GEAP-4200

EVESR - NUCLEAR SUPERHEAT FUEL DEVELOPMENT PROJECT

Third Quarterly Report, December 1962 - February 1963

Prepared by
R. T. Pennington

March 1, 1963

Atomic Power Equipment Department
General Electric Company
San Jose, California
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EVESR-NUCLEAR SUPERHEAT FUEL
DEVELOPMENT PROJECT
THIRD QUARTERLY REPORT
DECEMBER - FEBRUARY, 1963

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ATOMIC POWER EQUIPMENT DEPARTMENT
GENERAL ELECTRIC
SAN JOSE, CALIFORNIA
CONTRIBUTORS

R. T. Pennington, Project Engineer
EVESR-Nuclear Superheat Fuel
Development Project

S. Armour
R. A. Becker
F. J. Brutschy
F. A. Comprelli
A. G. Dunbar
E. L. Esch
J. R. Fritz
R. S. Gilbert
W. K. Green
V. E. Hazel
R. C. Higbee
E. T. Hubbard

P. W. Ianni
A. R. Kimball
R. F. Kirby
R. G. Kuhne
E. A. Lees
C. B. McKee
W. L. Pearl
J. M. Roberts
R. R. Roof
R. A. Schmidt
C. N. Spalaris
H. T. Wells
M. L. Weiss
# TABLE OF CONTENTS

<table>
<thead>
<tr>
<th>Section</th>
<th>Title</th>
<th>Page No.</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.0</td>
<td>INTRODUCTION AND SUMMARY</td>
<td>1</td>
</tr>
<tr>
<td>1.1</td>
<td>Introduction</td>
<td>1</td>
</tr>
<tr>
<td>1.2</td>
<td>Summary</td>
<td>2</td>
</tr>
<tr>
<td>2.0</td>
<td>EVESR REACTOR AND PLANT DESCRIPTION</td>
<td>3</td>
</tr>
<tr>
<td>3.0</td>
<td>SUB-TASK A-1 PROGRAM PLANNING AND EVALUATION</td>
<td>4</td>
</tr>
<tr>
<td>3.1</td>
<td>Installation of R&amp;D Program Instrumentation at EVESR</td>
<td>4</td>
</tr>
<tr>
<td>3.2</td>
<td>EVESR R&amp;D Instrument System</td>
<td>4</td>
</tr>
<tr>
<td>3.3</td>
<td>Planning and Review Activity at EVESR Site</td>
<td>9</td>
</tr>
<tr>
<td>4.0</td>
<td>SUB-TASK B-1 FUEL DESIGN</td>
<td>12</td>
</tr>
<tr>
<td>4.1</td>
<td>General Design Progress</td>
<td>12</td>
</tr>
<tr>
<td>4.2</td>
<td>Moss Landing Flow Tests</td>
<td>12</td>
</tr>
<tr>
<td>5.0</td>
<td>SUB-TASK B-2 FUEL DESIGN - ENGINEERING PHYSICS</td>
<td>14</td>
</tr>
<tr>
<td>5.1</td>
<td>Final Physics Report</td>
<td>14</td>
</tr>
<tr>
<td>5.2</td>
<td>EVESR Core Design and Fuel Following</td>
<td>14</td>
</tr>
<tr>
<td>5.3</td>
<td>EVESR Safeguards</td>
<td>15</td>
</tr>
<tr>
<td>6.0</td>
<td>SUB-TASK C-1 FUEL PROCESS DEVELOPMENT AND FABRICATION</td>
<td>16</td>
</tr>
<tr>
<td>6.1</td>
<td>Schedule</td>
<td>16</td>
</tr>
<tr>
<td>6.2</td>
<td>Procurement</td>
<td>16</td>
</tr>
<tr>
<td>7.0</td>
<td>SUB-TASK C-2 FUEL ELEMENT PROCESS DEVELOPMENT AND FABRICATION</td>
<td>17</td>
</tr>
<tr>
<td>7.1</td>
<td>Materials Procurement</td>
<td>17</td>
</tr>
<tr>
<td>7.2</td>
<td>Fabrication Development - Regular and Instrumented Assembly</td>
<td>19</td>
</tr>
<tr>
<td>7.3</td>
<td>Fabrication</td>
<td>20</td>
</tr>
<tr>
<td>7.4</td>
<td>ESADA Fuel Pre-Irradiation Measurements</td>
<td>20</td>
</tr>
<tr>
<td>7.5</td>
<td>Strain Cycle Tests</td>
<td>28</td>
</tr>
<tr>
<td>7.6</td>
<td>ESADA-VESR Coupon Irradiation Program</td>
<td>28</td>
</tr>
<tr>
<td>7.6.1</td>
<td>Objectives of the Experiment</td>
<td>29</td>
</tr>
<tr>
<td>7.6.2</td>
<td>Experimental Details</td>
<td>30</td>
</tr>
<tr>
<td>7.6.3</td>
<td>Data Expected from Investigation</td>
<td>31</td>
</tr>
<tr>
<td>7.6.4</td>
<td>Summary of the Irradiation Experiment</td>
<td>31</td>
</tr>
</tbody>
</table>
# TABLE OF CONTENTS (CON'D)

<table>
<thead>
<tr>
<th>Section</th>
<th>Description</th>
<th>Page No.</th>
</tr>
</thead>
<tbody>
<tr>
<td>8.0</td>
<td>SUB-TASK C-3 EVESR NEUTRON SOURCE</td>
<td>33</td>
</tr>
<tr>
<td>8.1</td>
<td>Status</td>
<td>33</td>
</tr>
<tr>
<td>8.2</td>
<td>Source Strength</td>
<td>34</td>
</tr>
<tr>
<td>9.0</td>
<td>SUB-TASK C-1 FUEL EVALUATION INSTRUMENTATION</td>
<td>34</td>
</tr>
<tr>
<td>9.1</td>
<td>Status</td>
<td>34</td>
</tr>
<tr>
<td>10.0</td>
<td>SUB-TASK C-5 FUEL TESTS - CHEMISTRY SAMPLING STATIONS, DESIGN AND INSTALLATION</td>
<td>36</td>
</tr>
<tr>
<td>11.0</td>
<td>SUB-TASK D-1 FUEL TESTS - ENGINEERING PHYSICS</td>
<td>37</td>
</tr>
<tr>
<td>11.1</td>
<td>Status</td>
<td>37</td>
</tr>
<tr>
<td>12.0</td>
<td>SUB-TASK D-2 FUEL TEST - SITE OPERATIONS</td>
<td>37</td>
</tr>
<tr>
<td>12.1</td>
<td>Status Report</td>
<td>37</td>
</tr>
<tr>
<td>13.0</td>
<td>SUB-TASK D-3 FUEL TESTS - TEST PROCEDURES</td>
<td>38</td>
</tr>
<tr>
<td>13.1</td>
<td>Status Report</td>
<td>38</td>
</tr>
<tr>
<td>14.0</td>
<td>SUB-TASK D-4 FUEL TESTS - DATA REDUCTION</td>
<td>38</td>
</tr>
<tr>
<td>14.1</td>
<td>Test Procedure</td>
<td>38</td>
</tr>
<tr>
<td>14.2</td>
<td>Setting Orifice Sizes</td>
<td>41</td>
</tr>
<tr>
<td>14.3</td>
<td>Analysis and Description of Orifice Sizing and Procedure</td>
<td>44</td>
</tr>
<tr>
<td>14.4</td>
<td>Data Retrieval System</td>
<td>53</td>
</tr>
<tr>
<td>14.5</td>
<td>Data Reduction - Advanced Fuel Irradiation</td>
<td>54</td>
</tr>
<tr>
<td>14.6</td>
<td>Using Instrumented Bundles to Determine Flow Split Between Fuel Elements</td>
<td>55</td>
</tr>
<tr>
<td>15.0</td>
<td>SUB-TASK D-5 FUEL ACTIVITY RELEASE AND COOLANT CHEMISTRY</td>
<td>58</td>
</tr>
<tr>
<td>15.1</td>
<td>Status Report</td>
<td>58</td>
</tr>
<tr>
<td>16.0</td>
<td>SUB-TASK E-1 POST-IRRADIATION EXAMINATION PLANNING AND EVALUATION</td>
<td>58</td>
</tr>
<tr>
<td>16.1</td>
<td>Pre-Irradiation Measurement Program for Mark II Fuel</td>
<td>58</td>
</tr>
<tr>
<td>17.0</td>
<td>SUB-TASK E-2 PRE- AND POST-IRRADIATION EXAMINATION</td>
<td>60</td>
</tr>
<tr>
<td>17.1</td>
<td>Status Report</td>
<td>60</td>
</tr>
<tr>
<td>18.0</td>
<td>SUB-TASK F-1 ADVANCED FUEL DESIGN</td>
<td>60</td>
</tr>
<tr>
<td>18.1</td>
<td>EVESR Advanced Fuel Design</td>
<td>60</td>
</tr>
<tr>
<td>Figure No.</td>
<td>Title</td>
<td>Page No.</td>
</tr>
<tr>
<td>-----------</td>
<td>----------------------------------------------------------------------</td>
<td>----------</td>
</tr>
<tr>
<td>3.1</td>
<td>Schematic of EVESR R&amp;D Instrumentation</td>
<td>5</td>
</tr>
<tr>
<td>3.2</td>
<td>EVESR R&amp;D Instrumentation</td>
<td>8</td>
</tr>
<tr>
<td>3.3</td>
<td>ESADA-VEESR Instrumentation for Fuel Performance Evaluation</td>
<td>10</td>
</tr>
<tr>
<td>3.4</td>
<td>ESADA-VEESR Instrumentation for Fuel Performance Evaluation</td>
<td>11</td>
</tr>
<tr>
<td>7.1</td>
<td>Dwg. 2125665 - Assembly Checking Fixture</td>
<td>23, 24</td>
</tr>
<tr>
<td>7.2</td>
<td>Dwg. 104B1587 - Tensile Specimen</td>
<td>32</td>
</tr>
<tr>
<td>14.1</td>
<td>EVESR Bundle Flow Control Diagram</td>
<td>42</td>
</tr>
<tr>
<td>14.2</td>
<td>Schematic Diagram of EVESR Reactor Station Designation</td>
<td>44</td>
</tr>
<tr>
<td>14.3</td>
<td>Schematic of EVESR Bundle Steam Piping</td>
<td>45</td>
</tr>
<tr>
<td>14.4</td>
<td>Data Reduction Code Output Variation of Flow Rate with Bundle Power</td>
<td>48</td>
</tr>
<tr>
<td>14.5</td>
<td>Data Reduction Code Output Variation of Reactor Pressure Drop with Flow Rate</td>
<td>49</td>
</tr>
<tr>
<td>14.6</td>
<td>Exit Steam Piping Orifices Variation of Orifice Pressure Drop with Flow Rate</td>
<td>50</td>
</tr>
</tbody>
</table>
### LIST OF TABLES

<table>
<thead>
<tr>
<th>Table No.</th>
<th>Title</th>
<th>Page No.</th>
</tr>
</thead>
<tbody>
<tr>
<td>7.1</td>
<td>Tubing for ESADA-VESR Fuel Fabrication</td>
<td>18</td>
</tr>
<tr>
<td>8.1</td>
<td>VESR Source Irradiation</td>
<td>34</td>
</tr>
<tr>
<td>18.1</td>
<td>Maximum Heat Flux and Power Density in Mark II Core</td>
<td>61</td>
</tr>
</tbody>
</table>
1.0 INTRODUCTION AND SUMMARY

1.1 Introduction

This is the third in the series of quarterly progress reports to be issued covering the work performed on the EVESR-AEC Nuclear Superheat Fuel Development Project under Contract AT(04-3)-189, P.A. 29. The EVESR plant design and the design of the first core load of fuel is described in detail in GEAP-4105, "The First Quarterly Report," and in APED-3958, "Final Hazards Summary Report for the ESADA-VEST." The third quarterly report, GEAP-4200, is intended to cover the technical progress on the AEC-sponsored EVESR Nuclear Superheat Fuel Development Project for the period between December 1, 1962 and February 28, 1963.

The EVESR-AEC Nuclear Superheat Fuel Development Project has been organized into the following work task areas:

Sub-task A-1: Program Planning and Evaluation
Sub-task B-1: Mark II Fuel Design
Sub-task B-2: Fuel Design, Engineering Physics
Sub-task C-1: Fuel Process Development and Fabrication
Sub-task C-2: Fuel Element Process Development and Fabrication
Sub-task C-3: Neutron Source Fabrication
Sub-task C-4: Fuel Evaluation Instrumentation
Sub-task C-5: Coolant Chemistry Sampling Equipment (Design & Installation)
Sub-task D-1: Fuel Tests, Engineering Physics
Sub-task D-2: Fuel Tests, Engineering Assistance for Site
Sub-task D-3: Fuel Tests, Test Procedures
Sub-task D-4: Fuel Tests, Data Reduction
Sub-task D-5: Fuel Activity Release and Coolant Chemistry
Sub-task E-1: Post-Irradiation Examination Planning and Evaluation
Sub-task E-2: Pre- and Post-Irradiation Examinations
Sub-task F-1: Advanced Fuel Design and Process Development
Sub-task F-2: Advanced Fuel Fabrication

These tasks will serve as a basic outline for the reporting of progress in this and all future quarterly progress reports.

1.2 Summary

The schedule for initial loading of fuel in the EVESR reactor has been delayed from February 1963, until the latter part of May 1963. This delay has resulted from an extension of the construction period due primarily to the addition of the R&D instrumentation and coolant chemistry systems and due to the inability to obtain the EVESR operating license on the schedule previously expected. Until such time as the fuel has been installed in the reactor, the EVESR R&D program activity will consist primarily of work in planning the development programs, and completion of the initial fuel design, development and fabrication.

Some of the significant results reached during this reporting period are as follows:

1. The subcontract was negotiated with Bechtel Corporation to install the R&D fuel evaluation instrumentation and coolant sampling systems.

2. Meetings were held between the General Electric Company and staff engineers of the Division of Licensing and Regulation of the AEC to discuss the EVESR Hazards Report (APED-3958). The ESADA-VESR Final
Summary Hazards Report and request for receipt of the operating permit are scheduled to be reviewed by the AEC and the ACRS during their meeting scheduled for March 21, 1963.

3. Design of the Mark II EVESR fuel has been completed, fabrication of this fuel has been initiated. The scheduled delivery of the 32 fuel bundles at the EVESR site is May 15, 1963.

4. Insertion approval has been received for the irradiation of the EVESR neutron source in ETR, Idaho Falls, and shipping and cask arrangements have been made.

5. Work is progressing on schedule in the planning activity, completion of engineering for the R&D coolant chemistry system, the R&D instrument system, the initial core loading and physics measurement program, and the fuel performance data handling procedures. In this regard significant progress has been made in the incorporation of a data retrieval system which will provide for orderly accumulation of fuel fabrication information, fuel design variables, fuel performance data and post-irradiation inspection.

6. Initial scoping activity in the advanced fuel program has indicated possible performance limitations for advanced fuel. Parametric evaluations will be made to establish the advanced fuel programmatic requirements.

2.0 EVESR REACTOR AND PLANT DESCRIPTION

The EVESR reactor and plant is described in Section 2.0 of the "First Quarterly Report," GEAP-4105, and in APED-3958, "Final Hazards Summary Report for ESADA-VESR."
3.0

**SUB-TASK A-1 PROGRAM PLANNING AND EVALUATION**

3.1

**Installation of R&D Program Instrumentation at EVESR**

Negotiations were completed and AEC concurrence obtained to contract with the Bechtel Corporation to install the fuel evaluation instrumentation (Task C-4) and coolant sampling (Task C-5) systems in the EVESR plant. It has been deemed the best course of action from both the schedule and cost standpoints to employ the Bechtel Construction forces now at the EVESR site for this installation work.

The installation work is divided into two phases as follows:

(a) **Phase A** - Installation of instrument sensing lines through the first block valve as required to permit hydrostatic pressure testing of the EVESR high pressure systems.

(b) **Phase B** - Completion of R&D coolant chemistry sampling system in containment and condenser building.

3.2

**EVESR R&D Instrument System**

The thirty-two fuel assembly flows, pressures, and temperatures are measured in the individual piping extending outside the reactor pressure vessel as shown in Figure 3.1. These control room recorded readings are corrected analytically for pressure and heat losses back to the bundle outlet. The local heat generation in the nine fuel elements in each of the fuel assemblies will be obtained from nuclear calculations compensated to agree with the data from the critical test facility and the ten flux wire monitors in the five typical bundle locations in the EVESR core. With the distribution of the heat generation set, the moderator heating established
from appropriate feedwater and moderator measurements, and the flow and exit enthalpy established from the readings in the external piping, a complete solution of the cladding and fuel temperatures is possible for the fuel assembly. The control room operated flow control valves can be used to set up individual test conditions in the various fuel assemblies.

The above data reduction procedure would permit a complete evaluation of bundle performance with essentially no in-core instrumentation. To be of value the accompanying analytical techniques must be checked out thoroughly and corrected where required to improve accuracy. To enable such a check to be made, extensive in-core instrumentation is planned for the EVESR Mark II core. There is one fully instrumented assembly in the core. Included in the fully instrumented fuel assembly are:

(a) Ten flux wire monitors for determining the axial and radial distribution of the neutron flux within the assembly.

(b) Nine mid-pass and nine exit steam temperature readings to permit a closer study of the power distribution between elements in the fuel assembly and to check the heat split between the inner and outer fuel claddings on all nine elements.

(c) The temperature in the inlet plenum is checked to determine the degree of superheating obtained in the downcomer.

(d) The temperature in the exit steam plenum will be used to check the analytical prediction based on corrected readings from the external piping.
(e) Six fuel cladding temperatures will be measured in this one assembly. This will require direct measurement of the cladding temperature in-pile.

(f) Three of the typical element positions in the fuel assembly will be checked for flow split between elements.

(g) A thermocouple will be attached to the bottom of the fuel assembly to measure the moderator inlet temperature to the bundle.

There are eight other bundles with lesser amounts of instrumentation of greater reliability which will be arranged as shown in Figure 3.2. The codes key the instruments in the fully instrumented fuel assembly to each of the eight assemblies. These eight fuel assemblies are located strategically in a quarter core symmetry. Some of the duplications in instrumentation are intentional to provide cross-checks on readings and to allow for spares in case of fuel failure.

As shown in Figure 3.1, there are flow meters installed in the moderator fuel water and the reactor inlet steam. Not shown are flow meters and thermocouples in the main and divert headers. All of this instrumentation permits gross heat and mass balances to be made which, in turn, can be checked against the data from the thirty-two fuel assemblies.

In Figure 3.1, the principal radiochemistry sample points around the reactor are shown. These are as follows:
Figure 3.2. EVESR R and D Instrumentation

- Fuel Element Steam Exit Temperature, 9
- Fuel Element Steam Temp. Between Passes, 9
- Flux Wires, 6
- Water Temperature, 4
- Gross Steam Inlet Temperature, 4
- Gross Steam Outlet Temperature, 4
- Midpoint on the Passes, 1
- Element Flow Rate, 1
- Radio Chemistry, 32
- In-Core Flux Monitors, 5
(a) Four fuel assembly inlet samples to determine if centrally located fuel assemblies suffer from more or less moderator water carryover than fringe located fuel assemblies.

(b) Exit steam samples from each of the thirty-two fuel assemblies will be drawn from the individual external steam pipes.

(c) Steam samples are also drawn from the incoming saturated steam and the mixed outlet steam in the main and divert headers.

In Figure 3.3 and Figure 3.4 the total EVESR instrumentation is listed showing that portion furnished by the EVESR plant and the additional instrumentation furnished by the R&D program.

3.3 Planning and Review Activity at EVESR Site

The following was accomplished in preparation for plant startup:

(a) Assisted in the re-vamping of critical test program fuel loading sequence to minimize chances of possible damage to instrumented fuel assemblies.

(b) Requested a change in source shape to facilitate its removal without removing surrounding fuel.

(c) Participated in discussions pertaining to preoperational fuel storage and post-operation fuel examination in the fuel pool.

(d) Continued preparation of R&D operating procedures as availability of information permitted.
Figure 3.3
ESADA - VESR Instrumentation for Fuel Performance Evaluation

SUPPLIED BY SUPERHEAT FUEL R & D PROGRAM
- - - - PART OF ESADA PLANT INSTRUMENTATION

SENSING POINTS
NEAR REACTOR

TEMPERATURE
INSTRUMENTATION

32 THERMOWELLS
IN OUTLET STEAM LINES

160 THERMOCOUPLES
IN 8 FUEL BUNDLES - QUARTER
CORE INSTRUMENTATION

PRESSURE
AND FLOW
INSTRUMENTATION

12 FLOW METERS
IN OUTLET STEAM LINES

20 FLOW METERS
IN OUTLET STEAM LINES

4 PRESSURE TAPS
IN OUTLET STEAM LINES

EQUIPMENT IN
CONTAINMENT BLDG

THERMOCOUPLE
REFERENCE JUNCTION
STATION

64 TC
LEADS

160

24

3/4"

PIPING

40

4

EQUIPMENT IN
CONTROL ROOM

APPROX 220

ALL ELECTRIC
CABLES

R & D
PROGRAM
INSTRUMENT
PANEL

BLOCK VALVES

PANEL
CA 08

12

ALL TUBING

TRANSUCER
RACK
4.0 SUB-TASK B-1 FUEL DESIGN

4.1 General Design Progress

During this period the design work on the Mark II fuel bundles has been confined to changes required to assist manufacturing.

The process development work associated with the fabrication of the instrumented fuel bundles has suffered a delay due to the long delivery of thermocouples. It is planned to fabricate the fuel rods that include cladding thermocouples using the same swaging and pressure expansion processes as are used for the other fuel rods of the Mark II type. Following the success obtained in the pellet grooving development work, a segment of a grooved pellet type of fuel rod was fabricated using grooved steel pellets to check the swaging and expansion operations and the effect of this configuration on potential wrinkles that could occur near the pellet groove. The sample rods were assembled and exposed to a pressure of 1250 psi at 570°F and 1000 psi at 1200°F. No wrinkling or other distortion occurred in the cladding over the grooved section of the pellets. The next step in the prototype fabrication program will include the assembly of a full-sized instrumented fuel rod using grooved UO₂ pellets.

4.2 Moss Landing Flow Tests

The destructive examination of the Moss Landing prototype fuel bundle has been completed. The general conclusion that has been reached regarding the operation of the prototype fuel bundle indicates that the bundle operated as anticipated and that the design changes that were made in
the latter part of the Mark II fuel design effort have been justified in light of what was learned in the Moss Landing Test Program.

A complete disassembly of the fuel bundle was made so that visual examinations could be made of the riser-downcomer and its internal flow passages; the interior and exterior surfaces of the internal and external reducer sections; the lower tube sheet and the support frame; the upper tube sheet; the individual fuel rods; and the process tubes, velocity boosters, and miscellaneous supports and spacers.

The inside reducer section showed signs of bending approximately 1/8" deep on each of its four sides due to the high pressure on the outside of this component during the test which simulated an outlet pipe break accident. In spite of the distortion, no failure of this component occurred during the test. The process tubes appear to have survived all testing in excellent shape. After splitting one of the process tubes into two halves to permit an examination to be made of the process tube inside surfaces, it was observed that the spiral wire spacer had made a rubbing contact with the process tube but no appreciable wear, if any, was observed. The velocity booster tubes were examined and appeared to be in very good condition. The process tube spacers, which consist of layers of wire grids welded to stainless steel bands, also appear to perform their function satisfactorily. Three fuel rods were removed from the process tubes without breaking the spacer wires attached to the rods. The remaining fuel rod spiral wire spacers were observed to have broken at the weld affected zone near the attachment of the spacer wire on the bottom and top end plugs. The three fuel rods that did not suffer damage to the spacer wires were
fabricated with the spacers spot welded to the outside cladding of the fuel rod, rather than to the end plugs, as was the case where the spacer wires were broken. However, it was also observed that the fuel rods on which the spacer wires were spot welded to the outside cladding had been subjected to water leakage at the spot welds for the spacer wires. It appears from this examination that the decision to use the dimpled process tube for the Mark II fuel bundle spacer configuration has been merited, since this type of spacer is not subject to either of the failure mechanisms described above. The other portions of the fuel bundle hardware all appear to have operated successfully.

5.0 SUB-TASK B-2  FUEL DESIGN - ENGINEERING PHYSICS

5.1 Final Physics Report

This report, APED-4074, "Final Physics Report, Empire State Atomic Development Associates Vallecitos Experiment Superheat Reactor," has been distributed.

5.2 EVESR Core Design and Fuel Following

Measured Mark II fuel process tube thicknesses are averaging between .030 and .031 inches, compared to a nominal design thickness of .028 inches. (The order to the vendor specified an .028 minimum tube thickness.) This thickness over nominal, results in a reactivity decrease of approximately 0.4%Δk/k (equivalent to a loss in burnup of about 200 MWD/T of uranium).

As a result of the decision to significantly increase the number of metal specimens being located inside velocity boosters of the Mark II fuel (for the purpose of determining irradiation effects of tensile and impact
properties), the reactivity decrease due to their presence may be as much as $0.3\% \Delta k/k$ (compared to an earlier estimate of $0.1\% \Delta k/k$).

The following tabulation summarizes the reactivity changes to the burnup life of the Mark II core which have occurred since the EVESR Final Physics Report (APED-4074) final draft was released to the printers:

<table>
<thead>
<tr>
<th>Change</th>
<th>Reactivity Effect</th>
</tr>
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<tbody>
<tr>
<td>Substitution of Incoloy for 304 SS Fuel Clad, Zone 5</td>
<td>- $0.4% \Delta k/k$</td>
</tr>
<tr>
<td>Received UO$_2$ enrichment 0.1% less than design values</td>
<td>- 0.6</td>
</tr>
<tr>
<td>Nominal moderator void content, 9% (down from 14%)</td>
<td>+ 0.5</td>
</tr>
<tr>
<td>.030-.031 process tube thickness vs. .028 design nominal</td>
<td>- 0.4</td>
</tr>
<tr>
<td>Increased quantity of metal specimens for irradiation testing</td>
<td>- 0.2</td>
</tr>
</tbody>
</table>

Net Reactivity Change

- $1.1\% \Delta k/k$

The net effect of these changes has been to take away the additional excess which was purposely requested in order to provide good confidence that initial operation at 12.5 MW thermal, with equilibrium xenon and samarium, would require full insertion of the four central control rods. The use of up to eight stainless steel channels to insure this operating pattern, or to assure meeting a one-stuck-rod criterion, now appears unnecessary. However, these alternate channels will be available in the event that the actual core excess reactivity in the hot condition significantly exceeds predictions.

5.3 EVESR Safeguards

Via meetings and letters, questions from staff engineers of the Division of Licensing and Regulation of the AEC were answered in the physics area
regarding the accident analysis, fuel storage, the adequacy of the neutron sources to provide reliable count rates on the startup instrumentation, the relative ability of observing the effect of control rod withdrawal on detector count rates, and technical specifications relating to advanced fuel design. Additional analyses are being made in the preceding areas as necessary. These staff engineers have indicated general acceptance of the information provided in the EVESR Hazards Report (APED-3958) and at subsequent meetings. However, relating to the Technical Specifications, they have requested a complete itemization of all omissions from the Appendix A outline.

6.0 SUB-TASK C-1 FUEL PROCESS DEVELOPMENT AND FABRICATION

6.1 Schedule

Completion of the first 32 bundles is scheduled for May 15, 1963.

6.2 Procurement

(a) Process Tubes

The initial order for process tubing was largely out of specification and was completely replaced. The replacement lot appears basically sound, and a portion has been released for dimpling.

(b) Riser-Downcomer

Ten units have been completed. The remaining 30 are in production.

(c) UO₂ Powder

Procurement of the additional 4.0% enriched UO₂, occasioned by a design change in the spare bundles, is unresolved. Commission approval for a blending operation is needed.
(d) **Steam Inlet Reducers**

These have been received and accepted. Sandblasting and passivation are required for removal of scale built up in the heat treating operation.

(e) **Adapter Tubes**

Delivery of the tubing stock has been delayed by vendor's difficulties in ultrasonic testing. The 304 and at least a majority of the Incoloy are now ready for shipment, subject to VQA approval.

(f) **Other Assembly Hardware**

Deliveries appear compatible with the production schedule. Problems exist, however, in the areas of bar stock, spacers, nose pieces and flux wire tubes.

7.0 **SUB-TASK C-2 FUEL ELEMENT PROCESS DEVELOPMENT AND FABRICATION**

7.1 **Materials Procurement**

Difficulties encountered by the vendor in fabrication of fuel cladding have resulted in delays in the schedule of deliveries. Therefore, in order to expedite the incoming inspection and the preliminary work required on all tubing, partial shipments of items have been made as tubing is completed. The status of all tubing is summarized in Table 7.1. All items will be completed by March 1, 1963, with the exception of the large sizes of Incoloy and of Type 310 VM. Losses during fabrication of these two items have made it necessary to obtain additional stacking material to complete the quantities required, and have therefore resulted in a projected completion date of about March 11, 1963.
<table>
<thead>
<tr>
<th>Alloy</th>
<th>Size</th>
<th>No. Pcs. Ordered</th>
<th>No. Pcs. Required</th>
<th>Fabrication Complete</th>
<th>U/S Test Passed</th>
<th>Shipped</th>
<th>Received at APED</th>
<th>Released by Q.C.</th>
</tr>
</thead>
<tbody>
<tr>
<td>Inconel</td>
<td>S</td>
<td>190</td>
<td>90</td>
<td>190 pcs.</td>
<td>151 pcs.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Inconel</td>
<td>L</td>
<td>190</td>
<td>90</td>
<td>39 pcs.</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Incoloy</td>
<td>S</td>
<td>190</td>
<td>171</td>
<td>114 pcs.</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Incoloy</td>
<td>L</td>
<td>190</td>
<td>171</td>
<td>68 pcs.</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>304 VM</td>
<td>S</td>
<td>190</td>
<td>27</td>
<td>190 pcs.</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>304 VM</td>
<td>L</td>
<td>190</td>
<td>27</td>
<td>146 pcs.</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>310 VM</td>
<td>S</td>
<td>72</td>
<td>36</td>
<td>65 pcs.</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>310 VM</td>
<td>L</td>
<td>72</td>
<td>36</td>
<td>21 pcs.</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>304 H</td>
<td>S</td>
<td>16</td>
<td>9</td>
<td>18 pcs.</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>304 H</td>
<td>L</td>
<td>18</td>
<td>9</td>
<td>18 pcs.</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
All of the bar stock required for end plug fabrication has been received, inspected, and released to the equipment shop for fabrication.

7.2 Fabrication Development - Regular and Instrumented Assembly

Fabrication techniques have been developed for the instrumented fuel rods (Drawing 612D997, pg. 26, Second Quarterly Report). Preliminary tests were conducted with 18 inch capsules to develop methods for attaining the required pellet-to-clad gaps. It was concluded that the swaging and pressurization processes could be used to reduce the outer and inner clad-to-fuel gaps without detrimental effects. At-temperature collapse tests on the capsules ($570^\circ F$, 1250 psi and $1200^\circ F$, 1000 psi) indicated no plastic deformation in the outer clad over the outer thermocouple groove. Thermocouple welding development pointed out the need for minor modifications in the top end plug design. These changes will be incorporated in the final design.

The assembly techniques for the instrumented fuel rod will be quite different than those used on the standard fuel rods. This difference is necessary to accommodate the two thermocouples which are attached to the inner and outer clad and leave the fuel envelope through the top end plug. These thermocouples will be approximately 25 feet long and special handling is required during welding and swaging operations.

An instrumented fuel rod prototype was fabricated using grooved steel pellets and 1/16 inch diameter wire to simulate thermocouples. This proved the basic feasibility of the assembly process and allowed tooling additions and refinements. A prototype with grooved UO$_2$ pellets and regular thermocouples is now being fabricated to check out tooling and finalize process instructions.
Final process instructions have been issued for fabrication of the standard fuel rods. Production travel cards have been prepared. Each fuel rod will have a travel card which will follow it through the production sequence to ascertain adequate quality control and collect data regarding pre-irradiation information.

7.3 Fabrication

Materials required for fabrication of all fuel segment hardware (end plugs, sleeves, shims, and vent plugs) have been released to the equipment shop. Shop follow-up work is being performed as required.

The present shop status of the components is as follows:

- **End plugs** - All bottom end plugs are complete. Fabrication of the top end plugs has been initiated and will follow a priority schedule with Inconel being the first material.
- **Sleeves** - Fabrication of all sleeves has been completed.
- **Shims** - Work on the shims has not as yet been started.
- **Vent plugs** - Work on the vent plugs has not as yet been started.

7.4 ESADA Fuel Pre-Irradiation Measurements

Mechanical stability of superheat fuel cladding may be one of the limiting factors in fuel overall performance. Bowing or swelling might make it difficult or impossible to use a fuel element for further irradiation service. It is necessary to have small coolant flow passages to achieve satisfactory heat transfer coefficient values. Bowing, warping, swelling, or collapse of the cladding could block or modify the flow passages to such an extent that portions of the cladding would be at temperatures
high enough to promote premature failure. Comparison of pre- and post-irradiation dimensions of a fuel element can be used to determine its mechanical stability during reactor service.

The following pre-irradiation dimensional measurements will be taken on the ESADA-VESR Mark II fuel:

(a) Outer clad outside diameter at five-inch axial intervals and at $90^\circ$ angular intervals

(b) Axial profile (bow) at the same intervals as outside diameter.

(c) Length.

(d) Inner clad (inside) diameter at six-inch axial intervals. This dimension will be measured separately from the others, using three-point micrometers.

The following criteria were applied in determining the type of measuring equipment necessary.

(a) Measurements should be reproducible and accurate.

(b) The equipment should be easy to operate rapidly for securing maximum data at minimum cost.

(c) The equipment should be adaptable for in-cell (FMI) use for post-irradiation measurements.

(d) Maximum expected amounts of bow, ovality, or deposits should not interfere with post-irradiation measurements.
Air gages on pneumatic probes can meet the first three requirements, but in order to ascertain the fourth requirement without going into a development program, a device using dial gages was chosen.

Design of the measuring equipment is shown in Drawing 212p865 (Figure 7.1). The equipment begins with a base piece which is made from a six-inch aluminum channel. A series of rollers is mounted on the base piece, and a carriage made from a five-inch channel rests on four of these rollers while other rollers bear against the side of the carriage to guide it in a straight line. Four pair of horizontally opposed dial gages are mounted on suitable fixtures attached to the carriage. The gage pairs are located at 15-inch intervals, so carriage travel of 15 inches permits outside diameter readings at any point along the active length of the fuel element (except for 1/2 inch at the bottom end). Diameter is measured by adding the dial gage readings on each gage pair, and profile is measured by considering each reading individually. The horizontal position of the gages minimizes gravitational effects on profile readings. Lever operated linkages on the carriage retract the gage pairs from the fuel element during carriage motion or fuel moves. This protects the dial gage spindles as well as minimizing scratches on the fuel element.

At each end of the base piece is a "V" block. The fuel element is positioned above the carriage and rests on the "V" blocks. A spring loaded plunger at one end of the fuel element seats a shoulder against the "V" block at the other end. A dial gage contacts the plunger and gives the length measurement. During fuel moves the plunger is retracted by a lever.
### Checking Fixture

**Title:**Checking Fixture  
**First Made For:** VESE Fuel Rod

<table>
<thead>
<tr>
<th>PART NO.</th>
<th>NAME</th>
<th>DRAWING NO., DESCRIPTION, MATERIAL, WEIGHT</th>
</tr>
</thead>
<tbody>
<tr>
<td>823</td>
<td>Shim</td>
<td></td>
</tr>
<tr>
<td>1629</td>
<td>Shim</td>
<td></td>
</tr>
<tr>
<td>1430</td>
<td>Shim</td>
<td>DETAIL</td>
</tr>
<tr>
<td>431</td>
<td>Tension</td>
<td></td>
</tr>
<tr>
<td>233</td>
<td>Bushing</td>
<td>STN STL BAR TYPE 324</td>
</tr>
<tr>
<td>433</td>
<td>Washer</td>
<td>1/20 X 1/2 ID X 1/2 TH SLN N40 P73</td>
</tr>
<tr>
<td>434</td>
<td>Washer</td>
<td>1/20 X 1/2 ID X 1/2 TH N90 P41</td>
</tr>
<tr>
<td>235</td>
<td>Bolt</td>
<td>1/2-20 X 1 1/2 HEX H10 SLN N28 P120</td>
</tr>
<tr>
<td>236</td>
<td>Locknut</td>
<td>FLEXLOC PT50 FASA 832 STL PRESSO</td>
</tr>
<tr>
<td>1237</td>
<td>Locknut</td>
<td>FLEXLOC PT50 FASA 1024 OR EQ</td>
</tr>
<tr>
<td>2833</td>
<td>SCR Fil</td>
<td>10-24 X 1/2 STN STL N55 P17 005</td>
</tr>
<tr>
<td>2839</td>
<td>SCR Fil</td>
<td>10-24 X 1/2 N55 P17 010</td>
</tr>
<tr>
<td>1641</td>
<td>Washer</td>
<td>1/20 X 1/2 ID X 1/2 TH N40 P35</td>
</tr>
<tr>
<td>432</td>
<td>Washer</td>
<td>1/20 X 1/2 ID X 1/2 TH N90 P41</td>
</tr>
<tr>
<td>434</td>
<td>SCR Fil</td>
<td>6-32 X 1/2 N55 P13 005</td>
</tr>
<tr>
<td>2844</td>
<td>SCR Fil</td>
<td>6-32 X 1/2 N55 P13 010</td>
</tr>
<tr>
<td>1646</td>
<td>SCR Hex</td>
<td>1/2-20 X 1/2 N17 P21 005</td>
</tr>
<tr>
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<td>Washer</td>
<td>1/20 X 1/2 ID X 1/2 TH N40 P35</td>
</tr>
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<td>SCR Hex</td>
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</tr>
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<td>SCR Fil</td>
<td>1/2-20 X 1/2 N55 P21 005</td>
</tr>
<tr>
<td>150</td>
<td>SCR Hex</td>
<td>10-24 X 1&quot; STN STL N17 P17 014</td>
</tr>
<tr>
<td>151</td>
<td>Rollpin</td>
<td>CAT #79 048 250-2500 ESRA OR EQ</td>
</tr>
</tbody>
</table>

**Description of Groups:**

**Revisions:**

**Prints To:**

---

**Figure 7-1. Drawing 212E865 - Assembly Checking Fixture**

23
All dial gages will be 3-5/8 inch diameter. The length measurement gage will have 0.001 inch graduations and a one-inch range. The diameter gages will have 0.001 inch graduations and 0.4 inch range. This permits measurements on fuel elements having up to 0.2 inch bow. More bow than this may be present in irradiated fuel, so the design permits replacement of the 0.4 inch range gages with one-inch range gages having shorter contact points. Then up to 1/2 inch bow can be accommodated.* The attachements for the diameter gages have one inch vertical travel, so a bowed element can be measured at any desired angular orientation.

Additional information concerning physical description, special features, and operation of the measuring equipment follows.

(a) A dimensional standard will be made for use in making initial adjustments and periodic checks on dial gage readings.

(b) All critical moving parts are spring loaded to minimize effect of side and end play.

(c) Previous experience indicates that ovality in fuel cladding may spiral along the fuel length. Several 0° and 90° diameter readings will be spot checked for each element. If appreciable ovality exists, additional readings at 45° intervals will be taken to give more information on amount of ovality.

Dial gages have been procured and fabrication of parts has begun.

* Extrapolating results of previous superheat irradiations indicate that 1/2 inch should be an upper limit for bow of the ESADA-VESR fuel. See GEAP-4024, Figure 3.1.
Strain Cycle Tests

A strain cycle study was initiated to evaluate the wrinkling effect of fuel cladding under simulated reactor conditions. This wrinkling was observed in the ESH-1, 0.028 inch wall fuel elements. Some information is available on the strain cycle performance of 0.016 inch wall clad, but the present series of tests are to be performed on tubing of ESADA-VESR wall thickness (0.028 inch).

Four specimens of Inconel tubing (1.25 inch O.D. x 0.028 inch wall) have been prepared and will be cycled (without irradiation) at 1300°F with a fixed strain range of ± 1 percent. Cycling will be performed by pressurizing the tubing specimens to alternately collapse them against I.D. and O.D. mandrels. One specimen will be cycled to failure and the remainder will be cycled for lesser times, probably 1/2, 1, and 2 cycles to determine the number of cycles required to initiate wrinkling and also the extent and character of the wrinkles formed. This information will be interpreted in terms of fuel element design and performance, and will provide a guide for ESADA core operation procedures.

ESADA-VESR Coupon Irradiation Program

The use of a material in a reactor is dependent not only on its pre-irradiated mechanical properties, corrosion resistance to reactor coolant, weldability and acceptable nuclear properties, but also on the extent of the property changes induced by nuclear irradiation. Data of this type are necessary for both fuel element design and understanding of the fuel element irradiation performance. Radiation-induced mechanical property changes significantly influence the allowable operating stress levels
in fuel element designs. The proper choice of materials for cladding and structural applications can be made only if the effects of radiation damage are known.

Five alloys have been chosen as fuel cladding alloys for the ESADA-VESR reactor - the nickel base alloys Incoloy and Inconel, the vacuum melted austenitic steels 304 VM and 310 VM, and commercial 304 stainless steel. The selection of the nickel base alloys and vacuum melted steels was based upon the best available information regarding their resistance to stress corrosion cracking in superheat environments.

7.6.1 Objectives of the Experiment

The alloys selected for investigation are the ESADA-VESR cladding materials, Incoloy, Inconel, 310 VM, and 304 VM. Efforts will be concentrated on the nickel base alloys, Incoloy and Inconel.

The general objectives of this investigation are listed below.

(a) Study the effects of neutron flux upon the room temperature and 1100 °F mechanical properties of selected alloys which were subjected to irradiation at temperature of 660 - 800°F. Properties to be studied include 0.2 percent yield strength, ultimate tensile strength, percent elongation, and microstructural hardness. The integrated neutron exposure (energies >1 Mev) and pre-irradiation condition will be investigated as independent variables.

(b) Study the annealing kinetics of radiation damage in alloys of interest as a function of pre-irradiation structure and post-irradiation annealing conditions. This investigation will be conducted to
determine the amount of reversible damage due to irradiation and the thermal conditions necessary for property recovery.

(c) Study the effects of neutron bombardment on aging phenomena and microstructural changes and determine their effect on mechanical behavior and fracture characteristics.

7.6.2 Experimental Details

A large amount of information has been published concerning radiation-induced changes in the tensile properties of austenitic stainless steels. However, most of the irradiations were carried out at temperatures significantly lower than those expected in superheat reactors. At higher irradiation temperatures, partial annealing of the damage occurs simultaneously with irradiation, preventing the direct extrapolation of data obtained at lower temperatures. Specimens for this investigation will be inserted into the flow booster tubes of 7 central bundles of the core. The temperature of the superheated steam in contact with the specimens will range from 660 - 800°F at an average fast neutron flux of $1.57 \times 10^{13}$ nvt. Engineering property measurements (hardness and tensile tests) will be used to detect radiation damage affecting the mechanical performance of the materials. Target neutron flux exposures will be $10^{20}$ and $10^{22}$ nvt (above 1 mev).

All annealed tensile specimens of Incoloy, Inconel, 304 VM, and 310 VM will be fabricated from the tubing purchased for ESADA-VESR fuel cladding. Incoloy and Inconel will also be studied in the cold worked condition. These specimens will be machined from development stock. Due to the
space limitations of the flow booster tube, a sub-size tensile coupon will be used and is shown in Drawing 104R1587 (Figure 7.2). The specimens will be located inside the velocity booster of the center rod of each of 7 bundles: B-3, B-4, C-2, C-4, C-5, D-2 and E-3. The increased neutron absorption due to these coupons will not affect the operation of the reactor in that the interior of the flow booster is otherwise an unused spacer. The specimens will be attached to a holder with nickel flux wire and positioned in the flow booster tube. Radiochemical analysis of these wires after irradiation will give accurate fast flux measurements for each specimen. Each tube will contain 56 tensile coupons.

7.6.3 Data Expected from Investigation

(a) Effects of irradiation exposure upon hardness and tensile properties of the cladding materials.

(b) Combined effects of radiation damage and extended thermal exposure on microstructural characteristics of test coupons.

(c) Annealing kinetics of radiation damage.

7.6.4 Summary of the Irradiation Experiment

The selection of the ESADA-VESR cladding materials was based upon the best information available through 1962 concerning resistance of materials to stress corrosion attack. However, no information is available dealing with the effects of irradiation on the physical and mechanical properties of the cladding alloys. It is the purpose of this investigation to determine the effects of radiation damage on these alloys (1) for
Figure 7-2. Drawing 104B1587 - Tensile Specimen
application to fuel element design of future cores; and (2) a more comprehensive understanding of the ESADA-VESR fuel element performance.

The investigation will be conducted as follows:

(a) Fabricate tensile coupons from ESADA-VESR incoming tubing and development stock. Compile the complete pre-irradiation history of the materials.

(b) Determine pre-irradiation properties and physical structure.

(c) Place specimens in pre-selected flow booster positions. Both high and low neutron flux positions will be utilized to obtain a gradient in integrated neutron exposure. Total exposure will be monitored by flux wires connected with each specimen.

(d) Monitor total exposure of test coupons from available physics and reactor operations data. Withdraw specimens from flow booster tubes when technically feasible.

(e) Determine post-irradiation properties of test coupons and analyze the data.

8.0 SUB-TASK C-3 EVESR NEUTRON SOURCE

8.1 Status

During the quarter all materials were received and the fabrication of beryllium and antimony components was completed. Thus all hardware is ready for irradiation and ensuing assemblage after post-irradiation inspection.
AEC Form 22 was prepared and forwarded for irradiation of rods at NRTS, Idaho Falls, Idaho, in accordance with Commission consignment. Pertinent design data and drawings were forwarded to Phillips Petroleum Company, Idaho Falls, for review and acceptance for irradiation.

Insertion approval has been received for irradiation in ETR, Idaho Falls, and shipping and cask arrangements have been made.

8.2 Source Strength

The four rods will go to ETR for one cycle now selected as #54. ETR thermal flux of $0.8 \times 10^{14}$, fast of $0.6 \times 10^{13}$, gamma heat 1 watt/gm, compare favorably with GETR calculations and should give about 370 curies antimony-124 per rod on removal.

**Table 8.1**

<table>
<thead>
<tr>
<th>ETR Cycle #</th>
<th>Begin Date (Shutdown for Loading)</th>
<th>End Date</th>
<th>Ship to VAL</th>
<th>Arrive VAL</th>
<th>To VESR</th>
</tr>
</thead>
<tbody>
<tr>
<td>53</td>
<td>18 Feb.</td>
<td>1 Apr.</td>
<td>8 Apr.</td>
<td>13 Apr.</td>
<td>20 Apr.</td>
</tr>
<tr>
<td>54</td>
<td>1 Apr.</td>
<td>13 May</td>
<td>20 May</td>
<td>25 May</td>
<td>1 June</td>
</tr>
<tr>
<td>55</td>
<td>13 May</td>
<td>24 June</td>
<td>1 July</td>
<td>6 July</td>
<td>13 July</td>
</tr>
</tbody>
</table>

9.0 SUB-TASK C-1 FUEL EVALUATION INSTRUMENTATION

9.1 Status

The engineering and drawings for the quarter core temperature measurements have been completed. The thermocouples for instrumenting the initial
core have been received. Work has been started on those units which require preparation prior to assembly in the fuel bundles.

The engineering and drawings for the special bundle temperature measurements have been completed. The thermocouples are on order and will be available well ahead of assembly schedules.

Preliminary instructions for eight instrumented fuel bundle assemblies have been approved and the final draft is now being prepared.

Application and design engineering is well underway for the flow instrumentation of three elements in the special bundle. This is scheduled to be completed by March 5.

The vessel penetration designs and drawings have been completed. Two-thirds of the parts have been received and the balance are expected well in advance of scheduled requirement.

The containment penetration sleeves, flanges, and fittings have been delivered to the site ready for installation by Bechtel Corp. The Scotchcast cables for R&D will be delivered to the site by March 1.

Drawings have been completed for the routing of the signal cables and the primary piping to the flow transmitter station. The primary flow transmitters have been received and the installation of the flow signal piping has been started.

The flow instrumentation engineering has been completed and the specification for a flow measuring system are out for bids.
Cabinets for the R&D panel and the temperature reference junction have been received. Twelve valve position indicators and five multipoint recorders have been received for the R&D panel. Three recorders are awaiting delivery pending the selection of flow metering system now out for bids.

The temperature recorders for the quarter core instrumentation are on order, also the associated selector switches.

Detail panel drawings and wiring diagrams are pending the placement of the order for specific flow measuring instrumentation. Engineering release is expected by the end of February.

9.2 The complete R&D instrument system is described in Section 3.2 to show the relationship of this equipment to EVESR plant instrumentation.

10.0 SUB-TASK C-5 FUEL TESTS - CHEMISTRY SAMPLING STATIONS, DESIGN AND INSTALLATION

The detail design of the facilities for chemistry sampling has progressed with several drawings issued. The drawings were completed on the platform for the sample station in the containment building. The layout of the sample lines, coolers, and cooling water system at the 534 level was completed. Detail design of the cooling water supply system, the sample line routing to the sample platform, and the sample line terminations is in progress.

In the condenser building, the drawings have been completed on the sample taps and routing of the sample lines to the sample station. Details of most of the sample line terminations have been completed in check print
form. The cooling water system has been detailed and will include a tank fed with potable water such that the potable water system is isolated by an air break. A pump is required to pump the potable water from the tank to the sample coolers. The sample probe for the saturated steam sample from the gas fired boiler was redesigned to reduce the thermal stresses in the welds.

The question of materials for sample coolers in the containment building was reviewed. The three coolers associated with sampling of the three headers (inlet saturated, main superheat, and divert superheat) will be tubed with Inconel rather than stainless steel tubing. These coolers will see frequent or continuous service and as such, dependable performance is especially important. Inconel is preferred as a tube material because it is less susceptible to stress corrosion cracking than the stainless.

11.0 SUB-TASK D-1 FUEL TESTS - ENGINEERING PHYSICS

11.1 Status

That part of the physics pre-analyses of the initial core loading and critical testing program which could be carried out prior to the measurements is complete, barring unforeseen inconsistencies, errors, or program changes. The rough draft describing the results of the pre-analyses is also complete. This report will include basic cell and bundle lattice cross sections incorporating design data current to December 1, 1962.

12.0 SUB-TASK D-2 FUEL TEST - SITE OPERATIONS

12.1 Status Report

No work will be carried out in this work area until the start of the fuel loading activity.
13.0  SUB-TASK D-3  FUEL TESTS - TEST PROCEDURES

13.1  Status Report

Considerable engineering effort has been devoted in this area in connection with the EVESR planning effort and the EVESR safeguards effort, but the actual preparation of the detailed test procedures for the program is not scheduled to start until later.

14.0  SUB-TASK D-4  FUEL TESTS - DATA REDUCTION

14.1  Test Procedure

There are thirty-two fuel bundles in the EVESR core, each of which contains nine individual fuel elements for a total of 288 elements. The R&D development program objectives require that certain stress levels and peak clad temperatures be maintained for each bundle and that these two parameters be varied throughout the core for the various fuel cladding materials in order to provide a maximum of fuel development information. The stress level in the cladding depends on the as-built clearances between the fuel and the cladding plus the existing combination of heat flux and cladding temperature in the individual fuel element.

The individual element heat fluxes will depend on the bundle power which in turn is subject to whatever power level exists in the core as well as the relative distribution of the power between bundles.

Although it is desirable from an R&D test standpoint to maintain a constant total reactor power and a fixed relative power distribution axially and radially in the core, the individual bundles and fuel elements in the bundle will be subject to startup and shutdown cycles
and to shifts in power distribution resulting from rod movements and fuel burnup. The planned test procedure will consist of having the FVESR reactor operator maintain as nearly as practical a constant rated total reactor power level while the R&D operator regulates the flow rates to the individual fuel bundles in order to produce the desired maximum cladding temperatures. Thus the stress level will be subject to cycling and drifts but the maximum clad temperature in each bundle will be maintained at a fixed value except during low power operation where minimum specified flows must be maintained.

The problem facing the R&D operator is what flow to set for each of the bundles in order to establish a given maximum cladding temperature. This is difficult enough under steady state bundle power conditions and is doubly difficult when the power is constantly shifting in all thirty-two bundles.

Figure 3.1 shows a section through the reactor pressure vessel wall and part of the core consisting of one bundle and the associated piping and R&D instrumentation. Note that the temperature, pressure, and flow is measured in the bundle piping extending outside the reactor vessel. The steam entering the bundles comes from the steam dome in the reactor vessel.

With the bundle flow rate, the exit enthalpy, and the inlet enthalpy measured it is possible to establish the overall bundle power except for heat losses to the moderator and the pressure and heat losses from the bundle inlet and exit piping. An analysis has been developed for this
heat transfer system. The control room recorded readings of flows, pressures, and temperatures are corrected analytically for pressure and heat losses back to the bundle outlet. The local heat generation in the nine fuel elements in each bundle will be obtained from nuclear calculations compensated to agree with the data from the critical test facility and the ten flux wire monitors in the five typical bundle locations in the EVESR core. With the distribution of the heat generation set, the moderator heating established from appropriate feedwater and moderator measurements, and the flow and bundle exit enthalpy established, a complete solution of the cladding and fuel temperatures is possible for the fuel assembly. Thus, it is difficult, but not impossible, to reverse the above procedure and establish the control room readings of flow, pressure and temperature which will produce the desired maximum cladding temperature in the individual fuel bundles.

Since this analysis is complete and since the number of times it must be executed to both set and monitor performance in thirty-two bundles for the extent of the test period is so great, it was decided to program the analysis procedure for APED's Philco 2000 computer. This program will be used initially to construct operating charts which will guide the R&D operator in setting the proper flows in the bundles and will be used later in the test program to monitor day-to-day performance variations.

Except for the flux wire monitors the procedure for testing EVESR fuel as outlined above involved no in-core instrumentation. Since the computer analysis is subject to analysis discrepancies and accumulated errors,
one-quarter of the core has been highly instrumented to provide detailed
test checks against the machine calculations. This instrumentation is
detailed on Figures 3.1 and 3.2. Figure 3.1 shows the location of
this instrumentation in the bundle and Figure 3.2 shows the distribution
of this instrumentation in the core.

Figure 14.1 shows how the flow control valves for all of the thirty-two
bundles are distributed for the core. It will be noted that in some
instances as many as four fuel bundles are controlled by a single bias
flow control valve. If the test program requires different flow rates
in the four bundles controlled by a single bias control valve, it will
be necessary to orifice them differentially to produce the desired
flows.

14.2 Setting Orifice Sizes

During the report period the data reduction portion of the PATSE code
was studied to determine how it could be used to size orifices in EVESR
fuel bundles which empty into a common header. This is necessary in
cases where the flow must be varied between fuel bundles emptying into
a common header since the bias flow control valve is located on the
header.

The orificing procedure outlined below applies specifically to Mark II
fuel, but the same techniques could be applied to any advanced fuel
design. This assumes however, that the associated computer code for the
advanced fuel has the same maximum clad temperature and weight flow
options as presently incorporated in the expanded TAPS V code, PATSE.
Figure 14-1. EVESR Bundle Flow Control Diagram
Under the present EVESR exit steam piping design it is a relatively easy matter to change orifice sizes during shutdown by merely removing the jumper as the orifice is readily accessible in the top of the riser. Thus, if any error is involved in sizing an orifice it is relatively easy to correct the error.

The amount of orificing that can be done is limited by thermal pipe stresses due to a flow induced differential temperature in the four pipe "legs" attached to a common header. It may be deemed necessary to use small diameter orifices in the four bundles in a common header in order to secure a higher percentage of the total pressure drop across the orifice. This would result in better overall control and stability of the reactor exit steam flow, but it would reduce the accuracy of the data reduction techniques by introducing larger losses in the exit piping system. Also heavy orificing increases the temperature differential as noted above. The maximum temperature differential between any two reactor connections on the main superheated steam line is limited by the thermal stresses due to thermal expansion of the piping. The maximum $\Delta T$ is 200°F (code limit) except on the divert steam system (where the maximum differential is 250°F) to which the four center bundles C3, C4, D3 and D4 are connected. This orificing procedure does not apply to these four bundles as they each have an individual bias control valve. However, the maximum allowable $\Delta T$ between the main steam header and the divert header is 200°F also. Hence, it is reasonable to limit the orificing to such a degree that the temperature differential is less than 200°F.
14.3 Analysis and Description of Orifice Sizing and Procedure

Consider the schematic diagram, Figure 14.2, which shows four of the 32 reactor bundles emptying into a common header. The station designation is the same as that used in the data reduction option, CORET, of the DARE code. See Figure 14.3.

$P_1$ = Pressure of reactor steam

Note $P$, $W$, $t$ measured for each bundle at station A.

SCHEMATIC DIAGRAM OF EVESR REACTOR STATION DESIGNATION

FIGURE 14.2
It is possible to write an expression relating the pressure at the two stations where the pressure is identical for all four bundles, namely 1 and a. Thus, in general

\[(P_1 - P_a) = (P_1 - P_M) + (P_M - P_{\text{core}}) + (P_{\text{core}} - P_N) + (P_N - P_G)\]  
\[+ (P_G - P_F) + (P_F - P_A) + (P_A - P_a)\]  
(1)

Noting the bundle core, riser and downcomer pressure drops respectively as

\[(P_M - P_{\text{core}}) + (P_{\text{core}} - P_N) = \Delta P_C\]  
(2)
\[(P_N - P_G) = \Delta P_R\]  
(3)
\[(P_1 - P_M) = \Delta P_D\]  
(4)

it is possible to write equation (1) in a more convenient form.

\[(P_1 - P_a) = \Delta P_C + \Delta P_R + \Delta P_D + (P_G - P_F) + (P_F - P_A) + (P_A - P_a)\]  
(5)

Assuming that the pressure drop \((P_A - P_a)\) is negligible relative to the other pressure drops and noting the orifice pressure drop as,

\[(P_G - P_F) = \Delta P_O\]  
(6)
\[(P_F - P_A) = \Delta P_H\]  
(7)

it is possible to reduce Equation (5) to the following form.

\[(P_1 - P_a) = (\Delta P_D + \Delta P_R + \Delta P_H) + \Delta P_O + \Delta P_C\]  
(8)

or

\[(P_1 - P_A) = \Delta P_D + \Delta P_R + \Delta P_H\]

As seen in Figure 14.2, pressures \(P_1\) and \(P_a\) are equal for each of the four bundles hence, it is clear that

\[(P_1 - P_A)_1 = (P_1 - P_A)_2 = (P_1 - P_A)_3 = (P_1 - P_A)_4\]  
(9)

where the numerical subscript denotes a particular bundle number.

Combining Equations (8) and (9) yields
\[ \sqrt{\Delta P_T + \Delta P_o + \Delta P_{c-1}} = \sqrt{\Delta P_T + \Delta P_o + \Delta P_{c-2}} = \]
\[ \sqrt{\Delta P_T + \Delta P_o + \Delta P_{c-3}} = \sqrt{\Delta P_T + \Delta P_o + \Delta P_{c-4}} \]  \( (10) \)

In order to obtain the orifice size of each bundle such that Equation (10) be satisfied, it is necessary to consider separately the contributions from each of the involved computer code options. Various values of maximum clad temperature and bundle power can be used as input data for the PATSE option. With this input it is possible to obtain from the PATSE option output the bundle weight flow rate and bundle pressure loss \( \Delta P_c \). A characteristic plot of flow rate versus bundle power with the maximum cladding temperature as an independent parameter would be of the form shown in Figure 14.4.

![Data Reduction Code Output Variation of Flow Rate with Bundle Power](image)

**Figure 14.4**
The EVEREST computer data reduction option, CORET, of the DARE code, can be used to determine (for varying input data, i.e., $P_1$, $P_A$, $t_A$ and $W$) as output several values of fuel bundle downcomer and riser pressure drop, $\Delta P_T$, and bundle power. The code in this case would not have any orifice in the riser. It is now possible to combine the output of the above two code options and plot a family of curves as shown in Figure 14.5, i.e., reactor pressure drop, $\Delta P_T + \Delta P_c$, (exclusive of orifice $\Delta P_o$) versus flow rate, $W$, for varying bundle powers.

![Diagram showing variation of reactor pressure drop with flow rate](image)

**DATA REDUCTION CODE OUTPUT**

**VARIATION OF REACTOR PRESSURE DROP WITH FLOW RATE**

**FIGURE 14.5**
The contribution of the orifice pressure drop can now be investigated. Consider the governing pressure drop equation in this case.

\[
\Delta P_o = \frac{8 \cdot W^e}{C_0 \cdot \epsilon \cdot \eta^2 \cdot D_o^4 \cdot \sqrt{0.84 \cdot \left(\frac{B_o}{B_G}\right)^2 + 0.57 \cdot j^2}}
\]

(11)

Equation (11) can be plotted in terms of \(W\) versus \(\Delta P_o\) for various orifice diameters as shown in the illustrative plot in Figure 14.6.

The three sets of curves described above combined with Equation (10) provide enough information to size the orifices for a predetermined or desired maximum clad temperature and bundle power. The orifice sizing procedure is as follows.
(1) The sizing procedure should start with whichever one of the four bundles on a common header has the highest maximum clad temperature and bundle power; as an example suppose that this is bundle number one. Assume that this bundle does not have an orifice in the riser therefore, \( \Delta P_0 = 0 \). In order that Equation (10) be satisfied it is clear that \( \Delta P_T + \Delta P_C \) is the total pressure drop for bundle number one. Locate the power and maximum clad temperature of bundle number one on Figure 14.4 and read the corresponding flow rate, \( W_1 \). Now enter Figure 14.5 at \( W_1 \) and the same bundle power as above and read \( (\Delta P_T + \Delta P_C)_1 \) for bundle number one.

(2) Enter Figure 14.4 at the bundle power and maximum clad temperature of any of the other three bundles, say bundle number two, and read \( W_2 \). Enter Figure 14.5 at \( W_2 \) and its corresponding bundle power and read \( (\Delta P_T + \Delta P_C)_2 \). Now the required orifice pressure drop for bundle number two can be calculated from Equation (10).

\[
(\Delta P_T + \Delta P_C + \Delta P_0)_1 = (\Delta P_T + \Delta P_C + \Delta P_0)_2
\]

but

\[
(\Delta P_0)_1 = 0
\]

Therefore,

\[
(\Delta P_0)_2 = (\Delta P_T + \Delta P_C)_1 - (\Delta P_T + \Delta P_C)_2
\]

Figure 14.6 can now be used to obtain the orifice diameter of bundle number two since both its flow rate and orifice pressure drop are known.

(3) Repeat step 2 above for the other two bundles to obtain their orifice diameters.
In the example above, bundle number one (the maximum flow bundle of the four) was assumed to be unorificed. As an alternate procedure any arbitrary orifice size could have been selected for this bundle. In this case Figure 14.6 would be used to obtain \((\Delta P_o)_1\) corresponding to the arbitrary orifice size selected. The total pressure drop for bundle number one would then be \((\Delta P_T + \Delta P_C + \Delta P_c)_1\) instead of just \((\Delta P_c + \Delta P_T)_1\). The other bundle orifice sizes would then be determined relative to an assumed orifice size in bundle number one. This procedure may need to be repeated more than once in order to obtain reasonable and feasible orifice diameters for all four bundles.

**Nomenclature**

- \(g\) Gravitational constant, ft-lb/wt-lb・sec\(^2\)
- \(P\) Pressure, psf
- \(P_a\) Steam pressure at Station a, psfa
- \(P_A\) Steam pressure at Station A, psfa
- \(P_l\) Pressure of moderator steam inside reactor, psfa
- \(P_{core}\) Steam pressure in bottom of core, psfa
- \(P_G\) Steam pressure at Station G, psfa
- \(P_F\) Steam pressure at Station F, psfa
- \(P_M\) Steam pressure at Station M, psfa
- \(P_N\) Steam pressure at Station N, psfa
- \(\Delta P_c\) Pressure drop across core, psf
- \(\Delta P_R\) Pressure drop in riser, psf
- \(\Delta P_D\) Pressure drop in downcomer, psf
- \(\Delta P_o\) Pressure drop across orifice, psf
- \(\Delta P_H\) Pressure drop across header region, psf
$\Delta p_T$  Sum of the downcomer, riser and header pressure drops, psf

$W$  Flow rate, lbm/hr

t  Temperature, $^\circ$F

t_A  Temperature at Station A, $^\circ$F

$\rho_U$  Density, lbm/ft$^3$

$D_O$  Orifice diameter, ft

$D_G$  Pipe diameter at Station G, ft

14.4 Data Retrieval System

An effort has been initiated to automate all of the non-operating data pertinent to the fuel performance such that it can be combined with the operating data from the PATSE data reduction code and other pertinent operating information such as transients, power cycles, startups, shutdowns, coolant chemistry data, etc. All of this data will be arranged in matrix form in a manner which will allow automatic retrieval and plotting of any combination of the available information. A preliminary list of possible data classifications which will be incorporated into this data retrieval system are:

(1) Non-operational data (single-tabulation items)

(a) Fuel design parameters and information

(b) Fuel and cladding material parameters and information

(c) Fuel fabrication data and information

(d) Fuel quality control data and information

(e) Pre- and Post-irradiation data and information

(f) Fuel handling history
(2) Operation data (multiple-tabulation items)
   (a) PATSE data-reduction code data
   (b) Reactor data - operation history, power transients, scrams, rod run-ins, power level, burnup data, etc.
   (c) Radiation - chemistry data on coolant chemistry, system contamination, impurity levels, plant coolant chemistry control data, etc.

Detailed lists of parameters under each of the above categories are being prepared at the present time. These lists will be organized into tabulations which have machine and program compatibility and which can be key-punched directly. These lists will then be printed and issued to the respective performing components for execution.

14.5 Data Reduction - Advanced Fuel Irradiation

An effort has been initiated leading up to the development of a machine code for the analysis of the rod cluster type advanced fuel. It is important that this effort begin early to gain lead time for development effort in deficient analyses areas. There are three major areas of concern in the reduction of rod cluster fuel data. These are:

(a) The turbulent mixing between flow regions around the rods.
(b) The local heat transfer coefficients around the circumference of the rods.
(c) The radiant heat interchange between rods and between rods and process tubes.
The literature is being searched for useful test data and analysis correlations in these areas. If a satisfactory analysis technique cannot be obtained from the literature, it will be necessary to develop an experimental program specifically aimed at the above evaluations.

The present technique, as revealed by the literature, appears to consist of calculating average heat transfer coefficients and using this with no turbulent mixing to determine a nominal or average circumferential clad temperature at each axial position. A hot spot factor is then applied to account for the factors given above. This approach is adequate for design studies but is open to serious questions when applied to a fuel evaluation program where minor variations about the nominal 1250°F maximum cladding temperature can have a profound effect on the measured performance of the fuel.

These hot spot factors are determined experimentally in most cases for the fuel geometry of interest to the program in question. The geometry of the EVESR rod cluster fuel appears to vary considerably from the available data which is very sensitive to the geometry of the coolant spaces.

**Using Instrumented Bundles to Determine Flow Split Between Fuel Elements**

An area of uncertainty in the reduction of EVESR data is the actual flow split between the nine elements in a fuel bundle. It was this concern which led to the introduction of the specially instrumented bundle with its three Potter flow meters in the three typical fuel element locations in a bundle.
In the PATSE code for the solution of nine parallel elements, the pressure drop is balanced analytically across all nine elements to give a theoretical flow split. The specially instrumented bundle, if it is successful, will provide an important check on these calculations. However, there is a second experimental approach for determining the flow split between fuel elements which is available in the quarter core instrumentation which will be incorporated directly in the data reduction program. The analytical development of this procedure is given below:

Analytical Development

In the data reduction program the bundle power will be determined from an equation of the following form:

\[
BP = \frac{W (h_{out} - h_{in}) + N \nu_0 \Theta_M}{1 - HM} \tag{1}
\]

Where:

- \( BP \) = Bundle power - BTU/HR
- \( W \) = Total bundle flow - LBS?HR
- \( h_{out} \) = Bundle exit steam enthalpy - BTU/#
- \( h_{in} \) = Bundle inlet steam enthalpy - BTU/#
- \( N \) = Number of elements per bundle.
- \( L \) = Active fuel element length - FT
- \( \nu_0 \) = Overall heat transfer coefficient across process tube per foot of length - BTU/HR-FT-\(^0\)P
- \( HM \) = Fraction of total fission heat released as neutron and gamma heating directly in moderator

\[
\Theta_M = \frac{(T_{in} - t_{a2}) + (T_{MFTA} - t_{a3})}{2}
\]
\[ T_{in} = \text{Bulk inlet steam temperature for bundle} \quad 0^\circ F \]
\[ T_{a2} = \text{M\&erator temperature at top of the fuel} \quad 0^\circ F \]
\[ T_{MPTA} = \frac{1}{9} \sum T_{MPT}, \text{average of nine midpass superheat steam temperatures} \quad 0^\circ F \]
\[ t_{a3} = \text{Moderator temperature at bottom of fuel} \quad 0^\circ F \]

A similar equation to Equation (1) can be written for the individual fuel element heat powers. An example for element No. 1 is given below:

\[ EP_1 = \frac{W_1 (h_1 - h_{in}) + L U_0 \Theta_{M1}}{1 - H}\ ] \]

(2)

Where:

\[ W_1 = \text{Flow rate through element \#1} \quad \text{LBS/HR} \]
\[ EP_1 = \text{Element power is element \#1} \quad \text{BTU/HR} \]
\[ h_1 = \text{Outlet enthalpy from element \#1} \quad \text{BTU/#} \]
\[ \Theta_{M1} = \frac{(T_{in1} - T_{a2}) + (T_{MPT 1} - t_{a3})}{2} \]

But: \[ EP_1 = (\phi_1)(BP) \]

(3)

Where:

\[ \phi_1 = \text{Fraction of total bundle fission power released from element \#1 as determined analytically from nuclear considerations.} \]

Combining equations (1), (2) and (3) and solving for \( W_1 \) gives

\[ W_1 = \phi_1 \sqrt{\frac{h_{out} - h_{in}}{h_1 - h_{in}}} + \frac{LU_0 (\phi_1 N \Theta_{M} - \Theta_{M1})}{(h_1 - h_{in})} \]

(4)

Sufficient instrumentation is available on the eight instrumented bundles to evaluate everything on the right hand side of equation (4). It should be obvious that eight more equations similar to (4) could be written and evaluated for the remaining eight flow rates in the bundle.
In the term $U_0 \left( \phi_1 \Theta_M - \Theta_{M1} \right) / (h_1 - h_{in})$ in equation (4) notice that when the value of $\phi_1 = \frac{1}{9}$ (case of uniform heat distribution between all nine elements) that:

$$(\phi_1 \Theta_M - \Theta_{M1}) = (\Theta_M - \Theta_{M1}) = 0$$

and this term drops out of equation (4) leaving as the major term:

$$W_1 = \phi_1 W \left[ \frac{h_{out} - h_{in}}{h_1 - h_{in}} \right]$$

From Equation (5) it can be seen that the accuracy of this flow measurement method hinges directly on the accuracy of evaluation of $\phi_1$ and the measured enthalpies with only a minor dependence on the thermal resistance term $U_0$.

15.0 SUB-TASK D-5 FUEL ACTIVITY RELEASE AND COOLANT CHEMISTRY

15.1 Status Report

The proposed sampling and test equipment and the proposed types of coolant chemistry evaluations are discussed in Section 10.0. Actual evaluations under this task will not be initiated until after reactor operation.

16.0 SUB-TASK E-1 POST-IRRADIATION EXAMINATION PLANNING AND EVALUATION

16.1 Pre-Irradiation Measurement Program for Mark II Fuel

Dial gages will be used in the measurement fixture. Drawings are being prepared, the dial gages have been received, and special parts for shop work are on order. It is planned to take the measurements immediately after the pressurization step in the fabrication process. Diameter and profile will be measured at five-inch intervals at two
angular orientations, 90 degrees apart. One length measurement will be taken. Inside diameter will also be measured, but will be done in a separate step as part of the quality control.

The design features and operation of the fixture are as follows: The fuel elements will be positioned in V blocks on a base piece. An index point will be used to position the fuel at the desired angular orientation. A shoulder on the upper end plug will be seated against a reference surface by means of a lever. This permits a dial gage to indicate a reproducible length measurement.

A set of four pairs of horizontally opposed dial gages will measure diameter and profile simultaneously. The four pairs of gages are mounted at 15 inch intervals on a carriage, which in turn is mounted on rollers attached to the base piece. A straight reference surface on the side of the carriage assures proper alignment. It is planned to position the carriage at three axial positions (five-inch spacing) to obtain the diameter and profile readings at five-inch intervals along the five foot active fuel length of the element. However, the carriage motion permits readings to be taken at as close a spacing as required if it becomes desirable to do so. An index on the carriage indicates its axial position so that position can be duplicated and readings recorded properly.

Other features include: 1) spring loading of all moving parts to minimize effect of end and side play; 2) one inch of vertical travel in the diameter gage frames to accommodate up to 1/2 inch of element bow; 3) three levels to retract all dial gage spindles to prevent
scratching fuel cladding during positioning and to protect spindles during fuel moves.

17.0 SUB-TASK E-2 PRE- AND POST-IRRADIATION EXAMINATION

17.1 Status Report

Except for the pre-irradiation examination discussed under Task E-1, no effort in this area will occur until after the first EVESR fuel failure occurs.

18.0 SUB-TASK F-1 ADVANCED FUEL DESIGN

18.1 EVESR Advanced Fuel Design

Some of the advanced fuel assemblies for EVESR should be tested with maximum clad surface heat fluxes ranging from 300,000 to 500,000 Btu/hr-ft². A preliminary analysis indicates that heat fluxes in excess of 200,000 Btu/hr-ft², for advanced fuel assemblies in the EVESR Mark II (annular fuel rods) core at a power level of 12.5 MW thermal, will be difficult, if not impossible, to attain. A scoping study was made to determine the theoretical limits, without regard to hazards or thermal stress (due to gradients) limitations, of advanced fuel maximum heat fluxes in the EVESR Mark II core. The assumption was made that advanced fuel bundles are located only in zone 1 (central four bundles). Table 18.1 summarizes the results of this scoping study.

The next step is a determination of feasibility, based on hazards and thermal stress limitations.
Table 18.1

Maximum Heat Flux and Power Density in Mark II Core

<table>
<thead>
<tr>
<th>Case</th>
<th>Maximum Clad Surface Heat Flux</th>
<th>Maximum Specific Power Density</th>
</tr>
</thead>
<tbody>
<tr>
<td>(1)</td>
<td>Mark II core with 4 central rods in. Mark II fuel in zone #1. Core power: 12.5 MW thermal.</td>
<td>110,000 Btu/hr-ft(^2)</td>
</tr>
<tr>
<td>(2)</td>
<td>As above (1) except 4 central bundles replaced by advanced fuel bundles having water-to-fuel ratio of 3.2 and an enrichment of 10%.</td>
<td>106,000 Btu/hr-ft(^2)</td>
</tr>
<tr>
<td>(3)</td>
<td>As above (2) except having an enrichment of 20%.</td>
<td>148,000 Btu/hr-ft(^2)</td>
</tr>
<tr>
<td>(4)</td>
<td>As above (2) except water-to-fuel ratio (\approx 7.0).</td>
<td>178,000 Btu/hr-ft(^2)</td>
</tr>
<tr>
<td>(5)</td>
<td>As above (4) except having an enrichment of 20%.</td>
<td>237,000 Btu/hr-ft(^2)</td>
</tr>
<tr>
<td>(6)</td>
<td>As above (4) except 4 control rods withdrawn, 8 outer rods inserted (license amendment required.)</td>
<td>424,000 Btu/hr-ft(^2)</td>
</tr>
</tbody>
</table>