CEND 513

MASTER

ADVANCED BONUS CORE DEVELOPMENT PROGRAM

Quarterly Progress Report No. 4

August - November

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December 1964

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Quarterly Progress Report No. 4

August - November 1964

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I. SUMMARY

This report describes the technical progress achieved on the advanced BONUS core development program during the period from August 1, 1964 to November 30, 1964, inclusive.

As a result of the meeting held in Dunedin, Florida on 7/30-31/64 among representatives of the AEC and GNEC, further design effort on the annular multitube fuel element was suspended in favor of a focused design effort on the small solid multitube fuel element. Since that time, design and analysis has proceeded to define the specific upper and lower steam header arrangements for the solid multitube element, the steam and boiler cladding material and dimensions were selected, and the fuel material and manufacturing process was established.

In the header arrangement studies, as a result of the previous cost analysis performed on the U-tube small solid fuel assembly, it was clear that the U-tube design must be considerably simplified or abandoned in favor of a more economical design. Both approaches were pursued and a simplified U-tube design was evolved. However, the 2-pass element design which essentially eliminates the high cost features of the lower end of the U-tube arrangement, was developed and a satisfactory 2-pass upper header evolved. Based upon a review of the studies conducted, the 2-pass small solid element was selected.

The solid multitube assembly consists of four elements with four individual floating tube headers at the lower end and a single header system connecting the four elements at the upper end. There are two passes of steam flow in each element with four first-pass tubes conducting

flow downward and three second-pass tubes conducting flow upward. The existing superheated steam is collected at the upper end with an internal sleeved header and conducted upward in the interior passage of a steam riser. The external passage (annular space) is used as the downcomer for conducting the incoming saturated steam to the assembly. A fission gas space is provided at the upper end of each element for the accumulation of fission gas. In order to maintain a constant steam flow area at the exit from the second pass, the three tubes conducting superheated steam are enlarged at the exit end by lap welding to an enlarged diameter tube. The enlarged diameter so formed accommodates the thermal liner sleeves which insulate the upper tube sheet from the 915°F superheated steam exiting from the second pass. The thermal liner sleeves are roller-expanded at their lower ends into the enlarged steam tube to maintain a first to second pass steam seal.

The reference design is based on the use of Inconel 625 cladding for both boiler and superheated steam surfaces and the use of pellet fuel. No internal pressure is required in the fuel element. The use of Incoloy 800 as a cladding material was discarded in favor of Inconel 625 because of the low creep strength of Incoloy 800 and the excessive thickness of this material required in the ABC fuel element. The excellent creep strength of Inconel 625, in combination with its demonstrated fine welding properties, make it very well suited both as a superheater and boiler cladding material in a multitube fuel element.

The increased cost per foot of Inconel 625 tubing over more conventional nickel alloys is believed to be more than offset by the saving in

fuel pellet cost that will be realized from using the excellent strength properties of Inconel 625 to relax rigid pellet dimensional and chipping requirements. The higher poison cross section of Inconel 625 over Incoloy 800 and stainless steel can be accepted in the ABC multitube fuel element because of the inherently low cladding-to- UO_2 ratio of this style of superheating fuel element. The clad-to- UO_2 ratio is about one-third as large as the same ratio for the first BONUS superheater fuel element.

Recent results of 740°C and 280°D irradiation tests at Hanford demonstrate adequate properties of Inconel 625 after exposure to a fast flux of about 10^{20} nvt. These results, obtained by Mr. T. Claudson of HAP0 were described to CEND representatives in a meeting in Dunedin on 11/9/64. Further experiments on Inconel 625 material are planned at HAP0 and are to be described in detail in subsequent Hanford progress reports.

Preliminary welding procedures for making the steam tube to tube-sheet joints, the lap weld, and the boiler clad to tube-sheet joint are well established. Fatigue and tensile tests of the welded mock-ups at room temperature, 600°F , 750°F , and 925°F demonstrate excellent strength for each of the weld joints at its design operating temperature. Work is continuing at Southwest Research Institute on the weld development of the Inconel 625 joints to determine the effect of varying welding parameters and repair welding on the strength of the joints.

A meeting was held in Windsor, Connecticut on 11/12-13/64 among AEC and CEND representatives to select the cladding material and the type of fuel to be used for the first irradiation experiment in VESR and to review

the over-all budgetary aspects of the Advanced BONUS Core program. As a result of this meeting the reference design fuel element (as described in this report) was selected for testing in VESR. Three fuel assemblies of this type are to be fabricated for the irradiation test. CEND presented two alternate programs for discussion at this meeting. In the first program, work was proposed to continue in parallel on two design concepts, (a) using Zircaloy-2 boiler cladding and vibratory compacted fuel and (b) using Inconel 625 cladding and pellet fuel. These two efforts would be pursued in all design, manufacturing and development aspects completely in parallel and would result in a substantial increase in the originally estimated cost to the ABC program for this fiscal year. In the second approach, CEND proposed to focus full attention on the Inconel 625 boiler clad element with pellet fuel and defer all work associated with vibratory compacted fuel and Zircaloy-2 cladding at least until next fiscal year. It was expected that the latter program could be pursued at the originally estimated level of expenditure for this fiscal year. The conclusion of the meeting was to pursue the more modest program, (1) in view of budgetary limitations during the fiscal year, (2) the objectives of the ABC development program can be satisfactorily accomplished within the scope of the modified program.

An arrangement for locating and testing the ABC fuel assembly in VESR was developed and sent to General Electric Company, San Jose, California. The two positions available in VESR are in zone 3 of that reactor. Nuclear studies indicate that locating the ABC fuel assembly in a skewed position in the cell improves the specific power obtainable and flattens

the power distribution in the ABC assembly, both of which are desirable. The effect of enrichment on the local power distribution within the element was evaluated. It was shown that above 15% enrichment, further increasing enrichment resulted in increasing power almost solely to the boiling side of the fuel element due to the self shielding effect. A preliminary estimate was made of the enrichment required in the boiler-superheater assemblies to achieve an average depletion of 10,000 Mwd/T. It was concluded that an initial enrichment of about 4% is required. Reactivity and power split calculations with various rod positions were performed on the ABC core depleted to 7500 Mwd/T. From these calculations it appears that an adequate range of power split exists over the current design lifetime of the reactor.

Although full attention has been directed to the design and development of the Inconel 625 clad fuel element for the thermal cycling test and the VESR in-pile irradiation test, the orderly termination of existing programs related to Zircaloy-2 clad multitube fuel elements is necessary. It is important to note that the design arrangement of the reference fuel assembly could permit substitution of Zircaloy-2 boiler cladding at a very late date in the design (or even manufacturing) schedule, without causing any change in lattice spacing, pellet size, steam tube or header arrangement. Thus, the development work on manufacturing of Zircaloy-Inconel joints, the Zircaloy-2 strength collapse tests, the irradiation tests on Zr-Inc. joints and the procurement of Zircaloy-2 tubing are of continued interest to the longer range considerations in the Advanced BONUS core program.

Satisfactory Zircaloy-Inconel joints have been produced by the explosively formed process at Battelle Memorial Institute and by the tandem extruded process at Nuclear Metals, Inc. and Metalonics. The BMI joints were 3-1/2 in. in diameter and representative tensile test samples have been machined from these joints and placed in the Hanford Loop at ETR for irradiation exposure. The samples will be removed in December and tested for strength and ductility. Samples of the tandem extruded joints have also been placed in ETR and will be tested at the same time. One explosively formed joint was exposed to 1500 thermal cycles between 150°F and 750°F without any apparent damage when metallographically inspected. A sample of one of the small (1-1/2 in. dia.) Zircaloy-Inconel joints made by Metalonics Corp. was thermal cycled in steam and water from 200°F to 950°F at 500 psi for 8000 cycles with no apparent failure when examined visually and with liquid dye penetrant. The test is being continued to 10,000 cycles and will then be terminated and the part metallographically examined.

The manufacturing investigation of perforated UO₂ pellets has continued at NUMEC. A lot size of 55 pellets was inspected with full dimensional data and found to comply with drawing requirements. The pellets were stacked on reference size tubing and found to be readily assembled. Investigation of the methods for clean-up grinding on the O.D. and faces is to be performed. The pellets are strong and can be handled with minimum care as long as they are not exposed to impact loads. Because a slight change was made in the steam tube size and tube circle diameter in establishing the reference design, two additional 50

pellet lots of natural UO_2 pellets were made at NUMEC. An inspection report on these pellets recently received at CEND indicate good dimensional control. A specification for the pellets to be made for the thermal test elements was prepared.

The manufacturing development work on the vibratory compaction of the annular multitube fuel elements was brought to completion and the results of the studies are reported in detail herein. The investigation has shown that annular multitube fuel elements of the type investigated for the Advanced BONUS Core can be vibratory compacted to a UO_2 fuel density of 85% of theoretical. With further post-compaction slight additional density can be obtained. Adequate control of steam tube position during compaction was obtained by the use of spacers and incremental loading of the fuel.

The corrosion-aging tests on superheater cladding materials were concluded in Dunedin at a total exposure of about 6000 hours. Results of hardness, measurement, weight gain, and tensile tests are plotted versus time of exposure at 1200°F for the period up to 5000 hours. The data for Inconel 625 shows that the ultimate strength increased with time of aging at both 1000°F and 1200°F . The yield strength increased with time at 1000°F but increased more markedly at 1200°F for the first 2000 hours and then remained constant or decreased slightly to 5000 hours. With an exposure to 2000 hours elongations decreased at 1200°F from about 60% to 20% with little change from 2000 to 5000 hours or perhaps a slight increase. This data indicates that while aging effects take place rapidly in Inconel 625, as previously reported, effects on the important properties of yield strength and elongation seem to reach saturation at, or before, 2000 hours with little change thereafter.

Tensile tests of Inconel 625 at room temperature, 600, 800, 1000, and 1200°F were performed using 0.262 in. O.D. and 0.396 in. O.D. tubing. The results of these tests are used directly in the stress analysis of the fuel element.

The work on the leaching tests of vibratory compacted UO_2 was discontinued as a result of the meeting in Windsor, Connecticut on November 12-13, 1964 among CEND and AEC representatives, wherein it was decided to focus attention on pellet fuel for the first irradiation test.

The out-of-pile thermal cycling test loop was under construction during this report period. The loop is about 90% complete and check-out using a dummy test specimen will be performed next month.

The work on the embedded tube tests was brought to completion during this report period. Although additional tests were performed with a controlled atmosphere within the test fuel elements no significant differences in the conclusions given in GNEC 361 were obtained. Embedded loads of vibratory compacted fuel on thermally expanded steam and boiler clad tubes are significant and must be accounted for in the design of these fuel elements.

II- MECHANICAL DESIGN

A. General

Subsequent to the meeting in Dunedin on 7/30/64 among AEC and GNEC representatives wherein the design was directed toward the further development of the solid multitube fuel element and the discarding of the annular multitube concept, work has since proceeded to establish the detailed cross-sectional dimensions, cladding thickness, pellet size, lattice spacing, etc. of the solid multitube fuel element. In addition, two header arrangement studies were concluded, the first having to do with improving the U-tube lower header arrangement previously described in GNEC 361, dated August 15, 1964, and the second, a new arrangement, intended to eliminate the U-tube altogether. The latter arrangement employs two steam passes in a single solid multitube fuel element and is thereby able to considerably simplify the design of the lower end of the fuel element. At the conclusion of the arrangement studies, the 2-pass solid multitube fuel element arrangement was selected as the reference design.

Although a detailed description of the studies leading to the selection of the reference design is given in subsequent paragraphs of this section, a general description of the current reference design is given as follows. The overall reference core mechanical arrangement is shown in the core cross-section drawing, Fig. II-1. The core consists of 36 boiler fuel assemblies of the type currently used in BONUS and 52 boiler-superheater fuel assemblies of the type shown in Fig. II-2. The control rods and rod locations are identical to the first BONUS core. Shim arrangements, although shown in Fig. II-1 to be identical to the first core, are currently under study by the nuclear analysis group and are subject to change as the studies progress. It appears that shims will be used, however. The core structure from the first BONUS core will be used to house the Advanced BONUS core.

The solid multitube fuel assembly shown in Fig. II-2 consists of four multitube fuel elements located in a square array with a 1.875 in. lattice spacing. Each fuel element contains seven 0.281 in. OD x 0.025 in. wall thickness Inconel 625 steam tubes located in a triangular array with a pitch spacing of 0.455 in. Four of the seven steam tubes are first pass and the remaining three are second pass tubes. On the boiler side, the cladding is Inconel 625 having an inside diameter of 1.445 in. and a wall thickness of 0.033 in. The fuel element does not require internal pressurization for structural adequacy. The fuel consists of UO_2 pellets having a minimum density of 94% of theoretical (theo. density being equal to 10.97gr/cc). The dimensions of the fuel pellets are given in Fig. II-3.

The consideration of vibratory compacted fuel for the irradiation test in EVESR was deferred at this time due to fiscal reasons and current cost estimates which indicate that pellets can be economically produced for the 7-tube small solid fuel element.

The active fuel length is 54.6 inches, the same as the first BONUS core. The upper header arrangement shown in Fig. II-2 was adapted from earlier studies on a similar arrangement for the annular multitube fuel element. Concentric inlet and outlet standpipes are used with the centermost pipe handling the superheated outlet steam and the outer annular space handling the inlet saturated steam. Thus, only one thermal liner is required. This liner is located between the superheated and saturated steam spaces. All structural material in the fuel assembly, other than the boiler and superheater cladding and the tube sheets, is composed of Inconel 600 in order to minimize the possibility of chloride stress corrosion in the fuel assembly.

B. Solid Multitube Fuel Element Cladding Design

The objectives of the cladding design are to obtain free standing clad at operating pressure, to have the combined bending and pressure stresses no greater than $2/3$ of the yield strength at the appropriate temperature, and in the case of the nickel alloys, to have the thinnest clad possible in order to keep the poisoning effect to a minimum, and

to have the cross-sectional areas of the boiler clad and superheat clad such that the axial strains are approximately evenly divided between the two.

The wall thickness in items (1), (2), and (3) of Tables II-1 and II-2 was developed by establishing the minimum wall thickness so that the critical pressure for elastic collapse into two lobes is just equal to the maximum pressure which the tube will see in service. The stresses were then calculated for each case. The bending stress was calculated by assuming that the clad deforms in a two-lobe manner until intimate contact was made with the fuel. The total stress is the sum of bending and pressure stress and this combined stress was compared with the allowable stress of $2/3$ the yield strength of the material at temperature. The stresses in the Incoloy 800 material are not acceptable when compared with the yield strength of this material. The stresses in the Inconel 625 are substantially lower than the allowable value which indicates that full advantage of its high strength are not realized. Item (4), Table II-1, is a wall thickness which was determined by allowing the cladding to collapse into two lobes but is thick enough to prevent collapse into more than two lobes which might produce wrinkling. This is believed to be an acceptable condition for Inconel 625 because of its high creep strength. In this case the combined bending and pressure stress is very near the allowable value, the poisoning effect is greatly reduced as evidenced by the k_{oo} values, and the ratio of boiler clad cross section to steam tube cross section (AB/AS) is very nearly one (1).

Since it is not possible to obtain an Incoloy 800 collapsed clad design without exceeding the allowable stress values, it appears that the only design in which this material may be used is a completely free-standing clad. Table II-3 gives the data for free-standing cladding for all three materials. The 0.075 in. wall required for the Incoloy 800 is not attractive for several obvious reasons. The AB/AS ratio is much too high, the poisoning effect is great enough that the k_{oo} is less than the present BONUS superheater, the thick cladding would have very high temperatures on the inside surface, and local

Table II-1. CLAD DESIGN FOR UNPRESSURIZED ELEMENT WITH PELLET FUEL

<u>Material</u>	<u>k_{oo}</u>	<u>AB/AS</u>	<u>Minimum wall thickness</u>	<u>Cold Condition</u>				<u>Hot Condition</u>		<u>Thermal stress</u>
				<u>Allowable stress*</u>	<u>Hoop stress</u>	<u>Bending stress</u>	<u>Combined stress</u>	<u>Allowable stress*</u>	<u>Hoop** stress</u>	
(1) Incoloy 800	1.2089	1.40	0.042	18,750	22,900	13,600	36,500	15,000	18,300	16,400
(2) Inconel 625	1.1778	1.36	0.041	46,000	23,500	13,800	37,300	40,000	18,700	18,200
(3) Zircaloy-2	1.3466	1.74	0.052	30,000	18,500	9,000	27,500	14,000	14,900	4,200
(4) Inconel 625	1.2167	0.99	0.030	46,000	31,500	11,600	43,100	40,000	25,000	13,300

k_{oo} for present BONUS superheater = 1.1425

* 2/3 yield strength at 0.2% offset at the temperature considered.

** These stresses assume a gap exists between fuel and clad in the hot condition as might occur in a low heat flux region. The hoop stress and the thermal stress were not combined because they do not occur at the same point.

Table II-2 CLAD DESIGN FOR ELEMENT PRESSURIZED TO 500 psi USING PELLET FUEL

<u>Material</u>	<u>k_{oo}</u>	<u>AB/AS</u>	<u>Minimum wall thickness</u>	<u>Cold Condition</u>				<u>Hot Condition</u>		<u>Thermal stress</u>
				<u>Allowable stress*</u>	<u>Hoop stress</u>	<u>Bending stress</u>	<u>Combined stress</u>	<u>Allowable stress*</u>	<u>Hoop** stress</u>	
(1) Incoloy 800	1.2309	1.16	0.035	18,750	16,500	11,000	27,500	(	negligible	)
(2) Inconel 625	1.2029	1.12	0.034	46,000	17,000	11,200	28,200	(	negligible	)
(3) Zircaloy-2	1,3480	1.47	0.044	30,000	13,100	7,400	20,500	(	negligible	)

k_{oo} for present BONUS superheater = 1.1425

* 2/3 yield strength at 0.2% offset at the temperature considered.

Table II-3 FREE-STANDING CLAD DESIGN PRESSURIZED TO 500 psi VIPAC FUEL

<u>Material</u>	<u>AB/AS</u>	<u>Minimum wall thickness</u>	<u>k ω</u>	<u>Room Temperature Condition</u>				<u>Assumed ovality</u>
				<u>Allowable stress*</u>	<u>Hoop stress</u>	<u>Bending stress</u>	<u>Combined stress</u>	
(1) Incoloy 800	2.56	0.075	1.1066	18,750	8,000	9,000	17,000	0.008 in.
(2) Inconel 625	1.50	0.045	1.1619	46,000	12,800	31,400	44,200	0.008 in.
(3) Zircaloy-2	1.87	0.054	1.3455	30,000	10,800	20,600	31,400	0.008 in.

* 2/3 yield strength at 0.2% offset.

creep would probably occur. Based on these facts, it seems that the choice of nickel alloys for the boiler cladding must be the Inconel 625. If the Zr-2 cladding is used, it may be necessary to use the thicker cladding (0.060 in. nominal wall). This would permit a later choice as to whether or not the element is to be pressurized, without having two different wall thicknesses to consider. It was decided that the Zr-2 could not be used in the collapsed condition because of its very low creep strength.

Table II-4 is a summary of the cladding thicknesses required for each of the possible conditions of fuel element operation. The number in parenthesis following the clad thickness refers to the condition by which the thickness was set. The notes at the bottom of this table summarize the various conditions which must be met by the cladding.

It should be noted that the value of the bending stress depends on the gap which actually exists between fuel and cladding. A calculated value of 0.0045 in. radial gap was used for the purpose of these calculations.

C. U-Tube Fuel Element Design

The previous U-tube design described in GNEC 345 employs a slider bar to keep the bending stresses in the U-tube to an acceptable value. This slider bar presents a difficult and expansive fabrication problem which it seemed desirable to eliminate if possible. Toward this end, several alternative methods of limiting the U-tube stresses were examined. The method which was finally decided upon employs a supporting tube around the outside of the U-tube. This tube is placed over the steam tube with a close fit and the two tubes are bent into the U-shape together. This adds stiffness to the U and thus has a greater force-to-deflection ratio which causes a greater percentage of the differential expansion between the first- and second-pass sides to be absorbed along the straight length of the tubing as direct tension and compression strains.

A calculation of the maximum stress in the U-tube using a 0.020 in. wall support tube gave a value of 26,000 psi and appears to be quite

Table II-4 MINIMUM REQUIRED THICKNESS OF BOILER CLADDING
IN SMALL SOLID FUEL ELEMENTS

	Vipac Fuel		Pellet Fuel	
	<u>Pressurized</u>	<u>Unpressurized</u>	<u>Pressurized</u>	<u>Unpressurized</u>
Inconel 625	0.045 ⁽¹⁾	0.055 ⁽¹⁾	0.025 ⁽²⁾	0.030 ⁽²⁾
Incoloy 800	0.075 ⁽¹⁾	0.085 ⁽¹⁾	0.075 ⁽¹⁾	0.085 ⁽¹⁾
Zircaloy-2	0.054 ^{(1)*}	0.070 ⁽¹⁾	0.044 ^{(2)*}	0.052 ^{(2)*}

Rules which govern thickness:

1. Bending plus hoop stress less than $\frac{2}{3} S_{y.s.}$ at temp.
2. Elastic collapse pressure equal to maximum differential pressure across cladding:
 - a. Two lobe failure for high creep materials.
 - b. Three lobe failure for low creep materials.

* Modulus of elasticity x area ratio is not balanced between steam clad and boiler clad. Should reduce thickness of steam tubes, where possible, to get more favorable area ratio for axial strains.

acceptable. The use of a U-tube design, however, was discarded in favor of the 2-pass element described in section II-A.

D. Dimensional Tolerances of Fuel Element Cross Section and Fuel Pellets

Fig. II-3 shows a cross-section of the small solid multitube fuel element in the active fuel region for pellet type fuel.

The tubing ID dimensions and tolerance of 0.005 in. were agreed upon by the tubing manufacturer as an obtainable value. Although the normal ovality allowance for such tubing is governed by the diameter tolerance, it was decided that a 0.010 in. maximum ovality would have to be used because of the thin wall and large diameter of this tubing. The tolerance on the OD of the fuel pellet was obtained from the pellet manufacturer. The maximum clearance between fuel and boiler cladding (0.009 in.) was used because this is the value which has been calculated will be present in the element after the first thermal cycle even if the initial clearance were zero.

The tolerances on pellet hole location and diameter tolerance were determined from the experience gained from the initial batch of fuel pellets which were obtained and checked recently. Although it is possible to describe a worse case condition which would not permit fuel elements to be stacked on the steam tubes using the tolerances of Fig. II-3, based on the inspections made of the tubing and pellets already received, these tolerances are believed to be acceptable. (See section VI-B of this report.)

E. Axial Strains in Cladding of 2-Pass Element

Two possible steam tube arrangements were considered for the ABC solid multitube fuel element. One uses only one passage of the steam through the fuel per element and has both ends of the steam tubes locked to the boiler clad. The second arrangement uses two passes of the steam in each element. The strains which might be expected due to differential expansion between steam tubes and boiler clad in the one case and between first- and second-pass steam tubes in the other case have been calculated for an all Incoloy 800 element and an all Inconel 625 element. The results of the calculations are given in Table II-5

Table II-5 AXIAL STRAIN IN SUPERHEATED STEAM TUBES OF THE SOLID MULTITUBE FUEL ELEMENT

Type of element	First-pass steam tube thermal growth(in.)	Second-pass steam tube thermal growth(in.)	Boiler clad thermal growth(in.)	First-pass strain (in./in.)	Second-pass strain (in./in.)	First-pass stress (psi)	Second-pass stress (psi)
<u>Inc.-625</u>							
Single-pass	0.2755	0.3908	0.2220	0.000473	0.00149	12,800	38,700
Two-pass 3-down 4-up	0.2755	0.3908	-	0.00133	0.000774	35,900	20,100
<u>Inc.-800</u>							
Single-pass	0.3266	0.4576	0.264	0.000553	0.00171	14,400	44,500
Two-pass 3-down 4-up	0.3266	0.4576	-	0.00154	0.00087	40,000	22,600

F. Fission Gas Pressure

Fission gas pressure buildup in the ABC fuel element was computed for both pellet and vipac fuel. The calculations assumed a total burnup of 15,000 Mwd/ton. The gas release for pellet fuel was estimated by the method developed by W.J. Levedahl at GNEC in December 1961. The calculation yielded a gas release rate of 5%. This release rate would produce a pressure of 110 psi at the end of the core life if a one-inch void length exists above the fuel region or 84 psi for a two-inch void length.

The calculation of gas release from the vipac fuel is more uncertain due to the limited amount of information available on this type of fuel. The estimate of gas release was obtained by using data from the Savannah River Project (DP-875), which presents a curve of gas release as a function of thermal rating. The gas release for the ABC element was obtained by first computing the thermal rating of the element by an approximate method and then using the curve mentioned above. The value of gas release obtained was 10%, which would indicate that the pressure buildup will be twice as great for vipac fuel as for pellet fuel with the same void space available.

The higher value for vibratory compacted fuel is in the right direction because of its greater surface-to-volume ratio and its greater ratio of peak-to-average temperature, both of which would be expected to produce a higher gas release rate.

G. EVESR Irradiation Test Assembly

Fig. II-4 shows an arrangement of the small solid fuel assembly in zone 3 of the VESR reactor. General Electric Co. has indicated the need for boiling water rods of low enrichment to supplement the four ABC fuel elements in each cell. These rods increase overall core reactivity and act as water displacers to maintain an adequate water-to- UO_2 ratio. The arrangement shows the ABC assembly skewed in the cell to place the ABC fuel elements as far away from the partially inserted control rods as possible. This raises the average specific power above that obtained in a concentric arrangement. General Electric is proceeding with the

evaluation of this fuel assembly in preparation to establishing the final design and fuel enrichment.

- 52 BOILER - SUPERHEATER
FUEL ASSEMBLIES

- 36 BOILER FUEL ASSEMBLIES

Fig. II-1 Core cross-section using solid multitube fuel elements.

Fig. II-2 Solid multitube fuel assembly.

Fig. II-3 Fuel assembly cross-section for pellet fuel.

[illegible]

1.445 TUBE I.D.
1.430
1.441 PELLET O.D.
1.443

6 HOLES TO BE WITHIN .005 OF TRUE POSITION ON .910 TUBE CIRCLE. HOLE STRAIGHTNESS IS TO BE WITHIN .001 T.I.R.

WALL $\frac{.030}{.036}$ INCONEL 625

$\frac{.281}{.285}$ O.D. TUBE (OVALITY WITHIN DIA'S)
WALL TH'K. .025/.027 - INCONEL 625

CONCENTRIC WITHIN .005

$\frac{.290}{.295}$ HOLE

D TO
ENSITY
)

STRAIGHTNESS OF ALL TUBES SHALL
BE WITHIN .015/FT.

III. THERMAL DESIGN AND ANALYSIS

A. Fuel Element Summary

A summary of the ABC fuel element characteristics is given in Table III-1. All of the values listed are the best estimates available at this time; however, they are subject to revision as the design progresses. There are two reference designs tabulated, the same in all respects except that one has a Zircaloy-2 water cladding and the other has an Inconel 625 cladding. The reference designs employ Inconel 625 steam tubes and UO_2 fuel in pellet form. The use of vibratory compaction is still under consideration as an alternate method of fuel loading. The mechanical design of the 4-element assembly, with two-pass flow in each element, is given in Section II.

The third column in Table III-1 lists the characteristics of the present (core 1) BONUS superheater fuel. It should be pointed out that the present type of BONUS boiler fuel elements will continue to be used in core 2 with the ABC boiling-superheating elements.

B. Thermal Code MUTI-1 Summary

This computer code continues to be a very useful tool in analyzing the thermal performance of the ABC fuel elements. The code was briefly described in GNEC 327, the ABC Quarterly Progress Report No. 1, for November 1963 through January 1964.

A memorandum on the use of the code was recently prepared for the project staff in order to assure its continued usefulness during the coming transition period in transferring the project from the CEND

Table III-1 SUMMARY OF FUEL ELEMENT CHARACTERISTICS

(The values given in this table are the current best estimates and are subject to revision.)

CORE DESCRIPTION	REFERENCE DESIGNS		BONUS core 1 rod type element
	7-tube solid elements Zr-2 cladding	Inconel cladding	
Superheater elements			
Number of 4-in. x 4-in. assemblies	52	52	--
Number of 4-in. x 8-in. assemblies	--	--	32
Boiler elements	core 1 rod type	core 1 rod type	core 1 rod type
Number of 4-in. x 4-in. assemblies	36	36	64
Combined core			
Number of 4-in. x 4-in. assemblies	88	88	128
POWER			
Total power in reactor, Mw(t)	50.0	50.0	50.0
Boiling power, Mw(t)	38.1	38.1	38.1
Superheating power, Mw(t)	11.9	11.9	11.9
Power in boiling assemblies, Mw(t)	27.8	27.8	36.7
Conduction boiling, Mw(t)	27.2	27.2	36.0
Neutron and gamma boiling, Mw(t)	0.6	0.6	0.7
Power in superheating assemblies, Mw(t)	22.2	22.2	13.3
Conduction superheating, Mw(t)	11.9	11.9	11.9
Conduction boiling, Mw(t)	9.7	9.7	0.9
Neutron and gamma boiling, Mw(t)	0.6	0.6	0.5

Table III-1. (continued)

POWER (continued)	REFERENCE DESIGNS		BONUS core 1 rod type element
	7-tube solid elements Zr-2 cladding	Inconel cladding	
Power split			
Superheating fraction of reactor total	0.238	0.238	0.238
Superheating fraction of superheater total	.536	.536	0.90
Boiler power description			
Ave. specific power, kw/kg of UO_2	16.49	16.49	12.7
Ave. volumetric conduction power, Btu/(hr)(cu ft) of UO_2	16.74×10^6	16.74×10^6	12.5×10^6
Superheater power description			
Ave. specific power, kw/kg of UO_2	9.58	9.58	7.2
Ave. volumetric conduction power, Btu/(hr)(cu ft) of UO_2	9.72×10^6	9.72×10^6	6.99×10^6
Peaking factors in boiler region			
Over-all	2.75	2.75	3.46
Axial	1.77	1.77	1.77
Radial between assemblies	1.15	1.15	1.35
Local inside assemblies	1.35	1.35	1.45
Peaking factors in superheater region			
Over-all	3.00	3.00	2.70
Axial	1.70	1.70	1.60
Radial between assemblies	1.30	1.30	(1.69)
Local inside assemblies	1.36	1.36	(1.69)

Table III-1 (continued)

FUEL DESCRIPTION	REFERENCE DESIGNS		BONUS
	7-tube solid elements Zr-2 cladding	Inconel cladding	core 1 rod type element
Weight of fuel (UO ₂ pellets, 95 ± 1% density)			
Boiler elements, lb	3638	3638	6376
Superheater elements, lb	4972	4972	4061
Total, lb	8610	8610	10437
Volume of fuel (UO ₂ pellets)			
Boiler elements, ft ³	5.60	5.60	9.80
Superheater elements, ft ³	7.65	7.65	6.25
Water-to-fuel volume ratio			
Boiler elements	2.4	2.4	2.4
Superheater elements	2.57	2.69	3.85
Parasitic cladding-to-fuel volume ratio	0.133	0.271	0.605
<u>BOILER FUEL ELEMENT</u>			
Fuel elements per assembly	32	32	32
Cladding material	Zr-2	Zr-2	Zr-2
Cladding outside diameter, in.	0.500	0.500	0.500
Cladding thickness, in.	0.025	0.025	0.025
Fuel length, in.	54.0	54.0	54.0
Pellet diameter, in.	0.445	0.445	0.445
<u>SUPERHEATER FUEL ELEMENT</u>			
Fuel elements per assembly	4	4	32
Steam passes per element	2	2	1
Steam passes through assembly	2	2	4

Table III-1 (continued)

SUPERHEATER FUEL ELEMENT (continued)	REFERENCE DESIGNS		BONUS core 1 rod type element
	7-tube solid elements Zr-2 cladding	Inconel cladding	
Steam cladding material	Inconel 625	Inconel 625	Inconel 600
Outside diameter, in.	0.283	0.283	0.540
Thickness, in.	0.026	0.026	0.018
Tube circle diameter, in.	0.910	0.910	---
Water cladding material	Zr-2	Inconel 625	---
Inside diameter, in.	1.450	1.450	---
Thickness, in.	0.061	0.033	---
Full length, in.	54.6	54.6	54.6
Heating surface per assembly			
Steam surface, ft ²	7.706	7.706	20.58
Water surface, ft ²	7.490	7.223	---
Steam free area per assembly			
First pass, ft ²	0.00466	0.00466	0.00927
Second pass, ft ²	0.00349	0.00349	all passes
Steam flow through assembly			
First-pass tubes	16	16	---
Second-pass tubes	12	12	---
BOILER THERMAL PERFORMANCE			
Average heat flux, Btu/(hr)(sq ft)	133,800	133,800	102,700
Maximum heat flux, Btu/(hr)(sq ft)	368,000	368,000	355,000
Maximum rod power, kw/ft	14.4	14.4	13.6
Maximum fuel temperature, °F	< 4,000	< 4,000	< 4,000

Table III-1 (continued)

SUPERHEATER THERMAL PERFORMANCE	REFERENCE DESIGNS		BONUS core 1 rod type element
	7-tube solid elements Zr-2 cladding	Inconel cladding	
Steam surface			
Average heat flux, Btu/(hr)(sq ft)	98,700	98,700	64,800
Maximum heat flux, Btu/(hr)(sq ft)	296,100	296,100	175,000
Nominal maximum clad temperature, °F	995	995	1,010
Water surface			
Average heat flux, Btu/(hr)(sq ft)	82,700	82,700	---
Maximum heat flux, Btu/(hr)(sq ft)	248,000	248,000	---
Maximum fuel temperature, °F	3,000	3,000	2,450
<u>STEAM DATA</u>			
Total steam flow through superheater, lb/hr	152,000	152,000	152,000
Inlet pressure, psig	950	950	950
Inlet temperature, °F	540	540	540
Pressure drop from inlet to exit manifold, psi	84	84	74
Average exit velocity, ft/sec	200	200	121
Exit temperature, °F	900	900	915
Average heat transfer coefficient			
First pass, Btu/(hr)(sq ft)(°F)	730	730	650
Second pass, Btu/(hr)(sq ft)(°F)	910	910	all passes
<u>DESIGN LIMITS</u>			
Boiler fuel elements (core 1 limits)			
Maximum heat flux, Btu/(hr)(sq ft)	450,000	450,000	450,000
Maximum rod power, kw/ft	14.4	14.4	14.4
Maximum fuel temperature, °F	5,000	5,000	5,000
Superheater fuel elements			
Maximum heat flux, water surface, Btu/(hr)(sq ft)	500,000	500,000	---
Maximum steam clad temperature, °F	1,250	1,250	1,250
Maximum fuel temperature, °F	5,000	5,000	5,000

Florida Branch location to CEND. Different computing facilities will be used after the move to CEND.

The MUTT-1 is a two-dimensional thermal conduction code for steady-state conditions through irregular solids with non-uniform internal heat generation. It is a numerical code, essentially a modified PDQ. It allows a completely variable mesh structure and regionwise variation of thermal conductivity and internal heat generation over a maximum of 50 regions. In using an IBM-7094 computer 1300 mesh points are available; with an IBM-704, 2500 mesh points are available.

C. Superheated Steam Pressure Drop

The superheated steam pressure drop, as listed in Table III-1, is defined as the drop in static pressure of the steam, starting from the pressure vessel saturated steam pressure to the superheated steam pressure in the octant manifolds which connect to the steam nozzles. (The design of the plumbing in the BONUS reactor is shown in Figs. 301-1, 301-2, and 301-3 of the Hazards Report, PRWRA-GNEC 5.) The pressure drop can be subdivided into the following major categories:

1. Δp through the mechanical dryer and preheater.
2. Δp through the inlet standpipe, including a spiral shield plug.
3. Δp through the assembly inlet manifold into the steam tubes.
4. Δp through the first pass steam tubes, the bottom turn-around, and the second pass steam tubes.
5. Δp through the assembly outlet manifold.
6. Δp through the outlet standpipe, including a spiral shield plug.

7. Δp through the orifice at the inlet to the octant manifold.

The orificing philosophy is to orifice each assembly in order to obtain the same superheated steam temperature from each assembly; in other words, the orifices are selected to match the steam flow to the radial power distribution of the superheater assemblies. In order to minimize the over-all pressure drop the assembly with the highest steam flow rate (the highest power assembly) has no orifice in the line (item 7 above). The superheater pressure drop is therefore calculated by using the assembly (in each octant of the symmetrical core) which has the highest flow rate. In Table III-1 the radial peaking factor between assemblies is 1.30 for the reference designs. Therefore, the maximum steam flow through an assembly is $(152,000 \times 1.30/52)$ 3,800 lb/hr.

On this basis the Δp through the fuel element steam tubes (item 4 above) is computed to be 60 psi, the other Δp 's lumped together are estimated to be 24 psi, giving a total Δp of 84 psi.

D. Axial Steam and Clad Temperature Patterns

Axial steam and clad temperature patterns have been studied for 3 fuel element arrangements, all based on the use of 7-tube solid elements, 4 elements per assembly.

The results of the first arrangement are given in Fig. III-1. This arrangement is the previous reference design in which steam flows downward through two elements (14 steam tubes), crosses over in a bottom manifold, and then flows upward through the other two elements (14 steam tubes) in the assembly. The maximum clad temperature is 1007°F, and the difference in average clad temperature between pass 1 and pass 2 is 240°F.

Figure III-2 shows the temperature patterns for the current reference design, an arrangement that utilizes two-pass steam flow in each individual element. In this design steam flows down 4 tubes and then up 3 tubes of each element, which on an assembly basis is 16 tubes down and 12 tubes up. The result is a maximum clad temperature (995°F) that is a few degrees lower, and a smaller temperature difference (203°F) between the average clad temperature of pass 1 and pass 2.

Figure III-3 illustrates the results of a two pass element in which steam flows down 3 tubes and then up 4 tubes of each element. The maximum clad temperature (1010°F) is a few degrees higher, and the average clad temperature difference between the two passes is larger (253°F).

The temperature patterns of all three arrangements are shown together in Fig. III-4. The thermal and hydraulic design significance of the study is as follows:

1. The maximum clad temperature is very nearly the same for all three cases.
2. The two-pass element reference design has the smallest temperature difference between the average clad temperature of the first pass and the second pass.
3. The two-pass element with 3 tubes down and 4 tubes up gives the lowest exit steam velocity.

Concerning the calculational work in this study, all of the figures are based on an average power element with an average steam flow; almost identical curves are obtained at higher and lower powers as long as the steam flow is properly matched to the power. In the two-pass designs

the power split between adjacent hot and cold steam tubes is 95% power to the hot tube and 105% power to the cold tube. This power split value was determined with a sample problem computed with the MUTT-1 code.

E. Preliminary Calculations for the EVESR Test

Thermal and hydraulic analyses have been performed in support of the irradiation test program which is in the final planning stage at this time. In particular, attention has been given to the flow performances, both on the boiling water side and the superheated steam side. Also, the power split characteristic has been studied; its performance is slightly different than in the reference design since the fuel enrichment is increased from about 4% to 15%. This increase causes the radial power distribution inside the element to be much steeper, resulting in a higher power fraction in the outer perimeter, which in turn creates a lower power split to superheating (or a higher power split to boiling).

Fig. III-1 Axial steam and clad temperature patterns with steam flow in one direction through an individual element, the previous reference design.

Fig. III-2 Axial steam and clad temperature patterns with steam flow in two directions through an individual element, the current reference design.

Fig. III-3 Axial steam and clad temperature patterns with steam flow in two directions through an individual element, similar to the current reference design.

Fig. III-4 Axial steam and clad temperature patterns of three different steam flow arrangements.

IV. NUCLEAR ANALYSIS

A. Fuel Element In-Pile Test

Review of General Electric's Analysis¹ of the irradiation testing of ABC fuel assemblies in the EVESR reactor has indicated that some compromise between the average specific power attainable, and the power split within the element, must be made. The objective of GE's analysis was to obtain a minimum average specific power of 9.25 kw/kg of UO_2 , without regard to the local power distribution within the ABC element. According to this analysis a fuel enrichment of approximately 25% was required to achieve a specific power of 10.2 kw/kg in the small solid element. The local power distribution at this enrichment indicates that a large portion of the power will be to boiling, giving boiler clad heat fluxes which might be unacceptable. The question of how much is sacrificed in average specific power for lower enrichments, which give reasonable power distributions within the elements, has been investigated. Fig. IV-1 gives the power distribution within the small element (Zircaloy clad - dimensions from ABC No. 193) for three enrichments, namely 5%, 10%, and 15%. The water to UO_2 volume ratio is 1.4. This is the water to UO_2 one would get with the small elements on a 3 x 3 array within the EVESR test channel.

The effect of enrichment on the average specific power within the element has been estimated in one-dimensional geometry. Fig. IV-2 gives the power distribution in the 3 x 3 ABC element array in cylindrical

¹ Memo to J.L. Murray from M.L. Weiss, July 8, 1964, File No. 29F7.

geometry for the three enrichments, at the center of a cylindrical array of EVESR fuel cells. Tabulated on the figure for each enrichment is the fraction of the total power in the system being produced in the ABC elements. Although the actual magnitude of these power fractions are not meaningful to the EVESR test, the percentage difference for the different enrichments should be a good estimate of the percentage difference one should expect in the average specific power in the test. That is, if the average specific power in the actual test is 10 kw/kg for a fuel enrichment of 15%, then the average specific power for an enrichment of 10% would be $8.22/8.95 \times 10$ kw/kg, etc. After reviewing heat transfer considerations, a fuel enrichment will be chosen and a more accurate two-dimensional calculation of the actual test will be made.

Homogeneous slowing down calculations indicate that, for the relatively high enrichments being considered for the irradiation test, a good fraction of the power is being produced by epithermal neutrons. Since the spatial distribution of these neutrons must be important to the total power distribution it was necessary to attempt a calculation which accounted for more than one group of neutrons. Although the results must still be considered approximate, a calculation in sixteen energy groups has been performed. The FAIM² computer code for the IBM-7090 was used for the calculations. Fig. IV-3 gives the local power distribution for enrichments ranging from 3% to 25%. The difference in these calculations and those reported previously is due to the FAIM code taking account

² D.C. Baller, The FAIM Code, A Multigroup, One-Dimensional Diffusion Equation Code, Report AMTD-118, Jan. 1962.

of the energy dependence in the power distribution. The sixteen-group cross-section library of Hansen and Roach was used for the calculations. It should be noted that the distributions given in the figure are normalized to the same total power. Future heat transfer calculations should be based on these distributions

B. Advanced BONUS Core Studies

A detailed PDQ calculation was performed, in order to evaluate the power distribution in the 7-tube solid element, at ABC operating conditions. Previous analyses had assumed the steam holes to be void. This region has now been represented as a homogeneous mixture of steam and cladding.

Even though this was a one-group calculation, it is believed to be adequate for the evaluation and analysis of thermal parameters. Fig. IV-4 shows the radial distribution of power in the fuel element.

Diffusion equation coefficients for the BONUS boiler and for the small ABC Inconel 625 clad element have been calculated as a function of burnup. The burnup calculations for the ABC element included four initial enrichments, namely, 3.0%, 3.5%, and 4.0%. In order for the ABC core containing 36 boiler assemblies and 52 boiler-superheater assemblies to achieve an average burnup of 10,000 Mwd/ton, the initial enrichment in the boiler-superheater must be about 4%. This estimate is based on an excess reactivity of approximately 2.5% Δk in the case with an average burnup of 7,500 Mwd/ton.

Table IV-1 gives reactivity and power fractions for various rod conditions in the case whose fuel (both boiler and superheater) has been

burned to 7,500 Mwd/ton. Since the desired superheating power fraction is 0.238 one can see that the range of power split is adequate.

Table IV-1. REACTIVITY AND POWER SPLIT FOR ABC CORE
AT 7,500 Mwd/ton OVER-ALL

<u>Central rod</u>	<u>Blade rods</u>	<u>k_{eff}</u>	<u>% Boiler power</u>	<u>% BSH power</u>	<u>SH power</u>
Out	Out	1.028158	54.23	45.77	24.72
In	Out	.987107	46.53	53.47	28.87
In	In	.924911	61.31	38.69	20.89
Out	In	.987007	68.61	31.39	16.95

Fig. IV-1 Radial power distribution—small Zr-clad ABC fuel element.

Fig. IV-2 Power distribution in ABC test assembly (9 small elements) at the center of EVESR fuel cells.

Fig. IV-3 Radial power distribution (FAIM) small Zircaloy clad ABC fuel element on 3 x 3 array in the EVESR reactor.

Fig. IV-4 Radial distribution of power in 7-tube solid element at reactor conditions.

V. EXPERIMENTAL PROGRAMS

A. Materials Evaluation and Tests

1. Aging Tests

The aging tests at both 1000°F and 1200°F have attained 5000 hours exposure. The results of tensile tests performed on specimens removed after 2000 hours and 5000 hours are shown on Figs. V-1 through V-5. The data is presented in bar chart form. For each alloy the ultimate, yield, and elongation values are grouped together to permit easy comparison. Results are given as determined on the as-received material, as tested at temperature without aging, and after exposures of 2000 and 5000 hours.

For Inconel 625, Fig. V-1, the ultimate strength increased on aging at both temperatures. The yield strength increased on aging at 1000°F and increased markedly on aging at 1200°F for the first 2000 hours and then held constant, or decreased slightly, from 2000 to 5000 hours. Elongations are shown to decrease at 1000°F from more than 50% to 30% with little change from 2000 to 5000 hours. At 1200°F the decrease in elongation on 2000 hours exposure is even greater, from around 60% to less than 20%. However, no further decrease, and possibly a slight increase, in elongation took place between 2000 and 5000 hours aging at 1200°F.

The above indicates that while aging effects take place rapidly in Inconel 625, as previously reported, effects on the important properties of yield strength and elongation seem to reach saturation at, or before,

2000 hours, with little change thereafter. Over-aging is not as apparent as with the 1200°F elongation of Hastelloy X, Fig. V-4. The precipitation or other mechanisms which produce the changes in properties of Inconel 625 may merely stop in the 2000 hour range without the agglomeration or growth of carbides, or other phenomena, usually associated with over-aging. In any case, the decrease in elongation of Inconel 625 on aging at 1200°F is considerable, but apparently approaches a limit, and this limit, as shown in the figure, while low is still within the range of usable elongations. From the standpoint of the ABC program, the question of the combined effects of aging and irradiation is of primary importance. The only prediction which can be made safely regarding irradiation is that it will not improve the properties of material exposed for long periods at temperature in-pile. However, there remains a margin between 20% elongation and the minimum acceptable elongation which may not be entirely closed by irradiation effects.

With Incoloy 800, Fig. V-2, aging has little effect on the ultimate and yield strengths. Elongations decrease on aging at 1000°F, but at 1200°F the elongation decreased at 2000 hours and then increased. This could be attributed to over-aging at the higher temperature, a phenomena which could be expected to take much longer to become apparent at 1000°F. Considerably longer exposures at 1000°F might well show gains in elongation after a minimum was reached.

Incoloy 800, while a much weaker alloy than Inconel 625, shows no serious effects on aging in the test performed.

Inconel 600, Fig. V-3, changes little in ultimate strength on aging. The yield at 1000°F is also little changed, but some decrease at

1200°F is apparent. Elongations remain fairly constant to 5000 hours at 1000°F. At 1200°F the decrease in elongation on aging is marked (note scatter in the 1200°F-5000 hour results).

Hastelloy X shows fairly constant ultimate strengths at both temperatures on aging. The yield strength is little changed at 1000°F, but increases with exposure at 1200°F. The 1200°F elongation, in contrast, decreases and then increases on exposure, as previously discussed.

Over-aging, as indicated by decreased and then increased elongation, appears to be occurring with Type 406 stainless steel at 1200°F but the scatter in results is too great to justify conclusions. As discussed in previous reports, the strength of Type 406 at design temperatures is too low to make the alloy of interest.

From the above data, Incoloy 800 shows the least serious effects from aging. However, although Hastelloy X, Inconel 625, and Inconel 600 show marked changes in elongation on aging, their minimum values are still within the usable range, depending on design.

a. Hardness Data

The data on hardness variations in the 5000-hour, 1200°F test has been plotted on Figs. V-6 and V-7. It will seem that there has been a slow and limited decrease in hardness for six of the seven alloys in the last 2000 hours. The seventh alloy, Rene 62, shows a continuing slight increase in hardness. Except for IN-102 and the two 400-series stainless steels, this decrease in hardness amounts to a very few points and would not in itself indicate over-aging behavior of any importance. The most interesting hardness variations had been found to occur in the early hours

of exposure. Detailed data on the initial hardness changes up to 200 hours was presented and discussed in the May 1964 Monthly Technical Report of this series (OROO-3154-8, p. V-3).

b. Corrosion Data

Weight gain data for eight alloys in 1200°F, oxygen and hydrogen containing steam, has been brought up to date in Figs. V-8 and V-9. Here again, no dramatic changes are apparent in the last 2000 hours of exposure.

The curve for Inconel 625 continues to be quite flat at a low total weight gain. The curve for Incoloy 800 shows a decreasing weight gain rate in the last thousand hours, but Inconel 600 shows an increased rate. Both the Type 406 and Hastelloy X curves approach a linear weight gain. It must be emphasized that the scale of the ordinate used in plotting is such as to magnify small weight gains. The largest total weight gains shown, approximately 0.040 grams per square decimeter in 5000 hours, is quite small and should be acceptable.

2. Creep Tests

This work has been completed and reported (ABC Quarterly Progress Report No. 3, GNEC 361, p. V-7).

3. Tensile and Compression Tests

The purpose of the compression tests is to determine whether a superheat steam tube in the gas void space above the fuel column can be adequately supported by a surrounding coaxial tube to prevent buckling deformation of the steam tube when it is subjected to a compressive axial load induced by thermal expansion.

A fixture to mock-up the steam tube condition in the fuel element is shown in Fig. V-25. A test specimen 1/4-inch longer than the test fixture and containing two marks suitable for optical observation is inserted into the test fixture. The test fixture is then inserted into a standard tensile (compression) test machine. A compressive load is applied incrementally and the relative movement of the two tube marks measured optically. The experimental effort will attempt to obtain data to determine the load to which pure axial elastic compression takes place, the load at which the coaxial tube begins to provide lateral support, and the point at which permanent deformation of the tube occurs. The tests will be performed at room temperature, 400, 800, and 1200°F. The specimens will be heated either by resistance or by a Calrod inserted into the sample tube. Three samples at each temperature will be tested for one selected material.

Tensile tests were performed to determine the room temperature and elevated temperature tensile properties of Inconel 625 and Incoloy 800 tubing as part of the evaluation program to establish specific properties of superheater fuel cladding materials. Room temperature, 600, 800, 1000, and 1200°F tests have been completed on the 0.396 in. nominal OD by 0.039 in. wall Incoloy 800 and Inconel 625 and on the 0.262 in. nominal OD by 0.024 in. wall Inconel 625 tubing. The test results are shown in Table V-1 and the data are plotted in Figures V-26, V-27 and V-28.

Each of the data values are representative of a single test specimen. The as-received tubing was in the annealed and cold straightened condition. Figures V-27 and V-28 indicate good agreement between the

TABLE V-1Tensile Properties of
Incoloy 800 and Inconel 625 Tubing

<u>Material</u>	<u>Nominal OD Inches</u>	<u>Nominal Wall Inches</u>	<u>Temp. °F</u>	<u>Tensile Strength psi</u>	<u>Yield Strength psi</u>	<u>Reduction in Area %</u>	<u>Elongation in Inches %</u>
Incoloy 800	0.396	0.039	70	82,300	35,900	49.0	49
Incoloy 800	0.396	0.039	600	74,000	27,730	38.5	43
Incoloy 800	0.396	0.039	800	73,320	24,980	36.5	47
Incoloy 800	0.396	0.039	1000	59,000	22,200	44.5	47
Incoloy 800	0.396	0.039	1200	31,690	20,120	61.0	69
Inconel 625	0.396	0.039	70	131,400	66,620	48.0	58
Inconel 625	0.396	0.039	600	117,000	52,740	45.0	56
Inconel 625	0.396	0.039	800	112,800	50,000	40.0	54
Inconel 625	0.396	0.039	1000	104,000	48,300	30.0	36
Inconel 625	0.396	0.039	1200	70,300	45,100	22.5	33
Inconel 625	0.262	0.024	70	137,620	77,080	39.5	50
Inconel 625	0.262	0.024	600	123,300	64,930	30.0	39*
Inconel 625	0.262	0.024	800	120,400	61,680	32.5	39*
Inconel 625	0.262	0.024	1000	109,600	57,760	22.5	28
Inconel 625	0.262	0.024	1200	84,000	54,370	13.0	15*

* Sample failed at the gage marks.

0.396 in. OD and 0.262 in. OD Inconel 625 tubing. The small difference in tensile properties can be attributed to the different heats of material and to the different amounts of cold straightening which were required to obtain the desired dimensions.

The high temperature stress-strain curves for both Inconel 625 and Incoloy 800 exhibited a slight serrated effect at the yield point. This appears to be typical of materials, especially nickel-base alloys, which strain age rapidly at elevated temperatures.

4. Collapse Tests of Boiler Cladding

During this report period, collapse tests were performed on three specimens, of which two were collapsed at high temperature. The tests were conducted in the same general manner as those previously reported in Quarterly Progress Report No. 3, May-July 1964, Advanced BONUS Core Development Program, GNEC 361.

The primary purpose of these tests was to experimentally confirm analytical calculations on the collapse pressure of thin-walled Zircaloy-2 tubing. Initial ovality, end effects, and temperature were considered in the calculations. The results were very encouraging. Percentage errors between the calculated and actual collapse pressure range from 4 to 6%. If one allows for a 5% experimental error, it is seen that collapse pressures can be very accurately predicted.

The first tube tested, specimen number 7, shown in Fig. V-10, was collapsed at room temperature, while the other two tested, specimens number 2 and 8, shown in Fig. V-11 and Fig. V-12, were collapsed at temperatures greater than 500°F as given in Table V-2. A summary of dimensional data and collapse conditions is shown in Table V-2.

Table V-2 DIMENSIONAL DATA AND COLLAPSE CONDITIONS
FOR ZIRCALLOY-2 TUBES

	Specimen number 7 vacuum annealed seamless	Specimen number 2 15% cold-worked seamless	Specimen number 8 vacuum annealed seamless
Length, inches, ℓ	45	22.375	45
Average measured outside diameter, inches, OD	3.526	5.070	3.527
Average measured wall thickness, inches, t	0.1091	0.1404	0.1046
Length-to-radius ratio, $\frac{2\ell}{OD}$	25.52	8.826	25.52
Outside diameter-to-wall thickness ratio, $\frac{OD}{t}$	32.32	36.11	33.72
Average ovality, mils, (OD max - OD min) ave.	4.82	7.833	4.2
Maximum ovality, mils, (OD max - OD min) max.	8	11	6
Time to collapse	≈ 1 hr	≈ 4 hrs	≈ 4 hrs
Temperature at collapse	room	537.9°F	513.8°F
Calculated collapse pressure, psi	970.0	985.1	734.6
Actual collapse pressure, psi	910	945	770
Percentage error	6	4	5

The calculation methods were, in general, those given by Sturm.¹ The reduction of the modulus of elasticity due to temperature was taken from Zima.²

For specimen number 7, calculations predicted a reduction in collapse pressure of 7% due to initial ovality, based on an allowable stress of 60,000 psi. For specimen number 8, the reduction was predicted to be 5.7% based on the same allowable stress.

Analytically, the effect of out-of-roundness (ovality) varies with the ratio of initial deflection to the thickness of the wall, the $\frac{L}{R}$ ratio, the elastic limit of the material, and the bending modulus of failure of the material. An expression of the effect is obtained for an initial out-of-roundness of the same magnitude as the deflected tube at buckling.

5. Irradiation Tests of Cladding Materials

A meeting was held in Dunedin, Florida on November 9, 1964 to review the results of the irradiation data obtained in the Hanford Materials Program on the advanced BONUS core cladding materials. Mr. T. Claudson described the work being done at HAPO under the Alloy Selection portion of the program to determine the effects of irradiation on structural materials. The materials under investigation include:

Incoloy - 800	Hastelloy X-280
Inconel - 600	Hastelloy N
Inconel - 625	Hastelloy C
Inconel - 702	Hastelloy B
Inconel - 718	R-235
Inconel X-750	
S.S. Type 406	

The two materials of most interest to the ABC program are Inconel 625 and Incoloy 800. By reason of the strength requirements of the

¹ University of Illinois, A Study of the Collapsing Pressure of Thin-Walled Cylinders, Engineering Experiment Station Bulletin Series No. 329, pp. 23-34, Nov. 11, 1941.

² G.E. Zima, A Review of the Properties of Zircaloy-2, Hanford Atomic Products Operation, Number HW-60908, Fig. 11, p. 58, Richland, Wash.

multitube fuel element design selected, most interest centers on Inconel 625. Specimens had been irradiated at HAP0 at temperatures of 50, 280, 580, and 740°C. The two lower temperatures were in water and the two higher temperatures were in "contaminated" helium (i.e. helium containing small amounts of CO² and CO).

The specimens, after irradiation, were tested either at room temperature or 650°C. The specimens were flat, reduced section tensile specimens having a gauge length of 2 in. and measuring 0.200 in. x 0.100 in. at the center cross section.

The Incoloy 800 and Inconel 600 samples were tested in one of the following three conditions: (1) solution annealed; (2) 20% cold worked; or (3) mill annealed at 2050°F for 20 minutes. The Inconel 625 specimens were either in the solution annealed or annealed and aged condition.

The tests were conducted in ETR and at Hanford. Control samples were run in a loop at Hanford where temperature, Ph, etc., were controlled using an identical history to that obtained from ETR operating history. The tests at Hanford were at high temperature; the tests at ETR were at the low temperatures.

Two heats of Inconel 625 were tested to an exposure of about 9×10^{20} nvt (above 1 Mev). Tensile testing was performed 0.05 in./minute for the high-temperature tests and at 0.005 in./minute for the low-temperature tests.

All of the results obtained on the HAP0 program to date are very preliminary and were furnished by Mr. Claudson verbally in order to aid in the material selection for the ABC fuel element cladding.

The room temperature tensile tests for Inconel 625 specimens irradiated at 280, 580, and 740°C disclosed that the ductility diminished as the irradiation temperature increased. The 740°C irradiated samples displayed a higher strength than the unirradiated samples but the ductility was about the same. The results of one tensile test performed at 650°C after an irradiation of 9×10^{19} nvt at 740°C showed a characteristically low value of total elongation (about 3% in one inch) but a reasonably good reduction of area. The latter fact, coupled with the yield-to-ultimate ratio of about 0.83, indicates that the Inconel 625 retains an adequate ductility for use in the ABC superheater cladding after this long an irradiation exposure. Completion of additional tests at this temperature will be made and future tests will be performed at increased exposure time.

The results of one tensile test conducted at 650°C after exposure to 1.2×10^{20} nvt at 280°C showed excellent ductility and strength. These data lend technical support to the choice of Inconel 625 material for the boiler cladding of the Advanced BONUS Core fuel element.

Detailed description of the work performed on the Hanford materials program is to be described by Mr. Claudson in the next Hanford Quarterly Progress Report.

B. Component Evaluation and Tests

1. Thermal Cycling Tests of Zircaloy-Inconel Joints

The erection of the loop for thermal cycle testing the Zircaloy-Inconel joints was completed in the form essentially as described in the last Quarterly Progress Report, GNEC 361.¹ Checkout of the loop

¹ Advanced BONUS Core Development Program, Quarterly Progress Report No. 3, May-July 1964, GNEC 361, General Nuclear Engineering Corp., Aug. 15, 1964.

was done using a dummy specimen. This checkout indicated that test piece temperatures of greater than approximately 700°F could not be obtained for steady-state operation and that test piece temperatures of greater than approximately 600°F could not be obtained for cycling operation due to heat losses between the superheater and the test autoclave. In addition, it was found that the cooling cycle using dry air was excessively long, requiring five minutes for cooling from 600°F to 300°F and much longer if the lower temperature of 100°F is to be attained. The test equipment was modified by placing a resistance heater with a capability of heating to 1000°F within the test Zircaloy-Inconel joint. The details of the modified equipment are illustrated in Fig. V-30. A method for injecting water for the cooling cycle was developed and installed and found to reduce the length of the cooling cycle to 4 minutes.

The control circuitry for the loop has been modified to accommodate the new scheme. The electrical diagram is shown in Fig. V-31. A schematic diagram of the loop indicating design conditions and test pressures is shown in Fig. V-32. More than 8000 cycles from 250°F to 900°F were imposed on the first 1-1/2 in. diameter Zircaloy-Inconel joint, made by Metalonics, with no visible evidence of failure. Removal and inspection of the joint after 1000, 3600 and 6644 cycles showed no visible sign of failure. The joint was dye-penetrant inspected at each inspection period. The test will be discontinued after 10,000 cycles and the specimen forwarded to CEND Windsor for metallurgical inspection and evaluation.

2. Irradiation Tests of Zircaloy-Inconel Joints

Arrangements have been completed for the irradiation of Zircaloy-

Inconel tensile specimens in a Hanford loop at ETR. Orders were placed with Nuclear Metals for 24 tensile specimens machined from the 1-1/2 in. tandem-extruded joints, and with Battelle for the same number of tensile specimens machined from 3-in., explosively welded joints. The specimens were sent to Hanford on September 15 and were installed in ETR on the first of October. It is intended to remove the first specimens for test after one six-week cycle, so that the first data should be available in December. The test is scheduled to be completed by June of 1965. The test will not only be carried out under the proper water temperature conditions, and to reasonably high fluxes, but will also provide information on transition joints produced by two techniques. Fig. V-13 shows the specimens and the mounting fixtures for the irradiation test of the Zr-Inconel joints.

3. Fuel Element Thermal Cycling Test

The purpose of this task is to obtain information on the behavior of ABC fuel elements under thermal cycling conditions. An out-of-pile loop is being constructed which will be capable of subjecting a fuel element to temperatures and pressures it will encounter in reactor services. The loop has provisions for cycling the temperatures between simulated zero power and full-power reactor conditions.

At this time, the loop construction is nearing completion. It is expected that pre-operation checkout of the equipment can begin late in December. Present plans call for the use of a simple cooling coil in place of a final element during the checkout period.

Aside from checking performance of the loop's various components and instrumentation, one of the main reasons for the check-out is to

ascertain the loop's inherent thermal time constant; that is, how is the frequency of temperature cycling limited by the loop's characteristic thermal inertia. It is expected that two to three complete cycles per hour will be possible if there is a sufficient steam flow rate. If fuel elements are tested which require very small steam flow rates, the cycling frequency will probably have to be decreased or the loop modified to improve its time constant.

A flow diagram of the loop is shown in Fig. V-29. The loop as finally constructed may differ slightly, but no major changes are anticipated. A design for the fuel element manifold is nearly complete. This design provides manifolding which will route steam to and from a pair of 7-tube solid-type fuel elements (the current ABC reference design). Thermocouples have been provided for measuring the temperature of influent and effluent steam. There is also provision for measuring and recording the internal pressure in the elements as a means of detecting clad failure.

Detailed drawings will be released for fabrication and completion of manifolding is expected in January.

4. Embedded Tube Strain Cycling Tests

The objective of the embedded tube strain cycling experiments is to determine the deformation of the portion of the superheater tubes embedded in the fuel region of the fuel element, and to determine the extent of fuel "grabbing" or "sticking" that would occur under simulated reactor operating conditions. Each test specimen consists of a single 0.262-in. OD x 0.024-in. wall steam tube embedded in vibratory-compacted UO_2 which is encased in a 3/4-in. ID stainless steel cladding jacket. The first

four test specimens that were manufactured at CEND were completely described in GNEC 345, dated May 15, 1964; two of these specimens used an Incoloy 800 steam tube and the other two used an Inconel 625 steam tube. The test apparatus for performing the embedded tube tests was described in GNEC 345 and GNEC 361.

a. Test No. 14

Test fuel element ABC No. 3 with Incoloy 800 steam tube was heated for a total of 617 cycles, 551 of which were operated automatically between about 90°F and 675°F average steam tube temperature. At an average steam tube temperature of 490°F, the calibrated beam was connected at zero load. During cycling, the 675°F average steam tube temperature resulted in a steam tube expansion of ≈ 0.0262 in. against an axial compression load of 350 pounds. The 90°F average steam tube temperature resulted in a steam tube contraction to ≈ 0.0443 in. below the point where the beam was connected, creating a steam tube tension of 587 pounds. The cycles were repeated every five minutes. Failure of the steam tube occurred 1/8 in. inside the embedded tube area at the bottom. Fig. V-14 shows a graph of the total cycles on ABC No. 3 indicating the maximum average steam tube temperatures and the maximum and minimum expansions for each cycle. For comparison, Fig. V-16 in GNEC 361 shows a similar graph for test element ABC No. 1 with an Inconel 625 steam tube. Note the Inconel 625 steam tube indicated a total shrinkage of only 0.106 in. whereas there was a shrinkage of 0.279 in. for the Incoloy steam tube in ABC No. 3.

b. Test No. 15

Test fuel element ABC No. 4 with Incoloy 800 steam tube was tested in a manner similar to test No. 14; however, ABC No. 4 displayed considerable temperature instability at the ends, and during the seventh cycle burned off at the lower end at the point where the steam tube enters the embedded area. Fig. V-15 shows a graph of these cycles.

Fuel element ABC No. 1 which had been thermally cycled at CEND-Florida Branch was sectioned longitudinally at Windsor. Inspection of the fuel revealed a change of the brown color of the UO_2 powder at both ends of the element for approximately one-inch and along the center steam tube to a jet black. Examination by X-ray diffraction showed an appreciable amount of U_3O_8 in these regions. The oxidation occurred because the tests at elevated temperatures were conducted under normal atmospheric conditions. Four additional elements were fabricated for these tests and are equipped with fittings to permit blanketing of the element with argon during test.

Inspection of the center steam tube of ABC No. 1 showed that the tube diameter increased from 0.263 to 0.280 in. and to 0.267 in. at the two ends of the tube where the tube emerged from the fuel. This was probably caused by local overheating and yielding. The outside diameter of the outer clad remained unchanged, and no structural changes were observed.

The second set of four test specimens, ABC No. 5, 6, 7, and 8, are similar to the first four described in GNEC 345 except for modifications which include improved end fittings, copper sleeves top and bottom to strengthen and protect the hot-spots where the steam tube

enters the UO_2 embedded area, and a 1/8-in. NPT threaded boss on the outside clad tube for maintaining an inert gas pressure on the fuel side of the element. The test elements were fabricated to GNEC Drawing 090-90-056 (see Fig. V-16). Fabrication data for these elements are tabulated below:

(1) Fuel Shell Assembly

<u>Fuel shell assembly No.</u>	<u>Center steam tube material</u>	<u>Outer water tube material</u>
ABC-5	Incoloy 800	347 Stainless
ABC-6	Inconel 635	347 Stainless
ABC-7	Inconel 625	347 Stainless
ABC-8	Incoloy 800	347 Stainless

(2) UO_2 Fuel Powder Formulation

<u>U.S. mesh size</u>	<u>As-Loaded wt%</u>
-6 + 8	22
-8 + 10	11
-10 + 12	11
-12 + 16	16
-30 + 60	3
-60 + 100	12
-200	25

(3) Density Data

<u>Fuel shell assembly</u>	<u>Fuel Height (in.)</u>	<u>Fuel weight (gms)</u>	<u>Fuel volume (cc)</u>	<u>Density (gms/cc)</u>	<u>Density* (% theoretical)</u>
ABC-5	37.957	2,195	233.23	9.41	86.0
ABC-6	38.023	2,193	232.59	9.43	86.2
ABC-7	38.026	2,191	233.11	9.40	85.9
ABC-8	37.839	2,202	232.15	9.49	86.7

*Based on the as-received UO_2 particle density of 10.94 gms/cc as certified by the vendor.

c. Test No. 16

During the period of time that the replacement test elements were being fabricated, it was decided to test a dummy

element identical to the four modified UO_2 fuel elements except that compacted sand was used instead of UO_2 . An Inconel 625 steam tube was used. Test No. 16 was conducted on this sand-filled dummy element mounted with the ends air-cooled as shown in Fig. V-17.

Twenty-three slow cycles were performed without the calibrated beam load. Cycling between zero and 11.0 volts/110 amperes with an average steam tube temperature from ambient to about 1175°F , the steam tube expansions were 0.600 in. to 0.305 in.

The calibrated beam was connected (neutral) at 630°F average steam tube temperature. Seven slow cycles were run between zero and 11.0 volts/112 amperes with an average steam tube temperature from ambient to about 1080°F , resulting in calibrated beam deflection from -0.072 in. to 0.047 in. below and above neutral. The steam tube contraction of -0.072 in. created a tension load of 955 pounds on the tube. The steam tube expansion of 0.047 in. caused a compression load of 626 pounds.

Finally the sand-filled dummy element was cycled 1191 times on 8-minute automatic cycles between zero and 11.0 volts/110 amperes. The average steam tube temperature cycled from ambient to about 1170°F and the beam deflected about -0.078 in. below and 0.042 in. above neutral resulting in steam tube tension of about 1000 pounds and compression of about 570 pounds.

Figure V-18 shows a graph of the total cycles for Test No. 16 indicating the maximum average steam tube temperatures and the maximum and minimum expansions for each cycle.

d. Test No. 17

This test was conducted on a bare Inconel 625 steam tube with powdered asbestos packed between the two air-cooling end inserts to give basic comparative data for tests run on the improved test fuel elements. Figure V-19 shows the curve for the expansion versus temperature and Figure V-24 is a plot of the temperature profile.

e. Test No. 18

This test is similar to Test No. 17 except that an Incoloy 800 bare steam tube was used. Expansion versus temperature is shown in Figure V-20 and the temperature profile is shown in Figure V-21.

f. Test No. 19

An improved test fuel element ABC No. 6 with Inconel 625 steam tube was installed in the test apparatus as shown in Figure V-17. Twenty-three cycles were made without calibrated beam load. Figure V-22 shows typical tube and clad expansions versus average steam tube temperatures. Six manually-operated cycles were made with the beam connected. Comparison of Figure V-23 with Figure V-22 shows the effect of the calibrated beam load on the expansion and contraction of the steam tube and cladding under otherwise identical conditions. These six manually-operated cycles were made between zero and 24 volt/252 amperes. The average steam tube temperature cycled from ambient to 1015°F with an expansion of 0.034 in. and a contraction of 0.046 in. from neutral beam, resulting in a compression load on the steam tube of 472 pounds and a tension load of 639 pounds.

g. Test No. 20

In this test, improved test fuel element ABC No. 6 was installed as in Figure V-17 and cycled 1,985 times. These were 8-minute automatic cycles. Figure V-23 indicates the maximum average steam tube temperature and the maximum and minimum expansions. At the beginning of this test a power of 24 volts/250 amperes produced an average steam tube temperature of only about 1000°F for a steam tube expansion of ≈ 0.036 in. and a compression load of 500 pounds. After 700 cycles, 21.5 volts/224 amperes produced an average steam tube temperature of about 1230°F for a steam tube expansion of ≈ 0.032 in. and a compression load of 444 pounds.

5. Tube Fatigue Tests

This work, as previously described, is exploratory in nature. The test consists of cycling a tube with fixed ends between 140 and 950°F in a 1-3/4-minute cycle. The specimen of Inconel 625 tubing now under test is subject to an approximate 6-mil strain per inch of length for each cycle and has withstood more than 10,000 of such cycles. A relatively high resistance of fatigue of this type is indicated, but completion of the tests is required before much can be said regarding the significance of the work.

6. Fuel Ratchetting and Binding Tests

The object of this task is to fabricate an internally electrically heated solid multitube element for fuel binding and ratchetting tests.

A tentative design layout for the electrical heating units for a solid multitube element was established. Various insulator coatings

for the electrical heating element are being evaluated. These coatings are necessary to prevent electrical current channeling through the uranium dioxide powder at elevated temperature.

Adapters are being designed to permit the fabrication of this element with the jigs and fixtures that will be used in the fabrication of the solid multitube elements for thermal cycle tests and irradiation tests.

7. Bottom Plug Seal Tests

The purpose of this task is to develop a flexible plug seal to permit steam tube movement relative to the fuel in the Zircaloy boiler clad solid multitube elements.

A bottom plug seal will be fabricated from stainless steel wool or some other suitable material similar to those fabricated for the annular multitube elements. Fabrication of the seal assembly is more difficult because the steam tube to boiler clad spacing in the solid multitube is much smaller than in the annular multitube. It is the purpose of this task to resolve the fabrication uncertainties.

8. Leaching Tests

The purpose of this task was to determine the extent to which leaching of fuel powder will occur from slit and hole defects in the clad for elements fabricated to fuel densities ranging from 80 to 90% of theoretical. Because of the decision at the November 13, 1964 meeting in Windsor, Connecticut to defer vibratory compacted fuel development, this test has been discontinued.

Fig. V-1 Bar chart showing changes in the tensile properties of Inconel 625 on aging at two temperatures in oxygen and hydrogen containing steam.

Fig. V-2 Bar chart showing changes in the tensile properties of Inconel 800 on aging at two temperatures in oxygen and hydrogen containing steam.

Fig. V-3 Bar chart showing changes in the tensile properties of Inconel 600 on aging at two temperatures in oxygen and hydrogen containing steam.

Fig. V-4 Bar chart showing changes in the tensile properties of Hastelloy X on aging at two temperatures in oxygen and hydrogen containing steam.

Fig. V-5 Bar chart showing changes in the tensile properties of Type 406 stainless steel on aging at two temperatures in oxygen and hydrogen containing steam.

Fig. V-6 Hardness changes with time for three alloys exposed to 1200°F, oxygen and hydrogen containing steam.

Fig. V-7 Hardness changes with time for four alloys exposed to 1200°F, oxygen and hydrogen containing steam.

Fig. V-8 Weight gain behavior to 5000 hours for five alloys in 1200°F, oxygen and hydrogen containing steam.

Fig. V-9 Weight gain behavior to 5000 hours for three alloys in 1200°F, oxygen and hydrogen containing steam.

Fig. V-10 Collapsed 3-1/2-in.-OD, 0.109-in.-wall Zircaloy-2 tube.
Specimen No. 7.

Fig. V-11 Collapsed 5.07-in.-OD, 0.140-in.-wall Zircaloy-2 tube.
Specimen No. 2. Difficulty in removing the specimen from
the test chamber produced the deformation seen at the ends.

Fig. V-12 Collapsed 3-1/2-in.-OD, 0.104-in.-wall Zircaloy-2 tube.
Specimen No. 8. Note the steam-roller effect on the tube.

Fig. V-13 Zr-Inconel Transitron Irradiation samples.

Fig. V-15 Maximum and minimum expansions for Incoloy 800 steam tube in test No. 15 with ABC No. 4

Fig. V-16 Improved vibratory compacted single tube test specimens showing copper sleeves and inert gas inlet.

Fig. V-17 Setup for tests using water-cooled jacket and air-cooled steam tube ends.

Fig. V-18 Maximum and minimum expansions for Inconel 625 steam tube in sand-filled dummy fuel element Test No. 16.

Fig. V-19 Test 17, Run 3, Bare Tube Inconel 625, Asbestos Filled. Air Cooled Ends.

Fig. V-20 Test 18, Run 1, Bare Tube Incoloy 800, Ends Air Cooled, Asbestos Filled.

Fig. V-21 Test 18, Run 2, Bare Tube Incoloy 800 Asbestos Filled Tube - Ends Air Cooled.

Fig. V-22 Test 19, Run 8 (1st cycle) ABC 6 Inconel 625.

Fig. V-23 Test 19, Run 9 (2nd cycle) ABC 6 Inconel 625.

Fig. V-24 Test 17, Run 1, Temperature Profile Bare Tube - Asbestos Filled Inconel 625, Cooled Ends.

Fig. V-25 Compression test fixture.

Fig. V-26 Tensile properties of annealed Incoloy 800 tubing, 0.396" O.D. x 0.039" wall.

Fig. V-27 Tensile properties of annealed Inconel 625 tubing, 0.262" O.D. x 0.24" wall.

Fig. V-28 Tensile properties of annealed Inconel 625 tubing, 0.396" O.D. x 0.039" wall.

Fig. V-29 Fuel element thermal cycling flow diagram.

Fig. V-30 Zircaloy-Inconel transition joint thermal cycling test element.

Fig. V-31 Wiring diagram Zircaloy-Inconel joint thermal cycling test loop.

Fig. V-32 Schematic diagram of Zircaloy-Inconel transition joint thermal cycling test loop.

VI. MANUFACTURING DEVELOPMENT

A. Zircaloy-Inconel Transition Joint Development

Work on the preparation of transition joints is completed at all three of the suppliers; at NMI and Metalonics for the tandem extruded joints, and Battelle Memorial Institute for explosively welded joints.

1. Explosive Welding

Work at Battelle concentrated on the production of large tubular joints from which tensile specimens for the irradiation test could be machined. A total of more than 40 large joints were prepared during the course of the program. The last seven of these were intended for testing as well as for the preparation of the tensile specimens. All of this group appeared to be quite uniform and of acceptable quality. One of the joints was tested by thermal cycling between 150°F and 750°F at a heating time of 3 minutes, and an equal cooling time. After 1500 cycles the specimen was removed, apparently unchanged, and sectioned for metallographic examination. Little or no sign of separation at the interface was observed and the joint appeared sound.

While all the Battelle effort has been directed toward producing large (3.3 in. ID) joints, it was the opinion of the experimenters that the techniques developed could be successfully applied to producing 1-1/2 in. joints.

BMI has supplied Hanford with 24 tensile specimens prepared from explosively formed tubing. These samples will be irradiated before tensile testing.

The final report remains to be completed and issued.

2. Co-Extruded Transition Joint Fabrication

a. Metalonics

All of the eight small and eight large joints were finished and inspected. Seven completed large joints were received at Dunedin for examination. A photograph of one joint is given in Fig. VI-1. It will be observed that the interface between the Zircaloy and Inconel does not show as a smooth circumferential line but is roughly in the form of a large sine wave. This variation of the interface location

(shift) might have little effect on the strength of the joint, but could give rise to problems in fuel-element fabrication.

b. Nuclear Metals

Four of the small joints were successfully prepared and sectioned for the preparation of the tensile specimens required for the irradiation test at Hanford. Fig. VI-2 shows a longitudinal section of a 1-1/2 in. dia. Zr-Inconel joint produced at NMI. The remaining four small joints were shipped on the first of November. NMI encountered some difficulty with the extrusion of the large joints; however, one satisfactory large joint was produced and examined. The transition joint appears satisfactory in all respects, and there is essentially no shift in the interface on the surfaces of the joint.

Metallographic examination is being made of one tube from each group of large and small transition joints. Results of these tests will be reported during the next period.

Radiographs indicate no inclusions or defects in the band areas of the joints. Visual examination of the joints indicate they are sound and that dimensions are within expected tolerances. Due to the deferral of work on Zircaloy-2 clad fuel elements this fiscal year, the manufacture of the remaining eight large transition joints at NMI was discontinued.

B. Perforated UO₂ Pellets

An investigation was made of the lot of 55 perforated UO₂ pellets received from NUMEC. Full dimensional data, which included measurements of hole diameters of all holes in all pellets, the center hole to radial hole spacing of all pellets, the pellet diameter, and the pellet density, was received from NUMEC. A major objective was to find the maximum diameter tubing which could be inserted into a stack of pellets. Three sizes of tube were available, 0.251 in., 0.258 in., and 0.262 in. OD. Seven-tube frames which simulated the steam tube arrangement in a solid fuel, 7-tube, fuel element were prepared, one for each size of tubing. It was found that proper location of the holes in the base plug (corresponding to a header in the actual element) presented some

machining difficulties, and that meeting the hole tolerances of the UO_2 pellets as produced by ceramic techniques required considerable care in machining the base plug.

Placing the pellets on the 0.251 in. OD tube frame was quite easy. When assembled, the pellets were relatively loose. On the 0.258 in. OD tube frame, assembly was also easy. Placing the pellets on the 0.262 in. OD tube frame was readily done, but not quite so easily as with the smaller tubes. Placing the first pellet on the tubes required some manipulation, and is more difficult as the tubes become larger. Once placed, the first pellet is moved down the tubes one-half inch which maintains the tube spacing, so that it is easy to insert the next, and subsequent, pellets. An inexperienced operator was able to insert 20 pellets on the largest tubes in a 5-minute period after the first pellet had been placed. The pellets average approximately 107 gms UO_2 per pellet. The assembly of pellets is shown on Fig. VI-3. It will be seen that the OD of the pellet stack is not smooth and that small gaps are occasionally present between the faces of pellets. It is intended to investigate the clean-up grinding of both the OD and the pellet faces to prevent gap formation.

Table VI-1 lists the pellets as assembled and gives the minimum hole diameter, or diameters, for each pellet. The worst radial hole spacing for each pellet is also given. It will be seen that pellets with holes as small as 0.264 in. could be assembled on 0.262 in. tubes, and that pellets with holes as much as 0.008 in. from true position in the radial direction could also be assembled. The pellets were assembled in the large tubes by keeping the index marks on each pellet aligned, although with many pellets this is not essential.

The pellets appear surprisingly strong with regard to resistance to forces imposed on them in handling. Resistance to impact is very small. Several pellets have been broken in dropping a few inches down the tube to the base. A usable quantitative index of pellet strength is difficult to describe, but qualitatively the pellets appear to have more than adequate strength as long as they are not subjected to impact during assembly.

Table VI-1 CRITICAL DIMENSIONS OF THE PERFORATED UO₂
 PELLETS AS ASSEMBLED ON THE 0.262 in. OD TUBES

<u>Pellet No.</u> <u>Top to bottom</u>	<u>Smallest holes</u>	<u>Worst radial hole spacing</u> <u>(0.444 in. required)</u>
19	0.269	0.440
44	0.270	0.448
41	0.268	0.448
7	Not given	
1	0.269	0.440
32	0.267-0.268	0.439
12	0.267-0.270	0.447
9	0.265-0.266	0.440
10	0.267-0.267	0.438
28	0.269	0.488
45	0.268	0.450
42	0.267-0.270	0.450
5	0.264-0.266	0.439
2	0.264-0.269	0.446
55	0.270	0.449
50	0.269	0.448
34	0.266-0.267	0.441
4	0.266-0.266	0.436
24	0.267-0.269	0.447
33	0.268	0.443
20	0.269	0.446
30	0.269	0.4485
31	0.267-0.268	0.4425
40	0.268	0.4485
26	0.270	2- 0.450
43	0.268	0.450
39	0.269	0.4485
16	0.268	0.449
37	0.269	0.448
23	0.269	0.4485
3	0.266-0.266	0.439
18	0.268	0.447
25	0.268	0.449
52	0.268	0.450
49	0.267-0.270	0.4505
13	0.268	0.448

A second order was placed with NUMEC for two 50-pellet lots of experimental perforated pellets whose dimensions conform to the latest ABC, seven-tube design. These pellets were completed on November 10, 1964 and an inspection report submitted to CEND for evaluation. The steps used in the preparation of the pellets is briefly described by NUMEC as follows:

- Step 1. Receive and certify UO_2 to the specification.
2. Pre-press UO_2 powder to a slug density of 4-5 mg/cc.
3. Granulate the slugs through a 14-mesh screen.
4. Add 0.2 to 0.4 w/o Sterotex to the granular material and blend.
5. Press pellets to a green density of 4.5 gm/cc (nominal) using a hand-filled or automatic die on a hydraulic press.
6. Sinter the pellets at 1600-1700°C for 4-8 hours in hydrogen.
7. Inspect the pellets and prepare a final dimensional analysis.
8. Package pellets in rolls and ship in a 5-gallon steel container.

A visit to NUMEC will be made on December 3, 1964 to examine the pellets, review inspection records, and to discuss pellet specifications and fabrication problems.

C. Welding Development

Substantial progress was made in the welding development effort during this report period for two reasons: (1) the design alternates were eliminated and work was directed toward the single reference design as shown in Fig. II-21 of this report, (2) experimental welding was performed concurrently at CEND-Florida Branch and at Southwest Research Institute (Sw.R.I.).

Sample weld specimens were made as shown in Fig. VI-4. These double ended specimens contained a tube to tube-to-sheet weld at each end and a lap weld at the center between the large and small diameter tubes. The specimens are suitable for tensile tests and reversed constant load fatigue tests both of which were performed at Southwest Research

Institute. Because the tube sheet material was changed from Inconel 600 to Inconel 625, a significant improvement in the tensile strength and fatigue strength was experienced over that reported in GNEC 355, dated July 20, 1964. For instance, Fig. VI-5 is a plot of the fatigue strength of the Inconel 625 to Inconel 600 tube joints recently tested (which are identical to the previous test samples in all other respects) are tabulated in Table VI-2 and plotted on the same figure as the previous fatigue tests (Fig. VI-5). Room temperature and high temperature tensile test results of Inconel 625 to Inconel 625 weld specimens demonstrate that the test welds are strong enough to develop the full strength of the base tube in short-time tension tests. Table VI-3 shows the tensile test results obtained at Sw.R.I. on welded samples fabricated as described in Fig. VI-30.

Table VI-3A TENSILE TESTS OF INCONEL 625 TO INCONEL 625
TUBE TO TUBE-SHEET JOINTS

Spec. No.	Temp. of test (°F)	Yield strength at 0.2% offset (psi)	Ultimate strength (psi)	Remarks
29	R.T.	77,800	134,400	Failed at end B
30	R.T.	77,800	135,600	Failed at end A
31	925	54,200	106,700	Failed at end A
32	925	54,200	108,300	Failed at end B

The strength advantage of using an Inconel 625 tube-sheet is clearly demonstrated by the above tests, and the small increase in cost incurred by using the higher priced tube-sheet material is more than offset by the reduction in number and type of welding procedures that will be required for this fuel element. A preliminary welding procedure for the tube to tube-sheet joints was prepared by Sw.R.I. and forwarded to Manufacturing for information and preparation of plans for making test

Table VI-2 FATIGUE TESTS OF COMPOSITE STEAM TUBE WELD SAMPLES

<u>Spec. No.</u>	<u>Test Temperature °F</u>	<u>Applied Cycling Stress</u>	<u>Cycles to failure</u>	<u>Where</u>
3	925	50,000	2,378	Root of weld Small tube to header joint
4	925	39,000	4,356	" "
6	925	33,000	19,801	No failure
11	925	37,500	6,537	Small tube to header joint
12	750	55,000	3,545	Lap joint Failed in heat effected zone of tube on small tube side
10	754	60,000	3,909	" "
8	750	47,500	14,780	Large tube to header weld

fuel elements. The lap weld joint is undergoing further evaluation as to the effect of variation of welding parameters on the integrity of the joint. Based upon information obtained so far, the Inconel 625 material appears to be readily welded by the techniques used and it is unusually insensitive to variations in welding parameters.

Because the design selected as the reference design employs Inconel 625 as the boiler cladding material, mock-ups of the boiler clad to tube-sheet welds were fabricated as shown in Fig. VI-6. Several of these mock-ups were subsequently tested in tension at ambient temperature and at 600°F. The results are given in Table VI-3B.

Work on the boiler clad weld is continuing at Sw.R.I., and it is expected that the welding procedures for this weld will be established during the next report period.

D. Powder Compaction

The purpose of this investigation was to obtain the maximum compacted density of fused uranium dioxide powder in superheat fuel element geometries of the type shown in Fig's. II-3 and VI-7. Evaluation of several selected particle size distributions and of various fuel loading and electromechanical compaction conditions demonstrated that densities of 85% of theoretical are obtainable for the annular multitube geometry. A second objective of the task was the determination of the influence of steam tube ligament spacing on compacted powder density. Eleven, fourteen and seventeen steam tube elements were compacted. Except for the seventeen steam tube assembly in which the steam tube ligament spacing was much smaller than the largest powder particle size, 0.055 inches vs. 0.132 inches, ligament spacing had little effect on compacted density of the powder.

Three particle size formulations of fused uranium dioxide powder were evaluated. These formulations are shown in Table VI-4.

Formulations A and B were selected on the basis of extensive prior development work at CEND¹ to obtain high compacted fuel density and also

¹ Murtha, B.E., Chernock, W.P., "The Development and Testing of UO₂ Fuel Systems for Water Reactor Applications" CEND-178 Combustion Engineering, Inc. - Period ending March 31, 1963.

Table VI-3B TENSILE TESTS OF INCONEL 625 BOILER CLAD TO INCONEL 625 TUBE-SHEET JOINTS

Spec. No.	Temperature	Prop. Elas. Limit (psi)	Tube Area = 0.141 in ²	
			Ultimate strength (psi)	Where
3 (Lap joint)	R.T.	76,800	136,000	Failed in tube 1/4 in. from weld
2 (Lap joint)	R.T.	- - -	134,100	Failed in tube 1/4 in. from weld
1 (Lap joint)	600°F	- - -	109,000	In weld through zone
7 (Square butt)	600°F	- - -	112,000	In tube
6 (Square butt)	600°F	- - -	113,500	In tube

Table VI-4 FUEL PARTICLE SIZE FORMULATIONS

<u>US. Mesh size</u>	<u>A coarse</u>	<u>B medium</u>	<u>C Fine</u>
-6+8	50 w/o	22 w/o	6 w/o
-8+10	6	11	15
-10+12	6	11	) - 17
-12+16	10	16	)
-16+20	10	--	18
-20+30	--	--	3
-30+60	--	3	13
-60+100	--	12	--
-200	18	25	--
-200+300	--	--	10
-325	--	--	18

good fuel powder utilization. Formula C was selected on the basis of vibratory compaction studies at the General Electric Company².

The fuel loading and compaction apparatus, which has been used in the program, is shown in Fig's. VI-8 and VI-9. Loading of the fuel and compaction was accomplished under mechanical pump vacuum to prevent possible airborne segregation of the fines during loading of the powder. The loading funnel was designed to permit controlled powder loading into the inner annulus between the center water clad tube and the steam tubes and the outer annulus between the steam tubes and the outer water clad. Uniform loading was required to avoid differential fuel powder height in the annuli which then could produce lateral loads on the steam tubes and displace them from design position. Uniform loading was also required to avoid possible screening and particle size segregation by powder flow through the steam tube ligament spaces. To determine whether uniform blending of particle sizes depended on batch size, fuel elements were fabricated using 2, 7, and 14 kg incremental charges. After each 14 kg loading of fuel the loading funnel was removed and a compaction ram inserted. Table loads ranging from 50 lbs to 300 lbs were applied to the ram and compaction continued until densification ceased. The time required averaged about 30 minutes. Total fuel loadings for an element varied from 40 to 44 kg and the compacted fuel height averaged about 55 inches.

An MB electromechanical vibration machine, Type C10, was used for the compaction studies. Since the fuel element and the compaction hardware exceeded the table load limit of the MB machine, a spring suspension from a stationary platform carried the weight of the element and compaction hardware. The MB vibrator worked against the loaded springs and the inertial mass of the element. During loading and compaction the MB unit operated under maximum force output conditions,

² DeHollander, W.R., "Vibrational Compaction of Uranium Dioxide" GEAP-4032, General Electric Atomic Power Equipment Department March 1, 1962.

while sweeping a vibration frequency range from 150 to 1000 cycles per second at the sweep rate of one complete scan per minute.

A tabulation of the compacted densities for the various fuel elements is shown in Table VI-5. A review of this experimental data shows that formula B powder produced the highest densities. The particle size formulations of the three formulas are given on Table VI-4. Densities of 85% of theoretical can be obtained with this formulation. Based on post-compaction evaluation, it appears that higher densities were not achieved with formula A because:

1. The limited force output of the vibration machine did not permit the required amount of in situ communication required for such a coarse formulation.
2. A large fraction of the fines segregated to the bottom of the element. This may have been caused by the inability of the coarse particle to accommodate into a tight packing array around the steam tubes and thus prevent the drift of fines through the voids adjacent to the steam tubes.

The probable reasons why formula C did not achieve higher densities are:

1. The fraction of fines in the powder formulation was too large for the annular element geometry.
2. The specific vibratory compaction conditions used at General Electric could not be reproduced in the compaction of the annular element at CEND.

In summary, densities in excess of 85% of theoretical may be obtainable in the annular multitube element by some adjustments in the particle size distributions of formula B and by the use of a larger electromechanical vibration machine. These powder compaction methods developed on a laboratory scale are readily adaptable to full core quantity production use.

At the Advanced Bonus Core Design Selection Meeting held in July, 1964, the solid multitube element was chosen as the element for immediate development. Therefore, powder compaction experiments were planned with the solid multitube element using extrapolated particle

Table VI-5 COMPACTED DENSITIES OF ANNULAR MULTITUBE
FUEL ELEMENTS

<u>Element number</u>	<u>Number of steam tubes</u>	<u>Fuel* Formula</u>	<u>Compacted** density (% theoretical density)</u>	<u>Number of incremental loadings</u>
10-1-11	11	B	79.7	3
10-1-11	11	B	83.6 (Post impaction)	3
10-1A-14	14	B	83.3	6
10-1A-14	14	B	85.0 (Post impaction)	6
10-1-17	17	B	83.4	6
10-2-11	11	A	77.3	18
10-2-14	14	C	79.2	6
10-3-11	11	C	78.4	6
10-3-14	14	A	79.5	6

* Table VI-3.

** Based on a UO_2 theoretical density of 10.96 gms/cc.

size data and adjusted electromechanical vibration compaction conditions from the annular multitube compactions, to obtain proper compaction parameters. However, a cost comparison study of powder compacted versus pelletized fuel elements of the solid multitube geometry indicated that for the small Bonus reference design pellets were more economical. Further powder compaction studies will be conducted if the expected economic advantages of the pellet fuel do not materialize.

E. Spacer Washer Evaluation

The purpose of this investigation was to determine the degree of dimensional control that could be obtained by the use of spacer washers placed at regular intervals in the fuel column. The experimental studies were conducted with simple disc type stainless steel washers shown in Fig. VI-10.

Since the powder compaction method required the loading of a fuel increment, compaction and then insertion of a spacer washer before the loading of the next increment, it was necessary that the spacer washer accommodate the following:

1. The possible accumulation of dimensional variations due to clad tube straightness, ovality and tube wall thickness.
2. Variations in geometry produced by assembly and the compaction jig and fixture tolerance requirements.
3. Disturbances in element geometry caused by the fuel loading and compaction operation.

The first requirement was established from inspection data of the tubing purchased for the compaction studies. The dimensions and tolerances of this tubing are shown in Table VI-6.

The second and third requirements were established by trial and error, assuming initially zero tolerance requirements, fabricating the washer, and then modifying washer dimensions slightly until insertion of the washer was possible after a fuel increment loading. The resultant spacer washer dimensions and the maximum possible diametral clearances between washer and clad surfaces are shown in Table VI-7.

Table VI-6 ANNULAR MULTITUBE ELEMENT TUBING DIMENSIONS

<u>Dimension</u>	<u>Center Water Tube (inches)</u>	<u>Steam Tube (inches)</u>	<u>Outer Water Tube (inches)</u>
OD	1.564 $\pm$ 0.001	0.396 $\pm$ 0.001	- -
ID	- -	- -	3.316 $\pm$ 0.006
Wall Thickness	0.065 $\pm$ 0.0005	0.039 $\pm$ 0.002	0.083 $\pm$ 0.000 - 0.008
Ovality	- 0.002	- 0.002	- 0.015
Straightness	- 0.015 $\frac{\text{inches}}{3 \text{ feet}}$	- 0.030 $\frac{\text{inches}}{3 \text{ feet}}$	- 0.010 $\frac{\text{inches}}{3 \text{ feet}}$

Table VI-7 SPACER WASHER DIMENSIONS
AND CLEARANCES

<u>Dimension</u>	<u>Dimension</u>	<u>Max. dimensional clearance</u>
OD	3.294 in.	0.022 in.
Steam tube hole	0.406 in.	0.010 in.
Center water clad hole	1.570 in.	0.006 in.

The center water clad tube was the design reference tube for the element. All compaction jigs and fixtures positioned the steam tubes and the outer clad with respect to the center water clad tube, and all inspection measurements were made using the axis of the center water clad tube at each end of the tube as the reference line. Measurements of center water clad tube variation in location from design position were made at six inch intervals along the full length of the tube. The data are shown in Table VI-8.

Except for a small .004-in. correction for sag at the 30-in. station and lesser corrections at the other stations, the data show the strong influence of tube selection. Although all of the tubing used was within the specification limits shown in Table VI-6, the most uniform piece of tubing was selected for the first element to be compacted and each subsequent element was fabricated with tubing with successively larger bow and straightness variation. Tube bow exerted the strongest influence on deviations from design position.

Steam tubes were positioned in the element during fuel loading and compaction by 5/16-in. diameter rods inserted through the steam tubes and through top and bottom position spacer washers. These washers were indexed to the center water clad tube for radial and azimuthal alignment. A tabulation of inspection data of the variations in steam tube position in radial and azimuthal directions for each element was made and reported¹ in GNEC-361.

Data for two additional elements is shown in Tables VI-9 through

¹ Gibbons, J.F., "Advanced Bonus Core Development Program" GNEC-361 General Nuclear Engineering Corporation, May-July, 1964.

VI-12. A summary of this inspection data for the variation of steam tube location in the radial direction is shown in Table VI-13. A similar summary of data for the azimuthal direction is shown in Table VI-14.

Inspection of the data in Table VI-13 shows that radial variation from design position in excess of 0.020 inches and 0.030 inches were far more numerous in element 10-1-11 which contained only one spacer washer 18 inches above the bottom tube sheet of the element. All of the remaining elements contained spacer washers at the 18-inch and 36-inch fuel height. In these elements the number of readings exceeding 0.030 inches is very small, and the number of readings exceeding 0.020 inches is less than 20% in all cases.

Inspection of the data in Table VI-14 shows that azimuthal variations from design position in excess of 0.020 inches and 0.030 inches is far more numerous in the elements lacking spacer washers. Although control of steam tube location to within ± 0.030 inches of design positions is definitely attainable with the disc type spacer washer inserted at intervals into the fuel column, it appears that control to better than ± 0.020 inches may not be attained. To obtain such control a spacer washer concept of the type previously shown in Fig. VI-15 of GNEC-361 was designed. This washer is assembled on to the steam tubes and water cladding before fuel loading and, therefore, the assembly tolerances could be reduced. Fuel during loading flows through the hole openings in the washer and, therefore, binding problems which were encountered with the disc type spacer washer should not occur. Dimensional control to ± 0.020 inches from design position appears feasible.

F. Plug Seal

The purpose of this investigation was to determine whether a stainless steel wool plug can serve as a satisfactory sliding seal to accommodate the differential thermal expansion between the steam tubes and the boiler clad of the fuel element.

Two stainless steel wool plug seals were fabricated by circumferentially wrapping coarse stainless steel wool matting on a center water clad tube between two retainer spacer discs set one inch apart. The

Table VI-8 CENTER WATER CLAD TUBE DEVIATION FROM DESIGN POSITION
IN COMPACTED ANNULAR MULTITUBE ELEMENTS

Element No.	Survey Stations	5/8"	6"	12"	18"	24"	30"	36"	42"	48"	54"	60"
10-1-11	0°	0.000	0.000	-0.002	-0.002	-0.003	-0.003	-0.002	0.000	+0.003	+0.002	0.000
	90°	0.000	-0.003	-0.007	-0.007	-0.010	-0.010	-0.011	-0.011	-0.008	-0.006	0.000
10-1A-14	0°	0.000	+0.002	+0.002	+0.000	+0.005	+0.007	+0.007	+0.007	+0.009	+0.007	-0.001
	90°	0.000	-0.002	-0.002	-0.005	-0.008	-0.010	-0.015	-0.012	-0.012	-0.008	+0.001
10-1A-14 Post Impact	0°	0.000	+0.002	+0.002	+0.006	+0.005	+0.008	+0.008	+0.010	+0.008	0.000	0.000
	90°	0.000	+0.005	0.000	-0.003	-0.009	-0.012	-0.013	-0.013	-0.012	-0.007	+0.003
10-1-17	0°	0.000	-0.007	-0.011	-0.010	-0.011	-0.010	-0.009	-0.006	-0.003	+0.001	0.000
	90°	0.000	-0.004	-0.007	-0.010	-0.009	-0.017	-0.023	-0.029	-0.033	-0.034	-0.007
10-2-11	0°	0.000	+0.001	-0.004	-0.002	+0.006	+0.013	+0.019	+0.020	+0.018	+0.012	
	90°	0.000	-0.010	-0.008	-0.009	-0.010	-0.011	-0.012	-0.013	-0.018	-0.018	
10-2-14	0°	0.000	-0.005	-0.013	-0.012	-0.009	-0.004	-0.000	+0.003	+0.004	+0.003	
	90°	0.000	-0.001	-0.004	-0.001	+0.002	+0.005	+0.008	+0.009	+0.007	+0.003	
10-3-11	0°	0.000	0.000	0.000	+0.006	+0.011	+0.008	+0.005	+0.001	+0.001	-0.001	
	90°	0.000	-0.010	-0.019	-0.025	-0.033	-0.036	-0.038	-0.037	-0.037	-0.032	
10-3-14	0°	0.000	-0.024	-0.024	-0.013	-0.001	+0.007	+0.011	+0.011	+0.006	+0.006	
	90°	0.000	-0.003	-0.004	-0.013	-0.022	-0.036	-0.044	-0.046	-0.042	-0.042	

Table VI-9 ELEMENT NO. 10-3-11 - VARIATION OF STEAM TUBE LOCATION
FROM DESIGN POSITION

Tube No.	Survey* Coordinate	5/8"	6"	12"	18"	24"	30"	36"	42"	48"	54"	60"
1	x	0.000	+0.006	-0.003	-0.027	-0.036	-0.042	-0.042	-0.049	-0.052	-0.039	-0.013
	y	0.000	+0.008	+0.012	+0.010	+0.008	+0.003	+0.002	+0.001	+0.001	-0.003	-0.008
2	x	0.000	-0.018	-0.025	-0.027	-0.034	-0.032	-0.033	-0.046	-0.053	-0.041	-0.010
	y	-0.008	-0.005	-0.003	+0.003	+0.003	0.000	+0.001	+0.006	+0.004	+0.005	+0.006
3	x	0.000	-0.029	-0.034	-0.025	-0.019	-0.022	-0.027	-0.036	-0.035	-0.030	-0.009
	y	-0.006	0.000	+0.003	+0.005	+0.001	-0.004	-0.001	+0.007	+0.001	-0.007	+0.005
4	x	0.000	-0.034	-0.038	-0.016	-0.006	-0.011	-0.021	-0.019	-0.016	-0.017	-0.014
	y	-0.009	-0.004	-0.005	+0.002	+0.002	+0.001	-0.001	+0.013	+0.012	+0.005	+0.003
5	x	0.000	-0.017	-0.013	-0.005	-0.005	-0.014	-0.016	-0.003	+0.006	-0.004	-0.009
	y	+0.006	-0.005	-0.010	-0.009	-0.004	+0.002	+0.001	0.000	+0.003	-0.003	-0.002
6	x	0.000	-0.004	-0.021	-0.013	-0.005	-0.004	-0.006	-0.008	-0.009	-0.012	-0.006
	y	-0.003	-0.008	-0.016	-0.015	-0.005	-0.003	-0.003	-0.007	-0.011	-0.013	-0.001
7	x	0.000	-0.012	-0.020	-0.020	-0.018	-0.015	-0.011	-0.016	-0.033	-0.026	+0.004
	y	-0.001	-0.001	-0.007	-0.016	-0.012	-0.007	-0.008	-0.013	-0.009	-0.012	-0.006
8	x	0.000	0.000	-0.007	-0.014	-0.035	-0.034	-0.026	-0.022	-0.022	-0.018	-0.003
	y	+0.001	-0.003	-0.008	-0.012	-0.013	-0.011	-0.016	-0.016	-0.014	-0.014	+0.001
9	x	0.000	-0.005	-0.020	-0.036	-0.050	-0.051	-0.040	-0.023	-0.011	-0.009	-0.015
	y	+0.001	+0.005	-0.005	-0.012	-0.009	-0.012	-0.014	-0.014	-0.014	-0.018	-0.008
10	x	0.000	+0.010	-0.009	-0.034	-0.047	-0.039	-0.037	-0.023	-0.007	-0.013	-0.022
	y	+0.002	+0.003	0.000	-0.006	+0.002	-0.007	-0.012	-0.016	-0.014	-0.013	-0.005
11	x	0.000	+0.012	-0.008	-0.032	-0.037	-0.030	-0.034	-0.037	-0.028	-0.024	-0.010
	y	+0.001	+0.004	+0.004	+0.003	+0.005	-0.001	-0.007	-0.011	-0.016	-0.017	-0.009

* x = Azimuthal, y = Radial

Table VI-10 ELEMENT NO. 10-3-14 - VARIATION OF STEAM TUBE LOCATION
FROM DESIGN POSITION

Tube No.	Survey* Coor- dinate	5/8"	6"	12"	18"	24"	30"	36"	42"	48"	54"	60"
1	x	.000	-.007	-.010	-.026	-.042	-.060	-.065	-.065	-.065	-.060	-.038
	y	-.003	-.011	-.011	+.002	+.004	+.001	+.006	+.011	+.011	+.008	+.001
2	x	.000	+.009	+.017	+.003	-.041	-.060	-.070	-.065	-.060	-.058	-.030
	y	-.003	-.006	+.001	+.007	.000	-.004	+.006	+.023	+.029	+.022	-.001
3	x	.000	-.003	-.015	-.025	-.040	-.055	-.060	-.065	-.053	-.048	-.043
	y	-.003	-.015	-.010	-.003	+.002	+.003	+.015	+.026	+.026	+.018	+.004
4	x	.000	-.018	-.039	-.036	-.017	-.018	-.030	-.044	-.045	-.041	-.027
	y	-.001	-.015	-.017	-.008	+.001	+.007	+.017	+.024	+.020	+.016	+.003
5	x	.000	-.023	-.043	-.035	-.019	-.016	-.021	-.036	-.041	-.039	-.015
	y	-.001	-.010	-.013	-.008	-.004	+.001	+.010	+.020	+.020	+.016	+.005
6	x	.000	-.017	-.030	-.030	-.019	-.020	-.027	-.033	-.036	-.030	-.025
	y	+.005	.000	-.005	-.003	-.006	-.011	-.004	+.007	+.009	+.007	+.001
7	x	.000	-.016	-.030	-.025	-.017	-.021	-.029	-.030	-.030	-.032	-.024
	y	+.006	+.003	+.001	.000	-.005	-.014	-.010	-.004	-.005	-.003	+.003
8	x	.000	-.021	-.041	-.040	-.021	-.010	-.018	-.040	-.050	-.044	-.017
	y	+.008	+.012	+.016	+.006	-.006	-.017	-.016	-.011	-.009	-.007	.000
9	x	.000	-.017	-.032	-.029	-.032	-.035	-.040	-.046	-.046	-.036	-.021
	y	+.004	+.011	+.014	+.009	.000	-.008	-.013	-.016	-.016	-.009	+.004
10	x	.000	+.002	-.008	-.023	-.040	-.048	-.053	-.060	-.060	-.047	-.034
	y	+.005	+.021	+.020	+.013	-.002	-.009	-.015	-.020	-.018	-.011	.000

Table VI-10 (continued)

Tube No.	Survey* Coordinate	5/8"	6"	12"	18"	24"	30"	36"	42"	48"	54"	60"
11	x	.000	+.014	+.011	-.010	-.040	-.055	-.060	-.070	-.065	-.056	-.035
	y	+.001	+.022	+.025	+.016	-.002	-.009	-.015	-.022	-.020	-.014	+.003
12	x	.000	+.020	+.026	+.001	-.050	-.070	-.075	-.065	-.065	-.052	-.044
	y	.000	+.010	+.014	+.012	-.002	-.010	-.015	-.015	-.010	-.008	.000
13	x	.000	+.019	+.025	-.005	-.049	-.065	-.070	-.065	-.065	-.065	-.033
	y	-.002	+.014	+.024	+.015	-.004	-.016	-.016	-.011	-.007	-.011	+.002
14	x	.000	+.004	+.008	-.012	**	**	**	**	**	**	**
	y	-.005	.000	+.010	-.019	**	**	**	**	**	**	**

* x = Azimuthal, y = Radial

** Burr obstructed movement of optical target.

TABLE VI-11

FUEL ELEMENT NO. 10-3-11
 VARIATION IN LIGAMENT SPACING
 FROM DESIGN VALUE

Ligament Between Steam Tubes	Stations										
	5/8"	6"	12"	18"	24"	30"	36"	42"	48"	54"	60"
1-2		+0.024	+0.022	.000	-.002	-.010	-.009	-.003	+0.001	+0.002	-.003
2-3		+0.011	+0.009	-.002	-.015	-.010	-.005	-.010	-.018	-.011	-.001
3-4		+0.005	+0.004	-.009	-.003	-.011	-.006	-.017	-.019	-.013	+0.005
4-5		-.017	-.025	-.011	-.001	+0.003	-.005	-.016	-.022	-.013	-.005
5-6		-.013	+0.008	+0.008	.000	-.010	-.010	+0.005	+0.015	+0.008	-.003
6-7		+0.008	-.001	+0.007	+0.013	+0.011	+0.005	+0.008	+0.024	+0.014	-.010
7-8		-.012	-.013	-.006	+0.017	+0.019	+0.015	+0.006	-.011	-.008	+0.007
8-9		+0.005	+0.013	+0.022	+0.015	+0.017	+0.014	+0.001	-.011	-.009	+0.012
9-10		-.015	-.011	-.002	-.003	-.012	-.003	.000	-.004	+0.004	+0.007
10-11		-.002	-.001	-.002	-.010	-.009	-.003	+0.014	+0.021	+0.011	-.012
11-1		+0.006	-.005	-.005	-.001	+0.012	+0.008	+0.012	+0.024	+0.015	+0.003

TABLE VI-12 FUEL ELEMENT NO. 10-3-14
 VARIATION IN LIGAMENT SPACING
 FROM DESIGN VALUE

Ligament Between Steam Tube	<u>5/8"</u>	<u>6"</u>	<u>12"</u>	<u>18"</u>	<u>24"</u>	<u>30"</u>	<u>36"</u>	<u>42"</u>	<u>48"</u>	<u>54"</u>	<u>60"</u>
1-2		-.016	-.027	-.029	-.001	.000	+.005	.000	-.005	-.002	-.008
2-3		+.012	+.032	+.028	-.001	-.005	-.010	.000	-.007	-.010	+.013
3-4		+.015	+.024	+.011	-.023	-.037	-.030	-.021	-.008	-.007	-.016
4-5		+.005	+.004	-.001	+.002	-.002	-.009	-.008	-.004	-.002	-.012
5-6		-.006	-.013	-.005	.000	+.004	+.006	-.003	-.005	-.009	+.010
6-7		-.001	.000	-.005	-.002	+.001	+.002	-.003	-.006	+.002	-.001
7-8		+.005	+.011	+.015	+.004	-.011	-.011	+.010	+.020	+.012	-.007
8-9		-.004	-.009	-.011	+.011	+.025	+.022	+.006	-.004	-.008	+.004
9-10		-.019	-.024	-.006	+.008	+.013	+.013	+.014	+.014	+.011	+.013
10-11		-.012	-.019	-.013	.000	+.007	+.007	+.010	+.005	+.009	+.001
11-12		-.006	-.015	-.011	+.010	+.015	+.015	-.005	.000	-.004	+.009
12-13		+.001	+.001	+.006	-.001	-.005	-.005	.000	.000	+.013	-.011
13-14		+.015	+.017	+.007	*	*	*	*	*	*	*
14-1		+.011	+.018	+.014	*	*	*	*	*	*	*

* Burr obstructed movement of optical target.

Table VI-13 SUMMARY OF RADIAL STEAM TUBE DATA
VARIATION FROM DESIGN POSITION

Element Number	Total Number of Recorded Measurements	Recorded Number of Readings $> + 0.020$ in.		Recorded Number of Readings $> \pm 0.030$ in.	
		Number	Percent	Number	Percent
10-1-11	110	23	23	9	9
10-1A-14	154	8	5	0	0
10-1A-14	154	1	< 1	0	0
Post Impaction					
10-1-17	187	57	30	26	14
10-2-11	109	14	13	1	< 1
10-2-14	154	29	19	5	3
10-3-11	121	0	0	0	0
10-3-14	147	11	8	0	0

Table VI-14 SUMMARY OF AZIMUTHAL STEAM TUBE DATA
VARIATION FROM DESIGN POSITION

<u>Element Number</u>	<u>Total Number of Recorded Measurements</u>	<u>Recorded Number of Readings $> \pm 0.020$ in.</u>		<u>Recorded Number of Readings $> \pm 0.030$ in.</u>	
		<u>Number</u>	<u>Percent</u>	<u>Number</u>	<u>Percent</u>
10-1-11	100	31	31	16	16
10-1A-14	140	9	6	3	2
10-1A-14 Post Impaction	140	11	8	3	2
10-1-17	168	40	24	10	6
10-2-11	84	9	11	3	4
10-2-14	140	0	0	0	0
10-3-11	99	8	8	0	0
10-3-14	126	12	10	2	2

outer water clad tube was then inserted over the plug seal assembly and the retainer spacer discs drawn together to a 1/2 inch steel wool thickness by three small through bolts. Spacer bushings 1/2 inch long inserted between the retainer discs were used to prevent further compression of the steel wool during compaction. A pointed steel adapter was then mounted on the steam tubes to pierce holes through the steel wool matting and insert the steam tubes. The experimental data obtained for both washers are shown in Table VI-15.

Satisfactory performance of stainless steel wool matting as a seal has been demonstrated, provided that the assembly of the fuel element during fabrication is such that the initial fuel sieved through the washer during loading and compaction can be removed.

G. Fuel Element Clad Procurement

Inconel 625 has been selected as the boiler and steam tube clad material for the solid multitube element. The clad dimensions and a clad materials specification have been established. Table VI-16 lists the dimensions, specification and vendor who is to produce the tubing.

Table VI-15 STAINLESS STEEL WOOL PLUG SEAL
EXPERIMENTAL DATA

<u>Description</u>	<u>Plug seal #1</u>	<u>Plug seal #2</u>
Density of steel wool packing	13 w/o*	18 w/o
Grams of UO_2 powder seived through the seal during loading and compaction	70	39
Grams of UO_2 powder sieved through the seal during one hour of vibration of the element on MB machine after completion of compaction	2	<.1
Grams of UO_2 retained in the seal	12.7	13.8

* Based on a theoretical density of steel at 450 lbs/ft³.

Table VI-16 DIMENSIONS AND SPECIFICATIONS FOR CLAD MATERIALS

<u>Clad surface</u>	<u>Dimensions</u>	<u>Specification</u>	<u>Vendor</u>
1) Boiler clad (Inconel 625)	1.447/1.452 inches ID 0.033 $\pm$ 0.003 inch wall	ABC-90	Pacific Tube
2) Steam tubes (Inconel 625)	0.283 $\pm$ 0.001 inches OD 0.026 $\pm$ 5% inches wall	ABC-90	Superior Tube
3) Steam tube extension (Inconel 625)	0.285 $\pm$ 0.001 inches ID 0.026 $\pm$ 5% inches wall	ABC-90	Superior Tube
4) Steam tube to second pass header (Inconel 600)	0.283 $\pm$ 0.001 inches ID 0.026 $\pm$ 5% inches wall	ASTM-167	Superior Tube
5) Steam tube bushings (Inconel 600)	0.375 $\pm$ 0.001 inches OD 0.030 $\pm$ 5% inches wall	ASTM-167	Superior Tube

Fig. VI-1 Large tubular tandem extruded Zircaloy to Inconel transition joint as fabricated by Metalonics.

Fig. VI-2 Section of a 3.316-in.-ID Zircaloy-Inconel tandem extruded transition joint produced by Nuclear Metals.

Fig. VI-3 The first lot of perforated UO_2 pellets assembled on a simulated steam tube frame of 2 large diameter tubes.

Fig. VI-4 Composite weld specimen for tensile and fatigue tests of superheated steam tube.

NOTE: ALL FITS ARE TO BE ROLLED
PRIOR TO WELDING.

ALL MATERIALS INCONEL 625

Fig. VI-5 Fatigue curve for welded Inconel 625 superheated steam tubes.

Fig. VI-6 Weld specimens for Inconel 625 boiler clad to Inconel 625 tube-sheets.

Fig. VI-7 14-tube annular fuel element cross section.

Fig. VI-8 ABC multitube fuel loading and densification assembly.

Fig. VI-9 ABC multitube fuel loading and densification assembly (perspective cut-away).

SPACER WASHER

Fig. VI-10 Spacer washer configurations.

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