(12) INTERNATIONAL APPLICATION PUBLISHED UNDER THE PATENT COOPERATION TREATY (PCT)

(19) World Intellectual Property Organization
International Bureau

(43) International Publication Date
19 September 2002 (19.09.2002)

PCT

(10) International Publication Number
WO 02/072902 A1

(51) International Patent Classification[7]: C22C 16/00, C22F 1/18, G21C 3/356

(21) International Application Number: PCT/US02/06158

(22) International Filing Date: 27 February 2002 (27.02.2002)

(25) Filing Language: English

(26) Publication Language: English

(30) Priority Data:
09/808,072 14 March 2001 (14.03.2001) US

(71) Applicant: WESTINGHOUSE ELECTRIC COMPANY LLC [US/US]; P.O. Box 355, Pittsburgh, PA 15230-0355 (US).

(72) Inventor: GARDE, Anand, M.; 2 Sherwood Lane, West Simsbury, CT 06092 (US).

(74) Agents: SPADACENE, Joseph, C. et al.; Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, PA 15230-0355 (US).

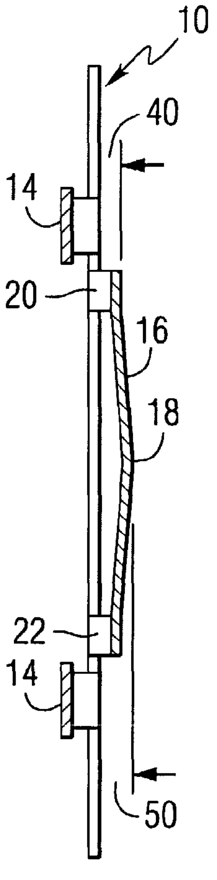
(81) Designated States (national): AE, AG, AL, AM, AT, AU, AZ, BA, BB, BG, BR, BY, BZ, CA, CH, CN, CO, CR, CU, CZ, DE, DK, DM, DZ, EC, EE, ES, FI, GB, GD, GE, GH, GM, HR, HU, ID, IL, IN, IS, JP, KE, KG, KP, KR, KZ, LC, LK, LR, LS, LT, LU, LV, MA, MD, MG, MK, MN, MW, MX, MZ, NO, NZ, OM, PH, PL, PT, RO, RU, SD, SE, SG, SI, SK, SL, TJ, TM, TN, TR, TT, TZ, UA, UG, UZ, VN, YU, ZA, ZM, ZW.

(84) Designated States (regional): ARIPO patent (GH, GM, KE, LS, MW, MZ, SD, SL, SZ, TZ, UG, ZM, ZW), Eurasian patent (AM, AZ, BY, KG, KZ, MD, RU, TJ, TM),

[Continued on next page]

(54) Title: ZIRCONIUM ALLOY FOR ADVANCED NUCLEAR APPLICATIONS

(57) Abstract: A zirconium alloy useful as a component (10, 14, 16) in a nuclear fuel assembly (100), particularly as a spring strip (16), said component having been cold worked and having a composition consisting essentially of, in weight %: 0.5 to 1.5 Sn; 0.2 to 0.8 Fe; 0.1 to 0.3 O; with the balance Zr, and where no more than 0.02 Cr; 0.01 Nb and 0.007 Ni are present in the composition.



European patent (AT, BE, CH, CY, DE, DK, ES, FI, FR, GB, GR, IE, IT, LU, MC, NL, PT, SE, TR), OAPI patent (BF, BJ, CF, CG, CI, CM, GA, GN, GQ, GW, ML, MR, NE, SN, TD, TG).

Published:

— *with international search report*

For two-letter codes and other abbreviations, refer to the "Guidance Notes on Codes and Abbreviations" appearing at the beginning of each regular issue of the PCT Gazette.

ZIRCONIUM ALLOY FOR ADVANCED NUCLEAR APPLICATIONS

BACKGROUND OF THE INVENTION

Field of the Invention

This invention relates to a new Zr-Sn-Fe-O alloy which is essentially free of Cr, Nb and Ni for use as a component in a fuel assembly, for example, as a low cold worked grid spring associated with a nuclear fuel rod assembly.

Background Information

In pressurized water reactor (PWR) nuclear fuel assemblies, zirconium alloy grids are used as a structural support for the nuclear fuel rods. The zirconium alloy commonly used usually contains, in wt.%: about 98.3 Zr; 1.3 Sn; 0.2 Fe; 0.1 Cr; and 0.1 O (Zircaloy-4 material). The grids are made of grid cells formed by welding zirconium alloy strips. Each grid cell has springs that contact the fuel rod and provide rod constraint. As the fuel assembly is irradiated, conventional grid springs relax due to thermal and irradiation stress relaxation and the fuel cladding moves initially away from the spring due to cladding creepdown. This can lead to the development of gaps between the spring and cladding tube. Under high coolant flow velocity conditions, such gaps can lead to vibratory sliding contact between the grid spring and the fuel cladding. This vibratory sliding contact can lead to wear of the fuel cladding, sometimes progressing to penetration of the cladding tube wall. It is thus highly desirable to avoid development of gaps between the grid spring and cladding to decrease the probability of fuel failure.

With the objective of avoiding development of gaps between the fuel cladding and grid strip springs, it is desirable to promote the inward movement of the grid spring towards the fuel cladding as irradiation proceeds, to maintain a hard contact between the grid spring and the cladding. Since cold working of zirconium alloy microstructure promotes irradiation growth, a cold worked I-spring was patented for grid applications in U.S. Patent Specification No.5,331,678 (Hatfield, et al.), where from about 1% to 7% cold working was used. Hatfield, et al., did not recognize that with only this minimum cold working, the usual Zircaloy spring material would still be subject to a negative irradiation growth, that is, contraction, rather than a positive irradiation growth, that is, elongation.

It is known that the irradiation growth behavior of annealed Zircaloy-2 with small amounts of cold work introduced prior to irradiation, exhibits growth strain transients, and such transients may produce growth strains in opposite direction compared to the expected growth strain based on the texture considerations alone as taught by R. A. Holt, J. Nucl. Materials 159, 1988, p. 310-338. Therefore, an I-spring made of Zircaloy with a small amount of cold work would not exhibit positive irradiation growth strain necessary for the permanent inward movement of a spring in the grid cell at low burnups.

An advanced zirconium alloy more suitable for this application (grid spring application) is badly needed. While other compositions of zirconium are known to provide corrosion resistance at over 350°C in lithiated water, such as those taught in U.S. Patent Specification No. 09/506913, filed on February 18, 2000 (Comstock, et al., Docket No. ARF 99-001C), which utilized from 0.6 to 2.0 wt.% Nb. Another advanced zirconium alloy having excellent corrosion resistance is taught in U.S. Patent Specification 5,278,882 (Garde, et al.), which utilized up to 0.4 wt.% Cr and up to 0.06 wt.% Ni. This material would have stable second phase particles and not significant irradiation growth. Neither of these inventions provide the requirements mentioned above as needed.

SUMMARY OF THE INVENTION

Therefore, it is a main object of this invention to provide an advanced zirconium alloy that exhibits permanent inward movement even if cold worked only from about 1% to 7%, thus eliminating expensive additional cold working steps and/or critical control of the degree of small cold work introduced in the I-spring.

These and other needs are met by providing a zirconium alloy useful as a component in a nuclear fuel assembly, said component having been cold worked and having a composition consisting essentially of, in weight %: 0.5 to 1.5 Sn; 0.2 to 0.8 Fe; 0.1 to 0.3 O; with the balance Zr, and no more than 0.02 Cr; 0.01 Nb and 0.007 Ni are present in the composition.

The invention also resides in a nuclear fuel assembly grid cell comprising a plurality of grid strips surrounding a fuel rod where at least one grid strip has an attached, associated spring strip contacting the fuel rod and helping to hold it in place, said spring strip being cold worked and having a composition consisting essentially of, in weight %: 0.5 to 1.5 Sn; 0.2 to 0.8 Fe; 0.1 to 0.3 O; with the balance Zr, and no more than 0.02 Cr; 0.01 Nb and 0.007 Ni are present in the composition. This composition will initially contain coarse second phase particles of Zr-Fe having sizes up to 3.5 micrometers, preferably from 0.2-3.5 micrometers, rough diameter which are not stabilized by Cr, Nb or Ni and which will dissolve substantially but not completely after irradiation. The component or spring strip need not be cold worked more than 10% to be effective. The preferred range of cold working is from about 5% to 10%, although a very useful range is from 3% to 10%.

BRIEF DESCRIPTION OF THE DRAWINGS

For a better understanding of the invention, reference may be made to the exemplary embodiments shown in the accompanying drawings, in which:

Fig. 1 is a sectional view of a grid strip containing the associated spring strip of this invention;

Fig. 2 is a plan view of a portion of a nuclear fuel assembly grid cell, here containing a plurality of grid strips surrounding a fuel rod where several grip strips have an associated spring strip;

Fig. 3 is a section view taken along line 3-3 of Fig. 2;

Fig. 4 is an idealized magnified view of the microstructure of the spring strip prior to irradiation growth, showing coarse second phase Zr-Fe particles in the matrix;

Fig. 5 shows the expanded microstructure of idealized Fig. 4 after positive irradiation growth where the Zr-Fe particles, though smaller, still remains in the matrix.

Fig. 6 shows comparative graphs of Zr alloy cell dimensions as a function of neutron fluence;

Fig. 7 shows comparative graphs of PWR corrosion resistance of non-heat flux strip material as measured by oxide thickness on different Zr alloy samples where a best estimate of a Zirc 4 strip (isothermal model) is shown as the line 200;

Fig. 8 shows comparative graphs of PWR hydrogen uptake as a function of the oxide thickness plotted in Fig. 7;

Fig. 9 shows comparative graphs of irradiation growth strain average, after one-cycle irradiation, of I-spring sample alloys, as a function of I-spring cold working prior to irradiation; and

Fig. 10 shows comparative graphs of I-spring growth strain average, after two-cycle irradiation, for various Zr alloy samples.

DESCRIPTION OF THE PREFERRED EMBODIMENTS

Figs. 1-3 show a portion of a nuclear fuel assembly 100 having a variety of metal components, including grid strips 10, which represents one of a plurality of inter-engaged strips that would make up an egg-crate type fuel assembly grid with cells as shown most clearly in Fig. 2 for supporting a plurality of nuclear fuel rods 12. Each strip 10 is initially sized and annealed as a substantially rectangular flat plate having length, height, and width dimensions. The plate is then stamped to form a plurality of cut outs, slots, and projections. The present invention is preferably implemented with components, such as strips 10 and spring 16, that are made from a special zirconium alloy.

A plurality of strips 10 can be interengaged to form well-known egg-crate configurations. Each strip which forms a cell wall has fuel rod support features or structure. In a preferred embodiment there are three cold formed fuel rod supports, upper and lower members 14, and spring 16 per cell wall, located respectively in upper, central, and lower regions of the cell wall. The support structure in each of the upper and lower regions includes a relatively stiff, arched stop members 14 which project in a first direction and a support structure 16, which acts as a spring, in the central region. The spring is a relatively soft, arched spring, which projects in a second direction opposite the first direction of stops 14.

In accordance with the invention, the spring 16 is meant to include spaced apart pedestals 20, 22 or similar projections, as taught in U.S. Patent Specification No. 5,331,678. The spring 16 is resilient and extends between and is rigidly supported by the pedestals 20 and 22 so as to project in the second direction beyond the projection of the pedestals. The pedestals 20 and 22 project from the central region base area a first distance 40, and the spring 16 has a crown 18 which projects from the central region base area a second distance 50 which is less than twice the first distance.

All the fuel rod components including the support features as described in connection with Figs. 1-3 can be formed during a single stamping operation, which cold-works the material. The projecting stop structures 14 and 22 necessarily experience a certain amount of straining (cold working) during formation. The more highly strained portions of the spring 16 undergo greater elongation and relaxation during exposure to radiation in the reactor core.

The strips of the type shown in Figs. 1-3 are assembled into an egg-crate structure that results in the creation of grid cells with the geometry shown best in Fig. 2. Insertion of the fuel rod 12 into a grid cell is shown in Fig. 2 and partially in Fig 3. Insertion of the fuel rod 12 into the cell, as shown in Figs. 2 and 3, deflects spring 16 and thus preloads the rod against the stops 14. The preload prevents relative motion between the rod and grid during handling and shipment.

Very importantly, the alloy of this invention, which can be the composition of any of the components shown in Figs. 1-3, but is particularly useful as the spring 16, has unstable, coarse-sized second phase particles (SPPs) that tend to dissolve with irradiation damage resulting in a high positive irradiation growth strain in the strip longitudinal direction. The growth strain is positive right from the start of irradiation without any growth strain transient in the opposite direction. At the same time the alloy has excellent in-reactor corrosion resistance and low hydrogen uptake, consistent with the other grid material requirements for desirable in-PWR fuel performance.

Fig. 4 illustrates in an idealized way, a high-powered magnified view of the second phase particles 30 in a Zr-Sn-Fe-O matrix 32 after minor cold working and before irradiation. A large amount of very small particles (unnumbered) are shown within matrix 32. Fig. 5 illustrates in an idealized way another high-powered magnified view of the growth of the matrix 32 and loss of size of second phase particles 32, which, however, still remain to a certain extent and are not completely dissolved by the irradiation experienced during operation in a nuclear environment. The particles 30 are not stable as is seen by comparison of Figs. 4 (larger particles) and 5 (smaller particles).

The zirconium alloy of this invention has high irradiation growth in cold worked state, excellent in-reactor corrosion resistance, low hydrogen uptake and good creep resistance and is designed for nuclear fuel assembly components, especially the grid strip spring material. Without cold work, the irradiation growth is low. Thus, it is possible to control the extent of irradiation growth of this alloy by controlling the cold work in the alloy. It can be used for single material transverse stamped grids where the I-spring length is oriented in the strip rolling (longitudinal) direction. Such grids will have lateral grid dimensional stability due to low irradiation growth in the strip transverse direction. The alloy can also be used for longitudinal stamped grids with two different alloys used for the grid cell walls and the springs. In such "bimetallic" grids, the proposed high growth alloy can be used for the cold worked springs stamped with strip rolling direction along the spring length. A different low growth alloy (with lower Fe content and with additions of Nb, Cr or Ni) can be used for the grid walls.

Coarse second phase particles ("SPP"), shown as 30 in Fig. 4, are preferred in the zirconium alloy of this invention so that dissolution of the SPP occurs over a wide neutron fluence range. To achieve high irradiation growth strain, the iron dissolution needs to continue as more point defects are generated due to irradiation damage. Coarse SPP containing iron are desirable for this purpose. The alloy impurity levels of Nb, Cr and Ni are to be kept as low as possible since incorporation of these elements in the SPP increase the irradiation stability of SPP and thereby decrease the irradiation growth strain of the alloy. It is critical that no more than 0.02 wt.% Cr; 0.01 wt.% Nb; or 0.007 wt.% Ni are present in the composition. Table 1 below describes the alloy composition of this invention:

TABLE I

Alloying Element	Range by wt.%	Preferred Aim Point
(Tin) Sn	0.5-1.5%	0.3% to 0.6%
(Iron) Fe	0.2-0.8%	0.3% to 0.5%
(Oxygen) O	0.1-0.3%	0.20% to 0.25%
(Zirconium) Zr	balance	balance
SPP Size, micrometers	0.2-3.5	0.3-2.0

Important features of the Zr alloy of this invention include:

(1) Growth strain transients at the start of irradiation that could decrease the irradiation growth stain are avoided by adjusting the alloy composition and processing to produce coarse iron bearing second phase particles (SPP) in the as-fabricated strip microstructure.

(2) The iron bearing SPP without the stabilizing elements such as Nb, Cr and Ni are unstable under neutron irradiation. The iron dissolution from such coarse particles occurs over a wide fluence range. The rate of iron dissolution keeps up with the rate of point defect generation due to neutron bombardment thereby enhancing irradiation growth strain of the alloy, probably by

promoting formation of c-dislocation vacancy loops starting at low fluences.

(3) Irradiation growth is enhanced by controlling composition and processing of the alloy without degradation of corrosion resistance and hydrogen uptake.

(4) Intrinsic irradiation growth of this alloy is low. It is possible to increase the growth by introducing small cold work in the alloy.

EXAMPLE

Experimental capsules containing strings of single grid cells of three zirconium alloys were irradiated in a PWR for two 2-year duration cycles. The chemical composition of the strip alloys investigated are listed in Table 2 below:

TABLE 2

Alloy	Alloying Element, wt.%					
	Sn	Fe	Cr	Nb	Ni	Oxygen
Zr-Sn-Fe-O	0.50	0.40	--	--	--	0.22
*Zircaloy-4	1.31	0.22	0.12	--	--	0.11
*Zr-Sn-Nb-Fe-Ni	0.90	0.50	--	1.05	0.1	0.10

	SPP Size, micrometers		
	Average	Maximum	Minimum
Zr-Sn-Fe-O	0.42	3.40	0.2
*Zircaloy-4	0.17	.81	
*Zr-Sn-Nb-Fe-Ni	0.62	2.90	

* = Comparative Examples

Each cell had four I-springs on the four walls of each grid cell. The clearance dimension between the springs (cell size) was measured using a rod gauge prior to irradiation and after irradiation in a hot cell. The change in cell dimensions as a function of neutron fluence is presented in Fig. 6. Grid cells with I-springs made of Zircaloy-4 (the triangle points) show either no change in the grid cell size or a moderate decrease in the grid cell size with increasing neutron fluence. The largest decrease in the grid cell size was observed in cells with Zr-Sn-Fe-O alloy (the round 400 points) I-springs with higher cold work. The low cold work (about 1% cold work) data points for Sn-Fe-Zr-O alloy in Figure 6 (points designated as 300) show low growth, confirming the low growth of this alloy intrinsically without cold work. The data in Fig. 6 show that Zr-Sn-Fe-O alloy (the round 400 points with high cold work (about 7% cold work)) I-spring would provide the best constraint to the fuel rod during irradiation.

The in-PWR corrosion resistance of the different grid strip alloys, as measured by the oxide thickness on different alloy strips, is shown in Figure 7. The corrosion resistance of Zr-Sn-Fe-O alloy is similar to that of Zircaloy-4. The best estimate of the Zircaloy-4 setup (isothermal model) is shown as line 200. The in-PWR hydrogen uptake of different strip alloys as a function of the oxide thickness is plotted in Figure 8. The hydrogen uptake of Zr-Sn-Fe-O alloy is similar to all the Zr alloys investigated and is expected to be less than 10% at high oxide thicknesses.

The irradiation growth strain of the I-spring after one 2-year irradiation cycle exposure is plotted as a function of I-spring cold work (introduced by deformation twinning prior to irradiation) in Figure 9. Cold work can be introduced by different techniques such as tensile deformation, punching, pressing or rolling. The deformation can occur by either dislocation motion or deformation twinning depending on the strain rate and temperature of deformation. The preferred means of introducing the cold work is the room temperature deformation at a strain rate such that deformation twins are introduced by the cold work. Presence of deformation twins in the cold worked spring is desirable to achieve positive irradiation growth. While the Zr-Sn-Fe-O alloy shows an increase of growth strain with increasing cold work for cold work levels below 7%, for the other two comparative alloys the growth strain at these low levels of cold work, decreases with increasing cold work. The growth strain for the cell wall region can be estimated from the intercept of the curve extension for an alloy to the zero cold work vertical axis in Figure 9. Difference between the growth strain of the spring (with a designed cold work level) and the cell wall (estimated by the intercept on the vertical axis) gives a measure of inward motion of the spring within the cell. The data in Figure 9 shows that for the one-cycle exposure, Zr-Sn-Fe-O material provides maximum inward movement of the I-spring. The other two materials do not show inward motion of I-spring for cold work levels of 1 and 7% (actually they show an outward motion). Zircaloy-4 shows a smaller inward motion of spring when the degree of cold work is increased to 10%. Thus only Zr-Sn-Fe-O material show positive growth at very small cold work levels less than 7%. The other two materials show a reversal of growth at very small cold work levels (<7%).

The I-spring growth strain after two 2-year irradiation cycles is plotted as a function of cold work in Figure 10. In Figure 10, the Zr-Sn-Fe-O alloy shows the strongest dependence of growth strain on the degree of cold work among the three alloys investigated. In other words, the benefit of cold work in increasing the irradiation growth strain of the I-spring is maximum for the Zr-Sn-Fe-O alloy of this invention.

The microstructures of the three zirconium alloys were characterized before and after irradiation regarding the size and composition of the second phase particles (SPP). The Zircaloy-4 as-fabricated microstructure contained fine (SPP size up to 0.81 micrometers) particles of composition $Zr(Fe-Cr)_2$. These SPP did not dissolve easily after irradiation. After two irradiation cycles, some of these SPP were present in the irradiated microstructure. The as-fabricated microstructure of Zr-Sn-Nb-Fe-Ni alloy contained particles up to 2.9 micrometers in size and the SPP contained Zr-Nb-Fe-Ni. Moreover, these SPP were also very stable with respect to the irradiation damage. Most of these particles were present in this alloy microstructure after two irradiation cycle exposure.

The as-fabricated microstructure of the Zr-Sn-Fe-O alloy of this invention contained coarse (size up to 3.4 micrometers) SPP containing Zr-Fe. Moreover, these particles easily dissolved, although not completely, in the matrix due to irradiation damage. In a metallograph at 500x magnification, only very few SPP were still present in the alloy microstructure after one irradiation cycle exposure. The number of SPP present in the two-cycle exposure microstructure were even fewer. Metallographic examination results showed that the second phase particles dissolve with irradiation damage significantly (although they do not disappear entirely) only in Zr-Sn-Fe-O alloy and not significantly in the microstructure of the other two alloys.

Important conclusions derived from the above experiment are: (1) Zr-Sn-Fe-O alloy I-springs will provide the best support to fuel cladding by movement of the I-spring towards the fuel cladding due to positive irradiation growth strain in the cold-worked I-spring. (2) Small amounts of cold work can most efficiently induce positive irradiation growth strains in the I-spring of Zr-Sn-Fe-O alloy. (3) A positive irradiation growth strain at all fluence levels is obtained by the controlled dissolution of SPP with increasing fluence. Coarse Zr-Fe SPP in the as-fabricated microstructure are most suitable for this purpose. (4) Additions of Nb, Cr or Ni to the Zr-Sn alloy increase the stability of SPP with respect to the neutron irradiation damage and thereby decrease the irradiation growth of the alloy. (5) The Zr-Sn-Fe-O alloy has excellent in-PWR corrosion resistance, low hydrogen uptake and low intrinsic irradiation growth in the recrystallized structure without cold work.

It should be understood that the present invention may be embodied in other forms without departing from the spirit or essential attributes thereof, and accordingly, reference should be made to both the appended claims and to the foregoing specification as indicating the scope of the invention.

What is Claimed is:

1. A zirconium alloy useful as a component in a nuclear fuel assembly, said component having been cold worked and having a composition consisting essentially of, in weight %: 0.5 to 1.5 Sn; 0.2 to 0.8 Fe; 0.1 to 0.3 O; with the balance Zr, and where no more than 0.02 Cr; 0.01 Nb and 0.007 Ni are present in the composition.

2. The zirconium alloy component of Claim 1, wherein the composition will initially contain coarse second phase particles of Zr-Fe having sizes up to 3.5 micrometers rough diameter which are not stabilized by Cr, Nb or Ni and which will dissolve easily after irradiation.

3. The zirconium alloy component of Claim 1, wherein the component is not cold worked more than 10%.

4. The zirconium alloy component of Claim 1 wherein the cold working is by deformation twinning.

5. The zirconium alloy of Claim 1 having a composition consisting of in weight %: 0.5 to 1.5 Sn; 0.2 to 0.8 Fe; 0.1 to 0.3 O; with the balance Zr, and where no more than 0.02 Cr; 0.01 Nb and 0.007 Ni are present in the composition.

6. The zirconium alloy of Claim 1 having a composition consisting essentially of in weight %: 0.3 to 0.6 Sn; 0.3 to 0.5 Fe; 0.2 to 0.25 O with the balance Zr.

7. The zirconium alloy of Claim 2 wherein the second phase particles have a size between 0.2 and 3.5 micrometers.

8. The zirconium alloy of Claim 2 wherein the second phase particles have a size between 0.3 and 2.0 micrometers.

9. A nuclear fuel assembly grid cell comprising a plurality of grid strips surrounding a fuel

rod where at least one grid strip has an attached, associated spring strip contacting the fuel rod and helping to hold it in place, said spring strip being cold worked and having a composition consisting essentially of, in weight %: 0.5 to 1.5 Sn; 0.2 to 0.8 Fe; 0.1 to 0.3 O; with the balance Zr, and where no more than 0.02 Cr; 0.01 Nb and 0.007 Ni are present in the composition.

10. The nuclear fuel assembly of Claim 9, wherein the spring strip composition will initially contain coarse second phase particles of Zr-Fe having sizes up to 3.5 micrometers rough diameter which are not stabilized by Cr, Nb or Ni and which will dissolve easily after irradiation.

11. The nuclear fuel assembly of Claim 9, wherein the spring strip is not cold worked more than 10%.

12. The nuclear fuel assembly of Claim 9, wherein the spring strip is cold worked by deformation twinning.

13. The nuclear fuel assembly of Claim 9, wherein the spring strip alloy has a composition consisting of in weight %: 0.5 to 1.5 Sn; 0.2 to 0.8 Fe; 0.1 to 0.3 O; with the balance Zr, and where no more than 0.02 Cr; 0.01 Nb and 0.007 Ni are present in the composition.

14. The nuclear fuel assembly of Claim 9 where the spring strip alloy has a composition consisting essentially of in weight %: 0.3 to 0.6 Sn; 0.3 to 0.5 Fe; 0.2 to 0.25 O with the balance Zr.

15. The nuclear fuel assembly of Claim 10 where the spring strip composition will have second phase particles having a size between 0.2 and 3.5 micrometers.

16. The nuclear fuel assembly of Claim 10 where the spring strip composition will have second phase particles having a size between 0.3 and 2.0 micrometers.

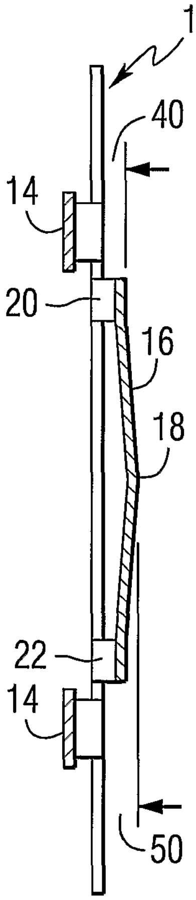


FIG. 1

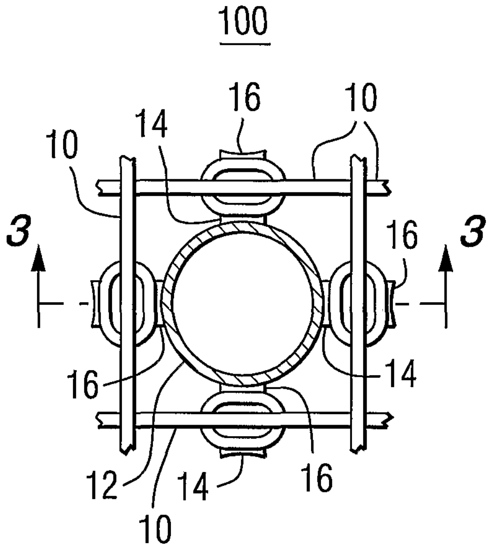


FIG. 2

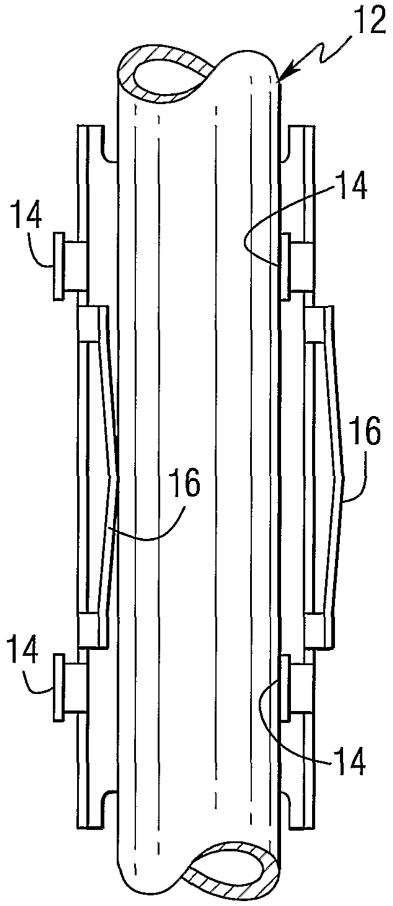


FIG. 3

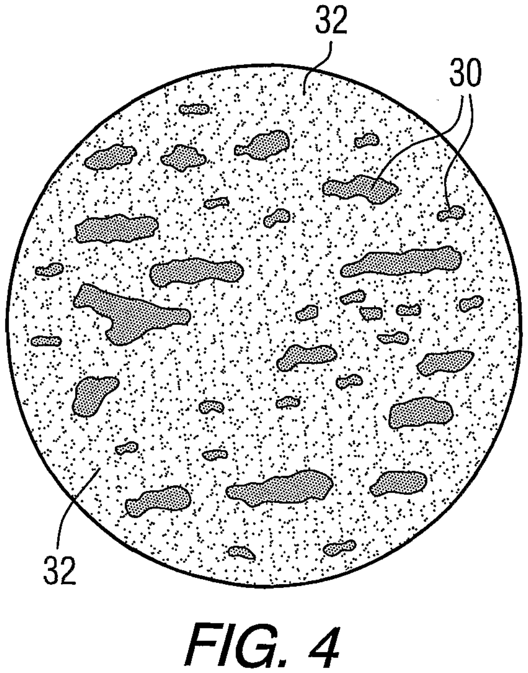


FIG. 4

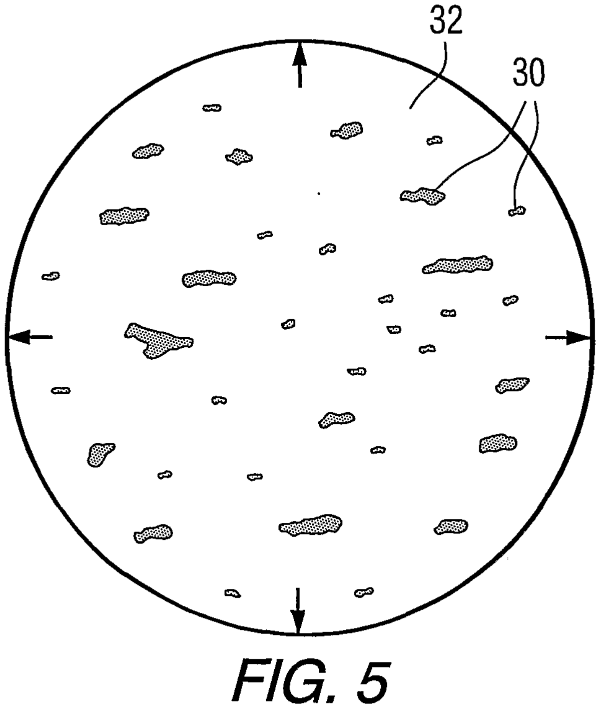
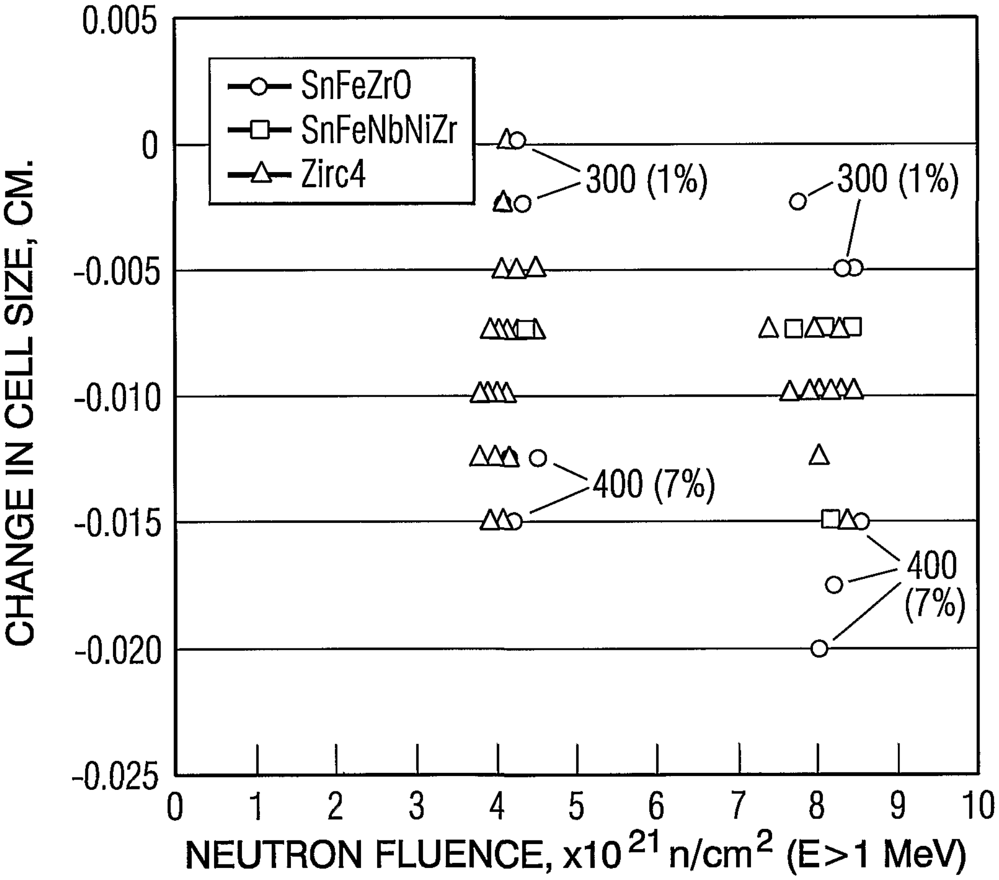


FIG. 5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(7%)
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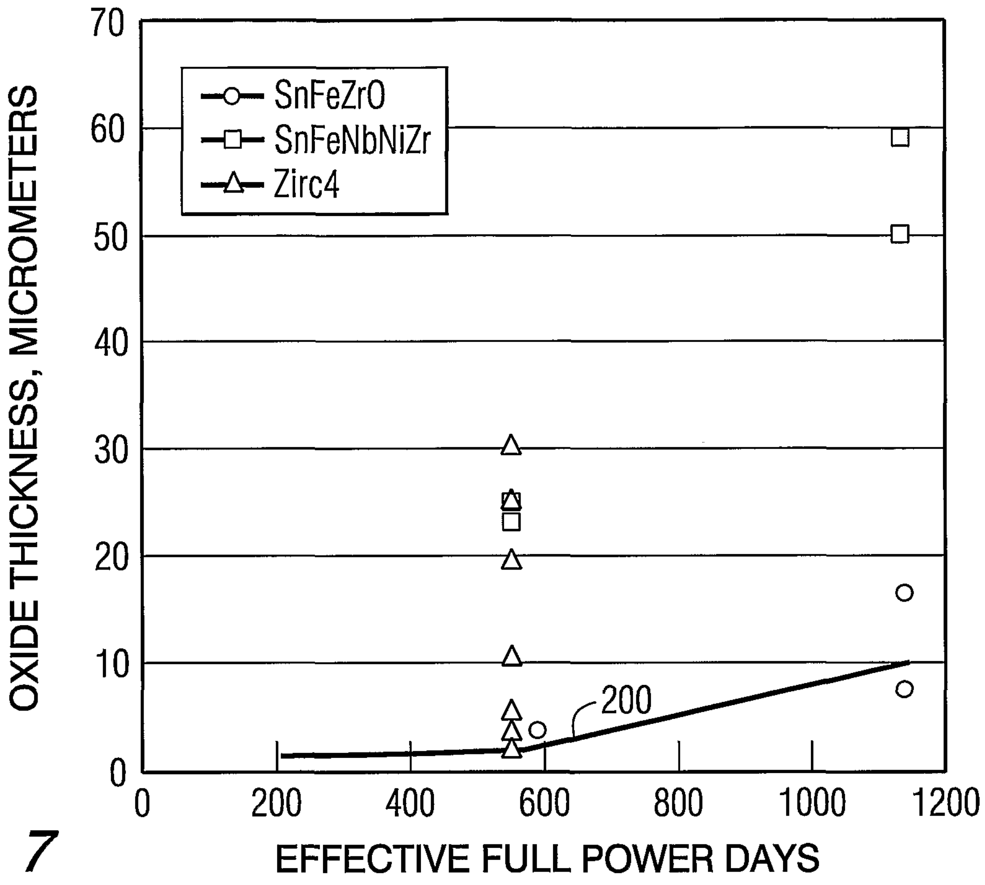
FIG. 6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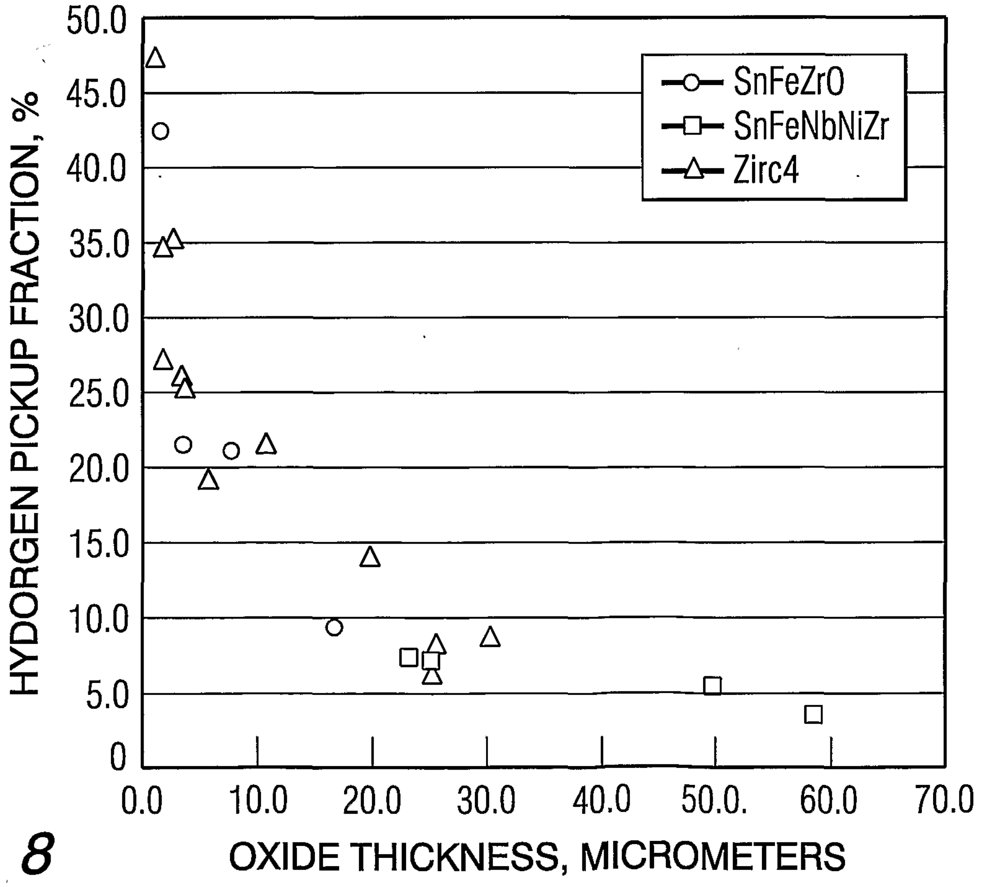
EFFECTIVE FULL POWER DAYS

FIG. 7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OXIDE THICKNESS, MICROMETERS

FIG. 8

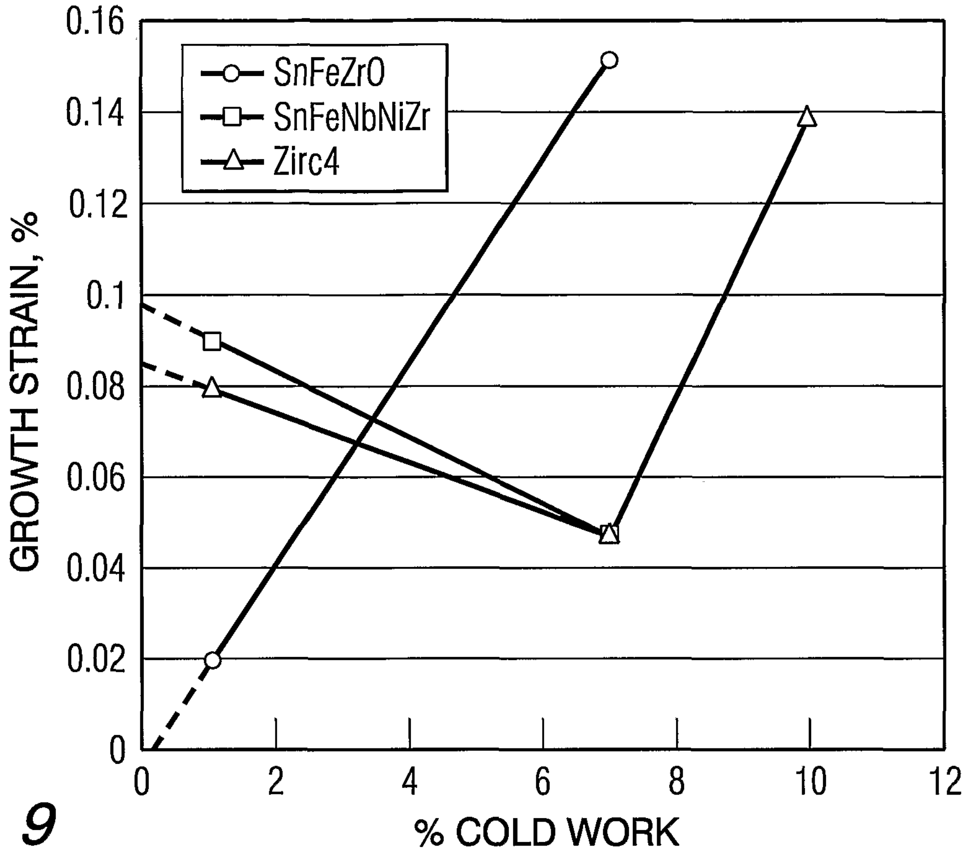
0.16
0.14
0.12
0.1
0.08
0.06
0.04
0.02
0
GROWTH STRAIN, %
0
2
4
6
8
10
12
% COLD WORK
SnFeZrO
SnFeNbNiZr
Zirc4

FIG. 9

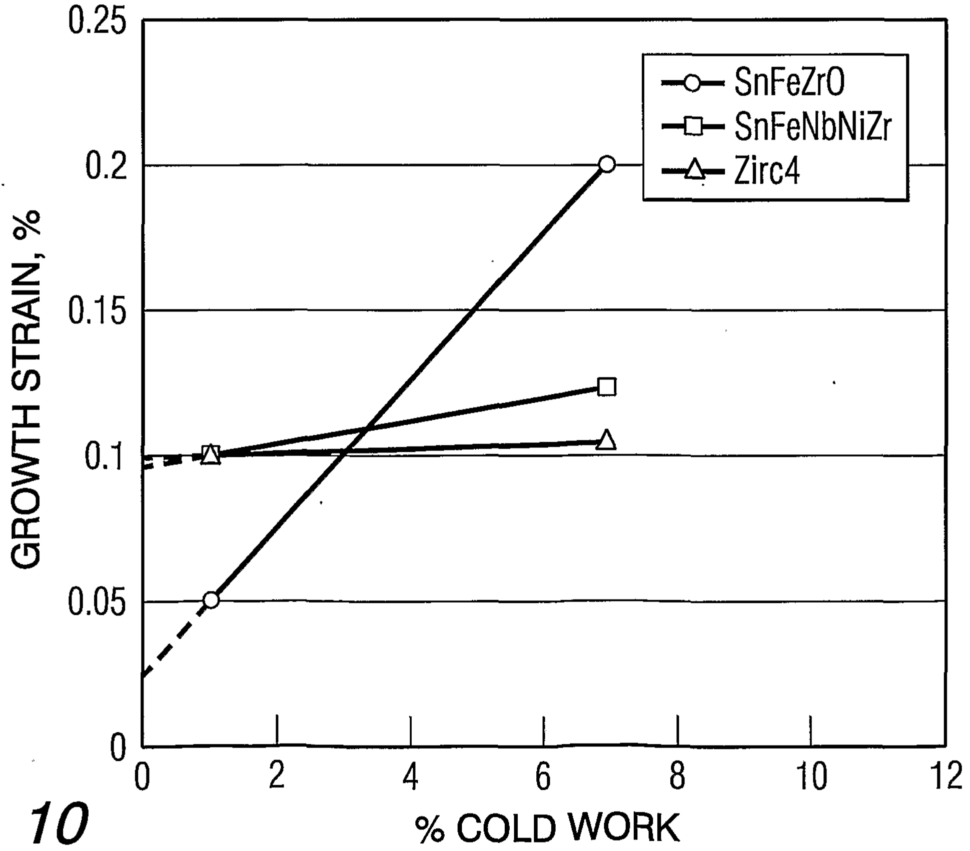
0.25
0.2
0.15
0.1
0.05
0
GROWTH STRAIN, %
0
2
4
6
8
10
12
% COLD WORK
SnFeZrO
SnFeNbNiZr
Zirc4

FIG. 10

INTERNATIONAL SEARCH REPORT

Int ınal Application No
PCT/US 02/06158

A. CLASSIFICATION OF SUBJECT MATTER
IPC 7 C22C16/00 C22F1/18 G21C3/356

According to International Patent Classification (IPC) or to both national classification and IPC

B. FIELDS SEARCHED

Minimum documentation searched (classification system followed by classification symbols)
IPC 7 C22C C22F G21C

Documentation searched other than minimum documentation to the extent that such documents are included in the fields searched

Electronic data base consulted during the international search (name of data base and, where practical, search terms used)

EPO-Internal, WPI Data, PAJ, CHEM ABS Data

C. DOCUMENTS CONSIDERED TO BE RELEVANT

Category °	Citation of document, with indication, where appropriate, of the relevant passages	Relevant to claim No.
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☒ Further documents are listed in the continuation of box C.

☒ Patent family members are listed in annex.

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"&" document member of the same patent family

Date of the actual completion of the international search
29 May 2002

Date of mailing of the international search report
06/06/2002

Name and mailing address of the ISA
European Patent Office, P.B. 5818 Patentlaan 2
NL – 2280 HV Rijswijk
Tel. (+31-70) 340-2040, Tx. 31 651 epo nl,
Fax: (+31-70) 340-3016

Authorized officer
Vlassi, E

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INTERNATIONAL SEARCH REPORT

Inte onal Application No
PCT/US 02/06158

C.(Continuation) DOCUMENTS CONSIDERED TO BE RELEVANT

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PCT/US 02/06158

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