NUCLEAR SUPERHEAT MEETING NO. 3, OCTOBER 13 AND 14, 1960, MILWAUKEE, WISCONSIN

By
Andrew E. Mravca

November 10, 1960

Chicago Operations Office, AEC
Chicago, Illinois
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NUCLEAR SUPERHEAT MEETING NO. 3

OCTOBER 13 AND 14, 1960

MILWAUKEE, WISCONSIN

CHICAGO OPERATIONS OFFICE
U. S. ATOMIC ENERGY COMMISSION

NOVEMBER 10, 1960

Prepared by
Andrew E. Mravca
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INTRODUCTION

The third of a series of AEC-sponsored nuclear superheat meetings was held in Milwaukee, Wisconsin, on October 13 and 14, 1960. These meetings are in the nature of technical seminars with particular emphasis on engineering data that have been developed since the last nuclear superheat meeting (April 7 and 8, 1960). The primary purposes of these meetings are to keep the AEC contractors engaged in nuclear superheat projects abreast of the over-all Commission's superheat program and to provide a means for the exchange of current technical information. It is not intended that these meetings be held for advisory or planning purposes.

The minutes of the meeting have been prepared to assist interested organizations in the task of keeping abreast of new results in the nuclear superheat program. The minutes are not meant to be a comprehensive abstract of the material discussed at the meeting. For the benefit of those who may be interested in more details than presented herein, a bibliography of all reports published to date under the superheat program is included. Readers are urged to consult the references in order to obtain the background of the work reported and to obtain the interpretation of the results given by the original author.
ATTENDANCE AT THE
AEC-SPONSORED THIRD QUARTERLY MEETING ON NUCLEAR SUPERHEAT PROGRAM
MILWAUKEE, WISCONSIN

OCTOBER 13 AND 14, 1960

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D. Crimmins
W. Farmer
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K. Gruenwald
D. Hall
R. J. Holl
R. W. Kiecker
C. E. Klotz
G. Kutsch
R. Lodzinski
R. Michel
D. Patterson
J. Patterson
N. Sher
J. Stone
D. Swanson
C. Tester
R. Vollmer
J. Wilson

ATOMICS INTERNATIONAL

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L. S. Mims
D. J. Stoker

COMBUSTION ENGINEERING, INC.

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L. O. Mayer
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NUCLEAR DEVELOPMENT CORPORATION OF AMERICA
G. A. Sofer

WESTINGHOUSE ELECTRIC CORPORATION
S. N. Tower
Joel Weisman
J. H. Wright
SUMMARY

1. ALLIS-CHALMERS MANUFACTURING COMPANY

The presentation involved a discussion of the Pathfinder R&D program and a status report on the construction of the Pathfinder reactor plant. Plant construction was about 34 percent complete as of October 15, 1960. The containment shell has been fabricated and is currently being leak tested. It appears that the constructor will be able to continue work throughout the winter season inside the various buildings as scheduled. Barring no unforeseen difficulties, the scheduled reactor criticality date of July 1962 should be met.

Allis-Chalmers has set a target of developing a low enrichment superheater fuel element for insertion in the third Pathfinder core (1963). The current low enrichment fuel design is based on a seven pin cluster fitted into a tube diameter such that the entire assembly is interchangeable with the high enrichment fuel elements.

The critical experiment program currently under way consists of the mock-up of the Pathfinder "at power" core.

2. ANGONNE NATIONAL LABORATORY

The BORAX V plant, which is currently being constructed at the NRTS in Idaho, was described. As of October 15, 1960, plant construction was about 82 percent complete versus 100 percent scheduled. Pressure vessel delivery has been the major item in delaying plant construction. Initial criticality with a boiling core is scheduled to begin April 1, 1961, and full power operation with an integral boiler-superheater core is scheduled for July 1, 1961. A decision has not yet been made as to whether a central or peripheral superheater region will be tested first.

3. GENERAL ELECTRIC COMPANY

The presentation involved a description of the design effort on the Separate Superheater and Mixed Spectrum Superheat Reactor concepts and the experimental research and development program. Sufficient experimental data have been obtained on Once-Through Heat Transfer to initiate a preliminary reactor design. In the "Once-Through" reactor, water enters the core in the subcooled state and exits in a superheated condition.

The modifications to the VEWR, which started September 23, 1959, to permit higher power operation have been completed. VEWR operation up to 1 MW has been initiated. It is expected that SABE loop operation will begin in November 1960.
Annular superheater fuel elements for the critical facility have been delivered, and experimentation is expected to begin the latter part of October 1960.

1. **COMBUSTION ENGINEERING, INC. - GENERAL NUCLEAR ENGINEERING CORPORATION**

The presentation involved a description of the BONUS reactor plant, supporting R&D program, and the general NUSU RAD program. Construction of the BONUS reactor plant was initiated August 23, 1960, and is approximately 2 percent complete as of October 15, 1960. The preliminary reference design of an integral boiler-superheater reactor capable of producing steam at 1200 psig and 1050°F for generating 200 MWe has been essentially completed, and the report is expected to be issued shortly.

5. **ATOMICS INTERNATIONAL**

A study was performed to evaluate the merits of using zirconium hydride as a solid moderator in the superheater region of an integral boiling superheating reactor. On the basis of this study, it was concluded that an economic incentive for substituting zirconium hydride for water moderator in an integral reactor does not exist. In addition, the technical problems encountered using zirconium hydride are more severe than those of a water moderated reactor.
EXPERIMENTAL PROGRAM

1. ALLIS-CHALMERS MANUFACTURING COMPANY

a. Critical Experiment Program - R. Vollmer

A critical experiment program in support of the Pathfinder reactor was initiated in November 1959.

The purpose of the first set of experiments which were completed on February 1, 1960, was to provide information on simple lattices of boiler fuel, in slab geometry, so that this may be compared with the theoretical analyses.

Summarizing the results of the slab experiments, in which cases of different water/uranium ratio, enrichment, and void content were run, the measured reactivity agreed with the calculated value to within 0.5% delta k/k. The theoretical analysis utilized two dimensional diffusion theory calculations, using thermal and fast constants from Sofocate and Muft, with appropriate transport corrections. Briefly, the results are shown in the table below:

<table>
<thead>
<tr>
<th>Enrichment</th>
<th>W/U</th>
<th>% Void</th>
<th>Loading (Kg U-235)</th>
<th>k (effective) Experimental</th>
<th>Calculated</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.8</td>
<td>3.1</td>
<td>18.5</td>
<td>33.3</td>
<td>1.0025</td>
<td>0.9994</td>
</tr>
<tr>
<td>1.8</td>
<td>3.8</td>
<td>0</td>
<td>25.2</td>
<td>1.0025</td>
<td>0.9960</td>
</tr>
<tr>
<td>1.8</td>
<td>4.8</td>
<td>15.2</td>
<td>33.8</td>
<td>1.0019</td>
<td>1.0053</td>
</tr>
<tr>
<td>1.8</td>
<td>5.7</td>
<td>0</td>
<td>26.4</td>
<td>1.0025</td>
<td>1.0081</td>
</tr>
<tr>
<td>1.59</td>
<td>3.8</td>
<td>0</td>
<td>37.7</td>
<td>1.0010</td>
<td>0.9973</td>
</tr>
</tbody>
</table>

Following these experiments, a full core loading of 96 boiler elements and 428 superheater elements was constructed for design experiments. With an enhanced superheater worth, a flooding coefficient of -1.4% delta k/k was measured. This compares with a -1.7% delta k/k calculated worth. With a normal superheater worth expected in Pathfinder, this would be reduced to approximately -0.5% delta k/k.

Again, with an enhanced superheater worth, the total rod worth was measured by uniformly distributed poison and rod drop. The total rod worth was measured to be -13.5% delta k/k. The calculated value for this configuration was -13.7% delta k/k. The total rod worth for Pathfinder will be higher than that measured for this configuration.
The experimental program presently under way consists of the mock-up of the Pathfinder "at power" core, with measurement of power distributions and splits, rod worth, flooding coefficients, void coefficients, and lattice parameters.

b. **Pathfinder Reactor Dynamics - D. H. Crimmins**

1. **Superheater Dynamics**

   The objectives of this task are to determine analytically: (1) the dynamic behavior of the reactor and related systems under normal and abnormal operating conditions and (2) the effectiveness of the reactor control systems and the nuclear instrumentation requirements.

   Mr. Crimmins gave a general description of Pathfinder transient simulator and the dynamic model used in studying the reactor performance. Among the advanced features that are considered in the dynamic analysis are the forced recirculation system and the integral nuclear superheater. The recirculation system includes three externally located centrifugal pumps. Gross reactor power changes will be accomplished by manual control rods, and smaller power adjustments will be made by varying recirculation rate.

   Slides (Figures 1 and 2) were shown of the analysis results of: (1) transients resulting from reactivity additions of 5%/second for 10 seconds at full power and (2) transients resulting from sudden failure of one recirculation pump.

2. **Dynamic Effect of Superheater upon the System**

   The superheater produces a negligible amount of reactivity feedback under steady state operating conditions for a highly enriched Pathfinder type superheater core. A low enrichment core may have an appreciable Doppler effect due to fuel temperature variations.

   The pressure drop across the superheater presents a significant effect on Pathfinder system dynamics, in that it acts as a thermal "buffer" between the reactor and turbine.

3. **Dynamic Effect of System upon Superheater Temperatures**

   If the feedwater temperature is lower at partial powers, the steady state superheater fuel surface temperatures may rise significantly as power level is decreased.

   Steam flow is generally incapable of following rapid power variations very closely, hence transients in superheater...
temperatures will occur. These may be reduced to very acceptable values by proper design of the control system.

c. Heat Transfer Experiments - N. Sher

(1) High Pressure Loop

The high pressure heat transfer loop was described and shown in slides. The loop is designed with a boiler and superheater test section in parallel with a common power supply system. Saturated steam generated in the boiler section will be channeled to the superheater section and superheated to simulated Pathfinder conditions.

The first superheater test section has been assembled and will be installed in the loop in the early part of 1961. The test section consists of two tubes simulating a highly enriched Pathfinder element. Both tubes will be heated independently. The outer tube is concentrically spaced from the inner tube by three straight wire spacers providing a 90 mil channel. The over-all section length is \( \frac{1}{4} \) feet. Approximately 1000 lbs/hr steam will pass through the annulus formed between the tubes. Tests will be run to experimentally determine the heat transfer coefficients and fluid flow characteristics at simulated Pathfinder conditions.

(2) Low Pressure Air Tests

Air flow tests were made through an unheated annular test section. The annulus was formed using an inner tube with 0.750 inch O.D. and spaced by:

(a) One 76 mil wire spiral with a 12-inch pitch.

(b) Three straight 76 mil wires.

Results of these tests indicate that the friction factor for the three parallel wires compares favorably with published data. Results with the spiral wire indicate that the friction factor is 50% higher at \( Re = 10^5 \) than with parallel wires.

(3) Shutdown Cooling Test

The purpose of this test is to determine the superheater fuel element heat losses through the double wall process tube. A heat transfer rig is currently being constructed and should begin operation shortly.
d. **Low Enrichment Superheater Fuel Development - D. Patterson**

The objective of this task is to develop fabrication and testing technique for low enrichment superheater fuel elements for insertion in the third Pathfinder core. Primary emphasis has been placed on three fuel configurations using uranium dioxide fuel with stainless steel cladding. The three fuel configurations consist of: (1) seven fuel rods in a septafoil cluster contained in a scalloped or round flow tube, (2) single annular fuel tube, and (3) a perforated fuel rod penetrated by multiple flow channels.

The current experimental effort has been concentrated on developing high density swage-compacted UO$_2$ fuel rods for the septafoil design. An intensive study of swaged fuel rods is being made to determine the effects of fabrication variables. The fabrication variables being considered include powder particle size, size distribution, preswaged density, reduction of swaging, the number of swaging passes, and strength of cladding. Fabricated rods are then checked for swaged density, clad wall thickness, and quality of compact and clad.

To date, eight 316L stainless steel clad fuel rods were swaged using UO$_2$ power with a different particle size range (-400 to -70 \(\mu\)m) for each rod. A 54\% reduction in cross sectional area was made in each case. One test result indicated that the element made with -400 mesh power resulted in the lowest ultimate density (86.3\% theoretical) but had the best cladding integrity following swaging.

The -70 \(\mu\)m mesh power gave the highest ultimate density (91.9\% theoretical) but serious deformation of the cladding was evident.

Future investigations will include swaging (54, 55, 60, and 65\% reduction) of fuel rods filled with UO$_2$ powder in eight narrower ranges of particle sizes in an attempt to achieve 92\% theoretical fuel density.

e. **Pathfinder Reactor Internals - J. Wilson**

**Mechanical Tests**

(1) **Tube-to-Tube Sheet Welds**

Various joint designs have been investigated for welding the type 316 superheater container tubes to the type 304 upper tube sheet. The designs investigated are shown in Figure 3 attached. Joints 1 and 6 provide the most satisfactory connections when consideration is given to distortion, ease of fabrication, and over-all appearance.
(2) **Inner-to-Outer Tube Seal**

A series of mechanical connections between the container tube and insulating tube have been investigated. Lip type, O-ring, and rolled joints were among those investigated. The most satisfactory design is the rolled joint which is similar to those used in condensers. Standard tube expanders and tube pullers can be used to make and break the connection.

(3) **Tube Collapsing Tests**

Tests have been run to determine the collapsing pressure of the superheater container tubes and to determine the effect of fabrication imperfections on this critical pressure. It has been found that the 1.076 inch O.D. by 0.025 inch wall tubes will withstand a pressure of 790 psig at 800°F. Out of roundness of the tubes seems to be the most detrimental fabrication imperfection. Small dents do not appreciably affect the strength of the tube.

f. **Steam-Water Separation - J. Wilson**

(1) **Natural Separation**

Experiments conducted at Allis-Chalmers show that natural separation will produce steam of 99% quality or better, provided the release rate is below a critical value.* This critical value is a function of system pressure, temperature, height above the interface, and up to a certain size the diameter of the vessel. Sterman** provides an expression for determining the approximate diameter at which entrainment is independent of vessel diameter.

\[
\frac{d}{\sqrt{\frac{\sigma}{\rho - \rho''}}} \left(\frac{\rho''}{\rho - \rho''}\right)^{-0.2} \geq 260
\]

\[d = \text{vessel diameter} \]
\[\sigma = \text{surface tension} \]
\[\rho = \text{specific wt. of liquid} \]
\[\rho'' = \text{specific wt. of vapor} \]


Results of the test for a two and four-inch diameter vessels are presented in detail in Allis-Chalmers Report ACNP-5921. In general, the change in moisture content from 1 to 7 percent or greater was very abrupt.

Typical results of an experiment carried out in a 19-inch diameter vessel, which by Sterman's criterion is a large vessel, are shown in Figure 4. This shows the relationship between quality measured 33 inches above the interface and the steam release rate for different system pressures. The rapid rise in the quality of the steam after the "break" point is reached was typical for all test vessels. Thus, if the release rate is below the critical value, very little mechanical separation is required.

(2) Mechanical Separation

Two types of centrifugal separators were tested by Allis-Chalmers at pressures and temperatures up to Pathfinder operating conditions (Types A and B in Figure 5). Results show that the overload condition for both "A" and "B" was reached considerably below the manufacturer's rated capacity. The results are shown in Figure 6 for the Type A centrifugal separator. Results obtained with the Type B were similar except that the break point for 600 psig occurred at approximately 22,000 lbs/hr.

Data on mesh type separators have been obtained from a 6-inch thick mesh with a density of 5 lb/ft³. Results of this test are presented in Figure 7. Inlet moisture varied from 1/2% to 7% during the test. In all cases, the outlet steam quality was 99.9%. Note the similarity to the curves obtained for both natural separation and the centrifugal type separator.

In order to establish a basis for comparing efficient use of reactor cross sectional area, an area requirement was established for each centrifugal separator. The maximum flow rate before the overload point was reached was divided by the area requirement to determine a permissible unit load factor. For the mesh separator, this factor is 27,500 lb/hr-ft². For the best centrifugal type separator, this factor is 10,100 lb/hr-ft².

The wire mesh could have a cost advantage of as large as 10 to 1 compared to either centrifugal type.

As a result of the tests described, it has been concluded that the mesh type separator is best suited for use in Pathfinder. Parameters considered most important in making the selection were:
(a) Separation efficiency, i.e., the outlet quality compared to the inlet quality.

(b) Separator operating characteristics when overloaded.

(c) Most efficient use of reactor cross sectional area.

(d) Maintenance and cleaning problems.

(e) Cost.

Future plans for the separator loop include the testing of more mesh type separators to determine the effect of thickness and mesh density on the operation of the dryer. These tests will be conducted up to the full capabilities of test loop.

Presently, a program to express the operation of the separator in terms of dimensionless constants is under way. With additional data obtained from future tests, this program can be carried successfully to completion. Another program to correlate the data on natural separation using the methods of dimensional analysis is also under way. Both programs involve the use of dimensionless groups which appear to more closely predict the phenomenon of natural and mechanical separators. With these equations, the critical flow rates can be predicted for any pressure.

g. Steam Carry-under - D. Swanson

In December of 1956, preliminary development work was started on developing steam separating devices to limit carry-under in a forced circulation boiling water reactor with internal nuclear superheat. A concept using a long tube with a tangential inlet nozzle extending along the length was investigated. A high capacity with a relatively low pressure drop was required. Cold water tests on 1-inch diameter models were started in December 1956 to determine the effects of varying the length-to-width ratio of the inlet nozzle. At approximately the same time work was started on a 5-inch diameter model separator test loop.

Parallel to the downcomer separator program, several other types of separating devices were evaluated and tested. An angle type separator, which consisted of several angle pieces placed in the flow path of the recirculation water above the downcomer, was tested. This test was modified several ways by changing the arrangement of the angle to form a chevron pattern, using perforated plates and other modifications. The results of the tests indicated that the method would not be satisfactory for the Pathfinder reactor.
Following the Allis-Chalmers development program, which resulted in the conception of the internal separator design for the Pathfinder reactor, an AEC-sponsored testing program was initiated on November 22, 1957. Under this program, a test loop was built to test a full-scale Pathfinder type separator.

A special summary report including several curves of test results was distributed at the meeting. Several drawings of upcomer type separators developed and tested under the Allis-Chalmers sponsored development program are also included in the summary report. A number of limitations must be evaluated when applying upcomer type separators. For example, hydraulic instability in the reactor core may be caused by imposing an added pressure drop on the core exit. Furthermore, upcomer separators cause structural interference with the above-the-core control rod drives of the Pathfinder reactor. Also, upcomer separators add complexity to the refueling process. The above considerations led to the decision to apply major development on downcomer separators.

Test results on the Pathfinder separator indicate a low carry-under rate (approximately 0.12% by volume, 0.0032% by weight steam at 600 psig) with a relatively high water flow rate (1330 gpm). In the test program, the separator was tested under hydraulic and geometric conditions similar to those of the reactor. An analysis of the factors affecting separator performance indicated that the influence of entrance conditions, flow patterns, and other geometric factors would have to be duplicated if test results are to be applied to the reactor design. The tests were performed with air and water to permit direct observation and to facilitate loop modifications which are necessary to test separator performance under reactor conditions. The results obtained with air-water tests for carry-under information are sufficient for predicting steam-water performance at reactor conditions. The air-water tests were run at a temperature of 180°F to lower the viscosity to approach reactor conditions. The effects of temperature on natural carry-under (carry-under without separators) are shown in the summary report.

Figure 8 shows test results of carry-under and pressure drop at water flows up to 1500 gpm for the Pathfinder type separator. Figures 9 and 10 show carry-under rates and pressure drops, respectively, obtained in an extended capacity test of the same separator with a wider inlet nozzle. These results show the potential capacity of these separators, and the relatively low pressure drops suggest possible use in natural circulation reactors.
Transients Resulting From Reactivity Addition of $5\text{f}/\text{second}$ for 10 Seconds at Full Power

1. Neutron Flux

2. Steam Flow Rate

3. Normalized Power to Steam Flow Ratio

4. Superheater Hot Spot Temperature $T_F$

Time - Seconds
Transients Resulting From Sudden Failure of One Recirculation Pump

Neutron Flux

Steam Flow Rate

Normalized Power to Steam Flow Ratio

Superheater Hot Spot Temperature °F

Time - Seconds

Fig. 2
Figure 3 Sketches of various tube to tube-sheet joint designs.
(Dwg. 51-433)
FIG. 4  MOISTURE ENTRAINED 33 INCHES ABOVE INTERFACE IN A 19-INCH DIAMETER TEST SECTION.
FIGURE 5

TYPE A AND B CENTRIFUGAL SEPARATORS
STEAM DRYER TEST

STEAM FLOW RATE - LB/HR x 10^{-3}

FIG 6  TYPE A SEPARATOR
OPERATING CHARACTERISTICS OF A 6 INCH THICK MESH SEPARATOR

FIG. 7
FIGURE 8
PRESSURE DROP AND PERCENTAGE CARRYUNDER BY VOLUME VS WATER FLOW RATE FOR PATHFINDER CENTRIFUGAL STEAM SEPARATOR
TEST CONDITIONS: AIR AND WATER AT 175°F, 5 PSIG
VOID FRACTION IN UPCOMER ≈ 50% CONSTANT
VOID FRACTION AT SEPARATOR INLET ≈ 25%.

WATER FLOW RATE - GPM

PERCENT CARRYUNDER

PRESSURE DROP

FEET OF WATER AT 60°F PSIG SAT.

CARRYUNDER % BY VOLUME IN DOWNCOMP
SEPARATOR STAND PIPE % CARRY UNDER VS WATER FLOW
CONSTANT AIR FLOW AND TEMP. 10-12-60 A.W.O.

DATA SHEET NO. 2 - 60 CFM 190°F
DATA SHEET NO. 4 - 90 CFM 9/10 CFM 190°F

FIGURE 9

WITH SEPARATOR, AT EXTENDED CAPACITY
WATER TEMPERATURE TAKEN FROM THERMOCOUPLE

ALLIS-CHALMERS MFG. CO.
NUCLEAR POWER DEPT.
FIGURE 10  SEPARATOR STANDPIPE TOTAL PRESSURE DROP VS WATER FLOW

THERMOCOUPLE TEMPERATURE 180°F
WITH SEPARATOR AT EXTENDED CAPACITY
2. ARGONNE NATIONAL LABORATORY

a. Steam-Water Separation - M. Petrick

The gravimetric steam-water separation program at ANL is being conducted in three distinct phases. Studies are or are scheduled to be carried out (1) within EBWR, (2) on air-water atmospheric loop, and (3) on the 2000 psi test loop.

The prime purpose of the air-water studies is to establish the pertinent factors which affect the carry-under problem and to attempt to develop analysis or correlations that can be used for extending ranges of the meager data available. In this respect, the results obtained to date from the study have been very encouraging.

The air-water loop is schematically illustrated in Figure 1. It is constructed of large sections of Lucite to permit visual and photographic studies. To date, a series of runs have been run over the following parameter range:

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Range</th>
</tr>
</thead>
<tbody>
<tr>
<td>Downcomer velocity</td>
<td>0.9 - 1.67 ft/sec</td>
</tr>
<tr>
<td>Riser velocity</td>
<td>1.5 - 2.75 ft/sec</td>
</tr>
<tr>
<td>Riser qualities X</td>
<td>0.0002 - 0.002</td>
</tr>
<tr>
<td>Riser steam volume fraction</td>
<td>0.08 - 0.45</td>
</tr>
<tr>
<td>Interface heights above riser discharge</td>
<td>4 - 19&quot;</td>
</tr>
</tbody>
</table>

The principal data derived from these tests are riser volumetric and weight gas fraction and downcomer volumetric and weight gas fractions. Typical data are shown in Figures 2 to 7. Figure 2 is a series of runs showing the effect of interface height and riser mixture quality on the volumetric ratio of voids in the downcomer to riser. The larger voids in the downcomer then in the riser stem from the unique two-phase flow characteristics in downflow. As the downcomer velocity approaches the buoyancy velocity of the gas bubbles, the volume fraction increases sharply. The same data plotted on a weight fraction basis are shown in Figure 3. Similar trends can be seen.

The apparent interface height effect on carry-under shown in Figures 2 and 3 may be an exaggerated one characteristic of an atmospheric air-water system resulting from the hydrostatic head effect.
The effect of temperature on carry-under is shown in Figure 4. As the temperature increases, the carry-under decreases. Since all the physical properties of the system remain virtually constant except for the viscosity, the change in carry-under is attributed to the change in viscosity. The strong viscosity effect was noted on an air-glycerine system where it was extremely difficult to separate the air from the very viscous glycerine.

An attempt to account for the various effects on carry-under is shown in Figure 5. All the data obtained to date are plotted as $X_D/X_p = f(V_g/V_{ent}, G, \lambda)$ where $V_g$ is the actual gas velocity in the riser and $V_{ent}$ is the relative entrainment velocity in the downcomer. This correlation appears to be promising and will be evaluated more thoroughly as more data are obtained over wider ranges.

Analytical studies of the carry-under problem being carried out concurrently with the experimental study have shown that carry-under should be very sensitive to the relative size and distribution of bubbles in the riser and downcomer. As a result, photographic studies were made. Typical bubble size data are shown in Figure 6. It is interesting to note that the bubble size is larger in the downcomer than in the riser due to the lower liquid velocity and quality. The bubble size data are being used in conjunction with the downflow slip data to establish the velocity versus bubble size curve for down flow. Trace photographs of the riser discharge into the upper plenum have proved to be very interesting and have shown that the majority of the carry-under comes from the peripheral riser area.

The air-water loop is now in the process of being revamped to extend the parameter ranges that can be covered. One of the major factors on carry-under that remains to be studied is to determine whether data taken on small scale systems are applicable to the larger systems.

The carry-over studies will be carried out primarily on the 2000 pound test facility and should commence within the next month. Simultaneously, carry-under studies similar to the one being carried out on the air-water loop will also be made. Data that will be taken in gravimetric carry-over will cover the following parameter ranges:

- **Pressure**: 600 - 2000 psi
- **Steam dome vapor velocity**: 0.25 - 4 ft/sec
- **Dome height**: 10 - 30"
Data on the gravimetric steam-water separation problem will also be obtained directly from EBWR as it is brought up in power. Instrument probes and sample tubes have been placed in strategic positions in the reactor as shown in Figure 6. Static differential pressure readings taken in the downcomer will yield data on carry-under before, during, and after the make-up water addition. Carry-over data will be obtained from sample probes at various heights above the expected interface level. Data on the vapor holdup and steam-water interface will also be obtained during the tests.

It is expected that the air-water studies will be completed by January 1961 and the high pressure loop program and reactor tests by April 1961.

b. EBWR Seal Leakage Test - R. Gariboldi

The Experimental Boiling Water Reactor (EBWR) was originally equipped with a special air-seal vapor recovery system to recover primary fluid leakage from the power plant's turbine, reactor feed pump, and major steam control valve seals. The system was installed in anticipation of possible D₂O operation. Recent interest stimulated tests to establish the operating efficiency of the design and a background of experimental evidence for possible future applications.

The general ideal operation of the system is as follows:

(1) Leakage is collected at each seal by means of a vacuum chamber toward the outboard end of the seal. A 6-inch water gauge vacuum in the chamber causes air inleakage to the chamber as well as vapor collection. To prevent inleakage of humid atmospheric air, which would contaminate the (hypothetical) D₂O with H₂O, a dry air barrier chamber is located outboard of the vacuum chamber.

(2) The air-vapor mixture is drawn from the seal to a vent condenser.

(3) The condensed vapor is accumulated in a sump from which it is pumped back to the turbine's condenser hot well.

(4) Air recovered from the condenser is drawn through an alumina bed desiccator to remove virtually all vapor from the air.

(5) The dry recovered air is returned, by a blower, to the dry air barrier chambers on the seals. A 6-inch water gauge pressure is applied to this chamber by the blower.

(6) The air barrier is at a positive pressure and its outboard end is not completely sealed; thus, there are losses of dry air to
the atmosphere. These air losses are made up by permitting the suction side of the blower to draw air into the system via a make-up air desiccator. This air enters the make-up desiccator from the atmosphere.

With proper operation of the system, all seal leakage is collected. The only vapor loss that can occur is the minute humidity entrained in the "dry" air that escapes from the dry air barrier to the atmosphere. The \((D_2O)\) vapor loss to the atmosphere is thus directly dependent upon the efficiency of the recovered air desiccator and the rate of dry air loss from the dry air barriers. Similarly, the amount of atmospheric \((H_2O)\) vapor entering the system depends upon the make-up desiccator efficiency and the make-up air flow rate.

For the tests, the system was instrumented to measure: (1) **pressures** at all seal chambers, (2) **air flow** recovered from the vent condenser, (3) **condensate flow** to the vent condenser, and (4) **humidity** of the air leaving the desiccators.

A rigorous series of tests were conducted. Care was taken to insure that test apparatus did not alter normal operation. Pressure measurements indicated that effective dry air barriers were maintained, and sufficient vacuum was developed in all associated seals. This ascertained that all vapor leakage from these seals is collected and ducted to the vent condenser. Samples of recovered air from the condenser indicated that the air leaving the condenser was saturated. "Dry" air sampled after the recovered air desiccator contained less than 40 ppm vapor. "Dry" make-up air entering the system via the make-up desiccator contained an average of 230 ppm. The vapor in the make-up air cycled between 30 ppm and 800 ppm because of poor reactivation. Dry air flow to the seals was 20 SCFM. Dry air was lost from the barrier chambers at a rate of 9 SCFM; 11.7 SCFM was recovered.

The vapor lost to the atmosphere, with the 9 SCFM of dry air above, is normally 0.3 lbs/month. This is the only "leakage" permitted when the vapor recovery system is operating properly.

Several severe abnormal conditions were observed, including complete failure of the recovered air desiccator operation. Even with such a condition, the system was found to be capable of holding leakage to less than 70 lbs/month. Complete test details may be found in ANL-6189.

c. **BORAX V**

(1) **Superheated Steam Heat Transfer Experiment - J. B. Heineman**

The objective of this experiment was to obtain data and correlations which determine convection coefficients applicable to superheat reactors, with particular attention focused upon the proposed BORAX V reactor design. The study has been completed and an extensive description is available as ANL-6213.
A rather extensive literature review revealed but one experiment which was applicable directly to the problem of steam convection, and this study is restricted geometrically to an annulus and the data runs to a Reynolds number of 40,000. Since the original BORAX V superheater fuel plates dictated a thin rectangular channel (1.250 inch x 0.047 inch) and Reynolds numbers were expected to be in the neighborhood of $3 \times 10^5$, it was deemed necessary to experiment first with a round channel (0.333 inch diameter x 12 inches long) and second with a thin rectangular channel of design dimensions (1.250 inch x 0.047 inch x 12 inches). It was felt that both tests were required to effect a geometric comparison. (The actual channel design was changed during experimentation and a spacing of 0.0625 inch is to be used in the reactor core.)

The circular channel was cooled by superheated steam in the following range of variables:

- **Steam pressure**: 300 to 1500 psia
- **Steam temperature**: 550 to 900°F
- **Initial steam superheat**: 73 to 142°F
- **Surface temperature**: 650 to 1290°F
- **Heat flux**: 50,000 to 287,000 B/hr·ft$^{-2}$
- **Steam mass velocity**: 144,000 to 792,000 lb/hr·ft$^{-2}$
- **Reynolds number**: 60,000 to 370,000

The thin rectangular channel experiment encompassed the following range of variables:

- **Steam pressure**: 300 to 1000 psia
- **Steam temperature**: 470 to 850°F
- **Initial steam superheat**: 5 to 160°F
- **Surface temperature**: 520 to 1125°F
- **Heat flux**: 23,000 to 96,000 B/hr·ft$^{-2}$
- **Steam mass velocity**: 197,000 to 553,000 lb/hr·ft$^{-2}$
- **Reynolds number**: 25,000 to 100,000
The question of steam conductivity and viscosity values is still unsettled, but an extensive review of water and steam properties by Nowak and Grosh (ANL-6064) indicated that the various Russian conductivity values were to be preferred over the data of Keyes and Sandell (1950). Accordingly, these values were incorporated into the data analysis.

Mr. Heineman believes that the experiments were quite successful, and the resulting correlations may be used to $\pm 10\%$. Figure 8 shows the results of the round channel experiment. Three major comparisons arising from the experimental results may be used to illustrate the conclusions.

(a) Comparison of round tube results with MacAdams 1949 annular data (Figure 9).

Since the slopes of the correlation lines in the two experiments are 0.89 and 0.84, no direct comparison can be made due to the difference in the influence of the $L/D_e$ parameter. However, a ratio of convection coefficients is instructive since it creates a comparison valid for the same heat transfer situation in both experiments. The differences noted at low values of $L/D_e$ are due to the different unheated entrance conditions found in each experiment.

(b) Comparison of rectangular channel data with the round channel data (Figure 10).

In Figure 10, the rectangular data and the round tube correlation are plotted on the same coordinates at an $L/D_e = 60$. Apparently, there is no difference in convective behavior, at least to an aspect ratio of 26:1.

(c) Comparison of general correlation line with the Dittus-Boelter equation, evaluated at average film temperature.

This comparison indicates that the Dittus-Boelter equation yields values of the coefficient which are 11.8 percent higher at $N_{Re} = 25,000$ and 1.8 percent higher at $N_{Re} = 370,000$ than those values predicted by the present study.

The correlations produced are believed to be satisfactory for design calculations, provided that the values of steam conductivity and viscosity are taken from the tables found in the text of ANL-6213, which values are those recommended by Nowak and Grosh in ANL-6064.
An experimental program was carried out by R. P. Anderson of Reactor Engineering Division, ANL, to determine the natural circulation performance of a system hydrodynamically similar to the final design of BORAX V. An attempt was made to mock up as closely as possible the final boiler core design. This core consists of forty-nine (49) 3/8-inch OD rods in a 7 x 7 matrix. The spacing between the center plane of the outer row of rods and the wall was made larger than half the center distance between adjacent rows of rods to decrease the detrimental effect of the wall on the flow in the outer channel. Using 1/2-inch center distance between rows and 3/8-inch from the center of the edge row to the wall, the equivalent diameter of the corner flow segment was 0.433 inch; of the edge segments was 0.527 inch; and of the central segments was 0.472 inch. The equivalent diameter of the tubing used in this test corresponded to that of the central segments.

To approximate the core axial power distribution, the wall thickness of the heated tubes was varied to give an axial power distribution. Figure 11 shows the power distribution in the test section.

Table I compares dimensions of the reactor design and the test system.

<table>
<thead>
<tr>
<th>Reactor (Flow Outside Rod)</th>
<th>Test (Flow Through 2 Parallel Tubes)</th>
</tr>
</thead>
<tbody>
<tr>
<td>De = 0.433 in. corner segment</td>
<td>De = 0.466 in.</td>
</tr>
<tr>
<td>0.527 in. edge segment</td>
<td></td>
</tr>
<tr>
<td>0.472 in. central segment</td>
<td></td>
</tr>
<tr>
<td>Length = 2 ft.</td>
<td>Length = 2 ft.</td>
</tr>
<tr>
<td>Flow area/element = 0.14 in.²</td>
<td>Flow area/element = 0.171 in.²</td>
</tr>
<tr>
<td>Heat trans. area/element = 28.2 in.²</td>
<td>Heat trans. area/element = 35.2 in.²</td>
</tr>
<tr>
<td>Coolant volume = .0549 liter</td>
<td>Coolant volume = .0674 liter</td>
</tr>
<tr>
<td>Tube diam./core diam. = 1.075</td>
<td></td>
</tr>
</tbody>
</table>
One tube of the test section had a burnout tap and seven thermocouples attached to the tube to be used for detection of burnout. The tap was connected to a burnout detector which is a power tripping mechanism. It was felt burnout would likely occur a short distance after the maximum power location which is located eight inches from the inlet, but the burnout actually occurred at the outlet end of the tube and was not detected by the burnout detector.

A differential pressure transducer was connected across the pressure taps of the downcomer venturi measuring total recirculation rate. The inception point of unstable operation was defined as the point where oscillations in the transducer output first became evident. On this basis, the inception of instability occurred at 100 kw total power. The amplitude of the oscillations increased with every increase in power until burnout occurred at 115 kw.

Although an actual burnout did occur during these tests and caused one of the tubes to fail, it seems fairly certain the burnout point was predominantly affected by the unstable operation of the system. Based on these experiments, loop stability during operation appears to be the reactor power limiting mechanism.

The loop began to oscillate at an average flux of 220 watts/cm² and a maximum flux of 298 watts/cm². The power was thereafter increased three times before burnout occurred. Each power increase increased the amplitude of oscillation. The final step in power caused an increase in flow oscillation amplitude which we believe was the cause of the burnout.

The stability of the loop has been shown to be partly dependent on the geometry of the system external to the test section and riser. Considering the dependence of instability inception measurements on the external geometry of the test loop, it is not valid to place great confidence in the tests of reactor core mock-ups in test loops. Until the basic causes of hydrodynamic oscillations are
known, application of test results to reactor design must be "taken with a grain of salt."

(3) **In-Core Instrumentation Development - W. R. Wallin**

The primary efforts since the last meeting have been in the area of thermocouple development for measurement of fuel temperatures and the evaluation of flow meters.

Thermocouples for superheat fuel plate temperature measurement have been received from two manufacturers, the Aero Research Company and Continental Sensing Company. The thermocouples are chromel-alumel couples having aluminum oxide insulation and stainless steel sheath. At the request of one of the fabricators, part of the couples will be flattened to .020 of an inch rather than .015 because the fabricator feels that he cannot assure that the junction will be maintained at the tipped position by flattening beyond .020 thickness. The chief problem is to develop a method for inserting this couple into the .030 inch thick superheat fuel plate. The method being experimented with, at present, consists of cutting a rectangular hole in one side of the .030 thick fuel plate, then expanding the hole to .025 inch with a drift pin. The thermocouple tip with Microbraze compound is inserted into this hole and the assembly is microbrazed. Since the thermocouple must penetrate to the meat of the fuel, a seal must be re-established by the Microbraze process. If a satisfactory thermocouple can be developed, three fuel assemblies will be fitted with 20 thermocouples each in order to measure the temperature at various points in the fuel element.

One set of sample thermocouples for measuring center temperature in the boiling fuel rods has been received from Aero Research Company. The couple is tungsten versus tungsten 26% rhenium with magnesium oxide insulation and a tantalum sheath.

The balance of the thermocouples and thermopiles will be stainless steel sheathed chromel-alumel. Design of these couples and the method of mounting them is in progress. The temperatures to be measured include boiling assembly water channel temperatures, superheat steam exit temperatures, saturated steam temperatures, and subcooled inlet water temperatures.

An air-water test loop which operates at room temperature has been built to study the characteristics of the flow meters.

The loop can operate with air flow rates from 0 to 9,000 SCFH which is equivalent to a maximum of 50 percent exit voids at the maximum water flow rate of 500 gpm.
Turbine type flow meters, manufactured by Potter Aeronautical, have been tested for entrance and exit flow measurement, and a drag type flow meter manufactured by Physical Scientific Corporation has been ordered. Figure 12 shows the data obtained in measuring the pressure drop due to the addition of the turbine flow meter. These data indicate that this device will be satisfactory for measuring flows from 100 to 500 gpm. The flow rate for a differential pressure of 5 psig is reduced from 350 gpm to 325 gpm by the addition of the inlet flow meter. This is not sufficient to cause burnout in an instrumented flow assembly and will cause only a small amount of change in the power produced in the assembly. Assuming that no difficulties arise due to the instrument being in high temperature liquid and in a radiation field, we should be able to measure water flow rates with an accuracy 1.5%. Further testing in the low flow range is being conducted.

As a check of the feasibility of measuring flow with the two-phase flow, air and water were introduced while using the entrance flow meter.

The tests thus far have shown that the turbine type flow meter can be used for measuring a two-phase flow. Further testing is planned with the exit void meter.

The change in pressure drop through an element with and without entrance and exit flow meters at 20% voids is approximately 0.32 psi at 100 gpm and approximately 2.2 psi at 500 gpm.

(4) Superheater Fuel Element Seal Test Program – W. R. Wallin

The superheating section of the BORAX V reactor is a two-pass system. Half of the fuel elements conducts the steam from the steam chamber at the top of the reactor vessel down to the plenum chamber which is welded to the core support plate. The other half of the fuel elements conducts the steam from this plenum up through the reactor and out through the superheat steam nozzles. Water surrounds the outside of these fuel elements. Since the fuel elements must be removable, a seal must be formed between the fuel element and the core support plate. To select this seal, a test program was initiated in which several types of seals were tested. Figure 13 shows a schematic diagram of the seal test vessel.

The vessel is made up of a six-inch diameter pipe approximately 13 feet long with flanges at either end. Electrical strip heaters attached to the outside of the pipe serve as
the means of bringing the vessel and contents up to the operating pressure and temperature; 600 pounds, 489°F.

The earlier tests reported at the April meeting consisted of trying several types of seals, varying the sealing pressure, and varying the differential pressure across the seal. As a result of these tests, the silver plated hollow metal O-ring was selected as the most desirable for two main reasons. The first was that a good seal could be maintained with small differential pressures. The second was that this was the most economical seal. This seal does, however, require a substantial hold-down force. Tests confirmed the fact that 800 pounds pressure was required, this pressure being slightly less than that recommended by the manufacturer.

Two tests, each of approximately one month duration, have been conducted on this seal in which an average leakage of 1 to 2 cubic centimeters per hour was obtained. This is well below the $2\frac{1}{2}$ gallons per hour per fuel assembly which would be required to drop the steam temperature 10 degrees if the reactor were operating at 20 megawatts producing 850°F steam. The flange surfaces were carefully machined to have concentric grooves in the surfaces as recommended by the seal manufacturer. Since some of the components in the vessel were carbon steel, a considerable amount of rust collected from earlier runs. This rust was removed by wiping with a clean rag prior to each of these one-month tests. It is, therefore, anticipated that simple swabbing of these flange faces would be satisfactory in providing the clean surface for this seal prior to insertion of the fuel elements. These tests were discontinued due to excessive leakage through our safety valve. Further tests of longer duration are planned.
AIR-WATER ATMOSPHERIC LOOP

FIGURE 1

AIR SYSTEM

AIR RELEASE CHAMBER

RISER TUBE

AIR MIXING CHAMBER

AIR SEPARATION TANK

PUMP

BAROMETRIC LEG

DOWNCOMER TUBE
Figure 2 Effect of height and quality on the Volumetric carryunder ratio
Figure 3 Effect of Height and Quality on the Weight Carryunder Ratio
Figure 4  Effect of Temperature on Carryunder
Figure 5 Preliminary Correlation of Carryunder Data
Figure 6  Effect of the Liquid Velocity on the Average Bubble Size
Figure 7 Location of EBWR Instrument Probes

Probe | Type of Data
--- | ---
1 | Carryunder
2 | Carryunder
3 | Carryunder
4 | Vapor Holdup
5 | Vapor Holdup
6 | Riser Voids
7 | Riser Voids
8 | Carryover
9 | Carryover
10 | Carryover
11 | Interface Height
12 | Interface Height
13 | Interface Height
14 | Interface Height
\[ Y = 0.0157 \left( N_{Re} \right)^{0.84} \]

**FIG. 8**

**FINAL CORRELATION FOR SUPERHEATED STEAM FLOWING IN A TUBULAR CHANNEL.**
FIG. 9

COMPARISON OF CONVECTION COEFFICIENTS

\( h_a - \text{PRESENT STUDY} \)
\( h_m - \text{REF. 4} \)

\( N_{Re} = 100,000 \)
\( N_{Re} = 50,000 \)
\( N_{Re} = 250,000 \)
\( N_{Re} = 370,000 \)
\[ Y' = 0.023(N_{Re})^{0.80} \]
\[ Y'_{1} = 0.0133(N_{Re})^{0.84} \]

(DITTUS-BOELTER EQUA.)
(EQUA. 21 AT \( L/De = 60 \))

FIG. 10
COMPARISON OF RECTANGULAR CHANNEL DATA WITH EQUA. 21.
A—boiling region—central superheater core based on Fig.
B—test section with axial distribution

FIG. 11
REPRESENTATIVE LOCAL TO AVERAGE AXIAL FLUX
PRESSURE DROP ACROSS FULL 49 ROD BOILING FUEL ASSEMBLY VS. WATER FLOW RATE

FIG. 12
600 PSIG SEAL TEST VESSEL
BORAX \textdegree

FIG. 13
3. GENERAL ELECTRIC COMPANY

a. General Review of Research and Development Program - R. T. Pennington

Mr. Pennington stated that the General Electric Company superheat R&D program being performed for the AEC is very broad in scope and is not tied to a single specific reactor concept. The objective of the program is to investigate in depth major problems of technical feasibility for economic application of nuclear superheat for power generation. At the present time, the following reactor concepts are being investigated:

(1) Boiling water reactor in series with a separate superheat reactor.

(2) Integral mixed spectrum boiler-superheat reactor.

(3) Once-through nuclear superheat reactor.

(4) A thermal integral nuclear superheat reactor.

The specific objectives of the R&D program are to develop reactor design criteria to be used as a basis for the above reactor designs.

Critical Experiment Program

All the fuel for the superheat critical experiments has been fabricated and has been delivered to the site at the critical facility. Startup of experiments is expected to begin in the latter part of October 1960.

SADE Loop

Modifications to the VBWR reactor have been completed. Initial operating license to operate the VBWR up to 1 MW has been received. After operation at 1 MW has been satisfactorily demonstrated, request will be made to operate the reactor at higher power levels. The SADE loop will resume operation after the above experimental runs in VBWR are completed which is expected to be during the month of November 1960. The next superheater fuel element to be irradiated in the SADE loop will be a non-free standing clad annular type element.

A conceptual design arrangement has been completed for an expanded SADE facility. The modification would incorporate nine SADE type fuel elements replacing 4 VBWR fuel elements. Design conditions assumed for the loop are as follows:
Total power output 675 KW
No. of fuel elements 9
Design steam flow 13,500 lbs/hr
Design steam outlet temperature 1000°F
Design steam outlet pressure 1000 psi

Corrosion Loop Experiments

The superheated steam test loop was placed in operation in June 1960. The first superheat corrosion tests were performed to evaluate recombination of H₂ and O₂. Preliminary data obtained to date indicate that a significant amount of free H₂ and O₂ gasses recombined in traversing the superheated steam test section.

b. Once-Through Heat Transfer Experiments – S. Levy

The objective of this program is to study once-through heat transfer and to obtain sufficient information for a preliminary once-through reactor design. In the "once-through" reactor, water enters the core in the subcooled state and exits in a superheated condition. The evaluation of the once-through superheat concept has lagged behind that of other superheat reactors primarily because its heat transfer performance could not be predicted.

In order to design a once-through reactor, data of the type shown in Figures 1 and 2 are needed. The behavior of the heated rod beyond the critical steam quality point is controlling. Heat transfer coefficients, temperature increases, and oscillations must be known to obtain a sound design from the heat transfer viewpoint. A preliminary search of literature showed that data in the region of interest were not available at low or high pressure upon which the basis for a once-through reactor could be designed. This study was, therefore, undertaken to obtain such a basis.

Operation and Test Results

The once-through heat transfer tests were performed in the large heat transfer facility shown in Figure 3. The test section is shown in Figure 4. A typical operating temperature trace is shown in Figure 5. Figure 5 shows that at time zero all the thermocouples were slightly above the saturation temperature. At a time of 10 seconds, the power was raised and the temperature
at the top thermocouple No. 2 started to rise. Successive power increases caused thermocouple No. 6 below thermocouple No. 2 to become wavy at 40 seconds and, finally, thermocouple No. 11 became unsteady at 130 seconds. The power was turned off when thermocouple No. 2 approached 1000°F. Departure from boiling conditions did not always occur from top to bottom of the heated rod and that sometimes transition took place in the midsection and even inlet of the rod. This is understandable because the transition from boiling to film is expected to be sometimes delayed.

Similar traces were obtained at different flow rates, inlet subcooling, and pressure. A total of 140 runs with peak heater temperatures of approximately 1000°F has been taken up to October 10, 1960, and each of these runs has several preceding steady state runs associated with it. The range of variables covered includes pressures from 800 to 1400 psia, flow rates from 0.7 to 1.9 x 10^6 lbs/hr ft^2, steam qualities from 20 to over 100 percent. In addition, two heater rods producing different heat inputs per unit length were used to study the effects of heat flux.

Discussion of Results

Examination of Figure 5 shows that it is possible to operate at high steam qualities beyond previously called "burnout" points without reaching excessive heater temperatures.

The effects of flow, pressure, steam quality, and heat input per unit length in a once-through system can be obtained by plotting the heat flux which terminated the run against the corresponding exit steam quality. Figure 6A shows the effects of flow and pressure. It is noticed that the pressure effect is small for the two constant flow rates shown and that the limiting heat flux increases as the flow rate is raised. In both instances, the limiting heat flux is below those that could be obtained for 100 percent of vaporization of water and 1000°F heater rod temperature. In Figure 6B, similar runs are plotted to show the effects of steam quality. It is observed that the limiting heat flux remains relatively constant as the quality is increased from 40 percent to about 100 percent. In Figure 7, data are plotted for two test sections with different heat input per unit length. It is observed that the limiting heat flux increases as the heat input per unit length increases.

Physical bases for some of the above results are believed to be as follows:

(1) The limiting heat flux values are below those of saturated steam because droplets of water still present in the steam
could be attempting to conglomerate next to the wall. The water will have lower velocity than the main stream of steam, and the steam film next to the wall will reflect this by yielding a lower heat transfer coefficient.

(2) As the flow is increased, the film velocity next to the wall increases and higher heat fluxes can be sustained.

(3) With increased heat input per unit length, the steam escaping from the heater surface increases and tends to carry with it water drops into the main stream so that the heat transfer coefficient improves and approaches that of saturated steam.

(4) As the pressure is raised, the steam properties improve and the heat transfer coefficient should increase. However, the water velocity is lower at increased pressure for the same steam quality, and this should reduce the coefficient. The net effect could be to maintain the coefficient at a constant value.

While Figures 6A, 6B, and 7 show the effects of key design parameters, they do not describe in detail what occurs in a once-through system. The temperature oscillations and heat transfer coefficients are obtained from analysis of conditions at positions besides the exit point. Examination of temperature traces reveals that they follow three types of behavior. In the first type of trace, the temperature slowly increases from boiling temperature to film temperature. Temperature oscillations during such a run are small. A typical such transition is shown in Figure 8. The second type of transition is shown in Figure 5. The behavior follows that of Figure 8 except that the temperature oscillations are greater in the transition zone. A typical trace of this type has already been shown in Figure 5. The last type of trace is given in Figure 9. The temperature rises much more rapidly and does not stabilize before the power is reduced to protect the heater rod. Once-through system should not be operated with conditions that yield traces of the type of Figure 9. On this basis, a plot of all runs that gave traces of the first two types above can be used to define a design zone for once-through reactors. This is shown in Figure 10, and it is noted that the allowable heat flux value increases approximately with the square root of flow.

The wall temperatures within this acceptable operating zone can then be obtained from measured heat transfer coefficients. A typical plot of such coefficients at a mass flow rate of 0.77 x 10^6 lbs/hr ft² and 1400 psia is shown in Figure 11. The boiling, transition, and film regions are clearly visible in this plot.
Conclusions

(1) Design of a once-through reactor is possible, and data have been obtained upon which such a design can be based.

(2) Once-through reactors can be built to operate beyond the so-called "burnout" points because convective film coefficients are high enough to maintain the heated wall temperature below accepted values. High performance can be obtained on this basis and heat flux values of 300 to 1,000,000 Btu/hr-ft$^2$ can be used.

(3) The performance of all once-through systems can be improved through increased flow. It is relatively insensitive to pressure for pressures between 800 and 14,000 psia.

c. Steam-Water Separation Development - C. H. Robbins

The objectives of the General Electric superheat steam separation program are to develop primary and secondary type separators suitable for nuclear superheat application. Free surface testing has not been part of the superheat program, but results in this field obtained with General Electric money and with AEC funds on T-7 Tanker work were discussed.

Two interim reports (GEAP-3564 and GEAP-3563) giving results obtained on the superheat steam separation work were distributed at the meeting.

(1) Test Facilities

The major test facilities consist of a large steam-water loop located at the Moss Landing Power plant and an air-water loop at San Jose. Slides of the facilities were shown at the meeting. The steam-water loop design conditions include:

- Design pressure: 1250 psig
- Steam flow: Up to 60,000 lbs/hr
- Water flow: 600,000 lbs/hr or 1700 gpm at 150 ft head

The air-water facility is used to screen work for the Moss Landing loop. The water and air flows are comparable to the Moss Landing loop, and the test vessels are essentially the same dimensions. Results to date indicate that performance with the air-water loop gives good prediction of performance in the Moss Landing steam separator loop.
(2) **Free Surface Separation**

In work financed by General Electric, free surface separation tests were conducted at 1000 psig with 4-10 percent core exit quality. One configuration used had a 12-inch chimney in a 19%2-inch vessel. Another had a 24-inch chimney in a 36-inch vessel. Results were the same for both sizes.

(a) The maximum steam leaving velocity for good separation was about 1.1 ft/second.

(b) The minimum height above the indicated water level for good separation was about 32 inches.

These results will be reported in a forthcoming ANS paper by H. D. Ongman.

Further work on free surface separation was performed for the AEC for the T-7 Tanker. Tests agreed with the other on the maximum steam leaving velocities. A movable probe inside the vessel took samples to determine the variation of quality as a function of radius and height. Some tests on carry-under were made but with inconclusive results. Water level seemed to have more effect than downcomer velocity on carry-under. The program was terminated before enough data could be obtained to yield conclusive results. In tests conducted, General Electric got 1 percent carry-under with a water velocity as low as 0.5 ft/second.

(3) **Dryers**

Development has been conducted on two types of dryers.

(a) Large dryers without specific application.

(b) Dryers designed to handle steam before it enters a superheated fuel element and integral with the element.

Work financed by General Electric on 20-inch diameter wire mesh 12 inches thick showed that demisters were satisfactory.

In the superheat program, tests were begun with the same type of mesh, 12-inch diameter by 12-inch thickness. Performance was unsatisfactory when the dryers were tested in a horizontal position, but performance was improved when the lower ends of the dryers were tilted 15° toward the periphery. The tilted dryers appear to provide better drainage characteristics of the separated water than the horizontal dryers. The tilted dryer could handle 3 percent inlet moisture at an inlet steam velocity of 7 ft/second at 1000 psi saturated steam conditions.
Two different size (9" OD x 4.5" ID and 6 3/4" OD x 2-3/4" ID) dryers for a dummy fuel element were tested in the Moss Landing loop during the summer of 1960. Both dryers were 12-inch thick mesh. Tests were run with the dryers sloped 15° toward the periphery and 30° toward the centerline of the assembly. Test results of the dryers in these shapes did not perform as well as the 12-inch diameter demister tilted 15°. The dryers did not perform well with inlet steam velocities greater than about 5 ft/second. Dryer requirements are about three times this much for the reference 300 MWe design.

Future plans consist of developing a dryer for a fuel bundle which will handle higher inlet moisture and flow rates. General Electric is currently designing a two-stage dryer for a fuel bundle that will have a centrifugal dryer followed by a demister.

(4) Primary Type Separators

The objective of this task is to develop primary separators that handle the kind of steam-water mixture leaving the core of a BWR. The following separators were tested under the superheat program:

(a) Archimedes Spiral (Figure 12)

Mixture enters at the bottom and discharges into a spiral passage and eventually reaches a cylindrical shroud. A cap over the top with openings near the outside is provided to allow steam to escape. Water leaves at the bottom. Under air-water test, this separator handled about 13,000 ft³/hr with a pressure drop of about 7-8 psi and carry-under equivalent to about 0.2 percent steam at 1000 psi.

(b) Internal Tangential Upflow Separator

Mixture enters from the bottom and discharges against a cylindrical skirt. Water leaves at the bottom and steam leaves at the top of the separator. Under air-water test, this separator handled about 13,000 ft³/hr with a pressure drop of 11 psi and equivalent carry-under of approximately 0.2 per cent steam at 1000 psi.

(c) Turbo Separator

A Combustion Engineering separator was loaned to General Electric under the AEC Superheat Program to be tested in

* Volumetric flow combines the variables of steam flow and water flow.
the air water loop. This separator handled about 18,000 ft³/hr of air-water with a pressure drop of 8 psi and equivalent carry-under of about 0.6 percent steam at 1000 psi. Neither the Archimedes spiral nor the internal tangential separator could handle as much flow as the Combustion Engineering turbo separator. On basis of the air water tests, General Electric does not plan any further development on the Archimedes or internal tangential separators since they do not show enough promise of being better than separators that are already available.

(d) Multi-Arc Separator (Figure 13)

The separator consists of a series of 150° arcs arranged vertically around the periphery of the reactor vessel. Steam-water mixture from the core enters through nozzles. The centrifugal force is supplied by the arcs. Steam leaves from the top. Water can leave from the bottom or at the open end of the arcs where it is angled toward the reactor vessel. The reactor vessel thus provides a surface to aid in disposing of the water.

In air-water tests, various nozzles were tested which required about 16 inches between the inner and outer circumference. The arcs were spaced three inches apart and the ends discharged about two inches from the simulated reactor vessel. Air-water tests showed that the separator would handle about 2460 ft³/hr per inch of inlet circumference with a pressure drop of 3.5 psi and carry-under of 0.1 percent. Steam-water tests were subsequently made at Moss Landing at a pressure of 1000 psig. Preliminary results indicate that the maximum steam-water capacity is 2900 ft³/hr per inch with carry-under of 0.3 percent and pressure drop of 6.1 psi. These are very encouraging results and indicate that this type of separator is the best kind tested for the space available around the edge of a reactor vessel.

Further development work will continue on the multi-arc separator. Primary emphasis will be placed on determining the effects of reducing the arc size and inclining the arcs.

d. Fuel Element Development - C.N. Spalaris

(1) Fuel Capsule Irradiation Program

The objectives of this program are to: (1) study the irradiation behavior of fuel elements fabricated with swaging over sintered pellets, (2) study mechanical
stability and integrity of the cladding at high temperatures, and (3) determine fission gas release from UO₂ during irradiation.

The first of three capsules was irradiated in the GETR trail cable facility for 35 days in a thermal neutron flux of 1.85 x 10¹³ NEV. The capsule is 36 inches long and contains four fuel specimens of 304 SS cladding swaged over UO₂ pellets. The fuel specimens are annular and consist of different cladding thicknesses ranging from 0.016 to 0.019 inch.

Fission gas release was determined for the four specimens and the data are shown in Table I.

**TABLE I**

<table>
<thead>
<tr>
<th>Element</th>
<th>Q/A Max.</th>
<th>Max. Fuel</th>
<th>% F. G. Release</th>
<th>Ave. Exposures</th>
</tr>
</thead>
<tbody>
<tr>
<td>B</td>
<td>5.17 x 10⁵</td>
<td>0.57</td>
<td></td>
<td></td>
</tr>
<tr>
<td>D</td>
<td>3.81 x 10⁵</td>
<td>0.51</td>
<td></td>
<td></td>
</tr>
<tr>
<td>A</td>
<td>3.79 x 10⁵</td>
<td>0.14</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

The percent of the amount of fission gas theoretically produced was calculated from Kr-85 experimentally determined. As noted in Table I, the fission gas release is a function of the fuel operating temperature.

(2) **Trail Cable Irradiation Tests**

The objective of this program is to investigate the plastic strain cycling mechanism of cladding materials and to establish the magnitude of deformation of thin clad annular fuel elements.

The experimental program consists of subjecting thin cladding (collapsible) in a 1000 psi system to repeated cycles of growth and skrinkage during reactor operation. The variables being investigated consist of (1) UO₂ - jacket gap, (2) thickness of cladding, (3) heat flux, and (4) axial gap. Both aluminum and stainless steel specimens are being tested. The use of aluminum cladding is a device to simulate the characteristics of stainless steel at 1200°F in a room temperature test. Results obtained to date were shown in slides at the meeting. These are also presented in the fourth quarterly report, GEAP 3538.
(3) **Superheater Fuel Fabrication Development**

(a) **UO₂ Erosion Studies**

The objective of this task is to determine the leakage of UO₂ (fabricated by powder compaction) through clad defects under static and dynamic superheated steam conditions. The results from the static tests which were performed under several (15) rapid-decompression cycles at high temperatures from 1000 psi to atmospheric showed that no UO₂ powder escaped from a 0.040 inch hole in the cladding. The specimens were swaged to 70 percent theoretical density with high temperature fired grade of UO₂.

Dynamic tests were carried under superheated steam of 1050°F, 1000 psi, and velocities of about 100 ft/second. The test was run for a total exposure of 781 hours in an out-of-pile loop. During the run, additions of oxygen were made to maintain levels expected in nuclear superheat steam environments. The specimens contained about 125 grams of Arc fused UO₂ powders compacted and swaged to 91 percent theoretical density. The loss in UO₂ to the coolant stream varied from about 67 grams for a sample with a slit defect (3' x .040") to 2.6 grams for a sample with a round 50 mil hole defect.

Additional tests with UO₂ pellets and powder are currently under way and should be completed by the latter part of November 1960.

(b) **Process Tube Development**

The objective of this task is to develop an insulated, low neutron absorber tube to be used for separating superheated steam from the water moderator.

(c) **Stainless Steel Process Tube**

Two prototypes consisting of two thin wall stainless tubes in an annular configuration (100 mil annulus) have been fabricated. The annulus has been filled with a mixture of Al₂O₃ powders (three mesh sizes) and vibratory compacted to an average density of 77 percent of theoretical. Dimensions are as follows:

- **Outer tube**: 1-5/16" OD x .010" wall
- **Inner tube**: 1-1/2" OD x .006" wall
- **Length of tubes**: 30"
One of the prototypes sustained a room temperature hydrostatic pressure of 1000 psi without any significant dimensional changes or wrinkling of the outer tube.

The other prototype was given a room temperature hydrostatic pressure differential test of 60 psi and the measured volumetric changes in the center of the tube indicated that no ovality occurred. When this prototype was removed from the pressure vessel for inspection, two pin hole leaks were found in the outer (6 mil) tube at the weldment. Since the test had been of sufficient duration for pressure equilibrium to be attained, it may be presumed that the inner wall sustained the 60 psi pressure and constricted in a normal manner.

The next objective will be to duplicate these results at 650°F. In addition, the differential test pressure of the defected tube will be raised to 200 psi. A pressure difference of this magnitude may result from the combined effects of the design operating differential (50 psi) and the differential due to a rupture in the steam lines (150 psi). The ultimate objective is a tube utilizing two .007 inch walls with a 60 mil annulus of Al₂O₃, having adequate structural stability for use in superheat reactors.

(d) Fuel Irradiation in SADE Loop

The sequence and purpose of irradiating the next three fuel elements in SADE has been revised as follows:

SH-4 - This will be the first element operated in SADE since the VEWR modifications. This element has been conservatively designed and will be tested primarily to get a fix on the operating parameters in the reactor.

SH-1 - Will contain a purposely defected fuel jacket, 0.03 inch hole at the upper plenum. The purpose of this test will be to study fission product release into the system and the stability of UO₂ under conditions for erosion resistance.

SH-5 - The element will contain a fully nonself-supporting jacket, 0.016 inch thick, both on the inner and outer jacket. The purpose will be to study mechanical performance of the cladding and fretting corrosion.

Each of the above elements will be irradiated for about one month at full power conditions.
CONVECTION

NUCLEATE BOILING

CONVECTIVE BOILING

CONVECTIVE FILM BOILING

STEAM

HEAT TRANSFER COEFFICIENT

SUBCOOLED WATER

STEAM-WATER

SUPERHEAT STEAM

SATURATED WATER

ENTHALPY

SATURATED STEAM

HEAT TRANSFER IN ONCE-THROUGH SYSTEM

FIG. 1
VARIATION OF HEAT TRANSFER COEFFICIENT IN ONCE-THROUGH SYSTEM AT CONSTANT FLOW AND VARIOUS CONSTANT HEAT FLUX VALUES

FIG. 2
FLOW DIAGRAM

HEAT TRANSFER & FLUID FLOW TEST FACILITY

FIG. 3
P = 800 psi
G = 0.755 \frac{lb}{hr-ft^2}

\[
\frac{q/A}{\text{final}} = 325,000 \text{RTU/hr-ft}^2
\]
\[
\frac{q/A}{\text{initial}} = 275,000 \text{RTU/hr-ft}^2
\]
HEAT FLUX BASED ON COLBURN EQUATION AT SAT. STEAM AND 900°F HEATER

1. MAXIMUM FLOW VARIATION — ±20% .
2. TEST SECTION: 0.625 IN. O.D. HEATER, 40 IN. LG., RUN AT ABOUT 900°F, 0.120 IN. FLOW ANNULUS.

EFFECT OF PRESSURE AND MASS VELOCITY ON LIMITING HEAT FLUX

FIG. 6-A
STEAM QUALITY EFFECT

FIG. 6-B
1. MAXIMUM FLOW VARIATION - ±10%
2. TEST SECTION: 0.625 IN O.D. HEATER, 40 IN. LONG, RUN AT ABOUT 900°F
   0.120 IN FLOW ANNULUS

EFFECT OF HEAT INPUT PER UNIT LENGTH ON LIMITING HEAT FLUX

FIG. 7
P = 1100  G = 0.655  \frac{LB}{HR\cdot FT^2}  \quad q/A_{\text{initial}} = \frac{445,000}{HR\cdot FT^2} \quad q/A_{\text{final}} = \frac{554,000}{HR\cdot FT^2}

TYPICAL SUDDEN TRANSITION

FIG. 9
LIMITING HEAT FLUX IN TERMS OF MASS VELOCITY FOR RUNS WHERE HEATER ROD TEMPERATURE STABILIZED AFTER TRANSITION AT ABOUT 1000 °F

FIG. 10
TYPICAL HEAT TRANSFER COEFFICIENTS FOR SMOOTH TRANSITION

FIG. 11

- \(0.769 \times 10^6 \frac{LB}{HR\cdot FT^2}\) & 1400 psi
- \(0.772 \times 10^6 \frac{LB}{HR\cdot FT^2}\) & 1400 psi

COLBURN COEFFICIENT AT SATURATION
STEAM WATER MIXTURE

SHROUD

ARCHIMEDES SPIRAL

FIGURE 12

STATUS
DEVELOPED & TESTED UNDER AEC CONTRACT
STEAM SEPARATOR TYPES

STATUS

DEVELOPED & TESTED UNDER AEC CONTRACT. STILL IN DEVELOPMENT STAGE.

RADIAL FLOW
(2-ARC.)
(MULTI-ARC.)
STEAM SEPARATOR TYPES

FIGURE 13
**Intermediate Geometry Experiments**

The purpose of this program was to provide criticality and power distribution data using (BONUS) rod type fuel elements for comparison with present analytical techniques as applied to integral boiler-superheater reactors. This set of experiments was initiated in June 1960 and was completed in the early part of October 1960.

The boiler fuel elements for these experiments consisted of natural and 1.85 percent enriched UO$_2$ pellets clad with aluminum. The superheater elements contain 3.41 percent enriched UO$_2$ pellets clad with stainless steel. The types of cores investigated are as follows:

(a) Pure boiler type cores.

(b) Boiler type cores with an external superheat region.

(c) Boiler type cores with an internal superheat region.

A detailed description of the various core configurations and results obtained to date were presented at the meeting and are summarized in Table I, page 32.

**Clean Geometry Experiments with Rod Type Fuel Elements**

This set of experiments was initiated about October 10, 1960, and will continue until approximately December 1, 1960. The purpose of these experiments is to study single and two region cores, in slab geometry, with a uniform fuel rod spacing within each region. Work will be concentrated primarily on measurements leading to experimental values of resonance escape, thermal utilization, and fast fission effects. The following cores will be studied using the fuel elements from the intermediate geometry experiments.

(a) Boiler type core with 625 mil pitch.

(b) An internal superheat core with a 1.094 inch square pitch in the superheater region and a 625 mil pitch in the two boiler regions. This core will be studied under both flooded and voided superheater conditions.
### TABLE I
SUMMARY OF INTERMEDIATE GEOMETRY EXPERIMENTS

<table>
<thead>
<tr>
<th>Boiler Region</th>
<th>Superheater Region</th>
<th>Special Notes</th>
<th>Critical Height (In.)</th>
</tr>
</thead>
<tbody>
<tr>
<td>4 x 5</td>
<td>None</td>
<td></td>
<td>35.3</td>
</tr>
<tr>
<td>4 x 6 minus corner assemblies</td>
<td>None</td>
<td></td>
<td>35.2*</td>
</tr>
<tr>
<td>4 x 6</td>
<td>None</td>
<td></td>
<td>26.9*</td>
</tr>
<tr>
<td>4 x 6</td>
<td>None</td>
<td>16 - 1/2&quot; displacers per cell</td>
<td>33.8</td>
</tr>
<tr>
<td>4 x 6</td>
<td>None</td>
<td>4 - 1/2&quot; displacers per cell</td>
<td>31.6</td>
</tr>
<tr>
<td>4 x 6</td>
<td>None</td>
<td>34 element cells (2 diagonal corners missing)</td>
<td>34.5</td>
</tr>
<tr>
<td>4 x 6</td>
<td>None</td>
<td>36 element cells</td>
<td>26.3*</td>
</tr>
<tr>
<td>4 x 6</td>
<td>None</td>
<td>1/8&quot; x 8&quot; x 6' aluminum blade</td>
<td>27.3</td>
</tr>
<tr>
<td>4 x 8 minus corner assemblies</td>
<td>None</td>
<td>20 mil x 7&quot; Cd on 1/8&quot; aluminum blade rod</td>
<td>36.8</td>
</tr>
<tr>
<td>4 x 6</td>
<td>None</td>
<td>40 mil x 7&quot; Cd on 1/8&quot; aluminum blade rod</td>
<td>37.5</td>
</tr>
<tr>
<td>4 x 6</td>
<td>None</td>
<td>1/8&quot; x 7&quot; boron steel blade rod</td>
<td>39.0</td>
</tr>
<tr>
<td>4 x 6</td>
<td>None</td>
<td>1/4&quot; x 7&quot; boron steel blade rod</td>
<td>42.9</td>
</tr>
<tr>
<td>4 x 6</td>
<td>4 boron pins, one per cell at core center</td>
<td>35.1</td>
<td></td>
</tr>
<tr>
<td>4 x 6</td>
<td>8 boron pins, two per cell at core center</td>
<td>48.4</td>
<td></td>
</tr>
<tr>
<td>4 x 6</td>
<td>8 boron pins, two per cell at core center, 36 element cell</td>
<td>44.6*</td>
<td></td>
</tr>
<tr>
<td>4 x 8 minus corner assemblies</td>
<td>None</td>
<td>4 natural U fuel rods at corners of each cell</td>
<td>39.0*</td>
</tr>
<tr>
<td>6 x 6 minus corner assemblies</td>
<td>None</td>
<td>4 central assemblies with natural U fuel rods</td>
<td>42.6*</td>
</tr>
<tr>
<td>1 1/2 x 8 each side</td>
<td>Internal</td>
<td>8 full and 8 one-half assemblies</td>
<td>37.3</td>
</tr>
<tr>
<td>1-3/4 x 8 each side</td>
<td>Internal</td>
<td>12 full and 4 one-half assemblies on each side</td>
<td>28.0*</td>
</tr>
<tr>
<td>2 x 8 one side</td>
<td>1/8&quot; x 7&quot; boron steel blade on each side of superheater</td>
<td>31.6*</td>
<td></td>
</tr>
<tr>
<td>2 x 8 + 1/2 x 6 other side</td>
<td>None</td>
<td>23.5*</td>
<td></td>
</tr>
<tr>
<td>3 x 8</td>
<td>External</td>
<td>1&quot; channel</td>
<td>23.7*</td>
</tr>
<tr>
<td>3 x 8</td>
<td>External</td>
<td>1&quot; channel, two 1/8&quot; x 7&quot; boron steel blades</td>
<td>31.5*</td>
</tr>
<tr>
<td>3 x 8</td>
<td>External</td>
<td>2&quot; channel</td>
<td>29.7</td>
</tr>
<tr>
<td>5 x 8</td>
<td>External</td>
<td>Two 1/8&quot; x 12&quot; boron steel blades fully inserted plus four 1/8&quot; x 7&quot; boron steel cruciforms</td>
<td>39.8</td>
</tr>
<tr>
<td>5 x 8</td>
<td>External</td>
<td>As above, boron blades at 15.2&quot;</td>
<td>41.0</td>
</tr>
<tr>
<td>5 x 8</td>
<td>External</td>
<td>As above, boron blades fully withdrawn</td>
<td>30.4*</td>
</tr>
<tr>
<td>7 x 8</td>
<td>External</td>
<td>6 cruciform rods plus 2 - 12&quot; blades at 24&quot;</td>
<td>46.0</td>
</tr>
<tr>
<td>7 x 8</td>
<td>External</td>
<td>6 cruciform rods plus 2 - 12&quot; blades fully withdrawn</td>
<td>40.6</td>
</tr>
</tbody>
</table>

*Note: Asterisk indicates cores on which power distribution measurements were made.
Double Annular Fuel Element Experiments

This series of experiments is scheduled to begin in December 1960 and is primarily concerned with establishing the basic physics properties of double annular fuel elements in a single region of uniform pitch. Measurements of critical loadings, flooding coefficients, buckling, migration area, and lattice parameters will be made in arrays which simulate the cold and hot atomic ratios of the NUSU reactor.

The next phase of the experimental program will be to investigate the properties of 19 central NUSU clusters, each containing 19 fuel elements. The basic program will be to obtain criticality data, flooding coefficients, power distributions, and control rod worths.

BONUS

A series of experiments with a mock-up BONUS core is scheduled to begin in the early part of 1961. These experiments will be carried out in a second cell at the Windsor, Connecticut, critical facilities. At present, the control console is being installed, and hardware for the cell interior is being designed and fabricated.

(2) Gravity Steam Separation Tests - P. C. Zmola

Steam separation tests were required to establish separation arrangements for use in connection with NUSU. Gravity separation tests were initiated first since:

(a) The present NUSU reference design was based upon a gravity steam separation arrangement.

(b) Gravity separation is a yardstick by which performance of mechanical separation devices will be measured.

(c) Gravity separation may be combined with other separation apparatus to provide required steam conditions.

Tests were run at system pressures of approximately 100, 300, 500, 750, 1000, and 1250 psi. Steam release rates were varied from 1300 to 13,000 lbs/hr sq ft, circulation ratios ranged from five to 25, and initial water levels over the steam-water discharge nozzle were varied from one to 12 inches.

The steam-water mixture was introduced through a nozzle near the bottom of the test region. The cross sectional area of the test region was 1.97 square feet. The nozzle area was
1.07 square feet. The discharge area of the nozzle (122 3/4-inch holes) was .375 square feet. In the steam space above the water level, sample lines were installed to check steam conditions for progressive drop-out. System water conductivity was maintained at approximately 1500 microhms sufficient to give satisfactory conductivity values without foaming. Measured steam and water flows were established and conductivities of the various samples were recorded. Conductivity ratios (sample conductivity to loop water conductivity ratio) were used to determine the wetness of the steam at various elevations in the steam space.

A considerable quantity of data has been obtained and plotted. Figures 30-7 through 30-11 are representative curves which were selected to show, in brief form, the effect of such parameters as steam release rate, circulation ratio, system pressure and nozzle submergence on steam quality at different heights above the water level. It should be pointed out that this is recently obtained data and, because of this, a thorough evaluation of these results has not been made.

Figure 30-7 shows the effect of system pressure on fall-out for a constant steam release rate and discharge nozzle submergence. As system pressure increases, the height required for fall-out decreases. However, the pressure effect does not appear very strong between 500 and 1250 psi.

Figure 30-8 shows the effect of circulation ratio on fall-out for a constant nozzle submergence. At higher circulation ratios, the required disengagement height increases.

Figure 30-9 shows the effect of steam release rate on fall-out for a constant nozzle submergence. At higher steam release rates, the required disengagement height is greater.

The effect of nozzle submergence on fall-out is shown in Figures 30-10 and 30-11. It can be seen that the required height for steam-water disengagement increases for greater nozzle submergence. The reason for this is not completely understood at this time. It is hoped that testing with a modified nozzle design arrangement can further answer some questions which have been raised by this trend.

Some difficulty in operation at 1250 psi was experienced due to system leaks. Because of this, some of the 1250 psi tests at high steam release rates and circulation ratios were not completed. It was decided to proceed with modifications of the nozzle assembly to check out the "submergence effect" on a nozzle arrangement which will more closely mock-up the latest NUBU reference design arrangement. While these modifications are being made,
it is planned to initiate tests on a compact mechanical separation arrangement.

While it is premature to draw major conclusions from the gravity separation tests run thus far, the data indicate that the general concept of steam separation used in the NUSU reference design is reasonable. However, until more complete testing at NUSU conditions has been made and a better understanding of the effect of water level has been obtained, the final adequacy of the NUSU reference design cannot be completely established.

(3) Steam Purity Test Program – P. C. Zmola

A steam loop has been built to produce saturated steam at pressures up to 1350 psig with flow rates up to 400 pounds per hour. This steam can then be superheated electrically to 1060°F in a test section which simulates the central hole in the boiler-superheater fuel element for the NUSU reference design. The feedwater is demineralized and degassed prior to pumping into the boiler. The effluent steam is discharged to waste. Provisions have been incorporated for sampling the water from the boiler and the steam condensate before and after the test section.

The attached NUSU steam purity figures show the loop flow schematic, test section components, and axial temperature distributions obtained along the test section during operation.

<table>
<thead>
<tr>
<th>Figure No.</th>
<th>Title</th>
</tr>
</thead>
<tbody>
<tr>
<td>III-4</td>
<td>NUSU Steam Purity Flow Diagram</td>
</tr>
<tr>
<td>III-3</td>
<td>NUSU Steam Purity Test Loop During Assembly</td>
</tr>
<tr>
<td>III-4</td>
<td>NUSU Steam Purity Test Section Assembly</td>
</tr>
<tr>
<td>III-5</td>
<td>Detail of Steam Purity Test Section</td>
</tr>
<tr>
<td>1</td>
<td>NUSU Steam Purity Test Section Assembly</td>
</tr>
<tr>
<td>III-2</td>
<td>NUSU Steam Purity Test Section Enclosure Assembly</td>
</tr>
<tr>
<td>30-4</td>
<td>NUSU Steam Purity Test Section, Predicted and Experimental Temperature Distributions</td>
</tr>
<tr>
<td>30-3</td>
<td>NUSU Steam Purity Test Section, Outer Wall Axial Temperature Distributions for Constant Flow Rate and Varying Electrical Power Input</td>
</tr>
</tbody>
</table>
The objective of these tests is to determine the steam purity requirements for the 200 MWe NUSU boiling-superheating reactor. The results of these tests will aid in selecting the steam separator design and will help to determine reactor water purity requirements. The tests may also indicate certain conditions which are or are not feasible for superheating fuel element designs.

In addition, the transport of radioactive nuclides has been and will continue to be an important area of investigation for nuclear systems. The problems encountered in a nuclear superheating core due to transport into or out of the fuel element may be magnified because of the higher temperatures of operation and because of the tendency for plating out of water-borne impurities. The potential magnitude of the transport problem is being investigated as part of this program. Radioactive tracers are being used to aid in measurements.

Test Conditions

The test conditions are intended to simulate the operating conditions of the NUSU 1100 reference design. The operating parameters of the test loop are as follows:
(a) **Test Section Conditions**

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pressure</td>
<td>1350 psia</td>
</tr>
<tr>
<td>Steam temperatures</td>
<td></td>
</tr>
<tr>
<td>Inlet (saturated)</td>
<td>580°F</td>
</tr>
<tr>
<td>Outlet</td>
<td>1060°F</td>
</tr>
<tr>
<td>Wall temperature (max.)</td>
<td>1250°F</td>
</tr>
<tr>
<td>Heat flux (design)</td>
<td>114,000 Btu/hr/ft</td>
</tr>
<tr>
<td>Mass flow</td>
<td>220 lb/hr</td>
</tr>
<tr>
<td>Inlet velocity</td>
<td>63 ft/sec</td>
</tr>
<tr>
<td>Outlet velocity</td>
<td>125 ft/sec</td>
</tr>
<tr>
<td>Power input</td>
<td>DC resistance heating</td>
</tr>
</tbody>
</table>

(b) **Water Conditions**

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Resistivity</td>
<td>&gt; 1 megohm</td>
</tr>
<tr>
<td>pH</td>
<td>5.5 to 7.5</td>
</tr>
<tr>
<td>Total solids</td>
<td>10 ppm</td>
</tr>
<tr>
<td>Dissolved O₂</td>
<td>&lt; 0.5 ppm with no addition</td>
</tr>
</tbody>
</table>

(c) **Test Section Fabrication**

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Length</td>
<td>108 inches</td>
</tr>
<tr>
<td>ID</td>
<td>0.237 inches</td>
</tr>
<tr>
<td>OD</td>
<td>0.317 inches</td>
</tr>
<tr>
<td>Material</td>
<td>Type 347 SS</td>
</tr>
<tr>
<td>ASTM grain size</td>
<td>#10</td>
</tr>
<tr>
<td>Brazed joints</td>
<td>Gold-nickel (permabrace 130) alloy</td>
</tr>
<tr>
<td>Test coupons</td>
<td>3 half-inch lengths of test section tubing each end</td>
</tr>
</tbody>
</table>

**Raw Data**

The test loop has operated for approximately 200 hours at test conditions. The data are very preliminary and should not be interpreted as representative of expected results from the test program.

(a) **Data Summary from Radioactive Tracer p\(^{32}\)**

1. \[ DF = \frac{\text{Activity in autoclave water}}{\text{Activity in steam before test section}} \]

   Results range from \(.6 \times 10^3\) to \(30 \times 10^3\). This corresponds to a moisture content in the steam of from 0.003% to 0.2%.

2. Plating out of p\(^{32}\) through the superheat test section has ranged from 30% to 80%.
(b) **Total Solids**

1. Total solids in autoclave water - 8.5 ppm ≤ 2 ppm during run.

2. Total solids in steam, before and after test section - approximately 1.5 ppm.

The general trend during the run was a reduction of total solids in the system.

(c) **Water Conductivity**

The conductivity of the water ranged from 0.6 to 1.5 micromhos.

(d) **Test Coupons**

Test coupon measurements after 200 hours exposure:

<table>
<thead>
<tr>
<th>Test Section</th>
<th>Inlet</th>
<th>Outlet</th>
</tr>
</thead>
<tbody>
<tr>
<td>Net weight gain*</td>
<td>0.3 mg</td>
<td>1.2 mg</td>
</tr>
<tr>
<td>Net weight loss* (after cleaning)</td>
<td>0.4 mg</td>
<td>3.4 mg</td>
</tr>
<tr>
<td>Activity d/m/coupon</td>
<td>2360</td>
<td>3248</td>
</tr>
</tbody>
</table>

*Weights are referenced to the clean sample before the test began.

**Conclusions to Date**

The only conclusion which can be drawn from the data to date is that we are observing phenomena which are measurable within the accuracies readily available in the laboratory. Refinement in techniques and verification of observed results may lead to useful correlations in the design of nuclear superheat reactor systems.

**Future Plans**

A second test section has been fabricated and will be installed when the usefulness of the present test section has been exhausted. Conditions will be varied during each successive test to obtain the desired range of operating conditions to permit evaluation of the test results.
Fig. 1—NUSU STEAM PURITY TEST SECTION ASSEMBLY
Fig. III-2  NUSU STEAM PURITY TEST SECTION ENCLOSURE ASSEMBLY
FIG. III-3 NUSU STEAM PURITY TEST LOOP DURING ASSEMBLY
Fig. III-4. NUSU STEAM PURITY TEST SECTION ASSEMBLY
Fig. III-4 - NUSU STEAM PURITY FLOW DIAGRAM

SYMBOLS
SL SAMPLE LOCATION
SC SAMPLE COOLER
CI CONDUCTIVITY INDICATING
TW THERMOCOUPLE WELL
T I THERMOCOUPLE INDICATING
TIC TEMPERATURE INDICATING AND CONTROLLING
PC PRESSURE CONTROL
PR PRESSURE RECORDING
PI PRESSURE INDICATING
PCV PRESSURE REGULATING
RPC DIFFERENTIAL PRESSURE RECORDING & CONTROLLING
FI FLOW RECORDING & CONTROLLING
FI FLOW INDICATING
LG LEVEL GLASS
AMMETER, DC
VOLTMETER, DC
MULTI-POINT TEMP RECORDER
MULTI-POINT TEMP RECORDER
INJECTION PUMP

FLOW LEGEND
• WATER AMBIENT TEMP
• WATER 200°F
• WATER 575°F
• STEAM 1350 PSI CONTAMINATED
• STEAM 1050°F
• STEAM 1350 PSI 100% QUALITY
• PREHEATER VENT
• PREHEATER DRAIN

SAFETY INTERLOCKS
POWER SUPPLY SHUTDOWN
LOW FLOW
HIGH TEMPERATURE
COMPLETE LOOP SHUTDOWN
HIGH-LOW PRESSURE
HIGH-LOW FLOW
HIGH TEMPERATURE
Fig. 30-3  NUSU STEAM PURITY TEST SECTION—OUTER WALL AXIAL TEMP. DISTRIBUTIONS FOR CONSTANT FLOW RATE AND VARYING ELECTRICAL POWER INPUT
Fig. 30-4 NUSU STEAM PURITY TEST SECTION, PREDICTED AND EXPERIMENTAL TEMPERATURE DISTRIBUTIONS
Fig. 30-7  THE EFFECT OF PRESSURE ON FALLOUT

* CIRCULATION RATIO; WATER TO STEAM FLOW RATIO

DISCHARGE NOZZLE SUBMERSION = 1 INCH
STEAM RELEASE RATE ~ 7700#/HR-FT²
CIRCULATION RATIO 10⁻¹¹

○ PRESSURE = 300 PSIA
△ PRESSURE = 500 PSIA
■ PRESSURE = 1000 PSIA
◆ PRESSURE = 1260 PSIA

(SHADED SYMBOLS ARE EXTRAPOLATED DATA)
PRESSURE = 300 PSIA
DISCHARGE NOZZLE SUBMERGENCE = 1" 
STEAM RELEASE RATE ~ 7550 #/HR-FT^2

△ - C. R.* = 4.85
○ - C. R.* = 11.30
□ - C. R.* = 15.10

Fig. 30-8 THE EFFECT OF CIRCULATION RATIO ON FALLOUT

* C. R. : WATER TO STEAM FLOW RATIO
PRES sure = 505 PSIA
DIS CHARGE NOZZLE SUBMERGENCE = 1"

O-STEAM RELEASE RATE = 7,730 #/HR-FT²
□-STEAM RELEASE RATE = 8,390 #/HR-FT²
Δ-STEAM RELEASE RATE = 10,550 #/HR-FT²

CIRCULATION RATIO* = 10

Fig. 30-9  THE EFFECT OF STEAM RELEASE RATE ON FALLOUT

* CIRCULATION RATIO; WATER TO STEAM FLOW RATIO
PRESSURE = 1255-1270 PSIA
CIRCULATION RATIO* = 16.81-16.96
STEAM RELEASE RATE ~ 7700#/HR-FT²

- □ DISCHARGE NOZZLE SUBMERGENCE = 6"
- △ DISCHARGE NOZZLE SUBMERGENCE = 9"
- ○ DISCHARGE NOZZLE SUBMERGENCE = 12"

Fig. 30-10 THE EFFECT OF DISCHARGE NOZZLE SUBMERGENCE ON FALLOUT, 0.74 FT/SEC STEAM VELOCITY

* CIRCULATION RATIO; WATER TO STEAM FLOW RATIO
PRESSURE = 1255-1270 PSIA  
CIRCULATION RATIO* 8.29 - 8.45  
STEAM RELEASE RATE ~ 10,100#/HR-FT²  

- DISCHARGE NOZZLE  
  SUBMERGENCE = 6"  
  ○ - DISCHARGE NOZZLE  
  SUBMERGENCE = 9"  
  ▲ - DISCHARGE NOZZLE  
  SUBMERGENCE = 12"  

Fig. 30-11 THE EFFECT OF DISCHARGE NOZZLE SUBMERGENCE ON FALLOUT, 0.966 FT/SEC STEAM VELOCITY  

* CIRCULATION RATIO; WATER TO STEAM FLOW RATIO
b. BONUS Research and Development Program - F. Bevilacqua

(1) Structural Restraint of Fuel Elements

The purpose of this task is to establish the reference processes for fabrication of the BONUS superheater fuel elements.

A "stretch-forming" technique for collapsing the stainless steel cladding tube on the UO₂ pellets is being studied for applications to the fabrication of BONUS fuel elements. In a preliminary test, a type 347 stainless steel tube has been collapsed onto 57 inches of UO₂ pellets by the "stretch-forming" technique. The x-ray photographs of the collapsed element indicate that the stretching fixture was able to hold the pellet faces in contact during the stretch forming operations and that a satisfactory collapsed clad element was obtained. Further evaluation is required to determine the adequacy of this technique.

Tube-to-tube sheet welded joint specimens simulating a BONUS superheater fuel assembly have been made successfully by an internal welding technique. Successful mock-ups of microbraze tube-to-tube sheet joints have also been made. Microbraze shows promise of feasibility, plus the possibility of reduced costs relative to fabrication by welding.

As part of the structural restraint program, eight single rod fuel elements complete with concentric coolant tube and pressure tube will be constructed. These elements will then be tested in the environmental steam test loop under the BONUS R&D program. Fabrication of components for these elements is near completion.

(2) Heat Transfer Test Simulating Loss of Coolant

Zero Flow Shutdown Cooling Test

A single element test mock-up of the BONUS superheater geometry was built and operated to verify the ability of the superheater
to lose heat to the moderator water under conditions of zero steam flow at decay heat power levels and pressures from atmospheric to rated operating pressure. The facility and test element were shown in slides at the meeting.

Experiments were performed with the test element surfaces monitored for temperature as a function of heat flux at zero, 380, and 900 psig. Thimble and liner temperatures as a function of heat flux at 900 psig are shown in Figure 1.

The curves associated with the data points are calculated from a calculational model which considers two modes of heat transfer across the coolant and insulation annuli: radiation and natural convection where the natural convection component includes all conduction by definition. The radiation-convection heat-transfer split in the coolant annulus is characteristically 90-10 while in the insulation annulus, due to lower surface temperatures, the split is characteristically 50-50.

The calculational heat transfer model when applied to the exact BONUS geometry at 900 psig yields the temperature flux curves shown in Figure 2. The vertical dotted lines at 4000 and 12,000 Btu/hr-ft² clad heat flux represent the peak fluxes (i.e., heat flux at local hot spot) at a shutdown power of 2% and 6% of full power. The corresponding hot spot clad temperatures for these two points are 1000°F to 1500°F. A topical report covering this work will be published shortly.

(3) Water Injection Experiment

A test system was built and operated for the qualitative determination of the thermal, mechanical, and pressure effects on the superheater elements when the element is suddenly flooded with water. A general description of the system was given at the meeting.

The accidental flooding case simulated was the rupture of one of the eight superheater parallel flow annuli. Test section surface heat fluxes approaching 40,000 Btu/hr-ft² and exit surface temperatures of 1200°F to 1400°F were reached. Actual average surface fluxes and maximum surface temperatures in portions of the BONUS superheater region at full power fall within these ranges.

The initial conditions for two representative test runs at different injection pressures are shown below:

<table>
<thead>
<tr>
<th></th>
<th>Run No. 1</th>
<th>Run No. 2</th>
</tr>
</thead>
<tbody>
<tr>
<td>Average surface heat flux (Btu/hr-ft²)</td>
<td>35,800</td>
<td>35,800</td>
</tr>
<tr>
<td>Water injection pressure (psig)</td>
<td>48</td>
<td>20</td>
</tr>
</tbody>
</table>
Run No. 1 Run No. 2

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Run No. 1</th>
<th>Run No. 2</th>
</tr>
</thead>
<tbody>
<tr>
<td>Water injection temperature (°F)</td>
<td>175</td>
<td>130</td>
</tr>
<tr>
<td>Heater exit surface temperature, Tex (°F)</td>
<td>1,250</td>
<td>1,300</td>
</tr>
<tr>
<td>Heater inlet surface temperature, Tin (°F)</td>
<td>970</td>
<td>880</td>
</tr>
<tr>
<td>Exit steam temperature (°F)</td>
<td>400</td>
<td>390</td>
</tr>
<tr>
<td>Inlet steam temperature (°F)</td>
<td>240</td>
<td>240</td>
</tr>
<tr>
<td>Exit steam pressure (psig)</td>
<td>9.5</td>
<td>9.5</td>
</tr>
<tr>
<td>Inlet steam pressure (psig)</td>
<td>10.0</td>
<td>10.0</td>
</tr>
</tbody>
</table>

As a result of the qualitative nature of data expected from such a near-atmospheric mock-up of the 900 psig superheater geometry and, as a result of the simplified approach to duplication of superheater conditions (e.g., manipulation of hand-operated valves to simulate flow tube rupture), the values for time lapse between water injection and flooding of the inlet and outlet elevations of the test section should be considered as order-of-magnitude values. Listed below are the indicated time delays:

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Run No. 1</th>
<th>Run No. 2</th>
</tr>
</thead>
<tbody>
<tr>
<td>Between injection and flooding of heater exit (sec)</td>
<td>2.0</td>
<td>3.3</td>
</tr>
<tr>
<td>Between injection and flooding of heater inlet (sec)</td>
<td>3.1</td>
<td>6.0</td>
</tr>
<tr>
<td>Between exit flooding and inlet flooding (sec)</td>
<td>1.1</td>
<td>2.7</td>
</tr>
</tbody>
</table>

These tests indicate that a change in injection pressure from 148 psig to 20 psig causes a change in time delay for flooding the test section from 1.1 seconds to 2.7 seconds. Other runs at injection pressures approaching the 10 psig initial system pressure show increasing time delays beyond 3 seconds; during one run at an injection pressure of essentially 10 psig, the test section apparently reached a quasi-steady-state condition of incomplete flooding with a mixture flow of steam and water.

(4) Insulation and Seal Connector Development

A brief description was given of the various connectors (also shown in slides) under consideration for coupling the upper end of the BONUS superheater standpipe to the steam dryer and steam headers. The connectors under consideration are:
(a) Modified Marman Conoseal.
(b) Modified Dahl clamp.
(c) DSD temperature compensating coupling.

Upon completion of the shutdown cooling test, the above type and perhaps other types of seals will be leak tested at simulated BONUS steam conditions in the test vessel.

(5) Control Development

Tape Drive Control System

The purpose of this program is to establish the feasibility of this method of control rod operation at room temperature and pressure.

The following are the initial tests to which this drive was subjected and the results:

Tape drive was cycled and scrambled as follows:

I. 1250 scrams, channel aligned, no simulated boiling.
II. 1250 scrams, channel misaligned 1/8", no simulated boiling.
III. 1250 scrams, channel aligned, 1.5 cfm of air.
IV. 1250 scrams, channel misaligned 3/8", 1.5 cfm of air.

During these 5000 scramble cycles which represent in 20 years of operation one scram every other week, rod position versus tape tension versus time data were obtained to detect variations in performance. At the conclusion of each phase, the rod guide assembly was disassembled, inspected, and wear measured. Satisfactory behavior with little wear or change in performance occurred.

The final test phase of the tape-operated control rod consisted of 50 full travel scrams for each of the following conditions:

(a) Rod moving up under motor operation, pulley free, then stuck, stopping the motor while locking the pulley.
(b) Rod moving down under motor operation, pulley free, then stuck, stopping the motor while locking the pulley.
(c) Rod moving down under motor operation with the rod free, then sticking.
(d) Rod moving up under motor operation with the rod free, then sticking.

(e) Rod falling freely for part of its full travel, then sticking.

(f) Rod falling freely with damper disconnected.

The highlights of the information obtained are as follows:

(a) The control rod moves satisfactorily into and out of the guide structure with the pulley stuck under both motor and scram operation for all the pulley and tape surface conditions and different reactor atmospheres tested.

(b) For the worst conditions, the full-travel (51-inch) scram time was increased from 1.30 seconds with a free pulley to 3.50 seconds with a stuck pulley. However, the scram time to 37-inch travel was only 1.17 seconds for this condition.

(c) The instrument combination of rod position indicator and tape tension detector was capable of detecting either a stuck pulley or a stuck control rod.

(d) There was no measurable wear of the tape or pulley during any of the tests even though the same tape and pulley were used throughout.

(6) **Fuel Element Performance Tests**

Essentially, a steam loop test rig has been constructed which will allow superheated steam at 900°F and 850 psig to be passed through the coolant channel of the BONUS type fuel subassembly at near operational velocity. Isothermal performance of the test elements will be observed. Part of the test rig is also designed to surround the pressure tube of the subassembly with saturated water. This will allow checking of the structural stability of the subassembly with an external pressure up to 75 psi.

Fuel elements for this test rig are being fabricated as part of the NUSU program Activity No. 40-00-00. These elements should be available early in November 1960.

The test rig assembly is now being installed in the Florida Power Corporation, Bayboro Plant.

(7) **Fuel Unloading Experiments**

**General**

The superheater fuel assemblies for the BONUS reactor differ radically from conventional fuel assemblies used in water reactors.
Long inlet and outlet steam piping from each superheater fuel assembly must be connected to and disconnected from the saturated steam header and the superheated steam collection headers. In addition, these long pipes must be handled as part of the fuel assembly when loading or unloading the reactor. Since rapid and trouble-free operation during refueling is essential to maintaining a low operating cost and a high plant factor, it is important that superheater fuel handling devices be developed and tested under simulated operating conditions.

To accomplish the simulation of handling the fuel under reactor design conditions, a water tank mock-up of the BONUS pressure vessel will be constructed. In this tank, portions of the reactor core and other components will be mocked up.

Handling tools will be developed and constructed for use in the simulated fuel handling tests.

Functional reliability of the special tools will be checked by actually disconnecting and reassembling the superheater steam inlet and outlet connectors and by performing simulated fuel loading procedures.

Progress to Date

A 7' diameter x 22½' high water tank to simulate the BONUS vessel has been fabricated and erected outside the GNEC Laboratory. A weather structure is now being constructed.

Preliminary designs are now being made of simulated grid structure, fuel assemblies, and control rod structures.

Handling tools will be developed along with the detailed designs of the boiler and superheater fuel assemblies.

Superheater Fuel Emergency Cooling Tests

Normally, the core of the BONUS reactor is covered with approximately eight feet of water, and cooling of the core is provided by circulation of the water and removal of steam. However, if a rupture should occur in the lower portion of the vessel or in any one of the pipe lines entering the lower portion of the vessel, water could leak from the vessel and, unless emergency provisions are made, the core could be left dry and with no means of heat removal. Loss of water would result in termination of the fission process, but a substantial heat source in the form of radioactive decay of fission products and activated materials continues to liberate heat.

As a consequence of this shutdown decay heat, rapid heating of the fuel and fuel clad will occur. If an emergency heat removal system is not available, the fuel cladding will reach a temperature where a rupture will occur and allow the release of radioactive fission products.
The boiling region of the core is so designed that a water spray directed downward from above can effectively remove the heat from the region. The superheating fuel assemblies, however, are not open at the top and, therefore, are not so readily cooled with a water spray. Consequently, a need exists for the development of a system which can introduce water in an even distribution to all tubes of the superheater assemblies.

Because of the difficulty of spraying water horizontally into the superheater fuel assembly, a distribution pan type system appears to offer the best emergency cooling possibilities.

The distribution pan system would call for the incorporation of a shallow tube sheet like pan near the top of each fuel assembly. This pan would have orifice holes around the periphery of each tube and would have an opening on the side of the pan for water injection into the pan. With this arrangement, water would be injected into the pan through a relatively large nozzle and would leave the pan through the peripheral orifices, thus providing good distribution of water to all tubes.

Tests have been carried out with nine heated rod elements to which water is distributed by means of half circle 0.089 inch diameter openings around each heated rod. Water was fed to the distribution pan at 3.6 gpm (.04 gpm per rod) with the rods preheated to 1450°F and operating at an average decay power of 5 percent of full power (110 watts/ft). Under these conditions, the full length BONUS type elements will be cooled in approximately 1.4 minutes. The pressure tube temperature reaches a steady state value of 300°F.

Because of the possibility that the small holes being used in the distribution pan will clog due to corrosion produce accumulation, alternate systems were investigated. The most satisfactory system evolved used a rather large opening (1/4-inch), placed between each set of four pressure tubes to direct the flow of cooling water from the distribution pan to the fuel element pressure tubes.

This system has been tested in the nine tube hot test rig and found to be satisfactory. Cooling of the full length BONUS pressure tube is calculated to occur in four to five minutes.
Figure 1. Variation of Test Element Temperatures With Thimble Heat Flux (Shutdown Cooling Test)
Figure 2. Variation of BONUS Element Temperatures With Clad Heat Flux (Shutdown Cooling Test)

- Fuel Clad Surface Heat Flux (BTU/HR-FT²)
- Temperature (°F)
- Peak Flux at 2% Full Power
- Peak Flux at 6% Full Power
- PR. TUBE
- CLAD
- LINER
- R1 = 0.268 IN.
- R2 = 0.355 IN.
- R3 = 0.367 IN.
- R4 = 0.399 IN.
- P = 900 PSIG
NUCLEAR SUPERHEAT REACTORS UNDER CONSTRUCTION

1. PATHFINDER - ALLIS-CHALMERS MANUFACTURING COMPANY - R. W. Klecker

Construction of the Pathfinder plant at the Sioux Falls site in South Dakota is well under way and, barring no unforeseen difficulties, the scheduled reactor criticality date of July 1962 should be met. Full power operation is scheduled for the fall of 1962. Plant construction was about 34% complete as of October 15, 1960.

A considerable amount of construction work has been completed to date. The perimeter walls for all the buildings such as the turbine building, water treatment building, fuel handling building, etc., have been essentially completed. The containment shell has been fabricated and is currently being leak tested. It appears that the constructor will be able to continue work throughout the winter session inside the various buildings as scheduled.

The Pathfinder plant features a direct cycle, controlled recirculation, boiling reactor with a core having an internal nuclear superheater. The reactor is designed for a heat power level capability of 164 MWT in the boiler region and 39 MWT in the superheater region. Saturated steam at 600 psig and 489°F will be generated in the boiler region. The integral nuclear superheater will raise the steam temperature to 825°F with a pressure drop of about 60 psi. The net electric plant power output is 62 MWE.

Fuel for the boiler will consist of rod type elements containing 1.85 percent enriched UO₂ pellets clad with Zr-2. To help flatten the axial power distribution, 81 rods will be positioned in the lower half of the fuel assembly, and 64 rods will be positioned in the upper half of the assembly. For the superheater section concentric tubular elements, consisting of 93 percent UO₂ particles, were dispersed in a stainless steel matrix and clad with type 316L SS.

2. BORAX V - ARGONNE NATIONAL LABORATORY - R. E. Rice

The reference reactor design is essentially the same as discussed at the last superheat meeting on April 7 and 8, 1960. Briefly, BORAX V has a nominal power rating of 20 MW thermal and a plant heat dissipation capacity of 40 MW thermal. The reactor pressure is 600 psi, and the outlet superheated steam temperature is 850°F. Three separate core configurations: a boiler core with a centrally located superheater, a boiler core with a peripherally located superheater, and a pure boiling core will be tested. In all cases, it will be possible to operate with either natural or forced circulation of water through the core. Slightly enriched UO₂ pellets clad with type 304L SS and highly enriched UO₂ dispersed in stainless steel clad with stainless
steel in the form of plates will be used in the boiler and superheater respectively.

The reactor plant is currently under construction at NRfS, Idaho. On October 14, 1960, the over-all construction project was 82 percent complete versus 100 percent scheduled. Pressure vessel delivery has been the major item in delaying plant construction. The revised construction and operation schedule is as follows:

<table>
<thead>
<tr>
<th>Event</th>
<th>Date</th>
</tr>
</thead>
<tbody>
<tr>
<td>Construction complete</td>
<td>February 1, 1961</td>
</tr>
<tr>
<td>Reactor start-up boiler core</td>
<td>April 1961</td>
</tr>
<tr>
<td>Start superheat test</td>
<td>June 1961</td>
</tr>
<tr>
<td>Full power with superheat core</td>
<td>July 1961</td>
</tr>
</tbody>
</table>

A core with a central superheater will be tested first.

In view of the additional time now available prior to completion of construction, ANL proposes to run several critical experiments with the mock-up BORAX core at the Laboratory in Illinois. The boiler fuel elements are scheduled to be delivered in January 1961. The superheater elements may not be available in time for the critical experiments, but mock-up superheater elements will be substituted.

ANL is currently preparing the final hazards report and operating manual for BORAX V. These reports should be available in the early part of 1961.

3. **BONUS - GENERAL NUCLEAR ENGINEERING CORPORATION - J. M. West**

Construction of the BONUS reactor plant was initiated August 23, 1960, and was approximately 2 percent complete as of October 15, 1960. The plant is being constructed as a joint venture for the AEC and the Puerto Rico Water Resources Authority. Construction work to date has consisted essentially of clearing the site and excavating for the reactor dome foundation. The current schedule for criticality and full power operation is December 1, 1962, and February 1, 1963, respectively.

The plant features an integral boiler-superheater reactor of the pressure vessel type. The gross electric output and throttle steam conditions have been set at 17,300 kW, 850 psig, and 900°F to conform to the requirements of an existing standard turbine. The core consists of an internal boiler region surrounded by a peripheral superheater region. Saturated steam at 950 psig and 510°F will be produced in the boiler region, and the steam temperature will be raised to 900-950°F in the h-pass superheater region. A controlled attemperator will be installed in the steam line to reduce the temperature to 900°F before the steam reaches the turbine. The thermal and gross electrical plant ratings are 50 MW and 17.3 MW, respectively. Of the total heat generated, 13 MW are produced in the superheater region and 37 MW in the boiler region.
The boiler fuel elements consist of 1.85 percent enriched UO$_2$ pellets clad with Zircaloy-2. The clad thickness is about 25 mils. The fuel assembly consists of a square array of $6 \times 6$ pins with the four central pins missing.

The superheater fuel elements consist of 3.5 percent enriched UO$_2$ pellets clad with 347 SS. The cladding is 18 mils thick and is non-free standing. The fuel assembly consists of 32 fuel element positions in individual process tubes. The ends of each process tube are connected to an integral plenum which is divided into four steam flow paths.

The containment shell consists of a hemispherical steel dome about 167 feet in diameter, attached to a vertical steel cylinder which in turn is sealed to a 3.5 ft thick concrete foundation mat. The entire plant is housed inside the dome with the exception of a small administration building. The cost of this type of building is estimated to be equivalent to a high pressure small volume shell. The dome is designed to withstand a maximum internal pressure of 5 psig and is predicated on a maximum credible accident of a brittle failure of the largest recirculation line. Because of its low pressure and of the small number of electrical and mechanical penetrations, the BONUS building is considered to be safer, as well as more convenient, than a small volume high pressure building.

The plant design is progressing very satisfactorily and is now in the stage of optimizing the various plant parameters. During operation of the plant, critically needed information pertaining to long-term effects such as fuel integrity, corrosion, radioactive contamination, etc., will be obtained.

The BONUS plant is being built at Punta Higuera, a seacoast site at the westernmost tip of Puerto Rico.
NUCLEAR SUPERHEAT DESIGN STUDIES

1. **BOILING WATER – SEPARATE SUPERHEATER 300 MWE PLANT – R. T. Pennington**

The objective of this work is to perform a preliminary design of the reactor complex and associated equipment in enough detail to determine the probable reactor plant equipment costs, nuclear fuel cycle costs, and operation and maintenance costs for a plant which could be in operation by 1965.

The major components of the proposed power plant are shown in the Fourth Quarterly Report - CEPAP 3538. This study is scheduled to be completed by mid 1961. The major feature of the plant is incorporation of a separate steam cooled superheat reactor in series with the boiling water reactor. The major advantage of this concept is the ability to control power level in the superheat reactor independent of power in the boiler. This is expected to result in operational simplification for a start-up, decay heat removal, and control of superheat exit temperature over the fuel life. The power plant, boiling reactor, and superheat reactor conditions are listed in Table I.

TABLE I

POWER PLANT CONDITIONS

<table>
<thead>
<tr>
<th>Description</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Gross electrical output</td>
<td>312,000 kw</td>
</tr>
<tr>
<td>Turbine throttle conditions</td>
<td></td>
</tr>
<tr>
<td>Pressure</td>
<td>965 psia</td>
</tr>
<tr>
<td>Temperature</td>
<td>900°F</td>
</tr>
<tr>
<td>Steam flow</td>
<td>2,560,000 lb/hr</td>
</tr>
<tr>
<td>Feedwater temperature</td>
<td>1,400°F</td>
</tr>
<tr>
<td>No. of heaters</td>
<td>4</td>
</tr>
<tr>
<td>Condenser steam flow</td>
<td>1,840,000 lb/hr</td>
</tr>
<tr>
<td>Gross plant heat rate</td>
<td>9,050 Btu/kw-hr</td>
</tr>
</tbody>
</table>

BOILING WATER REACTOR

<table>
<thead>
<tr>
<th>Description</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal rating</td>
<td>615 MNT</td>
</tr>
<tr>
<td>Type</td>
<td>Forced single cycle</td>
</tr>
<tr>
<td>Steam separation</td>
<td>Internal</td>
</tr>
<tr>
<td>Vessel ID</td>
<td>138 inches</td>
</tr>
<tr>
<td>Fuel</td>
<td>UO₂ solid rod SS</td>
</tr>
<tr>
<td>Clad</td>
<td>317 in OD</td>
</tr>
<tr>
<td>Fuel enrichment</td>
<td>2.3%</td>
</tr>
<tr>
<td>Control rod</td>
<td>80 cruciform 2% boron-steel</td>
</tr>
<tr>
<td>Core power density</td>
<td>38 kw/lit.</td>
</tr>
</tbody>
</table>
SEPARATE SUPERHEAT REACTOR

<table>
<thead>
<tr>
<th>Specification</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal rating</td>
<td>212 MW (Mega Watt)</td>
</tr>
<tr>
<td>Type</td>
<td>Single pass</td>
</tr>
<tr>
<td>Vessel ID</td>
<td>102 inches</td>
</tr>
<tr>
<td>Fuel</td>
<td>UO$_2$ annular element</td>
</tr>
<tr>
<td></td>
<td>Stainless clad</td>
</tr>
<tr>
<td></td>
<td>ID .25 inch</td>
</tr>
<tr>
<td></td>
<td>OD .70 inch</td>
</tr>
<tr>
<td>Fuel enrichment</td>
<td>3.3%</td>
</tr>
<tr>
<td>Control rod</td>
<td>26 square shape boron-steel</td>
</tr>
<tr>
<td>Core power density</td>
<td>30 kw/lit.</td>
</tr>
</tbody>
</table>

For a 312,000 MWE gross plant output at 965 psia, 900°F turbine inlet and a 400°F feedwater temperature, the gross thermal efficiency is 37.7 percent. The thermal rating of the boiling water reactor is 615 MW. The thermal rating of the superheater is 212 MW.

The boiler core configuration is based on utilization of 180 assemblies of a 10 x 10 array of 0.442 inch diameter fuel rods clad with .019 inch stainless steel cladding. Each assembly contains 99 UO$_2$ rods and one fuel rupture detection tube.

The design incorporates 80 locking piston, cruciform control rods fabricated with 2 percent natural boron stainless steel. The average core void is 20 percent by volume with a maximum void at channel exit of 75 percent. The effective core height is 106 inches with an equivalent core diameter of 107 inches. This design incorporates internal axial flow steam separators and steam dryers.

The superheater features a single pass annular UO$_2$ fuel configurations with light water moderator. The steam coolant flows down through the process tube, flows on both inside and outside of the annular fuel in a single pass to the exit manifolding at the bottom of the reactor. The single pass arrangement was selected over the two-pass bayonet arrangement in large core sizes in order to achieve the following advantages:

a. The core power density may be increased since for the single pass, annular fuel geometry the volumetric heat generation may be increased within the clad surface temperature limit.

b. The reduction in flow area of the single pass as compared to the double pass reduces the core steam void fraction, thereby increasing power density and also minimizing the reactivity changes associated with flooding the steam passages with water.

c. The parallel flow configuration minimizes the temperature differential between the inside and outside fuel cladding as compared to a bayonet flow arrangement.
d. The single pass configuration permits more effective utilization of axial power shaping to reduce the maximum clad temperature for the same average superheat exit temperature.

e. Higher steam velocities and improved heat transfer at the same core pressure drop.

f. Improved drawing characteristics of the fuel bundles.

g. Simplified mechanical arrangement and improved refueling characteristics.

The core configuration for the nuclear superheater is based on utilization of 69 assemblies of a 6 x 6 array of annular single pass fuel elements arranged in an insulated Zr process tube. The fuel element is clad on the inside and outside with thin stainless steel which will require support from the UO2 to withstand the external system pressure. The fuel inside diameter is 0.25 inch, and the outside diameter is 0.70 inch. The design incorporates 26 square box type, locking piston control rods fabricated with 2 percent natural boron stainless steel. The effective core length is 96 inches with an effective core diameter of 75 inches. The core pressure drop is 75 psi; the maximum exit steam velocity of a central tube is 310 ft/second.

The basic superheat fuel element consists of a double jacketed tube (outer jacket Zircaloy-2 and inner jacket stainless steel) which encloses an annular fuel rod which is clad internally and externally with stainless steel. The annular gap between the double jacketed tubes contains stagnant steam which serves as a thermal insulation between the high temperature superheated steam and the comparatively low moderator temperature which surrounds the Zircaloy process tube. This thermal insulation permits the usage of a Zircaloy process tube, operating at practically the same temperature as the moderator.

In an effort to both measure and control power distribution with the superheater core, the following instrumentation is provided.

a. Temperature measurement at the outlet steam temperature of 35 of the fuel bundles.

b. In-core neutron flux monitors capable of traversing axially five fuel channels.

The temperature measurements will provide a means of determining radial power (or power per bundle) distribution. The flux monitors will provide information on the axial power distribution that is required to accurately determine maximum clad temperatures for a known bundle power. Outlet temperature measurements afford an opportunity for adjusting orificing and control rod pattern for
best conditions of fuel surface temperature while providing rated operating condition of bulk outlet steam temperature to the turbine.

2. **MIXED SPECTRUM SUPERHEATER (MSS) STUDY** - B. Wolfe

The objectives of this study are to form a technical evaluation and conceptual design of an integral mixed spectrum boiler-superheater reactor and determine its potential economic incentives with a standard boiling water reactor. This task was initiated in April 1960 and has been virtually completed. A topical report on the mixed spectrum reactor will be issued shortly.

The conceptual reference design mechanical arrangements and set of reference design reactor operating conditions are shown in Figure 1.1 and Table II respectively. The reactor consists of a central unmoderated superheating region surrounded by a buffer region and an outer boiling region. Water is boiled in the outer boiling core region. The steam-water mixture passes up through a series of steam separators. The separated steam passes through steam dryers and then travels down through a set of twisted pipes to enter unmoderated superheating core. After leaving the superheating region, the steam leaves the pressure vessel through pipes which penetrate the vessel bottom and go directly to the turbine. The reference steam conditions are 1400 psi and 950°F outlet from the superheater.

Among the major problem areas of a thermal superheat reactor are:

1. The requirements of thin clad high temperature fuel cladding caused by the conflict between requirements of clad integrity and parasitic absorption by the clad,
2. Mechanical difficulties due to the large number of process tubes and the complicated manifolding required, and the associated problems of refueling such a system,
3. The low power density in the superheating region which, in turn, may lead to relatively high cost for the superheater. In contrast, analysis of the MSS thus far showed the following:

   a. There is a relatively low sensitivity to structural material in the superheater region so that rugged structure and fuel cladding may be utilized.

   b. The MSS has a relatively simple mechanical design from the standpoint that the superheater is contained in only one large pipe. Manifolding problems are essentially eliminated and refueling significantly simplified.

   c. The power density in the superheater can be made higher than that in a boiling core so that the output from the MSS will be higher than from a boiling water core of the same size.

   d. As a result of the higher thermal to electric conversion efficiency of the MSS, its higher power density, and its compact design,
MSS has the potentiality of significantly reduced capital costs over those of a standard boiling reactor of the EBASCO type.

e. The superheating section has the potentiality of achieving high burnup, good neutron economy, and consequently low fuel costs.

Based on the analysis to date, General Electric believes that the mixed spectrum boiler superheat reactor is technically feasible and can be operated safely. The mechanical design of the reactor involves no great extrapolation from present technology. The major area of doubt is due to the nuclear complexity of the system and the limitations of available analytical tools. Experimental work in the areas of physics, fuel element performance, and heat transfer and fluid flow are required to verify the design calculations.
<table>
<thead>
<tr>
<th>Table II: Summary of Reactor Plant Characteristics of Preliminary Reference Design</th>
</tr>
</thead>
<tbody>
<tr>
<td>(1) Heat Balance</td>
</tr>
<tr>
<td>(a) Net plant power, MWE</td>
</tr>
<tr>
<td>(b) Total reactor power, MWT</td>
</tr>
<tr>
<td>(2) Turbine Cycle Conditions</td>
</tr>
<tr>
<td>(a) Throttle temperature, °F</td>
</tr>
<tr>
<td>(b) Throttle pressure, psia</td>
</tr>
<tr>
<td>(c) Total steam flow, lbs/hr</td>
</tr>
<tr>
<td>(3) Reactor Description</td>
</tr>
<tr>
<td>(a) Reactor core</td>
</tr>
<tr>
<td>(1) Active equivalent diameter, ft.</td>
</tr>
<tr>
<td>(2) Active height, ft.</td>
</tr>
<tr>
<td>(3) Active core volume, ft^3</td>
</tr>
<tr>
<td>(4) Loading of superheating core</td>
</tr>
<tr>
<td>(b) Reflector or blanket</td>
</tr>
<tr>
<td>(1) Material</td>
</tr>
<tr>
<td>(2) Axial thickness, ft.</td>
</tr>
<tr>
<td>(3) Radial thickness, ft.</td>
</tr>
<tr>
<td>(4) Thickness of buffer region, ft.</td>
</tr>
<tr>
<td>(c) Fuel elements (for each type)</td>
</tr>
<tr>
<td>(1) Fuel material</td>
</tr>
<tr>
<td>(2) Fuel element geometry</td>
</tr>
<tr>
<td>(3) Clad material</td>
</tr>
<tr>
<td>(4) Fuel &quot;Meat&quot; thickness or diameter, in.</td>
</tr>
<tr>
<td>(5) Clad thickness, in.</td>
</tr>
<tr>
<td>(6) Fuel-clad gap (cold), in.</td>
</tr>
<tr>
<td>(7) Gap filler material</td>
</tr>
<tr>
<td>(d) Fuel assemblies (for each type)</td>
</tr>
<tr>
<td>(1) Total number</td>
</tr>
<tr>
<td>(2) No. of elements (rods) per assembly</td>
</tr>
<tr>
<td>(3) Cross sectional dimensions, in.</td>
</tr>
<tr>
<td>(4) Lattice spacing, in.</td>
</tr>
<tr>
<td>(5) End fitting materials</td>
</tr>
<tr>
<td>(e) Reactor control</td>
</tr>
<tr>
<td>(1) Method control</td>
</tr>
<tr>
<td>(2) Absorber material</td>
</tr>
<tr>
<td>(3) No. of control elements</td>
</tr>
<tr>
<td>(4) Cross sectional dimensions, in.</td>
</tr>
<tr>
<td>(5) Effective length, ft.</td>
</tr>
<tr>
<td>(6) Type of Drive</td>
</tr>
</tbody>
</table>
3. **ZIRCONIUM HYDRIDE REACTOR - J. D. Gylfe**

This study was performed to evaluate the merits of using zirconium hydride as a solid moderator in the superheat region of an integral boiling water nuclear superheat reactor. The report is expected to be issued in December 1960. Conceptual designs were developed for both water-moderated and zirconium hydride moderated reactors in order to compare the two materials on a consistent, quantitative basis. Both designs produce 300 MWE with a net efficiency of 38.5 percent. The steam pressure and temperature at the turbine throttle are 1450 psig, 1000°F in each case.

The reference zirconium hydride moderated design features a centrally located superheat region which is composed of 45 moderator logs, each of which contains 16 stainless steel clad, UO₂ fuel elements arranged in 7-rod clusters. The boiling region contains 84 Zircaloy-clad UO₂ fuel elements which are cooled by forced circulation. An alternate concept wherein the zirconium hydride is used both as a moderator and as a thick-walled, free-standing, fuel clad material was also investigated.

The water moderated design is similar to the zirconium hydride, moderator log concept, except for the superheater. The superheat region is centrally located and contains 49 fuel assemblies, each of which consists of 16 insulated process tubes. A 7-rod cluster fuel element is placed within each process tube. Two variations of this design were also investigated; in one case stainless steel is used for the process tube and in the other, Zircaloy-2 is used.

A summary of the fuel cycle costs for these four concepts is tabulated below:

<table>
<thead>
<tr>
<th>Reactor Description</th>
<th>Water Moderated</th>
<th>Zr-Hydride Moderated</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>SS Process Tube</td>
<td>Zircaloy Process Tube</td>
</tr>
<tr>
<td>Burn-up</td>
<td>1.73</td>
<td>1.66</td>
</tr>
<tr>
<td>Pu credit</td>
<td>(.66)</td>
<td>(.67)</td>
</tr>
<tr>
<td>Fabrication</td>
<td>1.40</td>
<td>1.50</td>
</tr>
<tr>
<td>Inventory</td>
<td>.40</td>
<td>.33</td>
</tr>
<tr>
<td>Shipping</td>
<td>.17</td>
<td>.17</td>
</tr>
<tr>
<td>Reprocess &amp; conversion</td>
<td>.39</td>
<td>.40</td>
</tr>
<tr>
<td>TOTAL</td>
<td>3.43</td>
<td>3.39</td>
</tr>
</tbody>
</table>
The costs shown are based on an average burn-up of 10,000 MWD/MTU and batch refueling for each design. It is noted that the zirconium hydride moderator log concept has the lowest fuel cycle costs and shows a .12 mills/kwhr advantage over the water moderated design with the Zircaloy process tube. This difference is due entirely to the higher fabricating cost of the "throw-away" process tube design. The integral fuel moderator concept has good nuclear characteristics but suffers from high fabrication costs.

The capital cost of the zirconium hydride moderator log concept, however, is about $3,330,000 higher than that of the water moderator design, including indirect charges at 53 percent. Most of this difference is due to the high cost of fabricating the zirconium hydride logs. This difference amounts to a capital cost penalty of 0.22 mills/kwhr which yields a net deficit of .10 mills/kwhr.

On the basis of this study, it is concluded that an economic incentive for substituting zirconium hydride for water as moderator in this type of reactor does not exist. The two designs are approximately equivalent from a cost standpoint, but the technical problems encountered using zirconium hydride are more severe than those of the water moderated design.

4. INTEGRAL SUPERHEAT REACTOR - S. J. Weems

The purpose of this task is to develop a preliminary design of an integral boiler superheater reactor capable of producing steam at 1200 psig and 1050°F for generating 200 M of electricity. The preliminary reactor design is to serve as a guide for the research and development programs aimed at development of large reactors of the integral boiler-superheater type. The design work was initiated in July 1959, and the preliminary design was recently completed and will be published shortly in GNEC 136. As further results become available from related R&D programs, the preliminary reference design will be modified to take these results into account. A final reference design report is scheduled to be issued in December 1961.

The preliminary reference design consists of a boiling-water moderated reactor producing a total of 518 MWT with feedwater returned to the reactor at 450°F. The steam cycle consists of five feedwater heaters and the gross cycle efficiency is 38.6 percent.

The core (horizontal cross section shown in Figure 2) consists of a central combination boiler-superheater region (hereinafter called the superheater region) which is surrounded by a pure boiler region. The central superheater region is 6.62 feet in diameter and the boiler region is .95 foot thick. The active core height is 9 feet. The central region contains 61 hexagonal shaped fuel assemblies, each containing 19 fuel elements of the double annular type arranged on an equilateral-triangular-pitch-spacing. The peripheral region of
the core consists of 4.7 percent enriched UO\(_2\) fuel rods clad with type 304 SS which are also arranged on an equilateral triangular pitch spacing. The equilateral triangular pitch spacing was used to obtain the maximum power density per given size coolant channel.

The double annular fuel element is unique in that boiling occurs on the outermost surface of the elements while the interior surfaces are cooled by a single steam pass through the interior channels (one annular and one cylindrical). The fuel element consists of 2.3 percent enriched UO\(_2\) pellets. Cladding of both surfaces on the inner tube is type 347 SS while the outer fuel tube has type 347 SS cladding on its exterior surface and Inconel X on its interior surface. The use of Inconel X on the interior surface helps alleviate thermal expansion and strength problems associated with the high temperature difference at which the exterior and interior surfaces of the outer tube operate. One of the significant advantages of the double annular elements concept is the reduction of stainless steel to fuel volume ratio compared to rod type elements and elimination of the process tube used in many other reactors to separate the water moderator from the steam channel. Another significant advantage is the ability to dissipate shutdown heat directly to the water moderator.

For the final reference design, consideration is being given to alternate type combination boiler-superheater fuel elements. Currently, the alternate elements consist of: (1) a large cylindrical tube filled with UO\(_2\) and containing a number of very small tubes which will carry the steam while it is being superheated and (2) a single thick annular element filled with UO\(_2\) and containing a number of small tubes; steam would be superheated within the small tubes and water would be heated on the inner surface of the annulus.

A schematic of the main coolant flow through the reactor is shown in Figure 3. Saturated steam is generated in the boiler and superheater region of the core. Forced circulated water enters at the bottom of the peripheral region. The water for the central region of the core is circulated entirely by natural convection. The steam leaving the core exit passes through steam dryers (demisters) prior to reaching the upper portion of the reactor vessel where it enters the saturated steam downcomer pipes and is then channeled to the bottom of the superheater. The saturated steam is heated in a single pass through the superheater (bottom to top) from 573°F to 1050°F. The maximum inlet and exit steam velocities are 90 and 209 ft/second. The maximum superheater fuel surface temperature is estimated to be 1250°F.
For the past two years, ANL has been studying the concept of coupled fast-thermal reactor systems. These studies were directed toward a power breeder concept with a high breeding ratio and neutron lifetime characteristics comparable to a thermal reactor.

The study presented at the meeting has been completed and will be published shortly in ANL-6286.

The reactor design, shown in Figure 4, features an internal fast superheating region surrounded by a thermal pressurized water region. A radial buffer region is positioned between the fast and thermal regions to reduce flux peaking in the thermal region. Above and below both regions are axial fuel blankets. Moving fuel is used for control rods.

The reactor produces a breeding ratio of 1.4. Pressurized water at 1525 psi and 562°F enters at the bottom of the thermal region through pressure tubes and leaves at 575°F and 1400 psi. Saturated steam at 1400 psi and 587°F enters the top of the superheater and exits at 1100 psi and 850°F. The steam is used in a direct cycle system operating at an efficiency of approximately 30 percent to yield 65 MWE.

In order to get a ratio of 1.4, it was necessary to proportion the power such that 75 percent is produced in the fast region and 25 percent in the thermal region. (If the power split were readjusted such that the power generated in the thermal region would supply sufficient steam to cool the fast region, the breeding ratio would be reduced to 0.9.) A steam compressor recycles about 66 percent of the steam leaving the reactor and mixes it with the feedwater to generate the desired steam inlet conditions to the reactor in order to maintain the proper power split between the core regions. (See Figure 5.)

The fast region is fueled with 1/8-inch diameter pins containing PuO₂ and depleted UO₂ clad with stainless steel. The thermal region is fueled with natural UO₂ pellets clad with Zircaloy-2. The buffer region and axial blankets are fueled with depleted UO₂. The maximum surface temperatures were set at 1325°F for stainless steel and 600°F for Zircaloy-2. The maximum oxide fuel temperature was based on 4500°F. Tables 4 through 8 summarize the power, core dimensions, and other pertinent data.

Results of ANL's study indicate that the coupled fast-thermal breeder reactor can be designed with a 1.4 breeding ratio, has a neutron lifetime comparable to a thermal reactor, and can be designed to operate safely. However, it was concluded that the economic incentive for such a system is marginal for the following reasons:

a. In order to obtain a high specific power in the fast region, it was necessary to finely subdivide the fuel. This results in high fuel fabrication costs.
b. The performance of the reactor when compared with a sodium cooled reactor of the same size and power range is not significantly outstanding with regards to specific power and cycle efficiency.

c. The benefits of using a direct steam cycle are negated by the added complexity and capital costs associated with employing a steam compressor and the necessary feedwater heating equipment to produce steam for the reactor.
TABLE 4. GENERAL DATA

<p>| | |</p>
<table>
<thead>
<tr>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Total, Mwt</td>
<td>215</td>
</tr>
<tr>
<td>Elec., Mwe</td>
<td>65</td>
</tr>
<tr>
<td>Cycle Eff., %</td>
<td>30</td>
</tr>
<tr>
<td>Initial Breeding Ratio</td>
<td>1.4</td>
</tr>
<tr>
<td>PuO₂ Critical Mass, kg</td>
<td>275</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Coolant</th>
<th>Flow, lb/hr</th>
<th>Max. Inlet Vel., ft/sec</th>
<th>Temp., °F</th>
<th>Pressure, psia</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>In</td>
<td>Out</td>
</tr>
<tr>
<td>Steam</td>
<td>2.34 x 10⁶</td>
<td>150</td>
<td>587</td>
<td>850</td>
</tr>
<tr>
<td>Water</td>
<td>10⁷</td>
<td>30</td>
<td>562</td>
<td>575</td>
</tr>
<tr>
<td></td>
<td>Fast Core</td>
<td>Radial Buffer</td>
<td>Axial Blanket</td>
<td></td>
</tr>
<tr>
<td>-------------------------</td>
<td>-----------</td>
<td>---------------</td>
<td>---------------</td>
<td></td>
</tr>
<tr>
<td><strong>Power, Mwt</strong></td>
<td>155</td>
<td>5.0</td>
<td>1.5</td>
<td></td>
</tr>
<tr>
<td><strong>Dimensions, in.</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>O.D.</td>
<td>27.5</td>
<td>31</td>
<td>31</td>
<td></td>
</tr>
<tr>
<td>I.D.</td>
<td>-</td>
<td>27.5</td>
<td>-</td>
<td></td>
</tr>
<tr>
<td>Length</td>
<td>31.5</td>
<td>31.5</td>
<td>23.6</td>
<td></td>
</tr>
<tr>
<td><strong>Fuel</strong></td>
<td>PuO₂ + UO₂ (Dep'l)</td>
<td>UO₂ (Dep'l)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Cans</td>
<td>48</td>
<td>13</td>
<td>61</td>
<td></td>
</tr>
<tr>
<td>Rods/can</td>
<td>547</td>
<td>61</td>
<td>61</td>
<td></td>
</tr>
<tr>
<td>Rod O.D., in</td>
<td>0.125</td>
<td>0.375</td>
<td>0.375</td>
<td></td>
</tr>
<tr>
<td>Clad Mat'l</td>
<td>Type 347</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Clad Thk., in.</td>
<td>0.005</td>
<td>0.017</td>
<td>0.017</td>
<td></td>
</tr>
<tr>
<td>Pellet Dia., in.</td>
<td>0.113</td>
<td>0.337</td>
<td>0.337</td>
<td></td>
</tr>
<tr>
<td>Heat Transfer</td>
<td>Fast Core</td>
<td>Radial Buffer</td>
<td>Axial Blanket</td>
<td></td>
</tr>
<tr>
<td>-------------------------------</td>
<td>-----------</td>
<td>---------------</td>
<td>---------------</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Upper</td>
<td>Lower</td>
</tr>
<tr>
<td>Max. Heat Flux, B/hr ft(^2)</td>
<td>5.6 \times 10^5</td>
<td>1.6 \times 10^5</td>
<td>7 \times 10^4</td>
<td>7 \times 10^4</td>
</tr>
<tr>
<td>Max. Temp., °F</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Surface</td>
<td>1325</td>
<td>1325</td>
<td>680</td>
<td>1200</td>
</tr>
<tr>
<td>Fuel</td>
<td>2620</td>
<td>2100</td>
<td>800</td>
<td>1450</td>
</tr>
<tr>
<td>Hot Channel Factors</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fluid</td>
<td>2</td>
<td>1.6</td>
<td>1.9</td>
<td>1.9</td>
</tr>
<tr>
<td>Film</td>
<td>2.4</td>
<td>1.9</td>
<td>9.8</td>
<td>9.8</td>
</tr>
<tr>
<td>Fuel</td>
<td>2</td>
<td>1.6</td>
<td>7.9</td>
<td>7.9</td>
</tr>
<tr>
<td>Overall Max./Avg.</td>
<td>1.78</td>
<td>1.66</td>
<td>6.6</td>
<td>7.2</td>
</tr>
<tr>
<td>Avg. Power Density, kw/ℓ</td>
<td>506</td>
<td>5.8</td>
<td>4.8</td>
<td>4.8</td>
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## Table 7. Pressurized Water Zones

<table>
<thead>
<tr>
<th></th>
<th>Thermal Core</th>
<th>Radial Blanket</th>
<th>Axial Blanket</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Oxide</td>
<td>Metal</td>
</tr>
<tr>
<td><strong>Power, Mw</strong></td>
<td>30</td>
<td>4</td>
<td>13</td>
</tr>
<tr>
<td><strong>Dimensions, in.</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>O.D.</td>
<td>40.9</td>
<td>45.4</td>
<td>77.8</td>
</tr>
<tr>
<td>I.D.</td>
<td>31.5</td>
<td>40.9</td>
<td>45.4</td>
</tr>
<tr>
<td>Length</td>
<td>31.5</td>
<td>31.5</td>
<td>78.7</td>
</tr>
<tr>
<td><strong>Fuel</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pressure Tubes</td>
<td>UO(_2) (Nat)</td>
<td>42</td>
<td>24</td>
</tr>
<tr>
<td>Rods/Tube</td>
<td>UO(_2) (Dep'tl)</td>
<td>61</td>
<td>61</td>
</tr>
<tr>
<td>Rod O.D., in.</td>
<td>UO(_2) (Dep'tl)</td>
<td>0.309</td>
<td>0.309</td>
</tr>
<tr>
<td>Clad Mat'l</td>
<td>Clad Thk, in.</td>
<td>0.02</td>
<td>0.02</td>
</tr>
<tr>
<td>Pellet Dia., in.</td>
<td>Clad Thk, in.</td>
<td>0.265</td>
<td>0.265</td>
</tr>
</tbody>
</table>
### TABLE 8. PRESSURIZED WATER ZONES

<table>
<thead>
<tr>
<th>Heat Transfer</th>
<th>Thermal Core</th>
<th>Radial Blanket</th>
<th>Axial Blanket</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Oxide</td>
<td>Metal</td>
</tr>
<tr>
<td>Max. Heat Flux, B/hr ft²</td>
<td>$3.4 \times 10^5$</td>
<td>$9.7 \times 10^4$</td>
<td>$2.2 \times 10^5$</td>
</tr>
<tr>
<td>Max. Temp., °F</td>
<td>2600</td>
<td>860</td>
<td>600</td>
</tr>
<tr>
<td>Surface Fuel</td>
<td>2600</td>
<td>860</td>
<td>600</td>
</tr>
<tr>
<td>Hot Channel Factors</td>
<td>1.45</td>
<td>1.8</td>
<td>Variable</td>
</tr>
<tr>
<td>Fluid</td>
<td>1.8</td>
<td>2.2</td>
<td>23.3</td>
</tr>
<tr>
<td>Film</td>
<td>1.8</td>
<td>2.2</td>
<td>23.3</td>
</tr>
<tr>
<td>Fuel</td>
<td>1.8</td>
<td>2.2</td>
<td>23.3</td>
</tr>
<tr>
<td>Overall Max./Avg.</td>
<td>1.3</td>
<td>1.6</td>
<td>17</td>
</tr>
<tr>
<td>Avg. Power Density, kw/ℓ</td>
<td>106</td>
<td>25</td>
<td>3</td>
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</tbody>
</table>
ISOMETRIC VIEW OF MIXED SPECTRUM SUPERHEATER

FIGURE 1.1
BOILER - SUPERHEATER REGION
6.62 FT. DIA., 9 FT. HT.
19 FUEL ELEMENTS PER HEXAGONAL CELL
1159 TOTAL FUEL ELEMENTS

13'10.75" I.D., 7.5" WALL, PRESSURE VESSEL
11'-6" O.D., 2" WALL,
10'-8" O.D., 1" WALL.
3 5/8" O.D., 1/4" WALL SATURATED STEAM DOWNCOMER

SATURATED STEAM MANIFOLD

BOILER - SUPERHEATER FUEL CELL (CENTRAL CORE REGION)

BOILER FUEL CELL (PERIPHERAL CORE REGION)

CONTROL ROD

9.69" CONTROL ROD DRIVE PITCH

BOILER REGION
5.4 FT. EQUIVALENT DIA.,
9 FT. HT. (.948 FT. THK.)
168 FUEL ELEMENTS PER HEXAGONAL CELL
6690 TOTAL FUEL ELEMENTS

CORE
8.51 FT. DIA., 9 FT. HT.
84 CONTROL RODS
61 HEXAGONAL CELL BOILER - SUPERHEATER REGION
40 HEXAGONAL CELL EQUIVALENT BOILER REGION

Figure 2. Reactor Horizontal Section
Figure 3. Coolant Flow Schematic
FIGURE 4. REACTOR ELEVATION
FIGURE 5. FLOW DIAGRAM
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Part B - Site Description

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