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DESIGN AND FABRICATION OF MARK IV SUPERHEAT FUEL

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ADVANCED PRODUCTS OPERATION

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H.C. \$ 3.00; MN .65DESIGN AND FABRICATION OF MARK IV SUPERHEAT FUEL
FOR EVESR FUEL DEVELOPMENT PROGRAM

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ABSTRACT

The design and fabrication of the ESADA-VESR* (EVESR) Mark IV fuel and fuel bundles are described herein. Mark IV fuel was designed to demonstrate the performance of superheat elements operating at high heat flux, high specific power, and high burnup typical of conditions desired in advanced steam-cooled reactors. One fuel element features turbulence promoters for improved heat transfer and another was spaced with projection spot-welded wires. Pellets and powder filled fuel rods were used. Alloys selected for fuel cladding were Inconel-625, Incoloy-800, and Hastelloy-X.

*Empire State Atomic Development Associates - Vallecitos Experimental Superheat Reactor, located at the Vallecitos Nuclear Center near Pleasanton, California.

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SECTION I

INTRODUCTION

The Mark IV superheat fuel elements were designed and fabricated to demonstrate high specific power fuel performance found in steam-cooled reactors. Maximum power densities and accumulated exposures anticipated for Mark IV fuel are approximately 150 kW/kg of UO_2 , and 40,000 MWd/T U. Previous fuel element designs were based on 50 kW/kg maximum power densities and 15,000 MWd/T U terminal accumulated exposures. The higher power densities are required in advanced thermal spectrum, mixed spectrum, or fast spectrum reactor concepts, where maximum gains can be made in the improvement of the fuel cycle economics and/or increase in thermodynamic efficiency.

The main design and fabrication features for the Mark IV fuel rods are (a) three types of cladding materials (Incoloy-800, Inconel-625; Hastelloy-X), (b) the use of UO_2 powder compaction and pellets to achieve a range of densities to study swelling, and (c) the incorporation of alternate spacer designs and turbulence promoters (for increased surface-coolant heat transfer coefficient with a concurrent lowering of bulk cladding temperature). The reasons for selecting the cladding materials, and the details of the process used to incorporate powder in the fuel rods are discussed in later sections of this report.

There are four complete bundles in the Mark IV experiment. Each Mark IV fuel bundle consists of 28 boiling water fuel rods and 4 high performance superheat fuel rods. The arrangement of the fuel rods is shown in Figure 1-1. The boiling water fuels are used only to increase neutron flux and maintain sufficient bundle reactivity. It is the purpose of this report to present the details of the materials selection, rod design, and fabrication of the Mark IV EVESR fuel experiment.

FIGURE 1-1. ARRANGEMENT OF FUEL RODS IN MARK IV ASSEMBLIES

SECTION II

SUMMARY

The experimental EVESR fuel designated as Mark IV is described herein. This fuel was designed and fabricated as a part of the EVESR advanced superheat fuel development program to investigate high power density fuel performance in superheat environments. The design and fabrication techniques used are expected to be applicable to fuel for thermal as well as mixed spectrum or fast spectrum steam-cooled reactors.

Selection of cladding materials was made on the basis of information available at present, taking into account stresses at near-end of fuel life, localized and general corrosion effects, and mechanical stability requirements for the fuel elements.

The alloys selected for fuel cladding were Incoloy-800, Inconel-625, and Hastelloy-X. Both pellets and vibratory compacted UO_2 powder were used in this experiment to determine relative fuel performance as a function of UO_2 swelling at the high accumulated exposures (40,000 MWd/T) planned to be attained with this fuel. Out of a total of 16 superheat rods, 11 contained pellets and 5 contained powder, with a density range 81% to 89% of UO_2 theoretical. Vibratory and swaging techniques were used to obtain the above densities. Two swaged rods were annealed at 1950°F to restore the cladding microstructure to the stress-free, fine-grain condition.

Two other areas are of importance in the Mark IV fuel design. These are (a) turbulence promoters to improve heat transfer coefficient at the steam-cladding surface interface and, thereby, decrease bulk cladding temperature, and (b) an alternate fuel rod spacing scheme.

Based upon data developed in Great Britain, at General Electric, at ORNL, and at Aerojet General Nucleonics, turbulence promoters were machined on the outside fuel cladding. The main objectives were to study the rate of impurity accumulation, corrosion, and erosion of the fins machined on the fuel cladding surface. The effectiveness of the fins in improving heat transfer depends upon maintaining their geometry throughout the desired reactor service. Turbulence promoters were fabricated in one rod with fins 2 mils high by 2 mils wide, spaced 40 mils apart, and formed by grinding the cladding surface. If tests prove successful, a more practical method must be developed for incorporating the fins on tubes to replace the cumbersome and expensive machining method presently available.

Although helical wires attached to end plugs have worked well in EVESR rod spacing designs, it was desired to test an alternate method of spacing. Therefore, one rod was fabricated with helically wrapped wire spacers which were projection spot-welded to the cladding surface at 5/8-inch intervals along the fuel rod. Although changes from the present reference spacing method (wires attached on end plugs only) will tend to increase fuel fabrication costs, it is possible to offset the increased costs by adaptation of automatic welding techniques.

SECTION IIIALLOY SELECTION FOR SUPERHEAT CLADDING MATERIAL

3.1 GENERAL

The goal in the selection of cladding for Mark IV fuel was to choose those alloys which appeared to have the greatest potential for successful service under the required conditions. The fuel cycle costs in a large power producing reactor are dictated by many factors, among which the most important general considerations are low fabrication costs and high reliability in performance. Fuel element cladding must be selected on the basis of its availability, cost, ease of fabrication, corrosion resistance (general and localized corrosion), and its ability to remain mechanically sound after 3- to 5-year reactor exposures at temperatures of 1100° to 1350°F. Neutron absorption cross section must be taken into account when fuel elements are destined for use in thermal spectrum reactors. For mixed spectrum (superheat section fast) or fast spectrum cores, the parasitic neutron absorption considerations are less important.

Past experience with nuclear superheated steam environments showed that localized corrosion attack, found to be assisted by stresses, ruled out the use of 300 series austenitic steels. ^{(1), (17)} If neutron cross sections are not a prime factor in high power density fuel, (especially in fast neutron spectrum cores) intermediate and high nickel content alloys are acceptable candidates. Literature surveys and work at GE showed that high nickel alloys are also less susceptible to localized intergranular corrosion attack, and in one particular test, Inconel-625 showed the best performance.

The four alloys selected as possible candidates are: Incoloy-800, Incoloy-801, Inconel-625, and Hastelloy-X. * Compositional requirements for these alloys are given in Table 3-1.

TABLE 3-1
ALLOY COMPOSITION, %

<u>Element</u>	<u>Incoloy-800</u>	<u>Incoloy-801</u>	<u>Inconel-625</u> (Nominal)	<u>Hastelloy-X</u>
Fe	Balance	Balance	3.0	17.00-20.00
Ni	30.00-35.00	30.00-34.00	61.0	Balance
Cr	19.00-23.00	19.00-22.00	22.0	20.50-23.00
C	0.10 max	0.10 max	0.05	0.05- 0.15
Mn	1.50 max	1.50 max	0.15	
S	0.03 max	0.03 max		
Si	1.00 max	1.00 max	0.30	1.00 max
Cu	0.75 max	0.50 max		
Co		1.0 max		0.50- 2.50
Mo			9.0	8.00-10.00
W				0.20- 1.00
Cb(Ta)			4.0	
Ti		0.75- 1.50		
Neutron Absorption Cross Section (cm ⁻¹)	0.292	0.292	0.343	0.307

*Incoloy and Inconel are trademarks of the Huntington Alloy Products Division, the International Nickel Co., Inc., Huntington, West Virginia. Hastelloy is a trademark of the Stellite Division, Union Carbide Corp., Kokomo, Indiana.

Examination of the available data indicates that although there is a definite lack of information applicable to superheat environments, especially with regard to post-irradiation properties, the four alloys selected have good potential as successful cladding materials. The specific reasons for considering each of the alloys shown in Table 3-1 are as follows:

3.1.1 Incoloy-800

This alloy has the greatest available information in regard to its performance in superheat environments, including fuel element irradiation. ⁽¹⁾⁽²⁾⁽³⁾⁽¹⁷⁾ All information indicated its performance to be acceptable so far for fuel cladding applications in either thermal or fast spectrum cores. In comparative tests with Type 304 stainless steel, Incoloy-800 exhibited excellent performance by factors of at least 10 (length of exposure without failure) over Type 304 stainless steel with no failure in service. ⁽⁴⁾ Mechanical properties after reactor exposure are acceptable with a minimum ductility of about 7% uniform elongation at 1300°F after 5000 hours of exposure at 1300°F with an accumulated neutron exposure of 2.0×10^{21} nvt ($E_n \geq 1$ MeV). ⁽⁵⁾ Aging studies to 10,000 hours at temperatures of 1100° to 1300°F show minimum ductilities of 15% to 20% uniform elongation, and other evidence of thermal stability based upon careful metallographic and electron microscope examination. ⁽⁶⁾ General corrosion rates were less than 2.2 mils of metal loss for a three-year period at 1350°F. ⁽⁷⁾ Cyclic imposition of stress at 600° to 1300°F in high velocity steam with chloride impurities present showed no intergranular attack even after 70 days when the test was terminated. ⁽⁸⁾ Types 304, 316, or 348 stainless steel failed after only 2 to 10 days in the cycling tests. ⁽⁸⁾

In comparative fuel cladding tests, Incoloy-800 performed satisfactorily for 135 days in an E-SADE fuel bundle without failure, whereas Type 304 failed in 29 days. Several intact fuel elements were transferred to an EVESR test assembly and were irradiated without failure for an additional 167 days. ⁽⁴⁾

3.1.2 Incoloy-801

This alloy is compositionally identical to Incoloy-800, but has a titanium content up to 1.5%. It was selected because of its improved mechanical properties over those of Incoloy-800, thus giving the fuel designer a 30° to 50°F advantage in stress to rupture without increase in neutron capture cross section. ⁽⁹⁾ This alloy was dropped from consideration, however, when irradiation exposure was found to reduce its ductility to less than 1% when tested at 1300°F after 5000 hours of exposure at 1300°F and neutron accumulated exposures of 2.0×10^{21} nvt ($E_n \geq 1$ MeV). ⁽⁵⁾

3.1.3 Hastelloy-X

A considerable amount of information is available for Hastelloy-X, including post-irradiation data, from sources other than General Electric. ⁽¹⁰⁾⁽¹¹⁾ However, only limited data are available under superheat conditions. Its strength and corrosion characteristics, along with the relaxed nuclear cross-sectional requirement for mixed spectrum service, make it a good candidate.

In long-term aging tests for 5000 hours at 1300°F, the minimum ductility found was 15% when tested at room temperature. Testing-aged Hastelloy-X at 1100° to 1300°F resulted in ductility values of 20% uniform elongation. Irradiation (2×10^{21} nvt $E_n \geq 1$ MeV) at 1300°F for 5000 hours and tensile testing at 1100°F resulted in ductility values of 4 percent (uniform elongation). ⁽¹³⁾ Although the terminal ductility could be lower after higher neutron exposures necessary in fast spectrum reactor environments, it is possible to offset further

drop through alloy improvements by removal of residual boron, grain refinement, and grain boundary strengthening. Its general corrosion resistance is at least as high as that for Incoloy-800, and although no definitive comparison has been made, its resistance to stress corrosion should be better than Incoloy-800 because of its higher nickel content.

3.1.4 Inconel-625

On the basis of preliminary strength and corrosion data and because of the reduced nuclear penalty resulting from cross section at higher neutron energies, this alloy appears to be very well suited for superheat applications especially in fast spectrum reactors.⁽¹⁴⁾ Of the four alloys selected, there are less data available for Inconel-625 than for the others. Irradiation test data are now being compiled for this alloy. This alloy has been included in the program primarily because of its resistance to rapid, localized intergranular attack. In an accelerated screening test, Inconel-625 performed best.⁽¹⁵⁾ Because of the rapid, stress assisted corrosion exhibited by Type 304 stainless steel in superheat reactor environments, it is highly desirable to retain an alloy in the program as a backup which displays a high resistance to localized corrosion. Aging and irradiation at 1300° F for 5000 hours for a total of 2×10^{21} nvt ($E_n \geq 1$ MeV) reduced its 1300° F ductility to 1%.⁽¹²⁾⁽¹³⁾ At 1100° F its ductility was about equal to that of Hastelloy-X. Alloying and metallurgical procedures can be used to improve its ductility. Its general corrosion properties are similar to or better than those exhibited by Incoloy-800 and Hastelloy-X.

3.2 MECHANICAL PROPERTIES

Mechanical property data applicable to fuel design have been collected for the four candidate alloys and have been plotted to allow comparisons of the alloys with each other.

Short time tensile data, including yield and ultimate strength and elongation values, are given in Figures 3-1, 3-2, and 3-3. Creep and stress-rupture data for the temperature range of interest are plotted in Figures 3-4 through 3-10. Figures 3-11 and 3-12 outline the available data on radiation effects⁽¹³⁾ and Figures 3-13, 3-14, and 3-15 indicate thermal effects on mechanical properties.

3.3 CORROSION PROPERTIES

Data concerning corrosion characteristics are divided into localized and general corrosion resistance. Results of bench tests for localized corrosion are given in Table 3-2.⁽⁸⁾⁽¹⁵⁾ During these tests, sensitized specimens were exposed to atmospheric boiling CuCl_2 (20,000 ppm Cl^-) for one hour and the mode of attack recorded.

TABLE 3-2
MODE OF ATTACK IN BOILING CuCl_2

<u>Alloy</u>	<u>Type of Attack</u>
Type 304 Stainless Steel	Intergranular
Incoloy-800	Intergranular
Incoloy-801	Not Tested
Hastelloy-X	Intergranular
Inconel-625	None

In addition, Inconel-625 was tested in boiling CuCl_2 with 40,000 ppm Cl^- and again no attack was noted.⁽¹⁵⁾ (Traces of CuCl_2 were found upon analytical examination of the fuel surface scale deposits scraped from failed VBWR-SADE loop fuel elements.)

A brief summary of general corrosion data is given in Figure 3-16.⁽¹⁶⁾ These data are based on specimens exposed in a simulated superheat environment (without irradiation), at 1150°F.

3.4 NEUTRON CROSS SECTION

Table 3-1 lists the nuclear cross sections for the candidate alloys. Although no values are given for Incoloy-801 and Inconel-625, they should be similar to Incoloy-800 and Inconel-600, respectively.

3.5 FABRICATION ABILITY

The fabrication ability of the candidate alloys should be considered from two aspects: tubing manufacture and fuel rod assembly. To date, a considerable amount of small diameter, thin-walled tubing has been produced of both Incoloy-800 and Hastelloy-X on a production basis. Tubing of Incoloy-801 and Inconel-625 has been produced on a more limited basis, but there appears to be no major tubing manufacturing difficulties inherent in their production.

A considerable amount of experience has been acquired at GE on the manufacture of fuel rods clad in Incoloy-800. Although experience with the other three alloys is more limited, preliminary information indicates no major areas of difficulty in fuel rod fabrication.

FIGURE 3-1. YIELD STRENGTH, 0.2 PERCENT

FIGURE 3-2. TENSILE STRENGTH

FIGURE 3-3. PERCENT ELONGATION IN 2 INCHES

FIGURE 3-4. SECONDARY CREEP RATE OF SELECTED SUPERHEAT ALLOYS - 1200°F

FIGURE 3-5. SECONDARY CREEP RATE OF SELECTED SUPERHEAT ALLOYS - 1300°F

FIGURE 3-6. STRESS-TEMPERATURE RELATIONSHIP FOR 1 PERCENT CREEP / 1000 HOURS

FIGURE 3-7. STRESS - RUPTURE DATA AT 1200°F

FIGURE 3-8. STRESS-RUPTURE DATA AT 1300°F

**FIGURE 3-9. STRESS-TEMPERATURE RELATIONSHIP
FOR RUPTURE IN 1000 HOURS**

FIGURE 3-10. STRESS-TEMPERATURE RELATIONSHIP FOR RUPTURE IN 10,000 HOURS

FIGURE 3-11. EFFECT OF NEUTRON RADIATION ON 1100°F TENSILE PROPERTIES

FIGURE 3-12. EFFECT OF NEUTRON RADIATION ON 1100°F TENSILE PROPERTIES

FIGURE 3-13. ROOM TEMPERATURE TENSILE STRENGTH AFTER EXPOSURE TO 1050°F AND 1150°F SUPERHEATED STEAM

FIGURE 3-14. ROOM TEMPERATURE ELONGATION AFTER EXPOSURE TO 1050°F AND 1150°F SUPERHEATED STEAM

FIGURE 3-15. ROOM TEMPERATURE YIELD STRENGTHS AFTER EXPOSURE TO 1050°F AND 1150°F SUPERHEATED STEAM

FIGURE 3-16. CORROSION DATA AT 1150°F

SECTION IV

FUEL DESIGN

4.1 FUEL DESIGN CONDITIONS

Superheat fuel calculations were based on (1) a maximum power of 150 kW/kg of UO_2 , (2) 1250° F maximum cladding temperature, and (3) the heat flux distribution of Figure 4-1 (1.17 axial peaking factor).

Boiler fuel design was based on a maximum power of 33 kW/kg and was adapted from current boiling water technology wherever possible.

4.2 FUEL DESCRIPTION

4.2.1 Boiling Water Fuel Rods

The EVESR Mark IV fuel consists of four bundles, each containing 28 boiling water fuel rods and four high performance superheat fuel rods. The purpose of the boiling water fuel rods is to increase neutron flux at the superheat rods and to maintain bundle reactivity at a level consistent with the overall core physics. The boiling water rods were made in two segments to eliminate spacers and decrease pressure drop for improved burnout margin. The boiler fuel is clad with 0.485-inch inside diameter $\times$ 0.036-inch wall Zircaloy-2, and contains sintered pellets of 5% enriched UO_2 . The boiler rod has an active fuel length (segmented) of 60 inches and a gas plenum 3/4-inch long.

4.2.2 Superheat Fuel Rods

The superheat rods have 0.280-inch outside diameter $\times$ 0.020-inch cladding. Fuel is both pellet and compacted powder UO_2 with enrichments of 25% and 50%. Active fuel length is 40 inches with a 1/2-inch natural UO_2 buffer at each end, and a 14-1/2-inch plenum at the lower end. Having the plenum at the lower end reduces pressure drop for Mark IV. One fuel rod has a helical wire spacer spot welded in place. Another rod has turbulence promoters on the cladding. Table 4-1 is a summary of the superheat cladding material and fuel type. Figures 4-2 and 4-3 are the drawings of the superheat and boiler rods, respectively.

Each superheat rod is enclosed in a velocity tube made of the same material as its cladding. The 14-1/2-inch plenum includes a spring, a connector, and an 11-inch empty plenum. The spring is to support the fuel column while allowing for differential expansion between the cladding and fuel.

4.3 HIGH PERFORMANCE SUPERHEAT FUEL DESIGN

4.3.1 Powder versus Pellets

Uranium dioxide powder is used in some of the Mark IV superheat rods because of potential fabrication cost advantage and, perhaps, performance advantage over pellet fuel. No comment will be made here regarding relative costs. Arguments for powder fuel are as follows:

- a. Powder fuel can be loaded to low (80% to 85% of theoretical) densities to allow space for swelling caused by fuel burnup, where the desired terminal fuel exposures are higher than 30,000 MWd/T.

FIGURE 4-1. EVESR MARK IV POWER DISTRIBUTION

TABLE 4-1

MARK IV HIGH PERFORMANCE FUEL DESIGN VARIABLES

<u>Assembly</u>	<u>Rod Number</u>	<u>% U-235</u>	<u>Peak Q/A (Btu/h-ft²)</u>	<u>Cladding Material</u>	<u>Fuel</u>	<u>Nominal Diametral Gap (mils)</u>	<u>Fuel Density (% TD)</u>	<u>Fabrication Method</u>
KB-42	01L25	25	600,000	Incoloy-800	Pellet	3	93.5	Standard
	02L50	50	480,000	Incoloy-800	Pellet	5	93.5	Standard
	03L50	50	440,000	Incoloy-800	Pellet	2	93.5	Standard
	10L50*	50	250,000	Incoloy-800	Powder	-	93.5	Standard
KB-43	04L50	50	440,000	Incoloy-800	Powder	-	83-85	Vi-Pak
	05L50	50	250,000	Incoloy-800	Powder	-	80-83	Vi-Pak
	06L50	50	480,000	Incoloy-800	Powder	-	86-88	Swaged
	09L25	25	600,000	Incoloy-800	Powder	-	83-85	Vi-Pak
KB-44	21V25	25	600,000	Inconel-625	Pellet	3	93.5	Standard
	22V50	50	480,000	Inconel-625	Pellet	5	93.5	Standard
	23V50	50	440,000	Inconel-625	Pellet	2	93.5	Standard
	24V50	50	250,000	Inconel-625	Pellet	5	93.5	Standard
KB-45	41X50	50	440,000	Hastelloy-X	Pellet	2	93.5	Standard
	42X25	25	600,000	Hastelloy-X	Pellet	3	93.5	Standard
	43X50	50	480,000	Hastelloy-X	Pellet	5	93.5	Standard
	25V50**	50	250,000	Incoloy-800	Pellet	5	93.5	Standard
Spare Rods	07L50	50	-	Incoloy-800	Powder	-	86-88	Swaged
	08L25	25	-	Incoloy-800	Powder	-	80-83	Vi-Pak

*Projection spot-welded wire wrap spacer on this element.

**Turbulence promoters used on this element — 2-mil height with 40-mil spacing.

FIGURE 4-2. SUPERHEAT FUEL DESIGN

FIGURE 4-3. BOILING WATER FUEL DESIGN

- b. Powder fuel does not impose as severe axial bending strains in the cladding as pellets do (circumferential ridges). Partial experimental support exists for this argument.⁽¹⁸⁾

Counterarguments:

- a. If a large sized defect occurs, low density powder may wash out in service. Going to higher density to relieve this problem reduces swelling space allowance.
- b. A further possible problem regarding powder, especially low density, is collapse behavior. Although isothermal tests⁽¹⁸⁾ indicate better creep collapse characteristics for powder filled rod than for pellets, it can be argued that the heat generation, cyclic conditions and temperatures of a superheat reactor environment may cause the powder particles to shift and permit excessive ovality to occur.

One of the purposes of the Mark IV test is to determine the comparative irradiation performance of powder-filled and pellet-filled rods under similar conditions.

4.3.2 Irradiation Effects

Cladding embrittlement and UO_2 swelling at long irradiation exposures are expected to be factors limiting the useful life of high performance fuel such as Mark IV. As the UO_2 swells because of burnup effects, it will push out and fracture the cladding if the strength and ductility of the latter are insufficient to resist or accommodate the expansion. Present information is insufficient to predict accurately at what burnup such fracture might take place. The Mark IV irradiation should help to resolve this question.

4.3.3 Spacers

Each fuel rod is spaced from its liner by a 0.062-inch wire helically wrapped around the cladding with a pitch of 5-1/4 inches. The wire ends, bent to 90°, are threaded through holes in the end plugs. The end of the wire is then melted into a ball of sufficient diameter to prevent removal. This arrangement puts the wire end weld in a low stress region. The span length was dictated by helical deformation of the fuel rods caused by the differential expansion stresses imposed on the fuel by the spacer.

The following assumptions were made in evaluating spacer distance required and the bending associated with the above stresses.

- a. The cladding behaves as though each support point is at a hinged joint.
- b. Maximum allowable deflection of cladding centerline is 25% of wire diameter.
- c. Compressive bending stresses in the cladding are completely relaxed at the line under the wire.

4.3.4 Spot-Welded Wire Spacer

The welded wire fuel rod is identical to the standard Mark IV geometry except that the wire spacer is projection spot-welded to the cladding at 5/8-inch intervals. The idea is that should the wire break, the spot-welds would hold it in place, whereas with the wire attached only at the end plugs, the wires would shift with potentially bad effects. A potential problem with the spot-welds is that stress concentrations are created on the cladding which might lead to early failure. As with the other parameters the actual irradiation is required to demonstrate performance.

4.3.5 Turbulence Promoter Cladding

The turbulence promoter rod has rectangular protrusions in a helical pattern on the cladding as shown in Figure 4-4. These are sized and spaced to trip the flow boundary layer for improved heat transfer and resultant improved thermal performance. In-reactor data will show whether deposit buildup on the cladding or general corrosion will smooth out the flow and defeat the purpose of the turbulence promoters.

4.3.6 Fission Gas Plenum Pressure

Space is provided at the bottom (low temperature region) of each fuel rod to accommodate fission gases. Fission gas release percentages were calculated using the following assumptions:

- a. All UO_2 above 3000°F releases 100% of the fission gas formed.
- b. All UO_2 below 3000°F releases 4% of the fission gas formed.
- c. Cladding surface temperature is assumed to be a constant 1250°F . This results in slightly high calculated release rates.
- d. Cladding-to-fuel gap conductance is $1000 \text{ Btu/h-ft}^2\text{-}^\circ\text{F}$.
- e. UO_2 conductivity of Lyons, et al.,⁽¹⁹⁾ applies for pellet fuel; powder conductivity is 85% of pellet conductivity.
- f. Maximum volumetric heat generation is $150 \text{ kW/kg (UO}_2\text{)}$.
- g. Axial power peak is 1.170 with heat flux of Figure 4-1.
- h. Volumetric heat generation within a fuel rod is a function of axial position only.

The $\int kdt$ method⁽¹⁹⁾⁽²⁰⁾ with constant volumetric heat generation was used in determining the volume of fuel above 3000°F for various maximum surface heat fluxes.

It is further assumed, in the case of an overpower transient, that operation has been at normal power for the entire Mark IV irradiation, and power suddenly increases 25% with an instantaneous release of fission gas. This instantaneous release is assumed to bring the total gas release to the value for steady operation at 25% overpower. The above assumptions result in the values of Table 4-2.

TABLE 4-2

CALCULATED FISSION GAS RELEASE

<u>Fuel Type</u>	<u>Power Level (%)</u>	<u>Fission Gas Release (%)</u>
Powder	100	57
Powder	125	71
Pellet	100	54
Pellet	125	68

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With release rates established, it is now necessary to determine the pressure caused by fission gas and residual gases. The following assumptions were made:

1. Average burnup is 40,000 MWd/T U.
2. Total fission gas formed is 1.34×10^{-3} gm-mole/MWd.
3. UO_2 density is 95% of theoretical in pellet and 91% of theoretical in powder.
4. Plenum gas temperature is 600°F.
5. The fuel contains 1.5×10^{-5} gm of H_2O and 40 μl of residual gases per gm of UO_2 at assembly.
6. The ideal gas law applies.

These assumptions lead to the following equations:

$$P = 30 + \frac{1}{V_p} (32.6 + 644F) \quad (\text{Powder Fuel})$$

$$P = 30 + \frac{1}{V_p} (33.9 + 676F) \quad (\text{Pellet Fuel})$$

where

P = plenum pressure, psia;

F = fraction of fission gas released;

V_p = plenum volume, in.³;

30 = pressure caused by helium back-fill, psi.

$\frac{32.6}{V_p}$ and $\frac{33.9}{V_p}$ = pressure caused by residual gas and moisture.

These equations are illustrated in Figures 4-5 and 4-6.

4.3.7 Stress Calculations

4.3.7.1 Internal Pressure Stresses

In calculating the internal pressure no credit was taken for fuel voids. The stress was then calculated using the standard hoop stress equation $\left(\sigma = \frac{PD}{2t}\right)$. The values of stress are tabulated in Table 4-3.

FIGURE 4-5. FISSION GAS PRESSURE-POWDER FUEL

FIGURE 4-6 FISSION GAS PRESSURE-PELLET FUEL

TABLE 4-3

CALCULATED INTERNAL PRESSURE STRESSES

<u>Type of Fuel</u>	<u>Power Level (%)</u>	<u>Stress*</u>
Powder	100	4850
Powder	125	5950
Pellet	100	4850
Pellet	125	6000

(For 600°F average gas temperature.)

*Assuming external pressure equals one atmosphere.

A plenum temperature of 600°F was used since calculations show that even for a no-flow shutdown condition of 1500°F maximum cladding temperature, the plenum wall average temperature is less than 600°F. Since the pellet fuel has the higher stress, it was used for calculating the time to rupture. The results are illustrated in Figure 4-7.

Rupture data were taken from References (2), (10), and (14).

4.3.7.2 Miscellaneous Stresses

It can be postulated that if a Mark IV bundle receives a shock loading, the plenum spring might be compressed by the fuel mass, and if UO_2 particles wedge certain pellets, then a gap between pellets could occur.

Stresses caused by a longitudinal temperature gradient induced by such a fuel void at the maximum heat flux region were calculated. The void was assumed to be 0.7 inch, but the results are applicable to any void of 1/2 inch or more. The stress for Incoloy was found to be 9700 psi which is less than the gradient stress below. There is no problem of pressure induced cladding collapse into the void because the theoretical elastic collapse pressure is more than 15 times the 1000 psi external pressure.

Stresses for a radial temperature gradient were calculated based on a 614,000 Btu/h-ft² heat flux at cladding outside diameter. These stresses are tabulated in Table 4-4.

TABLE 4-4

RADIAL GRADIENT THERMAL STRESSES

<u>Cladding</u>	<u>$\Delta T_{\text{cladding}}$</u>	<u>Stress (psi)</u>
Incoloy-800	92°F	15,800
Inconel-625	63°F	16,000
Hastelloy-X	81°F	11,900

FIGURE 4-7. FUEL CLADDING RUPTURE LIFE

Stress caused by flow induced fuel vibration was calculated to be 1000 psi at 250 ft/sec steam velocity using Burgreen's equations as modified by Shields.⁽²²⁾ These equations are for water flow. However, according to Quinn⁽²³⁾ natural frequency and fluid velocity are the dominant factors in parallel-flow vibration. Therefore, the results should be applicable. Axial stresses caused by differential thermal expansion of cladding-and-fuel and fuel-swelling should be relieved by the spring in the plenum until the cladding creeps into contact with the fuel and friction prevents free axial motion. Without the spring, these stresses were greater than 27,000 psi.

Strain cycling was investigated and should not be significant unless the operating cycle is between high-power and low-power where the cladding temperature is continuously in the 1400°F region with long cycle times so that creep and relaxation of the cladding can occur at all parts of the cycle.

4.4 BOILING WATER FUEL DESIGN

Most of the boiling water fuel design was adapted from prior boiling water fuel designs. The only new work was to determine plenum volume requirements and conditions affecting the 31-inch spacing of the segmented design.

Plenum requirements were determined on the same basis as for the superheat fuel except for the lower cladding temperature (assumed 650°F) and the assumption that the maximum power density of 33 kW/kg existed over the entire fuel length. Calculated fission gas release rate for 100% power was 4.2% and for 125%, 11%. Assuming all gases are contained in the free volume of the 3/4-inch-long plenum and assuming only atmospheric pressure acts on the fuel exterior, and hoop stresses caused by internal pressure are 12,800 psi at 100% power and 16,800 psi at 125% power.* These stresses are less than half of the cladding 650°F yield strength.

Horizontal deflection of the fuel segments was the remaining item considered in the boiler fuel design. Possible sources of deflection include: (1) vibration and (2) bowing. Deflection caused by bowing is caused by cladding wall eccentricity and temperature variation around the circumference.

The upper fuel segment was made approximately 1 inch shorter than the lower segment to give it a higher natural frequency than the lower segment. In addition, the higher quantity of coolant voids at the top of the core makes the effective mass increase of the upper segment caused by the coolant less than for the lower segment, resulting in a further frequency difference for the two segments. The coupling between segments will not be at the natural frequency of either.

Maximum calculated amplitude of horizontal vibration was less than 0.01 inch at a maximum coolant velocity of 9 feet per second.** Corresponding maximum bending stress is less than 400 psi.

If the cladding inner diameter is eccentric from the outer diameter 0.003-inch (maximum tolerance), the axial pressure does not act on the centroid of the tube cross section and a cladding bow of 0.011 inch results.

*The stresses are not in direct proportion to the release rates since much of the internal pressure is due to the initial gas and water in the fuel.

**According to a personal communication with C. M. Shields, December 11, 1964, this velocity is probably slightly high.

Temperature variation around the cladding outer circumference will be very small because of the high heat transfer coefficient to the boiling water. However, circumferential heat flux skewing will result in a corresponding variation in cladding average temperature because of the temperature gradient through the cladding. The cladding average temperature can be expected to differ approximately 25°F at diametric points. The resultant differential expansion causes a bow of about 0.006 inch. Combining the various deflections in a linear manner results in a total deflection of 0.026 inch which is less than 20% of the rod-to-rod spacing. This deflection should not interfere with fuel performance.

SECTION V

SUPERHEAT FUEL FABRICATION

5.1 GENERAL

The Mark IV superheat fuel rod fabrication consisted of the standard pellet, vibratory compacted, and swaged over powder fuel types of processes. In addition, one rod was fabricated with turbulence promoters (fins) in the cladding and one rod with wire spacers projection spot-welded onto the cladding. The characteristics of the fuel rods as-fabricated are given in Tables 5-1 and 5-2.

The drawings for the fabrication of these rods are listed below:

<u>Title</u>	<u>Drawing Number</u>
Fuel rod (pellet)	112C2013
Fuel rod (powder)	112C2016
Fuel rod (turbulence promoter)	112C2834
Fuel rod (spot-welded wire spacer)	112C2831
Upper cap	159A2708
Upper end plug	159A2709
Lower end plug	159A2710
Connector	159A2711
Spring	159A2739
Fuel pellet	159A2741

5.2 URANIUM DIOXIDE FUEL

The UO_2 powder was prepared at the department by blending 65% enriched and 10% enriched U-235 ceramic-grade powder in the proper proportions to obtain 25% and 50% enriched fuel material. The analyses of these blends are as follows:

<u>Batch</u>	<u>% Nominal Enrichment</u>	<u>Isotopic Content %</u>
-	25	25.40
1	50	49.45
2	50	50.01
3	50	50.90
4	50	49.90

Batches 1, 2, and 3 of the 50% enrichment and 1 kg of the 25% enriched UO_2 were pressed, sintered, crushed, and shipped to Battelle Memorial Institute, Pacific Northwest Laboratories, for "dynapaking" (pneumatic impaction) into a compactible grade powder for vibratory compaction and swaging.

The remainder of the blended ceramic grade material was used to make sintered pellets.

The chemical properties and analyses of the fuel used in Mark IV superheat rods are shown in Table 5-3.

TABLE 5-1
FUEL ROD CHARACTERISTICS

Characteristic	Rod No.	01L25	02L50	03L50	04L50	05L50	06L50	07L50	08L25	09L25	10L50
Fuel rod outside diameter, in.		0.280-inch Nominal									
Cladding wall thickness, in.		0.020-inch Nominal									
Cladding material		Incoloy-800									
Fuel rod length, in.		58-5/8	58-21/32	58-41/64	58-39/64	58-19/32	58-19/32	58-19/32	58-19/32	59-19/32	58-9/16
Fuel enrichment, %		25	50	50	50	50	50	50	25	25	50
Active fuel weight, gm		283.5	303.0	293.0	275.9	265.5	289.0	288.2	266.3	272.4	264.0
Active fuel length, in.		39-27/32	40-5/32	40-13/64	40-1/4	40-7/32	40-3/32	40	40-3/16	40-7/64	40
Total fuel weight, gm ⁽¹⁾		292.0	310.0	293.0	281.5	271.0	296.0	293.2	272.0	278.0	272.0
Total fuel length, in. ⁽¹⁾		40-63/64	41-3/64	40-13/64	40-63/64	40-61/64	40-31/32	41-1/32	40-59/64	40-27/32	41-1/64
Fuel fabrication		Pellet	Pellet	Pellet	Vipak ⁽²⁾	Vipak	Vipak	Vipak and Swaged ⁽³⁾	Vipak	Vipak	Vipak
Active fuel density, % TD		93.5	93.5	93.5	87.2	81.2	88.6	88.5	81.5	83.4	81.0
Nominal pellet diameter, in.		0.237	0.238	0.235	-	-	-	-	-	-	-
Pellet-cladding diametral gap, in.		3	2	5	0	0	0	0	0	0	0
Spacer		Incoloy-800 wire wrap									
Spacer wire, in.		0.065-in. diameter wire									

(1) Includes the natural (0.7% U-235) UO₂ buffer pellets at top and bottom of fuel column.

(2) Vibratory compaction with Dynapak UO₂ powder.

(3) Vibratory compacted and three-pass swaged.

TABLE 5-2

FUEL ROD CHARACTERISTICS

Rod No.	21V25	22V50	23V50	24V50	25V50	41X50	42X25	43X50
<u>Characteristic</u>								
Fuel rod outside diameter, in.	0.280-inch Nominal							
Cladding wall thickness, in.	0.020-inch Nominal							
Cladding material	Inconel-625					Hastelloy-X		
Fuel rod length, in.	58-41/64	58-41/64	58-41/64	58-21/32	58-19/32	58-39/64	58-19/32	58-19/32
Fuel enrichment, %	25	50	50	50	50	50	25	50
Active fuel weight, gm	284.0	293.0	305.0	295.0	292.0	301.5	286.0	292.0
Active fuel length, in.	39-53/64	40-5/32	40-1/8	40-3/32	40	39-7/8	39-27/32	39-13/16
Total fuel weight, gm ⁽¹⁾	293.5	299.5	312.0	301.5	300.0	309.0	294.5	300.5
Total fuel length, in. ⁽¹⁾	40-61/64	41-1/64	40-63/64	40-31/32	41-1/32	40-63/64	40-15/16	40-15/16
Fuel fabrication	Pellet							
Active fuel density, % TD	93.5% TD							
Nominal pellet diameter, in.	0.237	0.235	0.239	0.235	0.235	0.238	0.237	0.235
Pellet-cladding diameter gap, in.	3	5	1	5	5	2	3	5
Spacer	Inconel-600 wire wrap					Hastelloy-X		
Spacer wire, in.	0.062-in. diameter							

(1) Includes the natural (0.7% U-235) UO₂ buffer pellets at top and bottom of fuel column.

TABLE 5-3

CHEMICAL PROPERTIES OF UO₂ FUEL (SH*)

Enrichment	25%	50%	25%	50%
Fuel type	Pellet	Pellet	Dynapak	Dynapak
Isotopic content, %	25.23	49.67	24.84	50.14
Impurity content, ppm				
Al	6	100	35	-
Sb	<2	<2	<2	<2
B	<0.2	<0.2	<0.2	<0.2
Bi	<1	<1	1	<1
Cd	<1	<1	<1	<1
Ca	<10	<10	25	5
Cr	6	<6	95	100
Co	<2	<2	<2	10
Cu	2	2	20	11
Fe	10	10	400	200
Pb	<1	<1	1	2
Mg	<1	<2	4	10
Mn	<1	<1	22	22
Mo	<3	<3	<3	<3
Ni	5	<5	150	150
Si	<6	<6	34	60
Na	<20	<20	<20	<20
Sn	5	<1	10	10
V	<11	<11	<11	<11
Zn	<10	<10	<10	<10
Ag	-	<0.1	<0.1	<0.1
Gas content, ppm				
Cl	<3	2.2	<1	<1
F	2	2.6	3	<0.25
N	19	<10	<10	<10
Off-gas volume, μ l/gm				
Pellets	35.9	57.0		
28/150 mesh powder			28.8	
-150 mesh powder			177	
28/80 mesh powder				37.1, 44.5
150/270 mesh powder				105, 111
O/U ratio	2.015	2.002	1.986	2.007

*SH - Superheat

5.3 CLADDING

All incoming tubing was inspected for diameter and wall thickness measurements, and ultrasonically inspected for defects greater than 10% of the wall thickness. An identifying number was then electro-etched on the surface of each tube and maintained throughout the fabrication process. A sample from each lot of tubing was analyzed for chemical composition and mechanical properties at the department to confirm vendor certifications. The results are shown in Tables 5-4 and 5-5. Metallography of the as-received tubing is shown by Figure 5-1.

5.4 FUEL ROD FABRICATION PROCESS

Flow diagrams of the fabrication processes are shown in Figure 5-2. Process control was maintained by entering all pertinent data on a fuel rod travel card for each rod, as shown in Figure 5-3. Pre-irradiation measurements of the fuel rods are tabulated in Tables 5-6 through 5-9.

The rods were spaced by helically wrapping a 62-mil wire around the fuel elements with a pitch of approximately two turns per foot. The wire wrap equipment is shown in Figure 5-4.

Rod identifications were stamped on the lower end plug as shown in Figure 5-5. Some rods were stamped twice, with a small and a large sized number. It was thought that the small numbers might be difficult to read under water, consequently, the rods were restamped with larger numbers. The identification numbers were generated as follows:

<u>Alloy Symbol</u>	<u>Alloy</u>
L	Incoloy-800
V	Inconel-625
X	Hastelloy-X

The superheat fuel rod loading pattern is shown in Figure 5-6.

5.5 SWAGING AND ANNEALING

A number of the Mark IV fuel rods were fabricated by swaging powder-filled Incoloy-800 tubes. The swaging introduces a degree of cold work and associated grain orientation in the cladding which can have an adverse effect on the mechanical properties and corrosion resistance of the cladding material. The potential problems of the cold-worked material can be eliminated by annealing the fuel cladding to fine, stress-free grains. Work was performed to determine the effects of cold work and grain orientation on mechanical properties and to determine annealing parameters for the swaged tubes.⁽²⁴⁾ As a result of this work, these rods were swaged from 0.350-inch outside diameter to 0.280-inch outside diameter and annealed at 1950° F for 1 minute. The annealing was accomplished by moving the rod through an induction coil and quench tank at a rate of 1 ft/min. The annealed rods were then straightened by passing them through the swager.

TABLE 5-4
CHEMICAL ANALYSIS OF HARDWARE MATERIAL (SH)

	Heat No.	C	Mn	P	S	Si	Ni	Cr	Ti	Cu	Co	Fe	Al	Mo	Nb + Ta	Remarks
Incoloy-800 (0.280-in. outside diameter × 0.020-in.)																
Vendor certification	4154	0.06	0.89	-	0.007	0.34	30.87	20.94	0.47	0.33	0.04	46.54	0.31	-	-	
APED analysis		0.06	1.00	0.025	0.006	0.20	34.5	19.1	0.05	0.30	0.08	Balance	0.005	<0.001	-	As, Cd <100 ppm B <5 ppm Na, W <50 Zr <2 Ca, Ta, V, Nb <10 Ba, Be, Bi, Pb, Ag, Sn): <1
Incoloy-800 (0.350-in. outside diameter × 0.020-in.)																
Vendor certification	3659	0.03	0.80	-	0.007	0.33	31.53	19.95	0.57	0.39	0.09	46.54	0.14	-	-	
APED analysis		0.012	1.00	0.025	0.012	0.50	34.5	19.1	0.01	0.20	0.08	Balance	0.002	<0.001	-	As, Cd <100 ppm B <5 ppm Na, W <50 Zr <2 Ca, Ta, V, Nb <10 Ba, Be, Bi, Pb, Ag, Sn): <1
Incoloy-800 (Bar stock)																
Vendor certification	HH-3368-A	0.06	1.0	-	0.010	-	29.75	21.27	-	0.27	-	47.23	-	-	-	
APED analysis		0.053	1.20	-	0.010	0.50	31.8	19.0	0.050	0.50	<0.001	Balance	~0.50	~0.30	-	Sn 100 ppm Mg 10 ppm Cd, Na <100 Sb, B, Ba): <10 Bi, Zn <50 Ca, Pb, V) Ag <1
Inconel-625 (0.280-in. outside diameter × 0.020-in.)																
Vendor certification	NX-8636-A	0.03	0.10	0.009	0.008	0.25	62.71	22.13	0.21	-	0.03	1.89	0.18	9.10	3.35	
APED analysis		0.027	0.13	0.020	0.006	0.30	Balance	22.5	0.40	-	<0.020	3.0	0.40	9.8	4.02	Ta <0.020% Nb 4.0%
Inconel-625 (Bar stock)																
Vendor certification	NX-9011-A	0.03	0.15	-	0.007	0.30	62.00	21.41	0.27	0.05	-	2.73	0.24	8.64	4.15	
APED analysis		0.026	0.15	0.010	0.006	0.10	Balance	22.5	0.05	0.10	<0.001	4.0	0.050	9.5	-	Sn 500 ppm Sb, B, Ba): <10 ppm Mg 100 Ca, Pb, V) Ca, Na <100 Ag <1 Bi, Zn <50
Hastelloy-X (0.280-in. outside diameter × 0.020-in.)																
Vendor certification	X3-4306	0.13	0.68	0.015	0.006	0.63	Balance	22.16	-	-	1.76	17.91	-	9.00	-	W 0.75%
	X3-4360	0.10	0.82	0.010	0.003	0.78	Balance	22.18	-	-	1.52	17.87	-	8.94	-	W 0.48%
APED analysis		0.097	0.65	0.005	0.001	0.085	Balance	26.5	0.005	0.020	0.05	18.5	0.020	8.2	-	U <200 ppm Mg 25 As, Cd <100 La, Ta, Zn <10 Ca, V, Nb 50 Hf <5 Na <50 W 0.79% Sb, B, Be, Bi, Pb): <1 ppm Ag, Sn, Zr, Ba
Hastelloy-X (Bar stock)																
Vendor certification	X2-4904	0.10	0.68	0.010	0.007	0.69	Balance	22.60	-	-	0.09	17.70	-	9.07	-	W 0.03%
APED analysis		0.12	0.20	-	0.001	0.5	Balance	22.8	0.01	0.01	0.05	17.4	0.01	0.3	-	As, Cd <100 ppm Ta, B, Ca, La <10 ppm Na, Zn, Nb <50 Mg 5 V 20 Hf <5 Sn 10 Ag, Zr, Sb, Ba): <1 W 0.6% Pb, Be, Bi

TABLE 5-5

MECHANICAL PROPERTIES OF HARDWARE MATERIALS (SH)

	<u>Heat No.</u>	<u>Yield Strength 10³ psi</u>	<u>Ultimate Tensile Strength 10³ psi</u>	<u>% Elongation</u>	<u>Hardness</u>	<u>Grain Size ASTM</u>
Incoloy-800 (0.280-in. outside diameter × 0.020-in.)						
Vendor certification	4154	45.1	98.3	36	-	8-9
APED tests		56.6	84.8	32.7	30T 68(R _B 77)	9-1/2-10
Incoloy-800 (0.350-in. outside diameter × 0.020-in.)						
Vendor certification	3659	49.5	84.5	38	-	7-8
APED tests		45.5	86.4	37.6	30T 68(R _B 77)	8-1/2-9
Incoloy-800 (Bar stock)						
Vendor certification	HH-3368-A	41.0	87.0	51	R _B 83	-
APED tests		50.0	90.5	42.8	-	-
Inconel-625 (0.280-in. outside diameter × 0.020-in.)						
Vendor certification	NX-8636-A	77.1	132.9	41.7	-	7-1/2
APED tests		69.9	123.0	45.1	-	8
Inconel-625 (Bar stock)						
Vendor certification	NX-9011-A	104.0	149.0	50	R _C 27	-
APED tests		88.3	144.9	39.3	-	-
Hastelloy-X (0.280-in. outside diameter × 0.020-in.)						
Vendor certification	X3-4206	70.2	117.0	36	-	-
	X3-4360	74.5	120.4	36	-	-
APED tests		74.4	119	38	-	9
Hastelloy-X (Bar stock)						
Vendor certification	X2-4904	47.2	109.1	45	-	-
APED tests		50.5	114	42.4	-	-

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INCONEL-625 100x NEG No. 66 C 0019
HCl + H₂O₂

INCOLOY-800 100x NEG No. 65 C 0630
(0.280 in. o.d. x 0.020 in.) 10% OXALIC, ELECTROLYTIC

HASTELLOY-X 100x NEG No. 66 C 0018
10% OXALIC, ELECTROLYTIC

INCOLOY-800 100x NEG No. 65 C 0321
(0.350 in. o.d. x 0.020 in.) 10% OXALIC, ELECTROLYTIC

FIGURE 5-1. MICROSTRUCTURES OF CLADDING

FIGURE 5-2. FUEL ROD FABRICATION FLOW DIAGRAM

5
0

MARK IV FUEL ROD TRAVEL CARD (Powder)

Fuel Rod Number 06-L
Page Two

Helium Leak Detection (Calibrated Sensitivity 7×10^{-10} cc/sec)

Test Sequence	Test Pressure	Balance Reading	Test Reading	Leak Rate	Leak Location	Operator
Initial Weld	.01 mic.	0	0	none		N.D. 5/18/65
Insulation	.01	0	0	None		Q81 5/28
Re-test						
Retest						
Retest						

Diameter Measurements (0° orientation coincident with threaded hole)

Position	Orientation		Position	Orientation	
	0°	90°		0°	90°
Top Weld	.2825	.2830	36"	.2775	.2773
6"	.2790	.2790	42"	.2775	.2775
12"	.2770	.2769	48"	.2773	.2774
18"	.2778	.2777	54"	.2773	.2772
24"	.2775	.2774	Bottom Weld	.2812	.2805
30"	.2775	.2775			

Total Rod Length 58 19/32
Spacer Type wire .065
Enrichment Stamp 50%

Remarks: Pump Down Back pulled in chamber used Smith 106
Swaged over contact weld

MARK IV FUEL ROD TRAVEL CARD

(Powder)

Fuel Rod Number 06-L

Material Information

Tube Material and Number Incoloy L-1001
UO₂ Enrichment and Type 50% Dynapak
Fabrication Method 3-Pass Swaged

Weld Information

miller welder
Weld Number

	(1)	(2)	(3)	(4)
Current Recorder	24	24	24	24
Visual Examination (40X)	OK	OK	OK	OK
Radiography <u>Q. montia</u>				

Fuel Loading

Particle Size Distribution - (See Remarks)
Active UO₂ Weight (measured or calculated) 289 gm
Active Fuel Length 40 3/32"
Buffer Pellet, Length and Weight (Sum both pellets) .881 long and 7 gm
Total Fuel Length and Weight 40 31/32" and 296 gm
Plenum Length Measured 3.0
Disc and Spring Insertion Q81.

Remarks

20/35 → 131 gm
35/65 → 117
65/325 → 80
-325 → 17
345

FIGURE 5-3. MARK IV FUEL ROD TRAVEL CARD

TABLE 5-6

PRE-IRRADIATION DIAMETER MEASUREMENTS (SH)

(0° orientation coincident with wire weld bead on upper end plug)

Rod No. Orientation Position	01L25		02L50		03L50		04L50		05L50		06L50	
	0°	90°	0°	90°	0°	90°	0°	90°	0°	90°	0°	90°
Top Weld	0.2842	0.2840	0.2754	0.2775	0.2765	0.2765	0.2819	0.2830	0.2829	0.2828	0.2825	0.2830
6 in.	0.2793	0.2793	0.2758	0.2788	0.2790	0.2790	0.2790	0.2790	0.2792	0.2793	0.2790	0.2790
12 in.	0.2783	0.2781	0.2769	0.2768	0.2768	0.2767	0.2790	0.2795	0.2791	0.2790	0.2770	0.2769
18 in.	0.2792	0.2788	0.2750	0.2790	0.2788	0.2790	0.2790	0.2791	0.2789	0.2789	0.2778	0.2777
24 in.	0.2788	0.2789	0.2759	0.2788	0.2789	0.2789	0.2790	0.2789	0.2788	0.2791	0.2775	0.2774
30 in.	0.2792	0.2790	0.2759	0.2789	0.2790	0.2789	0.2789	0.2790	0.2790	0.2790	0.2775	0.2775
36 in.	0.2793	0.2790	0.2791	0.2790	0.2791	0.2790	0.2790	0.2788	0.2789	0.2791	0.2775	0.2773
42 in.	0.2791	0.2790	0.2789	0.2790	0.2790	0.2790	0.2789	0.2790	0.2789	0.2791	0.2775	0.2775
48 in.	0.2791	0.2791	0.2787	0.2788	0.2789	0.2788	0.2788	0.2790	0.2789	0.2790	0.2773	0.2774
54 in.	0.2792	0.2793	0.2788	0.2788	0.2789	0.2789	0.2789	0.2788	0.2790	0.2791	0.2773	0.2772
Bottom Weld	0.2828	0.2821	0.2823	0.2823	0.2813	0.2812	0.2823	0.2826	0.2828	0.2819	0.2812	0.2805
Maximum	0.2842	0.2840	0.2823	0.2823	0.2813	0.2812	0.2823	0.2830	0.2829	0.2828	0.2825	0.2830
Minimum	0.2783	0.2781	0.2769	0.2768	0.2765	0.2765	0.2789	0.2788	0.2788	0.2789	0.2770	0.2769
Average	0.2808	0.2797	0.2789	0.2789	0.2787	0.2787	0.2795	0.2797	0.2806	0.2797	0.2784	0.2783

Rod No. Orientation Position	07L50		08L50		09L25		10L50*	
	0°	90°	0°	90°	0°	90°	0°	90°
Top Weld	0.2821	0.2825	0.2843	0.2842	0.2832	0.2829	0.2796	0.2801
6 in.	0.2793	0.2793	0.2791	0.2792	0.2792	0.2790	0.2802	0.2805
12 in.	0.2765	0.2764	0.2769	0.2768	0.2768	0.2768	0.2802	0.2804
18 in.	0.2773	0.2773	0.2790	0.2790	0.2790	0.2792	0.2803	0.2804
24 in.	0.2773	0.2778	0.2789	0.2788	0.2790	0.2790	0.2803	0.2805
30 in.	0.2777	0.2779	0.2790	0.2790	0.2790	0.2790	0.2804	0.2803
36 in.	0.2773	0.2778	0.2790	0.2789	0.2789	0.2789	0.2803	0.2804
42 in.	0.2773	0.2778	0.2788	0.2788	0.2790	0.2790	0.2803	0.2804
48 in.	0.2773	0.2778	0.2788	0.2789	0.2790	0.2791	0.2804	0.2803
54 in.	0.2775	0.2773	0.2789	0.2790	0.2789	0.2790	0.2802	0.2802
Bottom Weld	0.2810	0.2803	0.2829	0.2828	0.2830	0.3320	0.2803	0.2802
Maximum	0.2824	0.2825	0.2843	0.2842	0.2832	0.2829	0.2804	0.2805
Minimum	0.2765	0.2764	0.2769	0.2768	0.2768	0.2768	0.2795	0.2790
Average	0.2785	0.2785	0.2796	0.2796	0.2795	0.2794	0.2802	0.2802

*Rod 10L50 had welded wire wrap prior to fuel loading, hence, measurements taken at available spaces.

TABLE 5-7

PRE-IRRADIATION DIAMETER MEASUREMENTS (SH)

(0° orientation coincident with wire weld bead on upper end plug)

Rod No. Orientation Position	21V25		22V40		23V50		24V50		25V50	
	0°	90°	0°	90°	0°	90°	0°	90°	0°	90°
Top Weld	0.2821	0.2825	0.2823	0.2828	0.2822	0.2825	0.2838	0.2822	0.2815	0.2817
6 in.	0.2798	0.2798	0.2798	0.2798	0.2783	0.2778	0.2763	0.2778	↑	↑
12 in.	0.2803	0.2805	0.2800	0.2800	0.2775	0.2775	0.2771	0.2773	↑	↑
18 in.	0.2797	0.2796	0.2796	0.2798	0.2796	0.2798	0.2795	0.2795	↑	↑
24 in.	0.2797	0.2797	0.2797	0.2798	0.2798	0.2798	0.2795	0.2794	↑	↑
30 in.	0.2797	0.2798	0.2798	0.2798	0.2797	0.2797	0.2795	0.2795	↑	↑
36 in.	0.2797	0.2798	0.2798	0.2798	0.2797	0.2798	0.2795	0.2795	↑	↑
42 in.	0.2795	0.2797	0.2798	0.2798	0.2795	0.2797	0.2795	0.2795	↑	↑
48 in.	0.2797	0.2797	0.2797	0.2796	0.2798	0.2795	0.2795	0.2795	↑	↑
54 in.	0.2797	0.2797	0.2793	0.2795	0.2797	0.2797	0.2795	0.2793	↑	↑
Bottom Weld	0.2818	0.2815	0.2818	0.2818	0.2839	0.2839	0.2820	0.2818	0.2842	0.2815
Maximum	0.2821	0.2825	0.2823	0.2828	0.2839	0.2839	0.2830	0.2822	0.2842	0.2817
Minimum	0.2795	0.2796	0.2793	0.2795	0.2775	0.2775	0.2763	0.2773	0.2797	0.2797
Average	0.2802	0.2802	0.2801	0.2811	0.2800	0.2800	0.2796	0.2796	0.2807	0.2803

Turbulence promoter
area

Rod No. Orientation Position	41X50		42X25		43X50	
	0°	90°	0°	90°	0°	90°
Top Weld	0.2830	0.2831	0.2820	0.2819	0.2820	0.2820
6 in.	0.2785	0.2788	0.2788	0.2785	0.2785	0.2785
12 in.	0.2783	0.2783	0.2789	0.2793	0.2799	0.2802
18 in.	0.2785	0.2785	0.2788	0.2788	0.2788	0.2787
24 in.	0.2787	0.2787	0.2785	0.2784	0.2788	0.2788
30 in.	0.2787	0.2787	0.2785	0.2788	0.2788	0.2788
36 in.	0.2785	0.2785	0.2785	0.2788	0.2786	0.2788
42 in.	0.2787	0.2785	0.2785	0.2785	0.2787	0.2788
48 in.	0.2786	0.2785	0.2785	0.2787	0.2787	0.2787
54 in.	0.2787	0.2787	0.2788	0.2788	0.2787	0.2787
Bottom Weld	0.2828	0.2828	0.2825	0.2815	0.2825	0.2827
Maximum	0.2830	0.2831	0.2825	0.2819	0.2825	0.2827
Minimum	0.2783	0.2783	0.2785	0.2784	0.2785	0.2785
Average	0.2794	0.2794	0.2793	0.2793	0.2795	0.2795

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TABLE 5-8

PRE-IRRADIATION LENGTH MEASUREMENTS (SH)

(measured tip-to-tip)

<u>Rod No.</u>	<u>Length, in.</u>
01L25	58-5/8
02L50	58-21/32
03L50	58-41/64
04L50	58-39/64
05L50	58-19/32
06L50	58-19/32
07L50	58-19/32
08L25	58-19/32
09L25	58-19/32
10L50	58-9/16
21V25	58-41/64
22V50	58-41/64
23V50	58-41/64
24V50	58-21/32
25V50	58-19/32
41X50	58-39/64
42X25	58-19/32
43X50	58-19/32

TABLE 5-9

PROFILOMETER MEASUREMENTS (SH)

<u>Rod No.</u>	<u>Average Diameter, in.</u>	<u>Maximum Ovality, in. (1)</u>	<u>Location of Maximum Ovality (2)</u>	<u>Remarks</u>
01L25	0.2792	0.0004	7-10	
02L50	0.2791	0.0004	41-weld	
03L50	0.2794	0.0005	13-15	
04L50	0.2794	0.0002	-	Uniform
04L50	0.2793	0.0004	13-15	Local cladding expansion of ~2 mils at fuel- plenum interface.
06L50	0.2780	0.0018	2-1/2	
07L50	0.2780	0.0005	-	Uniform
08L25	0.2794	0.0006	4-10	
09L25	0.2796	0.0003	-	Uniform
10L50	welded wire spacer rod			
21V25	0.2800	0.0002	-	Uniform
22V50	0.2799	0.0002	-	Uniform
23V50	0.2800	0.0002	-	Uniform
24V50	0.2798	0.0003	4-17	
41X50	0.2789	0.0007	3-4	
42X50	0.2780	0.0008	3-4	
43X50	0.2789	0.0003	22-24	

(1) Ovality = maximum diameter - minimum diameter (at the same cross section).

(2) Measured from weld at plenum, inches.

FIGURE 5-4. WIRE SPACER WRAPPING FIXTURE

FIGURE 5-5. ROD IDENTIFICATION MARK

5.6 WELDED WIRE SPACERS

One rod, KB-42 10L50, was fabricated with wire spacers welded to the tube. These wire spacers were projection spot-welded to the tube by Canadian General Electric. Corrosion tests with boiling MgCl_2 and 1300°F rupture burst tests showed no apparent detrimental effects caused by the spot-welds. The wire was crimped to make projections at $5/8$ -inch intervals, and the wires were spot-welded to the tubes at these points. Figure 5-7 shows the spacer wires welded to the cladding. The microstructure associated with the interfacial line in Figure 5-8 is an indication of a less than a perfect bond. However, where bulk fusion has occurred, i. e., in the dendritic zones, (Figure 5-8a) a perfect bond does exist. This area is larger than the tube wall thickness and, therefore, appears to be sufficiently strong. The root of the wire and the tubing beneath the weld are strained heavily and fine cracks are observed in the root (Figure 5-8b). The surface of the wire has an intergranular attack to a depth of 2 to 3 mils (Figure 5-8c). This appears to be caused by an incomplete removal of the drawing compound during the fabrication of the wire.

Stress rupture test performed on welded wire spacers are shown in Table 5-10. In all cases, the welded wire spacer had a longer life than the standard unwelded tubing. It may be concluded that the tube wall was not weakened by the wire attachment.

Because of the nature of the properties welding process, small dimples are impressed into the tube. These dimples prevent design size pellets from being loaded and consequently was loaded with compacted powder.

5.7 TURBULENCE PROMOTERS

One fuel rod, KB-45 Rod 25V50, was fabricated with turbulence promoters on the cladding (Figure 5-9). The fins are 2 mils high by 2 mils wide, spaced 40 mils from center to center. These fins were formed by grinding into the cladding surface. A great deal of difficulty was experienced in forming the fins. Consequently, the turbulence promoters were put in on only 30 inches of the fuel rod.

FIGURE 5-6. MARK IV SUPERHEAT FUEL ASSEMBLY LOADING DIAGRAM

FIGURE 5-7. PROJECTION SPOT-WELDS

TABLE 5-10
STRESS RUPTURE TEST AT 1350°F
 (time-to-failure)

<u>Specimen</u>	<u>Nominal Loop Stress, psi</u>			
	<u>13,000</u>	<u>9620</u>	<u>8130</u>	<u>6500</u>
Turbulence promoters	19 h	85 h	229 h	794 h
Welded wire spacers	43	-	-	914
Standard tubing	42	101	267	863

Stress rupture tests were performed on these tubings and the results are tabulated in Table 5-10. The tubing with turbulence promoters showed a somewhat lower life than the standard tube. However, it must be noted that the area used to calculate the stress is based on the nominal outside diameter before grinding. The actual area is somewhat smaller, consequently, the true stress applied is greater than that shown in the data.

(a)

Neg. No. 65C2054

(b)

(c)

Neg. No. 65C2056

FIGURE 5-8. WELDED WIRE SPACER MICROSTRUCTURES

FIGURE 5-9. TURBULENCE PROMOTERS

SECTION VI

BOILING WATER FUEL FABRICATION

6.1 GENERAL

The Mark IV boiling water fuel consists of 112 fuel rods with 5% enriched uranium dioxide pellets and clad with 0.485-inch i. d. $\times$ 0.036-inch wall Zircaloy-2. These rods are made up of two segments, G-1 and G-2. The upper segment, G-1, has a 29-inch $\pm$ 1/8-inch active fuel length and the lower segment, G-2, has a 30-5/32-inch $\pm$ 1/8-inch active fuel length. Each segment has a 3/4-inch spring supported plenum.

The drawings which describe the fuel rods are:

<u>Title</u>	<u>Drawing No.</u>
Boiling water fuel assembly and parts list	846D731
End fitting	159A2701
End plug	159A2702
End plug	159A2703
End plug	159A2704
Wafer	159A2705
Spring	159A2706
Pellet	159A2707
Fuel rod (boiling)	112C2007

6.2 URANIUM DIOXIDE

The 5% enriched UO_2 ceramic grade powder used conformed to specifications except that the bulk density was low. A sinterability test performed on this powder resulted in pellets of 94.7% to 97.4% of theoretical density. The pellet integrity was considered excellent. On the basis of this sinterability test, the low bulk density powder was accepted.

6.3 CLADDING

Cladding is made from Zircaloy-2 tubing material. All tubing was inspected for the following:

1. Inside diameter, 0.4850-in. $\pm$ 0.0015-in.;
2. Wall thickness, 0.036-in. $\pm$ 0.003-in.;
3. Bow tolerance, 0.010-in. per foot of length;
4. Ultrasonic test, $< 10\%$ of wall thickness.

6.4 FUEL ROD FABRICATION PROCESS

The boiling water fuel rods were fabricated according to the current standard pellet fuel rod fabrication procedure. Fuel weights of these rods were 885 ± 6 gms for the upper segments and 917 ± 10 gms for the lower segments. Each rod was identified by a number, 1 to 229, scribed on the lower end plug of each segment with a vibra-tool.

The rods were assembled in the bundle, and a 0.062-inch Type 304 stainless steel safety wire was threaded through adjacent cap screws to prevent loosening screws, caps, or twisting of the rods.

The boiling water fuel portion of the assembled bundle is shown in Figure 6-1.

FIGURE 6-1. BOILING FUEL ASSEMBLY – TOP VIEW

SECTION VII

FUEL BUNDLE DESIGN

7.1 CRITERIA AND DESIGN OBJECTIVES FOR THE MARK IV FUEL BUNDLES

The following objectives and criteria defined the configuration and requirements of the Mark IV fuel bundles:

1. The superheat test fuel would be High Performance Elements, in rod configuration, to test at high power densities and heat fluxes. A peak heat flux of 600,000 Btu/h-ft² and power density of 150 kW/kg UO₂ were set as initial design objectives.
2. The fuel bundles would be tested in the EVESR plant and must be compatible with that system.
3. EVESR core zones Nos. 1, 2, and 3 were to be considered for these fuel bundles.
4. Fuel design shall provide the power density and heat flux objectives without fuel melting.
5. A maximum fuel cladding temperature of 1250°F will be used for the reference design conditions.
6. High fuel burnup, consistent with a high performance fuel design, would be an objective. 40,000 MWd/T U average burnup was used for fuel design.
7. There must be no reactivity loss in the EVESR core when Mark IV fuel bundles are substituted for the Mark II bundles which these would replace. If possible, the Mark IV should provide some reactivity gain to compensate for core burnup.
8. The pressure drop across the fuel elements, at the reference design condition must not exceed 100 psi so that the Mark IV fuel elements, when installed in parallel with Mark II fuel bundles, will not impose unusual orificing requirements. EVESR core zones 2 and 3 might require such paralleling of Mark II and Mark IV fuel bundles.
9. The design shall minimize the steam requirement for the superheat rods, insofar as practical, to permit an increase in EVESR power level. A boiling water booster section to help maintain core reactivity would help boost the EVESR steam supply and meet this requirement.
10. The design of the boiling water booster section, if used to maintain EVESR core reactivity, shall be conservative. A critical heat flux ratio higher than the value of 1.5 used for boiling water reactor fuel design at the time of design of the Mark IV fuel bundles was desired.
11. The high performance fuel rods shall be replacable to permit flexibility in the test program.
12. The fuel bundle design must provide for measurement of the exit steam temperature from each high performance fuel rod.

7.2 MECHANICAL DESIGN

7.2.1 Assembly Arrangement

The Mark IV fuel bundle replaces a standard EVESR Mark II fuel bundle. The steam inlet and exit piping are a modification of the concentric riser-downcomer used for the Mark II fuel. This piping is removable from the Mark IV fuel bundles, utilizing flange coupling designs which were developed and tested in the E-SADE test program. Figure 7-1 illustrates the steam flow routing for the Mark IV fuel bundles, showing the inlet and exit to the high performance superheat fuel, as well as the boiling water fuel required to maintain the reactivity of these fuel bundles.

Figure 7-2 shows the detailed drawing of the entire Mark IV fuel bundle test assembly. The parts list for the assembly is given in Figure 7-3.

7.2.2 Mark IV Fuel Section

Figure 7-4 shows the fuel section of the Mark IV fuel bundles. The various assembly groups provide the proper fuel arrangements for different EVESR core locations. Provision is also made for a nonfueled assembly to permit shipment to the reactor site without fuel installed. This permits separate shipment of the boiling water assembly so that it may be adequately packed to protect against damage during handling and transit. Figures 7-5 and 7-6 show the parts lists for these assemblies. Figure 7-7 shows a completed assembly prior to loading of fuel. The shipping assembly is shown in Figure 7-8.

A pressure test cap is provided to facilitate handling of the fueled section, to protect the seal surfaces of the coupling between riser-downcomer assembly, to permit a pressure test of the assembly after the high performance fuel rods have been installed, and to prevent entry of foreign materials during shipment or storage. Figure 7-9 shows the pressure test cap installed on a fuel bundle.

7.2.3 Riser-Downcomer Assembly

The concentric arrangement used for the EVESR Mark II fuel bundles was adapted to the Mark IV program. This assembly is removable from the fueled section of the fuel bundles which is described in subsection 7.2.2 of this report. This removable feature permits the fuel section to be inverted in the EVESR fuel storage pool after irradiation to permit removal and reinstallation of high performance fuel rods.

Figure 7-10 shows the riser-downcomer assembly for use with Mark IV fuel bundles. Figure 7-11 shows a completed riser-downcomer assembly, after thermocouples have been installed and the assembly is ready for packing for shipment to the reactor site.

7.2.4 Boiling Fuel Assembly

The boiling water fuel assembly was designed so that it could be assembled to the Mark IV fuel bundle as a complete unit. Figure 7-12 shows the boiling water assembly. Figure 7-13 illustrates a completed assembly, ready for packing for shipment to the reactor site.

The separate assembly design of the boiling water fuel assembly permitted packing between the fuel rods and placing the entire assembly in a strongback to ensure that handling, vibration, shock, and other events associated with shipping would not cause damage to the assembly or to the fuel rods. The shipping strongback is shown in Figure 7-14.

FIGURE 7-1. SCHEMATIC ARRANGEMENT - MARK IV FUEL BUNDLE

197R202

GENERAL ELECTRIC

197R202

GEAP-5097

APPLIED PRACTICE	FINISH	TOLERANCES	REV.
145A5481	✓	FRACTION DECIMAL ANGULAR	0 X 2 3/4
			197R202

EXPERIMENTAL SUPERHEAT ASM
FIRST MADE FOR MARK IV EYESR

PL ISSUED

FCF 194 X 702

3	JUN 30 65	1	SECTION 14	REV.
4	JUN 30 65	2	SECTION 14	REV.
4	JUN 30 65	3	SECTION 14	REV.
4	JUN 30 65	4	SECTION 14	REV.
4	JUN 30 65	5	SECTION 14	REV.
4	JUN 30 65	6	SECTION 14	REV.
4	JUN 30 65	7	SECTION 14	REV.
4	JUN 30 65	8	SECTION 14	REV.
4	JUN 30 65	9	SECTION 14	REV.
4	JUN 30 65	10	SECTION 14	REV.

JUN 30 65
NOV 25-1964APO
SAN JOSE

197R202

FIGURE 7-2. MARK IV EXPERIMENTAL
SUPERHEAT ASSEMBLY

GENERAL ELECTRIC

PARTS LIST FOR

197R202

CONT ON SHEET

SH NO.

REV. NO. 0 X 23							TITLE		PARTS LIST FOR	
PARTS LIST FOR							EXPERIMENTAL SUPERHEAT ASM.		CONT ON SHEET	
197R202							FIRST MADE FOR		MARK IV EVESR FCF 194X702	
CONT ON SHEET							SH NO.			
GROUP NO. AND QUANTITY							PART NO.		DRAWING NO., DESCRIPTION, MATERIAL, WEIGHT	
1	1	1	1	1	1	1	1	RISER & DOWNCOMER ASM	237E801G1	
						1	2	SUPERHEAT FUEL BUNDLE	237E577G1	
						1	3		237E577G2	
						1	4		237E577G3	
						1	5	SUPERHEAT FUEL BUNDLE	237E577G4	
1	1	1	1	1	1	1	6	K-SEAL	#121150 PA 2000 HARRISON MFG.	OR ENG. APP. EQ.
1	1	1	1	1	1	1	7	K-SEAL	#12195 PA3937	
1	1	1	1	1	1	1	8	K-SEAL	#121150 PA3937 HARRISON MFG.	OR ENG. APP. EQ.
1	1	1	1	1	1	1	9	PROTECTION TUBE ASM.	129B2453 G1	
						1	10	IDENT. PLATE	159A2724 P2	
						1	11		159A2724 P1	
						1	12		159A2724 P3	
						1	13	IDENT. PLATE	159A2724 P4	
4	4	4	4	4	4	4	14	BOLT	M524674-24	
AR	AR	AR	AR	AR	AR	AR	15	SAFETY WIRE	.032 DIA. MAT'L 304 S'STL	
1	1	1	1	1	1	1	16	TC & PLUG ASM.	107C3918 G6	
						1	17	SUPERHEAT FUEL BUNDLE	237E577G10	
						1	18	IDENT. PLATE	159A2724 P5	
						1	19	ORIFICE	149A4568P(SPEC LATER)	
						1	20			
						1	21			
						1	22			
						1	23	ORIFICE	149A4568P(SPEC. LATER)	
1							24	ORIFICE	149A4568 P (SPEC. LATER)	
1							25	IDENT. PLATE	159A2724 PG	
1							26	SUPERHEAT FUEL BUNDLE	237E577, G11	

DESCRIPTION OF GROUPS						REVISIONS		PRINTS TO	
KB-51	KB-50	KB-45	KB-44	KB-43	KB-42	3	JUL. 21/65	1	DEC. 18-64
						DELETED P9	CHANGED: P16 QUANTITY FROM 4 TO 1 (50mg 1404)		
							ADDED: DWG. NO. P9		
							ADDED: P19, 20, 21, 22, 23		
						2	APR. 12/65		
							ADDED: 66 & P. 24, 25, 26		
							ADDED: DSCR. OF GROUPS		

MADE BY		APPROVALS		DIV. OR DEPT.		PARTS LIST FOR	
C. R. Nelson		10/11/24/64		APO		197R202	
Nov. 25-64				SAN JOSE, CALIF.		CONT ON SHEET	
						SH NO.	

FF-801-P 25M (3-57)
PRINTED IN U.S.A.FIGURE 7-3. PARTS LIST - MARK IV EXPERIMENTAL
SUPERHEAT ASSEMBLY

GENERAL ELECTRIC

PARTS LIST FOR

237E-577
CONT ON SHEET 2 SH NO. 1

REV. NO. 012 PARTS LIST FOR 237E-577 CONT ON SHEET 2 SH NO. 1							TITLE SUPERHEAT FUEL BUNDLE SECT. A FIRST MADE FOR MARK IV EYESR	
GROUP NO. AND QUANTITY	PART NO.	NAME	DRAWING NO., DESCRIPTION, MATERIAL, WEIGHT					
1 1 1 1 1 1	1	LOWER COUPLER	846D706G1					
1 1 1 1 1 1	2	TRANSITION CAP	129B2314P1					
1 1 1 1 1 1	3	STEAM TUBE ASSEMBLY	846D705G2					
1 1 1 1 1 1	4	FRAME	846D711G1					
2 2 2 2 2 2	5	CHUTE	159A2717G1					
2 2 2 2 2 2	6	PROCESS TUBE	112C2022G1					
2 2 2 2 2 2	7	PROCESS TUBE	112C2022G2					
1 1 1 1 1 1	8	CRUCIFORM CHANNEL	846D709G1					
1 1 1 1 1 1	9	BOILING FUEL ASSEMBLY	846D731G1					
4 4 4 4 4 4	10	TUBE PLUG (PIN TYPE)	129B2311G1					
1 1 1 1 1 1	11	LOCKING RING	129B2313G1					
4 4 4 4 4 4	12	RETAINER PIN	159A2718P1					
4 4 4 4 4 4	13	NUT	159A2716P1					
12 12 12 12 12 12	14	STRAP	159A2737P1					
8 8 8 8 8 8	15	SCR. SOC. HD. MS24674-21 1/4-20x3/8 SST.						
4 4 4 4 4 4	16	O RING	159A2731P1					
8 8 8 8 8 8	17	STOPNUT Z1200-066, 3/8-16, ESNA OR ENGRG. APPD. EQ.						
	3 18	FUEL ROD (SH.) POWDER	112C2016G1					
	1 19	FUEL ROD (SH.) POWDER	112C2016G2					
	2 20	FUEL ROD (SH.) PELLET (50%)	112C2013G1					
	1 21		(25%) 112C2013G2					
	3 22		112C2013G3					
	1 23		112C2013G4					
	2 24		112C2013G5					
	1 25	FUEL ROD (SH.) PELLET	112C2013G6					
1 1 1 1 1 1	26	STEAM TUBE ASSEMBLY	846D705G1					
1 1 1 1 1 1	27	STEAM TUBE ASSEMBLY	846D705G3					

DESCRIPTION OF GROUPS	REVISIONS	PRINTS TO
INCOLOY 800 NO-FUEL ASM. NO-FUEL ASM. HAST. X PELLETS LOC. E4-KB45 INCOLOY 800 PELL. LOC. B3-KB44 INCOLOY 800 PELL. LOC. B4-KB42 SWG INCOLOY 800 LOC. E3-KB43	3 DEC-6'65 H.A.T. 1 APR 12'65 PT. 24, CHG. QUAN. 2 FROM G3 TO G4 ADDED: G10 & G11 & DSCR. OF CAME. ADDED: 70 P20 & 21 ADDED DWG # FOR PTS. 3, 5, 20, 21, 26. ADDED PTS. DSCR. FOR P21 & 22 2 DEC-1'65 NE 11069	

APPROVALS	APO	DIV. OR DEPT.	PARTS LIST FOR
MADE BY: <i>Harry C. Stern</i> Aug 20, 1964 ISSUED: <i>Nov. 25-'64</i>	APO	SAN JOSE, CALIF.	237E-577 CONT ON SHEET 2 SH NO. 1

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S.H.

GENERAL ELECTRIC

PARTS LIST FOR

237E-577
CONT ON SHEET 3 SH NO. 2

REV. NO. 0123 PARTS LIST FOR 237E-577 CONT ON SHEET 3 SH NO. 2							TITLE SUPERHEAT FUEL BUNDLE SECT. A FIRST MADE FOR MARK IV EYESR	
GROUP NO. AND QUANTITY	PART NO.	NAME	DRAWING NO., DESCRIPTION, MATERIAL, WEIGHT					
1 1 1 1 1 1	1	LOWER COUPLER	846D706G1					
1 1 1 1 1 1	2	TRANSITION CAP	129B2314P1					
1 1 1 1 1 1	3	STEAM TUBE ASSEMBLY	846D705G2					
1 1 1 1 1 1	4	FRAME	846D711G1					
2 2 2 2 2 2	5	CHUTE	159A2717G1					
2 2 2 2 2 2	6	PROCESS TUBE	112C2022G1					
2 2 2 2 2 2	7	PROCESS TUBE	112C2022G2					
1 1 1 1 1 1	8	CRUCIFORM CHANNEL	846D709G1					
1 1 1 1 1 1	9	BOILING FUEL ASSEMBLY	846D731G1					
4 4 4 4 4 4	10	TUBE PLUG (WITH PIN)	129B2311G1					
1 1 1 1 1 1	11	LOCKING RING	129B2313G1					
4 4 4 4 4 4	12	RETAINER PIN	159A2718P1					
4 4 4 4 4 4	13	NUT	159A2716P1					
12 12 12 12 12 12	14	STRAP	159A2737P1					
8 8 8 8 8 8	15	SCR. SOC. HD. MS24674 1/4-20x3/8 SST						
4 4 4 4 4 4	16	O RING	159A2731P1					
8 8 8 8 8 8	17	STOPNUT Z1200-066, 3/8-16 ESNA OR ENGRG. APPD. EQ.						
	18	FUEL ROD (SH)						
	19							
3 3 3	20		(INCOLOY 800)((50%)) 112C2013G1					
1 1 1	21		(INCOLOY 800)((25%)) 112C2013G2					
	22							
	23							
	24							
	25	FUEL ROD (SH.)						
1 1 1 1 1 1	26	STEAM TUBE ASSEMBLY	846D705G1					
1 1 1 1 1 1	27	STEAM TUBE ASSEMBLY	846D705G3					

DESCRIPTION OF GROUPS	REVISIONS	PRINTS TO
INCOLOY 800 NO-FUEL ASM. NO-FUEL ASM. HAST. X PELLETS LOC. E4-KB45 INCOLOY 800 PELL. LOC. B3-KB44 INCOLOY 800 PELL. LOC. B4-KB42 SWG INCOLOY 800 LOC. E3-KB43	3 DEC-6'65 H.A.T. 1 APR 12'65 PT. 24, CHG. QUAN. 2 FROM G3 TO G4 ADDED: G10 & G11 & DSCR. OF CAME. ADDED: 70 P20 & 21 ADDED DWG # FOR PTS. 3, 5, 20, 21, 26. ADDED PTS. DSCR. FOR P21 & 22 2 DEC-1'65 NE 11069	

APPROVALS	APO	DIV. OR DEPT.	PARTS LIST FOR
MADE BY: <i>Harry C. Stern</i> Oct 27, 1964 ISSUED: <i>Nov. 25-'64</i>	APO	SAN JOSE, CALIF.	237E-577 CONT ON SHEET 3 SH NO. 2

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FIGURE 7-5. PARTS LIST FOR MARK IV BUNDLE

REV. NO. 0123		PARTS LIST FOR		GENERAL ELECTRIC		237E-577		TITLE	
CONT. ON SHEET 4		SH. NO. 3		CONT. ON SHEET 4		SH. NO. 3		SUPERHEAT FUEL BUNDLE SECT. B	
PARTS LIST FOR		237E-577		FIRST MADE FOR		MARK IV EVESR			
GROUP NO. AND QUANTITY		PART NO.		NAME		DRAWING NO., DESCRIPTION, MATERIAL, WEIGHT			
2	2	28	CHUTE	159A2717G2					
2	2	29	CHUTE	159A2717G3					
		30	STEAM TUBE ASSEMBLY	846D705G4					
		31	CHUTE	159A2717G4					
		32	FUEL ROD (SH. GERM.)	LATER					
		33	FUEL ROD (SH. GERM.)	LATER					
		34	PROCESS TUBE	112C2022G3					
		35	PROCESS TUBE	112C2022G4					
	1	36	FUEL ROD (WELDED WIRE)	112C2831G1					
	1	37	FUEL ROD (TURB. PROM.)	112C2834G1					
DESCRIPTION OF GROUPS		REVISIONS		PRINTS TO					
		1 7/28/65 MAY 18 '65							
		DELETED PTS 32 & 33							
		2 JUL 21 '65							
		ADDED P34 & P35							
		3 DEC 1 '65							
		NE 11069 11/30/65							
MADE BY		APPROVALS		PARTS LIST FOR		237E-577		TITLE	
HARRY C. STERN		APO		CONT. ON SHEET 4		SH. NO. 3		SUPERHEAT FUEL BUNDLE SECT. B	
NOV. 25-1964		SAN JOSE, CALIF.		LOCATION		CONT. ON SHEET 4		SH. NO. 3	

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REV. NO. 0123		PARTS LIST FOR		GENERAL ELECTRIC		237E-577		TITLE	
CONT. ON SHEET F		SH. NO. 4		CONT. ON SHEET F		SH. NO. 4		SUPERHEAT FUEL BUNDLE SECT. B	
PARTS LIST FOR		237E-577		FIRST MADE FOR		MARK IV EVESR			
GROUP NO. AND QUANTITY		PART NO.		NAME		DRAWING NO., DESCRIPTION, MATERIAL, WEIGHT			
2	2	28	CHUTE	159A2717G2					
		29	CHUTE	159A2717G3					
	1 1	30	STEAM TUBE ASSEMBLY	846D705G4					
	2 2	31	CHUTE ((SAYS 304SS))	159A2717G4					
		32	FUEL ROD (SH. GERM.)	LATER					
		33	FUEL ROD (SH. GERM.)	LATER					
2	2 2	34	PROCESS TUBE	112C2022G3					
2	2 2	35	PROCESS TUBE	112C2022G4					
		36	FUEL ROD (WELDED WIRE)	112C2831G1					
		37	FUEL ROD (TURB. PROM.)	112C2834G1					
DESCRIPTION OF GROUPS		REVISIONS		PRINTS TO					
		1 7/28/65 MAY 18 '65							
		DELETED PTS 32 & 33							
		2 JUL 21 '65							
		ADDED G12							
		3 DEC 1 '65							
		ADDED P34 & P35 11/30/65							
MADE BY		APPROVALS		PARTS LIST FOR		237E-577		TITLE	
HARRY C. STERN		APO		CONT. ON SHEET F		SH. NO. 4		SUPERHEAT FUEL BUNDLE SECT. B	
NOV. 25-1964		SAN JOSE, CALIF.		LOCATION		CONT. ON SHEET F		SH. NO. 4	

FF-801-P 25M (3-57)
PRINTED IN U.S.A.FIGURE 7-6. PARTS LIST FOR MARK IV
FUEL BUNDLE

FIGURE 7-7. FUEL SECTION – MARK IV FUEL BUNDLE

FIGURE 7-8. SHIPPING ASSEMBLY – MARK IV FUEL BUNDLE

FIGURE 7-9. PRESSURE TEST CAP – MARK IV FUEL BUNDLE

FIGURE 7-11. MARK IV RISER-DOWNCOMER WITH THERMOCOUPLE ASSEMBLY INSTALLED

FIGURE 7-12. MARK IV BOILING WATER ASSEMBLY

FIGURE 7-13. MARK IV BOILING WATER ASSEMBLY; READY FOR PACKING

7.2.5 Assembly of Fuel at Reactor Site

The high performance fuel rods, boiling water fuel assemblies, and hardware components were shipped to the reactor site as separate components to avoid damage during shipment. The superheat rods and boiling fuel assemblies were installed and checked at the reactor site in preparation for installation in the reactor.

The high performance fuel rods were first installed in the bundles and assembled in accordance with the detailed installation instructions. These assemblies were tested at 25 psig (using the EVESR breathing air supply as a pressure source) to check the process tube seals for tightness. The seals were immersed in deionized water to check for leaks. No difficulties were experienced during this work, either in installation and assembly, or in achieving satisfactory seals. Figure 7-15 shows a high performance fuel rod being attached to the process tube plug. The lower end of the Mark IV fuel bundle, with one high performance fuel rod nearly installed, is shown in Figure 7-16. The gold-plated metal O-ring used to seal this assembly can be noted in the socket of the process tube plug.

The boiling water assemblies were installed after installation of the high performance fuel rods was complete. Figure 7-17 shows a boiling water assembly being lowered into a Mark IV fuel bundle. In this picture, the boiling water assembly has been lowered approximately two-thirds of the way into the cruciform channel.

The position and orientation of the high performance fuel rods was checked during installation. Table 7-1 lists the pertinent data about these rods. Also, a record was made of the orientation of the boiling water assemblies in each fuel bundle. Since each boiling water fuel rod had a serial number, the position of each rod could be determined. This information was placed in the permanent EVESR records of the Mark IV assemblies.

7.3 FUEL BUNDLE DESIGN

7.3.1 High Performance Fuel Rods

A flux-trap design was selected for the Mark IV fuel bundles to achieve the desired power density in the superheat fuel rods. The cross section of the fuel bundle is illustrated in Figure 7-18. Because of flux limitations of EVESR, it was necessary to adopt the flux trap design to reach high power densities, even with high enrichment of the fuel. Since the flux-trap design and high enrichment required for the Mark IV design result in high "self-shielding" of the fuel, a rod cluster fuel assembly would not result in typical fuel exposure because of the extreme power differential across a fuel rod in a clustered configuration. The Mark IV design therefore provides for four high performance rods in each fuel bundle, one rod in each corner of a bundle, with individual velocity tubes and process tubes for each rod.

A detailed review of the EVESR fuel program at the time when the basic flux trap arrangement was selected for the Mark IV fuel bundles showed that all zone 1 positions of the EVESR core were committed to the Mark III program. All design and analysis subsequent to this determination were based upon the zone 3 location for Mark IV fuel bundles.

FIGURE 7-15. ATTACHING HIGH-PERFORMANCE SUPERHEAT FUEL ROD TO PROCESS TUBE PLUG

FIGURE 7-16. INSTALLING HIGH-PERFORMANCE SUPERHEAT FUEL
RODS IN MARK IV FUEL BUNDLE

FIGURE 7-17. INSTALLING BOILING WATER FUEL ASSEMBLY
IN MARK IV FUEL BUNDLE

TABLE 7-1

MARK IV FUEL BUNDLES - ASSEMBLY DATA

<u>Fuel Bundle</u>	<u>Planned Core Position</u>	<u>Boiling Water Assembly No.</u>	<u>Superheat Rod Position</u>	<u>Superheat Rod Serial No.</u>	<u>Orientation of Rod Serial No.</u>	<u>Fuel Enrichment (Percent)</u>	<u>Fuel Type</u>	<u>Superheat Rod Cladding</u>
KB-42	B-4	4	N (1)	02L50	W	50	Pellets	Incoloy-800
			E (2)	10L50 ⁽¹⁾	W	50	Powder Vi-Pak	Incoloy-800
			W (3)	01L25	W	25	Pellets	Incoloy-800
			S (4)	03L50	W	50	Pellets	Incoloy-800
KB-43	E-3	3	N (1)	06L50	W	50	Powder-Swaged	Incoloy-800
			E (2)	09L25	W	25	Powder Vi-Pak	Incoloy-800
			W (3)	05L50	W	50	Powder Vi-Pak	Incoloy-800
			S (4)	04L50	W	50	Powder Vi-Pak	Incoloy-800
KB-44	B-3	2	N (1)	24V50	E	50	Pellets	Inconel-625
			E (2)	23V50	W	50	Pellets	Inconel-625
			W (3)	22V50	W	50	Pellets	Inconel-625
			S (4)	21V25	E	25	Pellets	Inconel-625
KB-45	E-4	1	N (1)	42X25	W	25	Pellets	Hastelloy-X
			E (2)	43X50	E	50	Pellets	Hastelloy-X
			W (3)	41X50	E	50	Pellets	Hastelloy-X
			S (4)	25V50 ⁽²⁾	W	50	Pellets	Inconel-625

(1) Rod 10L50 has spacer wire welded to cladding.

(2) Rod 25V50 has turbulence promoters on cladding for improved heat transfer.

7.3.2 Boiling Water Booster

With the flux trap design selected for the high performance superheat rods, it was essential that boiling water rods be included to maintain the reactivity of the fuel bundle and to boost the flux in the central region between the high performance flux traps. The Zircaloy clad boiling fuel design used for an early loading of the Dresden reactor was adapted for the Mark IV boiling water booster. A cruciform channel is used to control flow of water and steam through the booster. A total of 28 boiling water rods, with each rod comprising an upper and lower section, are used for the boiling water booster.

Thermal-hydraulic performance of the boiling water booster fuel is given in subsection 7.3.6.

7.3.3 High Performance Fuel Rod Length

A preliminary heat transfer and pressure drop calculation showed that an unrealistically large flow annulus would be required and an unacceptable pressure drop would occur if a high performance fuel rod were the full length of the EVESR core (i. e., 60 inches) and operated at the desired peak power density of 150 kW/kg- UO_2 . The 60-inch length of the high performance rod would have required a coolant flow annulus of about 0.100-inch radial gap, with a pressure drop well above the 100 psi specified, and would have resulted in a greater steam requirement than desirable. The fueled length of the high performance rods was reduced to 40 inches and the gas plenum was arranged at the bottom of the fuel rod to reduce the pressure drop to an acceptable level.

7.3.4 Preliminary Thermal-Hydraulic Analysis

Note: Only a preliminary thermal hydraulic analysis was completed for the Mark IV fuel bundles, since the final analysis would be based upon physics calculations for the EVESR core configuration at the time of installation of Mark IV fuel bundles. The preliminary analysis was completed to ensure that the Mark IV design complied with the design objectives and criteria for the Mark IV fuel bundles.

The preliminary thermal-hydraulic analysis was based upon the mechanical configuration shown in Figure 7-19, and an axial power profile as shown in Figure 7-20. The fuel parameters for this analysis are listed in Table 7-2. Cladding temperature and pressure drop across the fuel bundle are given for the reference conditions in Figure 7-21.

TABLE 7-2

MARK IV SUPERHEAT FUEL PARAMETERS - REFERENCE DESIGN CONDITIONS

Superheat Fuel Rod Power	37.5 kW
Peak Heat Flux	615,000 Btu/h-ft ²
Peak Specific Power	150 kW/kg UO_2
Axial Power Factor	1.17
Circumferential Scallop	1.00
Steam Inlet Pressure	975 psia
Rod Diameter	0.280 inch
Cladding Thickness	0.020 inch
Cladding Material	Incoloy-800
Hydraulic Diameter	0.00904 ft
Fuel Length	40 inches

FIGURE 7-19. MARK IV FUEL GEOMETRY

FIGURE 7-20. AXIAL POWER PROFILE, MARK IV HIGH-PERFORMANCE FUEL RODS

ROD POWER - 37.5 kW

FIGURE 7-21. PERFORMANCE - MARK IV REFERENCE DESIGN

The total steam flow for a Mark IV fuel bundle will be about 2500 to 3000 lb/h, depending upon the power variation of the individual high performance rods, and the actual maximum cladding temperature selected.

7.3.5 Mark IV Fuel Performance – Reference Design

The fuel enrichments and powers for the reference design of Mark IV are listed in Table 7-3. (The values listed in this table are based upon physics calculations made early in the Mark IV program, and differ somewhat from the values listed in the physics section of this report.) These values are based on 15 MW core power, with a fully reversed EVESR control rod pattern (outer rods fully inserted and inner rods fully withdrawn), and the Mark IV fuel bundle in a zone 3 core location. The element listed as HPE-1 is located closest to the center of the core, and HPE-4 is farthest out.

TABLE 7-3

MARK IV FUEL PERFORMANCE – REFERENCE DESIGN

<u>Fuel</u>	<u>Enrichment (Percent)</u>	<u>Power (kW)</u>	<u>Peak Heat Flux (Btu/h-ft²)</u>
Boiling Water Booster	5	600	-
HPE-1	25	36.7	600,000
HPE-2	50	29.6	480,000
HPE-3	50	26.6	440,000
HPE-4	50	15.5	250,000

7.3.6 Thermal-Hydraulic Analysis – Boiling Water Fuel

The EVESR core was not designed for the purpose of irradiating boiling water fuel. Consequently, the performance of such fuel would be expected to be limited when compared to a conventional boiling water reactor. With natural circulation it is necessary to investigate the transition period following an EVESR scram followed by emergency cooling to determine that acceptable thermal-hydraulic performance is maintained throughout the transition period.

When the EVESR emergency cooling period is initiated, the reactor vessel steam pressure starts to drop rapidly, and this results in flashing or general boiling of the moderator throughout the vessel. This flashing, in turn, leads to a buildup of voids in the downpass region of the vessel and to a reduction in the hydraulic head and the recirculation rate through the core. The critical heat flux ratio might be adversely affected if the heat flux does not decrease as quickly as the flow diminishes. The analysis of this transient shows that the critical heat flux ratio does not diminish during the emergency cooling period provided reactor scram is initiated at the start of the emergency cooling period. Figure 7-22 shows the change in the heat flux and the critical heat flux ratio during the transient following an EVESR emergency cooling and reactor scram.

FIGURE 7-22. MARK IV BOILING FUEL PERFORMANCE

The initial conditions for the transient were:

Total Reactor Power Level	17 MW
Power to Moderator	8.8 MW
Mark IV Boiling Water Bundle Power	550 kW
Total Recirculation Flow	2.8×10^6 lb/h
Reactor Pressure Level	975 psi
Subcooling in Downpass	5.7 Btu/lb

Assumptions for the transient were:

1. Scram initiated with 0.2-second delay.
2. Neutron flux drops to zero at 0.5 second.
3. Reactor pressure drops 14 psi per second during the first 2.5 seconds, and then drops 5 psi per second for the next 2.5 seconds.

The change in the rate of pressure decrease is caused by eventual flashing of the moderator in the downpass and below the core where this water is initially subcooled. During the first 2.5 seconds, total flow will have a tendency to increase because of flashing which occurs above the core where the water is already at saturated temperature. Calculations show that the voids in the downpass reach about 25%, resulting in a flow reduction of about 50% during the next 2.5-second interval when flashing starts in this region. The critical transition period would be expected during this interval, when void formation in the downpass is reducing flow but heat flux has not dropped below 40%. The critical heat flux ratio does not drop below the value of two during this transient, even if one assumes that the buildup of voids in the downpass occurs without delay. The core moderator flow transient used for the analysis is shown in Figure 7-23.

7.3.7 Mark IV Velocity Tubes

The Mark IV fuel bundle, with its flux-trap arrangement to peak the power in the high performance fuel rods, offered an excellent means of obtaining additional irradiated samples of material which might be suitable for fuel cladding by utilizing the velocity tubes for this purpose. The velocity tubes would be exposed to the same high flux densities as the high performance fuel rods, at an exposure temperature in the range of 600° to 700°F. For convenience, it was decided to make the velocity tubes of the same material used for the cladding of the high performance fuel rods in the bundles. Therefore, bundles KB-42 and KB-43 have Incoloy-800, KB-44 has Inconel-625, and KB-45 has Hastelloy-X for these tubes. The tubing for these tubes was subjected to the same material specifications and quality control and receiving inspection requirements demanded of tubing for use as fuel cladding.

Use of these special materials for the velocity tubes required that a transition weld be made between each special material and Type 304 stainless steel. Service demands of the transition weld are not severe since the fuel bundle design is such that weld cracking or failure would result only in a slight by-passing of steam around the fuel section. However, a welding development program, including trial welds of actual samples and metallographic sections to examine weld structure, was undertaken to assure that satisfactory transition welds would be made. The transition weld micrographs for these materials are shown in Figures 7-24, 7-25, and 7-26. The steam velocity tube assembly includes a small pin which is welded to the

FIGURE 7-23. MARK IV BOILING FUEL TRANSIENT PERFORMANCE

FIGURE 7-24. TRANSITION WELD, INCOLOY-800
TO TYPE 304 STAINLESS STEEL

FIGURE 7-25. TRANSITION WELD, INCONEL-625
TO TYPE 304 STAINLESS STEEL

FIGURE 7-26. TRANSITION WELD, HASTELLOY-X TO
TYPE 304 STAINLESS STEEL

velocity tube to prevent the fuel orifice or a similar piece from completely blocking steam flow in case of a failure of these parts. These pins are also made of Type 304 stainless steel and require a transition weld similar to that for the velocity tube transitions. The parameters used for making these transition welds are listed below.

7.3.7.1 Transition Weld

A coil of copper foil is placed inside the tube to serve as a heat sink. For atmosphere control, helium is purged through the tube during welding. A tube expander is used to provide a snug fit of the tube into the transition piece prior to welding.

Welder	-	Vickers 0-300
Setting	-	Low range
Positioner	-	Glass lathe
Position	-	Horizontal
Speed	-	1/25 revolution per second
Current	-	12 amperes
Electrode	-	0.040-inch dia, 2% thoriated tungsten
Cup Size	-	No. 4
Gas Flow	-	10 cubic ft/h, argon

7.3.7.2 Pin Weld

No heat sink used. Purge through tube with helium for atmosphere control. Head of pin is formed to fit tube radius with a punch and die to prevent burn-through during welding. Weld is made free-hand in horizontal position.

Welder	-	Same as for transition weld
Current	-	6 amperes
Electrode	-	0.040-inch dia, 2% thoriated tungsten
Cup Size	-	No. 4
Gas Flow	-	10 cubic ft/h, argon

7.3.8 Improved Performance with Turbulence Promoters

Note: The information included in this section should be considered as preliminary. Additional experiments have been completed since the analyses reported in this section were made. See Reference (27).

A study was made to determine what changes would result from the use of turbulence promoters on the outside cladding of a high performance fuel rod. A spiral ridge or fin on the cladding would result in improved heat transfer because of increased turbulence and mixing in the film adjacent to the cladding surface. For a specific coolant flow, the pressure drop across the fuel element will be higher with the use of turbulence promoters, but the cladding temperature will be lower because of the increased heat transfer coefficient. For a given pressure drop, a lower coolant flow would occur with turbulence promoters, and consequently the cladding temperature would tend to increase. The combined effects on friction coefficient and heat transfer coefficient must be determined for a specific turbulence promoter configuration to determine final cladding temperatures at a fixed pressure drop. Likewise, with the lower coolant flow and a fixed heat input to the coolant, the coolant exit temperature would be increased.

The reference geometry and design parameters for the turbulence promoter heat transfer study are given in Table 7-4 and Figure 7-19. The data on the turbulence promoters were in the form of multipliers which were applied to the heat transfer and friction constants. The constants were applied only to the surfaces having the turbulence promoter, i. e., the high power portion of the fuel rod. Table 7-5 lists the values of the multipliers used in this study.

TABLE 7-4

MARK IV FUEL PARAMETERS FOR TURBULENCE PROMOTER STUDY

Rod Power	37.5 kW
Peak Heat Flux	615,000 Btu/h-ft ²
Peak Specific Power	150 kW/kg UO ₂
Axial Power Factor	1.17
Circumferential Scallop	1.00
Reynolds Number (600 lb/h)	183,000
Steam Velocity	280 ft/sec
Pressure	975 psi
Rod Diameter	0.280 in.
Cladding Thickness	0.020 in.
Hydraulic Diameter	0.00904 ft
Length, Fueled Section	40 in.

TABLE 7-5

MULTIPLIERS FOR HEAT TRANSFER AND FRICTION COEFFICIENTS
FOR TURBULENCE PROMOTERS

(0.0020 Fin Height)

Pitch $\left(\frac{\text{Fin Spacing}}{\text{Fin Height}} \right)$	Heat Transfer Multiplier	Friction Multiplier	*Effective Friction Multiplier
10	2.95	7.91	3.28
15	2.58	6.60	2.86
20	2.33	5.67	2.56

*Only about one-third of the friction surface has turbulence promoters; the remaining two-thirds of the surface is smooth.

The basic formula used to determine the heat transfer film coefficient was:

$$Nu = 0.019 Re^{0.8} Pr^{0.6} \left(\frac{T_B}{T_W} \right)^{0.575}$$

The heat transfer multiplier is applied to the 0.019 constant in this equation to obtain a new constant which is used in the calculation for that particular turbulence promoter design. Since the turbulence promoter is applied to the high power and hot spot region of the fuel, this calculation gives the hot spot temperature.

The basic equation used for the pressure drop calculation was:

$$f = 0.046 \left(\frac{1}{Re} \right)^{0.2}$$

The friction multiplier listed in Table 7-5 is applied only to the surface of the fuel which is 33.6% of the total friction surface for the Mark III high performance configuration. Thus, the constant of 0.046 applied to the basic friction coefficient calculation was replaced by a constant adjusted for the actual configuration.

$$\text{Adjusted Constant} = (\text{friction multiplier})(0.046)(0.336) + (0.046)(0.664)$$

The pressure drop in the saturated steam plenum, the downcomer piping and the fuel bundle downpass was calculated and an equivalent orifice coefficient was determined and added to the Mark IV calculational code. Thus, the fuel ΔP calculated and shown on Figure 7-27 is the pressure loss from the saturated steam plenum to the superheated steam plenum in the fuel bundle. At the flow rates and temperatures to be expected, the additional pressure loss from that point on to the exit from the reactor pressure vessel will be about 5 psi.

Because of the small size of the turbulence promoters needed for the small diameter characteristic of the high performance rods, there is concern that erosion, corrosion, or crud deposition might remove or block the effectiveness of the turbulence promoters during the long operating life expected from such fuel elements. By examination of Figure 7-27, it will be observed that if the rod is operated at a constant ΔP , there would be a compensating change in flow rate, which would prevent excessive cladding temperature as the effect of the turbulence promoters diminished. Of course, at a constant flow, cladding temperature would increase and pressure loss across the fuel rod would decrease, if the effectiveness of the turbulence promoters diminishes.

Figure 7-27 shows the effects of turbulence promoters with a 0.0020 height. Analysis of the effects of the turbulence promoters based upon data from tests which more closely approach the Mark IV configuration would be desirable to optimize the height and spacing of these fins. Reference (28) indicates that the effect of fin spacing is less than shown in Figure 7-27.

FIGURE 7-27. MARK IV FUEL ROD PERFORMANCE WITH 0.0020 - INCH TURBINE PROMOTERS

SECTION VIII

MARK IV EVESR PHYSICS

8.1 PHYSICS ANALYSES

Details of the calculated power distribution for the EVESR Mark IV core are given in Figures 8-1 through 8-8. Figure 8-1 illustrates the assumed core geometry and composition, in addition to the calculated power distribution, for the reversed normal control rod pattern. Figure 8-1 was synthesized from Figures 8-10 and 8-11, using the vertical power distributions from a Mark III core, three-dimensional, with reversed normal control rod pattern and outer rods 50 percent withdrawn.

Details of the power generation within the Mark IV high-performance elements are included in Figures 8-2 and 8-3. The calculated axial (vertical) power distribution is illustrated in Figure 8-2. The calculated skewing factors at the surface of the UO_2 in the superheat fuel rods are shown in Figure 8-3.

Shown in Figures 8-4 through 8-7 are details of the power generation within the Mark IV boiling water rods. Figure 8-6 illustrates the relative power generation for each BWR, calculated with the eight outer control rods 50% withdrawn (synthesized from Figures 8-4 and 8-5). Figure 8-7 shows the average vertical power distribution within the BWR section of the Mark IV fuel bundle with reversed normal control rod pattern (eight outer control rods 50% withdrawn). This figure shows a vertical peak/average of 1.84, whereas a range from 1.65 to 2.00 is expected over the total BWR section.

Finally, since it was noted that the exit voids in the BWR section of the Mark IV fuel bundle might be as high as 65%, an analysis was made to determine the effect on power distribution caused by an increase in the average voids from 37.1% to 65%. Figure 8-8 shows that the effect is essentially negligible. The axial shapes used in this synthesis do not include the effects of widely varying void content in the boiling elements which would tend to move the peak in Figure 8-9 more toward the bottom of the core and decrease its magnitude.

Similar power distribution details for the normal control rod pattern (four inner control rods 50% withdrawn) are given in Figures 8-9, 8-12, and 8-13. Figure 8-9 was synthesized from Figures 8-10 and 8-14, using the vertical power distributions from a Mark II core, three-dimensional, normal control rod pattern (Table 8-1).

Details of the generation within the Mark IV boiling water rods are given in Figures 8-12 and 8-13. Figure 8-12 illustrates the relative power generation for each BWR, calculated with the four inner control rods 50% withdrawn. This figure was synthesized from Figures 8-13 and 8-5, and Table 8-1.

The values of peak specific power for the various fuel elements and the two modes of control rod operation are summarized in Table 8-2.

**FIGURE 8-1. POWER DISTRIBUTION IN EVESR MARK IV CORE,
REVERSED NORMAL CONTROL ROD PATTERN
(OUTER RODS 50 PERCENT WITHDRAWN) –
POWER LEVEL 17.0 MW(t)**

FIGURE 8-2. ESTIMATED AXIAL POWER DISTRIBUTION IN MARK IV SUPERHEAT FUEL - ZONE 3 (TKO CASE 696)

CENTER OF CORE

1.05	0.98	1.01		1.05	0.97	1.04	OUT
1.05	0.99	1.01		1.10	1.00	1.03	IN
1.00		0.96		0.97		0.96	
1.00		0.96		1.00		0.93	
1.04	0.97	0.99		1.03	0.96	1.02	
1.03	0.97	0.99		1.04	0.93	0.97	
BWR							
1.07	0.97	1.02		1.05	0.97	1.04	
1.12	1.01	1.03		1.13	1.02	1.03	
0.99		0.94		0.97		0.96	
1.01		0.92		1.02		0.92	
1.05	0.86	1.00		1.03	0.96	1.02	
1.04	0.92	0.95		1.03	0.91	0.94	

UPPER NUMBER - ALL CONTROL RODS OUT (PDQ 700)

LOWER NUMBER - OUTER CONTROL RODS FULLY INSERTED - (PDQ 701)

FIGURE 8-3. CALCULATED VALUES OF AZIMUTHAL (CIRCUMFERENTIAL) POWER DISTRIBUTION AT UO₂ SURFACE OF MARK IV SUPERHEAT FUEL

FIGURE 8-4. RELATIVE POWER GENERATION WITHIN BOILING WATER ROD SECTION OF MARK IV FUEL BUNDLE (REVERSED NORMAL CONTROL ROD PATTERN)

FIGURE 8-5. RELATIVE POWER GENERATION WITHIN BOILING WATER ROD SECTION OF MARK IV FUEL BUNDLE (ALL RODS FULLY WITHDRAWN)

* BASED ON MEASUREMENT VERSUS CALCULATION, A CORRECTION FACTOR OF 1.10 ± 0.05 SHOULD BE APPLIED.

FIGURE 8-6. RELATIVE POWER GENERATION WITHIN BOILING WATER ROD SECTION OF MARK IV FUEL BUNDLE (REVERSED NORMAL CONTROL ROD PATTERN WITH EIGHT OUTER CONTROL RODS 50 PERCENT WITHDRAWN)

**FIGURE 8-7. ESTIMATED AXIAL POWER DISTRIBUTION IN BOILING
WATER RODS OF MARK IV IN ZONE NO. 3**

ALL CONTROL RODS OUT
POWER LEVEL: 17.0 MW (t)

624.7 (a)	708.0	446.7
626.7 (b)	725.0	465.7
(MARK III)		
618.3	604.7	362.3
552.3	613.6	376.3
(MARK IV)		
508.8	376.4	
508.0	382.4	

- (a) 37.1% VOIDS IN BOILING WATER RODS OF
MARK IV (PDQ CASE 585)
- (b) 65.0% VOIDS IN BOILING WATER RODS OF
MARK IV (PDQ CASE 706)

25.0		26.2
24.5		26.0
519.9 455.2		
25.9		21.3
25.4		21.2

MARK IV FUEL BUNDLE POWER
CORRECTED TO LENGTH AND
SITE OF THE SUPERHEAT FUEL
RODS (FACTOR 0.77)

**FIGURE 8-8. EFFECT OF HIGH VOID CONTENT IN BOILING WATER SECTION OF
MARK IV FUEL BUNDLE ON POWER IN MARK IV SUPERHEAT FUEL RODS**

0.98	0.97
136.3	135.1
1.07	0.98
149.5	137.5

1.21	1.17	0.97
93.0	90.0	74.8
1.07	0.88	0.93
82.3	67.8	71.2
0.99	0.88	0.89
75.9	67.4	68.7

1.06	1.12	0.92
51.2	54.0	44.0
1.00	1.02	0.84
47.9	49.0	40.3
1.13	0.98	0.93
54.2	47.2	44.9

1.01	1.02
30.1	30.2
582.9	
1.00	0.97
29.8	28.8

1.15	0.92	0.91
81.0	64.8	63.8
1.01	1.01	1.00
70.8	70.9	70.3
1.02	1.03	0.95
71.5	72.1	66.5

1.21	1.06	0.83
51.6	45.3	35.4
0.99	0.96	0.90
42.2	40.9	38.5
1.18	0.91	0.97
50.2	38.8	41.2

1.21	1.32	1.35
61.5	67.1	68.7
0.91	0.81	0.93
46.3	41.3	47.3
0.80	0.87	0.80
40.6	44.5	40.8

1.24	0.99	1.16
54.0	43.0	50.6
1.07	0.96	0.90
46.6	41.8	39.1
0.83	0.90	0.95
36.2	39.2	41.5

MARK IV SUPERHEAT FUEL RODS

ELEMENT POWER	AXIAL P/A*
30.1	1.42
30.2	1.32
29.8	1.29
28.8	1.23

558.4 1.05	691.1 1.30	432.7 0.81	kW
701.8 1.32	631.7 1.19	384.1 0.72	kW
458.1 0.86	392.0 0.74		kW

* BASED ON MEASUREMENT VERSUS CALCULATION, A
CORRECTION FACTOR OF 1.10 ± 0.05 SHOULD BE APPLIED.

**FIGURE 8-9. POWER DISTRIBUTION IN MARK IV CORE (NORMAL CONTROL ROD PATTERN
WITH INNER RODS 50 PERCENT WITHDRAWN)**

3%	3%	13.3%	7.8%
3%	3%		
0	0	11%	6%
37	1%		
0	0	10%	6%

VOID DISTRIBUTION USED

1.23	1.37	0.77
576.39	643.74	360.58
1.41	1.13	0.66
661.28	532.01	308.22
0.77	0.66	
358.77	308.95	

BUNDLE POWER (kW) AND RELATIVE BUNDLE POWER TO CORE AVERAGE

FIGURE 8-10. MARK IV CORE POWER DISTRIBUTION, ALL RODS OUT

10%	10%	16.7%	4.4%
8%	8%		
0	0	5.8%	2.1%
0	0		
4.4%	2.4%		

VOID DISTRIBUTION USED

1.74 814.21	1.50 705.37	0.71 331.04
1.58 739.44	0.79 371.72	0.51 237.20
0.68 319.44	0.49 231.57	

POWER BY BUNDLE (kW) AND BUNDLE POWER RELATIVE TO CORE AVERAGE

FIGURE 8-11. MARK IV CORE POWER DISTRIBUTION, OUTER RODS IN

FIGURE 8-12. RELATIVE POWER GENERATION WITHIN BOILING WATER ROD SECTION OF MARK IV FUEL BUNDLE (NORMAL CONTROL ROD PATTERN WITH INNER RODS 50 PERCENT WITHDRAWN)

FOUR INNER (TYPE-304 STAINLESS STEEL) CONTROL RODS IN
(PDQ 396)

FIGURE 8-13. RELATIVE POWER GENERATION WITHIN BOILING WATER ROD SECTION OF MARK IV FUEL BUNDLE (FOUR INNER CONTROL RODS FULLY INSERTED)

0%V	0%V	13.3%	11.5%
0%V	0%V		
0	0	13.5%	10.0%
0	0		
8.5%	9%		

VOID DISTRIBUTION USED

0.69 325.03	1.17 546.38	0.89 416.46
1.28 601.93	1.28 599.43	0.82 385.72
1.01 473.43	0.86 402.12	

POWER BY BUNDLE (kW) AND BUNDLE POWER RELATIVE TO CORF AVERAGE

FIGURE 8-14. MARK IV CORE POWER DISTRIBUTION, INNER RODS IN

TABLE 8-1

AXIAL POWER DISTRIBUTIONS MIDLIFE

EVESR Fuel Element	Position Above Bottom Fuel Plane (inches)										
	5.0	14.0	21.0	27.0	32.5	37.0	41.0	45.0	49.0	53.0	57.5
1	0.26	0.57	0.82	1.11	1.51	1.75	1.74	1.61	1.39	1.07	0.69
2	0.27	0.59	0.84	1.12	1.50	1.73	1.72	1.59	1.37	1.06	0.70
3	0.30	0.64	0.91	1.17	1.48	1.66	1.64	1.52	1.31	1.01	0.69
4	0.25	0.53	0.76	1.06	1.51	1.81	1.80	1.66	1.43	1.11	0.72
5	0.28	0.60	0.87	1.13	1.50	1.72	1.70	1.57	1.35	1.05	0.69
6	0.28	0.61	0.87	1.14	1.50	1.71	1.69	1.56	1.34	1.04	0.68
7	0.25	0.54	0.77	1.06	1.51	1.80	1.78	1.65	1.42	1.10	0.71
8	0.29	0.62	0.88	1.15	1.49	1.70	1.68	1.55	1.33	1.03	0.68
9	0.30	0.66	0.94	1.20	1.49	1.65	1.62	1.50	1.28	0.99	0.64
10	0.37	0.77	1.05	1.26	1.45	1.54	1.50	1.38	1.19	0.92	0.59
11	0.29	0.63	0.90	1.16	1.50	1.69	1.66	1.53	1.32	1.02	0.68
12	0.33	0.70	0.99	1.23	1.47	1.59	1.56	1.44	1.24	0.96	0.65
13	0.37	0.78	1.05	1.26	1.44	1.53	1.49	1.37	1.18	0.92	0.60
14	0.38	0.80	1.08	1.28	1.44	1.51	1.46	1.34	1.16	0.90	0.59
15	0.40	0.82	1.10	1.29	1.43	1.48	1.43	1.32	1.13	0.88	0.60
16	0.32	0.68	0.97	1.22	1.48	1.62	1.59	1.47	1.26	0.97	0.64
17	0.38	0.78	1.06	1.27	1.44	1.52	1.48	1.36	1.17	0.91	0.61
18	0.38	0.79	1.08	1.28	1.44	1.51	1.47	1.35	1.16	0.90	0.59
19	0.39	0.81	1.10	1.30	1.44	1.49	1.45	1.33	1.15	0.89	0.58
20	0.40	0.82	1.11	1.30	1.43	1.48	1.43	1.31	1.13	0.88	0.60
21	0.40	0.82	1.11	1.30	1.43	1.47	1.43	1.31	1.13	0.88	0.59
22	0.38	0.79	1.08	1.29	1.45	1.51	1.47	1.35	1.16	0.90	0.58
23	0.39	0.81	1.10	1.30	1.44	1.50	1.45	1.33	1.14	0.89	0.58
24	0.39	0.81	1.10	1.30	1.44	1.49	1.45	1.33	1.14	0.89	0.57
25	0.39	0.81	1.11	1.31	1.44	1.49	1.44	1.32	1.14	0.88	0.57
26	0.39	0.82	1.11	1.31	1.44	1.48	1.43	1.31	1.13	0.88	0.59
27	0.39	0.81	1.11	1.31	1.44	1.49	1.43	1.32	1.13	0.88	0.59
28	0.39	0.81	1.10	1.30	1.44	1.49	1.44	1.32	1.14	0.88	0.61
29	0.40	0.81	1.11	1.30	1.43	1.48	1.42	1.31	1.12	0.87	0.61
30	0.39	0.81	1.11	1.31	1.44	1.48	1.43	1.31	1.13	0.88	0.60
31	0.39	0.81	1.11	1.31	1.44	1.48	1.43	1.31	1.13	0.88	0.60
32	0.40	0.82	1.11	1.31	1.43	1.47	1.41	1.30	1.12	0.87	0.63
33	0.39	0.81	1.11	1.31	1.44	1.49	1.44	1.33	1.14	0.88	0.61
34	0.38	0.80	1.10	1.30	1.44	1.49	1.44	1.33	1.14	0.89	0.60
35	0.39	0.80	1.10	1.31	1.44	1.40	1.44	1.32	1.14	0.89	0.61
36	0.39	0.80	1.10	1.31	1.44	1.48	1.44	1.32	1.14	0.89	0.60
37	0.39	0.80	1.10	1.31	1.44	1.48	1.43	1.32	1.14	0.89	0.60
38	0.39	0.80	1.10	1.31	1.43	1.47	1.43	1.31	1.13	0.89	0.62
39	0.39	0.80	1.10	1.31	1.44	1.47	1.43	1.32	1.14	0.89	0.62

TABLE 8-2

PEAK SPECIFIC POWER VERSUS CONTROL ROD PATTERN*MARK IV CORE

(Mark III in Zone 1)
(Mark IV in Zone 3B)
(Mark II Elsewhere)

Power Level: 17.0 MW(t)

Control Rod Pattern

	<u>Normal Pattern</u> (Inner Rods 50% Withdrawn)		<u>Reversed Normal Pattern</u> (Outer Rods 50% Withdrawn)	
	<u>Power, kW</u>	<u>Position</u>	<u>Power, kW</u>	<u>Position</u>
<u>Average Power in Hottest Rod</u>				
Mark II Rod	93.0		119.2	
Mark III Rod (assuming $1.18 \times 1/7 \times P_{\text{Cluster}}$)*	24.0		32.3	
Mark IV Rod	30.1		34.7	
<u>Average Specific Power in Hottest Rod</u>				
Mark II Rod (8.8 kg of UO ₂ /Rod)	10.6 kW/kg of UO ₂		13.5 kW/kg of UO ₂	
Mark III Rod (1.24 kg of UO ₂ /Rod)	19.3 kW/kg of UO ₂		26.0 kW/kg of UO ₂	
Mark IV Rod (0.304 kg of UO ₂ /Rod)	99.0 kW/kg of UO ₂		114 kW/kg of UO ₂	
<u>Axial Peaking Factors</u>				
Mark II	1.00		1.66	
Mark III	1.75		1.66	
Mark IV	1.42		1.43	
<u>Peak Specific Power (Without Skewing)</u>				
Mark II	19.0 kW/kg of UO ₂		22.4 kW/kg of UO ₂	
Mark III	33.8 kW/kg of UO ₂		43.2 kW/kg of UO ₂	
Mark IV	141 kW/kg of UO ₂		163 kW/kg of UO ₂	
<u>Ratio (Reversed Normal versus Normal)</u>				
Mark II		1.18		
Mark III		1.28		
Mark IV		1.16		

* A correction factor of 0.95 is applied to correct for an error in the 3rd neutron group cross sections.

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