototype seals for testing has been started.

8.5.2 Summary

During this period work was initiated on the conception, design, and development of a device which will seal the joints between fuel rods and transition piping in a superheat reactor.

The general criteria for this seal has been defined as:

1. The media to be sealed are saturated water to saturated steam at approximately 540°F; and saturated to superheated steam at about 950°F.
2. The water to saturated steam leakage rate is not to exceed 20 lb/hr under a pressure differential of 10 psi.
3. The saturated to superheated steam leakage rate is not to exceed 75 lb/hr under a pressure differential of 75 psi.
4. The seal must allow the transition piping to be removed from the fuel rods with a minimum of effort.

In order to develop the required seal, the following program was set up:

1. Conceive and make detailed drawings of possible seal configurations.

2. Evaluate various seal designs and select the most promising for a test program.
3. Set up and perform feasibility and hot leakage tests on the seal design selected in item two.

8.5.3 Selection of Seal Configurations

Several types of seal configurations were investigated for use in this application.

Metal "O" ring or flat gasket type seals were considered, but rejected on two counts.

1. Since large axial bolt loads are required to effect the seal, seizure and galling of their threaded portions would probably be encountered.
2. Not enough space is available for the number of bolts necessary for this type of seal.

A type of seal which is activated by a change in temperature was considered. This seal has the advantage of sealing by radial expansion, and thus does not require high axial loads. One of these seals relies on the differential thermal expansion between the stainless steel piping and an iron-nickel alloy of the Invar type. This concept was rejected, however, due to the close manufacturing tolerances required. It was also determined that failure of the Invar due to high hoop stress is likely if temperatures much above design values are reached.

Another temperature actuated seal which was considered makes use of the expansion of a low melting alloy as it makes the transition from

the solid to the liquid phase. This alloy is contained behind a thin metallic membrane which is caused to deform against the joint and create a seal. Preliminary design calculations indicate that a satisfactory seal may be made by using a 50% Pb-50% Sn solder as the liquid metal. This seal concept appears to be the most acceptable to date, and it was selected for further study.

8.6 E-SADE Process Tube Seals

The lower, replaceable process tube connection on the E-SADE nine element bundle was mocked up for autoclave leak tests. (See Figure 8.7)

Work on the E-SADE process tube seals has been completed during this reporting period. Of the three sealing surface configurations tried, only the one with concentric grooves in the plug sealing face had zero leakage at all points. The drawings are now being changed to incorporate this surface. The minimum torque required to effect a room-temperature seal was found to be 150 ft. lb., and all seals were tested at this tightness. No galling or seizing was experienced during these tests.

8.7 Thread Galling Tests

In order to find a solution to the recent thread galling incidents in the SADE loop of the VBWR, a test program was initiated. Thread specimens size 5/8-11 will be assembled, aged in 545°F water for 300 hours, and checked for galling. It is hoped to find a thread form, material, and surface treatment combination from the following list which will not gall:

BOTTOM PROCESS TUBE CLOSURE FOR E-SADE

Fig. 8.7

<u>Thread Form</u>	<u>Material</u>	<u>Surface Treatment</u>
60° V	304	chrome plating
60° Stub	17-4 Ph	silver plating
29° Acme	A-286	nitriding
29° Stub Acme	AM-350	
	Al bronze	
	S - monel	

During this reporting period materials were accumulated and test samples fabricated for autoclave testing.

9.0 TASK H - SADE and E-SADE

9.1 SADE Operation with SH-4B

Final functional check out of the SADE loop was accomplished at the start of the report period. Condenser pressure during initial testing was higher than experienced during operations prior to shut-down for VBWR core modification. By change of the dilution air connection to the condenser it was possible to duplicate previous loop operating conditions. Apparently the dilution air had been partially blanketing the condenser, resulting in the higher condenser pressures experienced during the check-out period.

Fuel element SH-4B (Figure 9.1) was installed in the loop after satisfactory loop operation was achieved. Test of this element was initiated on January 15th and it was planned to continue the tests for approximately 1 month. However, because of recent difficulties experienced with 17-4 Ph material under certain heat-treated conditions, operation of the VBWR was subject to inspection of the critical applications of this material, particularly for the application for control rod shafts. During the inspection after the reactor shut-down on January 29th, it was discovered that cracking of these shafts was occurring and the decision was made to stop further operations of the reactor until modifications could be made to the reactor applications of this material.

A summary of the test conditions up to the time of reactor shut-down is shown on Table 9.1. (Occasional brief power changes are not included in this tabulation.) Design conditions for SH-4B are shown on Table 9.2.

TABLE 9.1

Date	VBWR Power	Superheat Temp	Approx. Days Tested	SH-4B Power*	Clad Temp*
1/15/61	Start-up - 10 am		$\frac{1}{2}$		
1/15/61	8-11 MW	600 to 635°F	2		
1/17/61	9-12 MW	600 to 652°F	4		
1/21/61	17 MW	700°F	4	55 kw	850°F
1/26/61	19-20 MW	740°F	3	65 kw	910°F
1/29/61	Shut-down - 8 am				

* Superheat power and clad temperatures determined by interpolation from a plot of operating conditions based upon a series of SHEBA code runs.

Maximum superheat temperature reached during test was 760°F, corresponding to a clad temperature of 930°F.

TABLE 9.2

DESIGN CONDITIONS FOR SH-4B

Power	100 kw
Fuel enrichment, nominal	4.5 U ²³⁵
Fuel enrichment, measured	4.55% U ²³⁵
Steam pressure	1000 psia
Active fuel length	36"
Outer clad, outside diameter	1.25"
Inner clad, outside diameter	0.75"
Clad thickness, outer and inner	0.028"
Clad material (weld-drawn tubing)	304 S.S.
Instrument tube, outside diameter	0.50"
Fuel (UO ₂) % theoretical density	95%
Total fuel loading (UO ₂)	3.76 kg
Maximum clad temperature (STEADY)	1200°F
Maximum superheat exit temp. (STEADY)	925°F
Maximum heat flux (STEADY)	332,000 Btu/hr-ft ²
Minimum steam flow rate (STEADY)	1032 lbs/hr
Maximum fuel temperature (STEADY)	3030°F
Maximum clad temperature (Transient)	1292°F
Maximum superheat exit temp. (Transient)	980°F
Maximum heat flux (Transient)	330,000 Btu/hr-ft ²
Minimum steam flow rate (Transient)	915 lbs/hr
Maximum fuel temperature (Transient)	3116°F

It is planned to resume testing of this element to complete the original test schedule, as soon as operation of VBWR is resumed, if it is possible to achieve a successful re-installation of a radioactive fuel element in this loop. It is planned that the test will follow through to the design conditions for this element since the previous testing was interrupted prior to reaching this objective.

9.2 Post Irradiation Examination of SH-4B

During the period of reactor shut-down, the element was transferred to the RML for inspection to determine the exact condition of the element at this time. Some slight changes in outside diameter and possible bowing were noted. The instrument tube could not be withdrawn during this check but it is believed that this resulted from too tight a fit of this tube when initially installed.

Fuel element SH-4B was visually examined, photographed, dimensioned, and gamma scanned at the VAL RML facilities.

9.2.1 Visual Examination and Photography

An orange-brown corrosion film was observed on the middle and lower portion of the fuel region cladding. The film appeared to be heaviest around flow discontinuities such as the middle spacing fins and bottom spacing fins. Some orange-brown corrosion was also observed on the tubing just below the flange at the top of the element. Photographs were taken.

9.2.2 Dimensional Measurement

Locations where measurements were taken and the fuel element orientation are shown in Figure 9.1. The same types of gages were

used for both pre and post-examination.

1. Outside Diameter - O.D. measurements were taken every 4" along the fuel section (see Figure 9.1). The lower section of the fueled length exhibited the most change, with a maximum of 0.007" decrease (Table 9.3).
2. Length Measurement - Results of length measurements are contained in Table 9.4 (see Figure 9.1). An 0.008" decrease was the largest change. Clad temperature was measured at 92°F, so there is no appreciable thermal expansion as a result of decay gamma activity heating the cladding.
3. Inside Diameter Measurement - The pre-irradiation examination values are shown in Table 9.5. No post-irradiation values could be taken since the instrument tube was wedged inside the element. Although the tube fit snugly prior to irradiation, there must be some deformation of the inner clad also contributing to the wedging.
4. Bow Measurements - Bow measurements were performed by placing three one-inch v-blocks on a surface plate, supporting the fuel on the v-blocks, and reading the height of the top clad surface with a dial gage. The element was supported on the plate identically for pre and post measurement. Results are listed in Table 9.6. Only the 0° orientation value change is significant, and this change is relatively small for a 40" long section.
5. Spacer Fins - Measurements over the spacer fins gave dimensions of 1.418" to 1.428". The design value for this dimension is 1.430, but since this measurement, prior to irradiation, had not been recorded for the element, this may or may not have changed

during irradiation.

6. Gamma Scan - The fuel element was gamma scanned several times. For convenience in reference, duplicate photographs were taken of the scan trace.

Every pellet-to-pellet gap shows clearly in the scan. The center inert ZrO_2 spacer is especially prominent as the large dip in the center of the trace. Flux peaking occurred in the fuel adjacent to the center and the end ZrO_2 spacers.

A Co 60 length calibration assembly, to determine distance on a scan, was also run several times. Quantitative results are given below:

<u>Component</u>	<u>Pre</u>	<u>Post</u>	<u>Δ</u>
Bottom ZrO_2 spacer			
Bottom fuel length (23 pellets)		17.85"	
Center ZrO_2 spacer		1.05"	
Top fuel length (22 pellets)		17.12"	
Top ZrO_2 spacer		0.62"	
Total fuel length		34.97	

The axial position monitor was discernible in a gamma scan trace taken at a much higher magnification of the γ -Activity ordinate. No accurate measurements were possible since there are many ways in which the scan can be interpreted, however, it is encouraging that even low clad exposures give rise to measurable Co 60 activity. Interpretation of the fuel movement will be possible

only at the conclusion of irradiation when the monitor region can be potted in Hysol and sectioned.

However, it would be worthwhile to electroplate pure cobalt on reference surfaces and then these surfaces only will activate strongly and show up as strong line sources in a gamma scan. Quantitative interpretation should then be very straightforward.

TABLE 9.3
OUTSIDE DIAMETER MEASUREMENTS

Position	0° - 180°			90° - 270°		
	Pre	Post		Pre	Post	
2"	1.253"	1.251	-0.002	1.253	1.252	-0.001
6"	1.251"	1.250	-0.001	1.253	1.250	-0.003
10"	1.253"	1.252	-0.001	1.253	1.246	-0.007
14"	1.253"	1.251	-0.002	1.253	1.249	-0.004
18"	1.253"	1.254	+0.001	1.257*	1.255*	-0.002*
22"	1.253"	1.251	-0.002	1.253	1.252	-0.001
26"	1.253"	1.253	0.000	1.253	1.252	-0.001
30"	1.253"	1.253	0.000	1.252	1.252	0.000
34"	1.253"	1.253	0.000	1.251	1.250	-0.001
36"	1.253"	1.253	0.000	1.252	1.251	-0.001

* near middle spacing fin weld

TABLE 9.4

LENGTH MEASUREMENTS BETWEEN WELD SHOULDERS OF FUELED SECTIONS

Position	Pre	Post	
0°	37.662" av	37.654 av	-0.008" clad temp.
90°	37.658 av	37.652 av	-0.006" of 92°F
180°	37.658 av	37.658 av	0.000"
270°	37.658 av	37.657 av	-0.001"

TABLE 9.5

INSIDE DIAMETER MEASUREMENTS

Position	Pre	Post
2"	0.694 av	
6"	0.694 av	
10"	0.694 av	
14"	0.694 av	Instrument tube wedged.
18"	0.694 av	
22"	0.694 av	No. I.D.
26"	0.694 av	measurements
30"	0.694 av	possible
34"	0.694 av	
36"	0.694 av	

TABLE 9.6

BOW MEASUREMENTS - MAXIMUMS

Position	Pre	Post	
0	+0.007"	+0.027"	0.020"
90	+0.008"	+0.009"	0.001"
180	+0.007"	+0.013"	0.006"
270	+0.006"	+0.008"	0.002"

+ = bow outward

9.3 SADE Delayed Neutron Counter

The delayed neutron leak detection system has been installed, but because the valve which was obtained to protect the delay tank was unsuitable, the system cannot be operated yet. This system is a fuel rupture detection system which should be more positive and faster in response than the present detection system.

The present system will indicate high radiation when there is a rupture in VBWR as well as when there is a defective superheat element. The system by itself does not identify the source of activity. The new system is based upon delayed neutron activity and should provide faster determination of SADE fuel ruptures and make their determination more positive

The following description and explanation applies to the delayed neutron system of detection shown on Figure 9.2. The rupture determination will be accomplished by monitoring delayed neutrons emitted by isotopes which are daughter products of the fuel fission process, primarily Br^{87} and I^{137} . These isotopes are soluble in water, therefore, Br^{87} and I^{137} coming from VBWR fuel ruptures should remain in solution with the boiling water.

114B5419
CONT ON SHEET SH NO.

TITLE
MONITORING SYSTEM FOR DELAYED NEUTRONS
FIRST MADE FOR SADE

Fig. 9.2

Br^{87} and I^{137} coming from a fuel rupture in SADE will be carried through the superheat loop to the desuperheater. If a monitor is placed somewhere between the superheat steam outlet and the desuperheater to measure delayed neutrons, due to the natural filtering of Br^{87} and I^{137} in VEUR, it will primarily measure activity coming from SADE.

There is another primary delayed neutron emitter - N^{17} . It is distinct from Br^{87} and I^{137} in that it is present all the time during reactor operation; whereas Br^{87} and I^{137} are only present at the time of a fuel rupture.

A monitor which counts delayed neutrons will not separate those emitted by N^{17} from the ones emitted by Br^{87} and I^{137} . The neutrons emitted by N^{17} must somehow be made negligible compared to those emitted by Br^{87} and I^{137} . The half-life of Br^{87} and I^{137} are 56 sec. and 22 sec., respectively. The half-life of N^{17} is 4.14 sec. If the steam sample can be delayed long enough, that N^{17} has decayed to the point that its neutron emission rate is small relative to the emission rate of Br^{87} and I^{137} after the same delay period, the monitor will be much more sensitive to the Br^{87} and I^{137} . The N^{17} will give a small background signal, while the Br^{87} and I^{137} present from a superheat fuel rupture will give a strong signal. The signal the monitor gives under normal operation will be small; when there is a fuel rupture the signal will show a significant increase which will indicate the fuel rupture.

There will obviously be a time at which the increase in the signal due to the presence of Br^{87} and I^{137} over the continuous signal produced by N^{17} will be a maximum. This point will be when the relative decay between the I^{137} , Br^{87}

and N^{17} is such, that the ratio of the number of neutrons emitted by I^{137} and Br^{87} to the number of neutrons emitted by N^{17} is maximum. It has been estimated this time will be somewhere between 30 sec. and 60 sec. The easiest way to determine the optimum delay time is by experiment. The system shown in the attached diagram is designed to determine this time, and also to determine the effectiveness of the delayed neutron system as a rupture detector for a superheated steam reactor system. Reference to the attached drawing, 114B5419, Figure 9.2, will be helpful in the following discussion. The SADE loop has a steam by-pass line with an orifice, RO-1102, which allows 200#/hr steam flow when the up-stream temperature and pressure are 625°F and 1000 psi. An orifice, RO-1103, can be installed down stream of orifice RO-1102, to raise the pressure between the two to 100 psi. The proposed delay system is to have its entrance connected between these two orifices and will terminate in the desuperheater. Between the entrance and exit there will be a shut-off valve V-1143, delay tank, drain valve LCV-1105, filter S-1108, and an orifice RO-1104 and metering valve V-1145 connected in parallel. Shut-off valve V-1144, will be connected in series with the metering valve V-1145, because the metering valve is not to be used as a shut-off valve. The two orifices which are to be installed (one down stream of existing orifice and one in the delay system) are to have a combined area such that the pressure at the entrance to the system will be 100 psi when steam is flowing through only these two orifices. The orifice RO-1104 will be so sized, that it will cause a 60 sec. delay when shutoff valve, V-1144, is closed. A calculation for this situation has been made, and the diameter of the orifice RO-1103 must be 0.226 inches, while orifice RO-1104 must have a diameter of .0551 inches. The above sizes were determined assuming a constant enthalpy expansion

across orifice RO-1102, this assumption makes the steam between the orifices 470°F at 100 psi. The tank has a volume of one cubic foot and is baffled so there will be no direct flow through it; when the delay time is set for 60 sec. (shut-off valve V-1144 closed) steam flow will be 11.12#/hr. The other extreme will be when shut-off valve V-1144 and metering valve V-1145 are wide open; at this condition the delay time will be approximately 29.8 sec. and the pressure between the orifices will have dropped to 94.38 psi. The flow will then be 21.2#/hr through the delay tank. Flow rates and delay times between these two extremes will be obtained by adjusting metering valve V-1145.

9.4 SADE TEMPERATURE ISOLATION CIRCUIT

Four of the SADE loop critical temperatures are recorded on single pen strip chart potentiometer recorders. The ranges of the recorders make a very accurate reading of temperatures difficult and subject to personnel interpolation. A multi-input precision indicator is associated with the over-all temperature system. This instrument provides a means of obtaining more accurate indicated and readout temperatures.

A problem, inter-action between two potentiometers devices, occurs when a single sensor (thermocouple) is loaded by two measuring instruments. As mentioned above, the need for a parallel measuring type of instrumentation is very important to perform heat balances and system operational performance limit calculations.

An isolation circuit was developed which permits the single thermocouple input to be alternately associated with the precision indicator and the individual single pen recorders. Eight channels of the isolation circuitry

were provided for the SADE loop. Four of these channels are in service and ready for loop operation. The remaining four channels are spares.

Principal of operation of the isolation system is a single pole double throw convertor (chopper) which alternately connects the measuring thermocouple, first to the recorder, and then to the precision indicator. A 100 micro farad capacitor provides an electrical potential storage signal for the recorder and the multi-point indicator input. Figure 9.3 gives the schematic for the isolation circuit.

9.5 NUSU Fuel Element Irradiation

Extensive studies and design have been completed to determine how the NUSU superheat element could be irradiated in the SADE test loop. (Because of the complexity of the E-SADE fuel assembly, it had been decided that the only practical method of successfully irradiating the NUSU element would be in the single element SADE system). Figure 9.4 shows how the NUSU fuel element must be fabricated in order to install it in the SADE system.

In a letter dated February 3, 1961, from S. E. Turner of GNEC to A. E. Mravca of the COO of the AEC, the desired irradiation conditions for the NUSU element were stipulated as follows: "GNEC requested that G.E. attempt to set the fuel enrichment and steam flow such that the maximum heat flux to the steam would be 340,000 Btu/hr-ft² for a maximum clad temperature of 1200°F."

To meet these operating conditions, approximately 150 KW must be generated in the NUSU element. Two particular operational problems result from this power level for this element.

1. Preliminary Physics calculations indicate that the fuel enrichment

SADE TEMPERATURE ISOLATION CIRCUIT

FIG. 9.3

6120850	
TITLE ASSEMBLY	
NCSU FUEL ELEMENT	
FIRST MADE FOR NUSU	
FLF 1048634	
NAME	DRAWING NO. DESCRIPTION MATERIAL WEIGHT
PROCESS TUBE FLANGE 198E230G1	
ADAPTER 145A556RPI	
P.E. ASSEMBLY G.W.E.C. DWG. NO. 057-90-192

Fig. 9.4

might exceed 4.5%, the maximum enrichment implied for the SADE loop (pg. 229, SG-VAL 2, 3rd edition).

2. The maximum heat flux to the water would be 400,000 Btu/hr-ft², and this heat flux exceeds the VBWR heat flux limitation imposed on new or experimental boiling water fuel elements of 250,000 Btu/hr-ft² (pg. 147, SG-VAL 2, 3rd edition).

The first problem can be solved by obtaining AEC approval to operate an element whose enrichment exceeds 4.5%. The second problem can be solved by using the procedure outlined on pg. 147 of SG-VAL 2.

The following table summarizes the important fuel element design conditions for rated operation and for the first step VBWR operation at 250,000 Btu/hr-ft² water side heat flux.

Element power	150 kw	94 kw
Maximum heat flux, steam side	340,000 Btu/hr-ft ²	212,000 Btu/hr-ft ²
Maximum heat flux, water side	400,000 Btu/hr-ft ²	250,000 Btu/hr-ft ²
Minimum steam flow rate (STEADY)	1100 lbs/hr	610 lbs/hr
Axial peaking factor	1.42	1.42
Maximum clad temperature (STEADY)	1200°F	1200°F
Steam temperature	765°F	785°F

Study of the modification of SADE as necessary to satisfactorily irradiate the NUSU element also showed the necessity of controlling the boiling water side conditions. This showed that it would be necessary to provide a flow tube around the element to simulate the NUSU application and to prevent excessive by-pass leakage in the VBWR core.

Four sizes of tubes were investigated at two different powers. The inside

diameter of these tubes was 2.0, 1.95, 1.9 and 1.85 inches. The NUSU element was assumed to be at total powers of 150 and 125 KW, of which 95 and 79 KW respectively were dissipated in the cooling water. The physical conditions for the element were chosen to correspond to a total core power of 36 MW and a core flow of approximately 10,000 gpm. The inlet enthalpy of the coolant was 535.3 Btu/lb and the total pressure drop through the core was assumed to be 4.0 #/in².

The following table gives the results for the 8 cases which were run on the computer.

I Power = 150 KW

<u>I.D. of tube</u>	<u>W #/hr</u>	<u>Steam Prod. #/hr</u>	<u>Exit Quality %</u>	<u>Avg. Void Fraction</u>	<u>Min B.O. Ratio</u>
2.00	22,740	163	0.71	0.027	3.18
1.95	18,540	227	1.22	0.053	3.06
1.90	14,610	285	1.96	0.089	2.90
1.85	11,070	339	3.06	0.140	2.72

II Power = 125 KW

2.00	23,710	64	0.27	0.006	3.91
1.95	19,450	127	0.65	0.024	3.77
1.90	15,440	188	1.22	0.053	3.59
1.85	11,790	243	2.06	0.094	3.38

On the basis of 150 KW element power, and limitations of VBWR operating license, a preliminary selection of 1.90 I.D. flow tube for the NUSU element has been made.

9.6 E-SADE Design and Installation

Work has continued rapidly on this project during this period. The many problems being encountered are being resolved as expeditiously as possible. The major problems and action being taken in each case are discussed in the specific sections following.

9.6.1 E-SADE External System

A mock-up of the main condenser was employed to determine that the location of the condenser would be satisfactory, and that connections and nozzles were practical. The vendor drawings for this equipment have been approved for construction on the basis of this mock-up.

(It was necessary to change location of one nozzle to provide accessibility to this condenser).

A detailed schedule for design and installation of the reactor external system required for the E-SADE loop is being prepared. This schedule is being based on a premise of achieving a minimum interruption of VBWR operations while installing this system, and still achieving the earliest practical installation date. Thus, extra effort is being devoted to providing the information necessary (drawings) to permit pre-fabrication of equipment with a minimum of time involved during reactor shut-down for the installation of this equipment.

The Piping and Instrumentation drawing is now being finalized.

Since this drawing represents the basic philosophy of operations convenience, reactor safety, and test requirements, it has required the co-ordination and co-operation of several functional groups to arrive at an acceptable arrangement. Most points in question have been satisfactorily resolved so that it will be possible to issue this drawing for the basic system shortly.

Difficulties were encountered with the plate material obtained to provide the additional penetrations of the VBWR enclosure. The

material obtained failed to qualify for the low temperature ductility required, and so it was not possible to complete this modification during the period while the reactor was shut-down to correct the 17-4 Ph material difficulties. Steel plate has now been ordered direct from the mill and it is expected that a satisfactory patch assembly will be ready for installation early in May. The new penetrations provide inlet and outlet for the E-SADE condenser cooling water and the necessary electrical, instrumentation, and control circuits for this system.

9.6.2 E-SADE Loop Safeguards

Work has progressed during the reporting period to the point where the analytical portion is approximately 50 percent complete.

A representative safeguards element was chosen and the design conditions of this element are as follows:

Power per element	100 kw
Fuel enrichment	variable
Steam pressure	1000 psia
Active fuel length	36"
Outer clad, outside diameter	1.25"
Inner clad, outside diameter	0.75"
Clad thickness, outer and inner	0.028"
Clad material (weld-drawn tubing)	304 S.S.
Instrument tube, outside diameter	0.5000"
Instrument tube thickness	0.028"
Fuel (UO ₂) % theoretical density	95%
Total fuel loading (UO ₂)	4.10 kg
Process tube, outside diameter	1.562"
Process tube thickness	0.060"
Max. clad temperature (STEADY)	1250°F
Max. superheat exit temp. (STEADY)	910°F
Maximum heat flux (STEADY)	344,000 Btu/hr-ft ²
Min. steam flow rate per element (STEADY)	1032 lbs/hr
Maximum fuel temperature (STEADY)	3160°F
VBWR Power	40 MW
Axial peak to average power	1.43

The following presentation summarizes the analytical work that has been completed in the accident analysis domain.

Flooding - Unflooding

Conservative physics calculations indicate that flooding all nine fuel elements when the reactor is hot (545°F) will produce a positive reactivity addition of 0.25% Δk . This is approximately 36 cents, and is the worst possible case, since flooding at any other reactor water temperature produces a smaller positive reactivity addition than the above value.

Normal Reactor Scram

If the reactor scrams for some reason not involving an important failure (e.g. due to instrument failure), the superheat test loop and the test section will be undamaged. Figures 9.5 and 9.6 show the power versus time curve used for all calculations in this section (unless otherwise indicated). This curve and the TIGER digital computer code were used to evaluate the maximum clad temperature during a normal reactor scram. The results indicate that the maximum clad temperature during the transient will not exceed 1250°F until the steam flow is shut off completely, and then it will rise to a maximum of 1350°F.

Loss of All Electrical Power Supplied to the Site

a) Elements Operating at Full Power:

Back-up air supply will keep the bypass steam valve open long enough that this accident will be essentially the same as a normal reactor scram as applies to maximum clad temperature.

FIGURE 9-5

WPS 9-57.61

POWER VS TIME FOR \$5 SCRAM FIRST 45 MINUTES

FIGURE 9-6

b) During Start-Up Operations:

1. Elements are steam cooled - This presents no problem since the elements will be operating at low powers and high steam flow.
2. Elements are water cooled - The cooling water pump will be knocked out, but the combination of flow coast down and high heat capacity of the water will dissipate the relatively high short-time heat. The long term decay heat is so low that the element could dissipate it even if stagnant steam filled the coolant passages.

Loss of Cooling Water Flow During Start-Up

This can be caused by shaft seizure or local power failure.

Whichever the mechanism, it is identical to the accident discussed above.

Loss of Cooling Water Flow to the Condenser

Loss of cooling water flow to the test loop condenser can occur if power to the pump is lost, or if the pump shaft seizes. This will cause a reactor scram and the steam will be discharged to the fuel storage pit through the condenser relief valve.

E-SADE Loop Pipe Rupture

Since the steam is taken directly from the reactor dome, the only credible pipe rupture is considered as being downstream of the fuel bundle. This means that the loop will not lose steam flow due to a pipe rupture.

Reactor Cold Water Accidents When E-SADE Loop Is Steam Cooled

The percent flow will always lead the percent power by a large margin in the E-SADE loop. Therefore, since the fuel time constant is large (approximately 20 seconds), the starting peak fuel temperature is low, and the amount of heat that can be picked up by the steam is large, a reactor cold water accident will not damage the E-SADE loop fuel or clad.

9.6.3 Reactor Internals

Drawings of a nine element assembly were completed and released for comment. Because of space limitations, this arrangement provided that spacers for the fuel element have to be a part of the process tube and cannot be attached to the fuel element. The nine elements were removable through a threaded end plug to provide flexibility of operation. After a detailed review of this arrangement, it was decided that this limitation on spacer locations should be avoided on the first assembly provided. A new arrangement is being made where only the center of the nine elements will be removable through a threaded plug. The remaining elements will be welded into the assembly. This change completely revises the process tube drawings and the lower ends of the assembly. Detailed drawings of these changes are now being prepared.

A test program to evaluate the E-SADE process tube seals is nearly complete. Seals were tested in the Engineering Development autoclaves. Leakage rates were measured at 50-70 psi from the outside of the seal at both 300 and 545°F water environment. Three types of sealing surfaces were tested; plain, spiral groove, and con-

centric groove on plug only. Only the concentric grooved seal had zero leakage at all points. The other seals leaked at rates from 35 to 70 cc/hr. At present the spiral grooved seal is being retested to check the repeatability of its performance. The minimum torque required to effect sealing at room temperature was found to be 150 ft.-lb., and all seals were tested at this tightness. No thread galling or assembly difficulties were experienced during these tests.

10.0 TASK J - MIXED SPECTRUM SUPERHEAT REACTOR DESIGN STUDY

During the quarter, work on the Mixed Spectrum Superheater consisted of:

- (a) Completing minor points on the reference 300 MW plant design for the steam cooled reactor study, and (b) Further evaluation of flooding effects.

The work on the flooding problem and the use of epithermal poisons involved the use of europium as an epithermal poison instead of hafnium which was utilized in the initial work. The results were in agreement with the earlier work with hafnium in that they showed that flexibility was available for adjusting the flooding coefficient through the use of epithermal poisons.

Figures 10.1 and 10.2 graphically present a summary of the important results.

Shown in Figure 10.1 is the reactivity effect of completely flooding the superheating core as a function of core size and europium content. In all cases, the unflooded operating core is loaded so that it is exactly critical and the reactivity values shown represent the reactivity in the completely flooded case. Except for the 3-1/2 ft. core, all calculations assume that the U-235 was used. In the 3-1/2 ft. core, calculations were performed for both U-235 and plutonium. It can be seen that in all cases, the total reactivity effect of flooding can be adjusted to zero or negative values by proper choice of europium content.

Figure 10.2 is a plot of the reactivity change as a function of the fractional flooding of the core. The 3 cases considered are a 1-1/2 ft. core containing 3% of europium and a 3-1/2 ft. core containing 0.2 and 0.3% of europium. In each case, the flooding is assumed to occur uniformly throughout the core. One notes that the reactivity change with flooding is highly sensitive to the size of the core and the poison content. In particular, whereas the reactivity in the small core rises slightly as the core is initially flooded, the reactivity in the larger core decreases with partial flooding and then increases with further

Figure 10.1

REACTIVITY EFFECT OF FLOODING A SUPERHEATER AS A
FUNCTION OF EUROPIUM CONCENTRATION AND CORE SIZE.
THE DIMENSION S REFERS TO THE LENGTH OF THE SIDE
OF A CUBICAL CORE. THE FUEL USED IS INDICATED IN PARENTHESES.

EFFECT OF *BURNABLE* POISON ON REACTIVITY DURING FLOODING

FIG.10.2

flooding. These effects can be qualitatively understood on the following basis. The small core has a relatively hard spectrum and the decreased leakage due to initial flooding increases reactivity. As the reactor is further flooded, the effect of degradation of neutron spectrum is such that the europium becomes effective and the increased absorption in the europium more than offsets the decreased leakage. In the case of the large reactors, the spectrum is soft enough and the leakage small enough so that even during the initial flooding the increased absorption in the europium has a greater effect than the decreased leakage.

One anomaly which is not understood at this time is that the large core with 0.3% europium has a lower reactivity than the core with 0.2% europium at the minimum of the reactivity curve; i.e., at a flooding fraction of 0.5. There is still some question as to whether the small difference may be due to inaccuracies in the calculations.

The rise in reactivity in the region of almost complete flooding is due to the degradation of the spectrum into the thermal region. At thermal energies, the europium is not an effective poison and the fissionable material cross section is very high relative to that of the other materials in the core. Thus, essentially all of the neutrons which are thermalized result in capture in the fuel. The shape of the curve can be considerably changed by the addition of a thermal poison as well as an epithermal poison in the core.

The possibility of having a minimum in reactivity during partial flooding may be important in an unflooding accident. The desirability of this is unclear at this time. Under some circumstances, one could envision increased safety since an unflooding accident results initially in a dip in reactivity. Indeed, it appears possible that the dip in reactivity can be made so large that there is

enough excess reactivity in the core to get past the minimum. On the other hand, the dip means that once the point of minimum reactivity occurs, further flooding increases reactivity. The situation is further complicated by the fact that in an unflooding accident, the reduction of water density will not occur uniformly through the core but will, in general, involve the unflooding of the upper part of the core before the lower part becomes substantially effected.