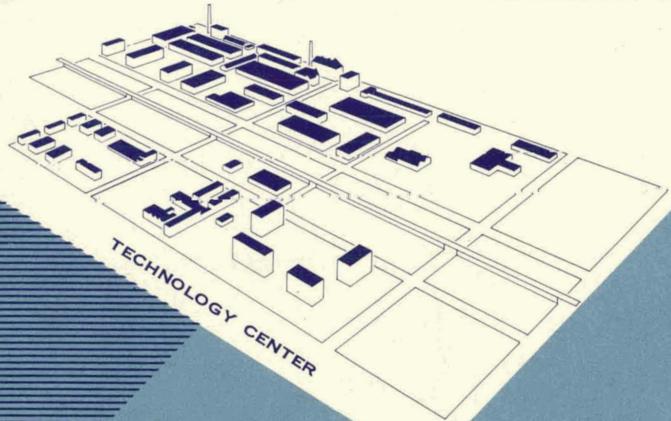
JOINT U. S. - EURATOM RESEARCH AND DEVELOPMENT PROGRAM

ARF

ARF 2198-26

ARMOUR RESEARCH FOUNDATION OF ILLINOIS INSTITUTE OF TECHNOLOGY



IMPROVED ZIRCONIUM ALLOYS

U. S. /EURATOM Program

Contract No. AT(11-1)-578
Project Agreement No. 1

25 years of research ARF

ARF-2198-6
(EURAEC-291)

ARMOUR RESEARCH FOUNDATION
of
ILLINOIS INSTITUTE OF TECHNOLOGY
Technology Center
Chicago 16, Illinois

IMPROVED ZIRCONIUM ALLOYS

D. Weinstein
F. C. Holtz

ARF 2198-26
Technical Progress Report
February 1, 1961 - February 28, 1962

Contract No. AT(11-1)-578
Project Agreement No. 1

U.S./EURATOM Program
Joint Research and Development Board
51-53 rue Belliard
Brussels, Belgium

and

Division of Reactor Development
U.S. Atomic Energy Commission
Washington 25, D. C.

February 28, 1962

Legal Notice

This document was prepared under the sponsorship of the U.S. Atomic Energy Commission pursuant to the Joint Research and Development Program established by the Agreement for Cooperation signed November 8, 1958, between the Government of the United States of America and the European Atomic Energy Community (Euratom). Neither the United States, the U.S. Atomic Energy Commission, the European Atomic Energy Community, the Euratom Commission, nor any person acting on behalf of either Commission:

A. Makes any warranty or representation, express or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this document, or that the use of any information, apparatus, method, or process disclosed in this document may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method or process disclosed in this document.

As used in the above, "person acting on behalf of either Commission" includes any employee or contractor of either Commission or employee of such contractor to the extent that such employee or contractor or employee of such contractor prepares, handles, disseminates, or provides access to any information pursuant to his employment or contract with either Commission or his employment with such contractor.

IMPROVED ZIRCONIUM ALLOYS

The United States and the European Atomic Energy Community (EURATOM), on May 29 and June 18, 1958, signed an agreement which provides a basis for cooperation in programs for the advancement of the peaceful applications of atomic energy. This agreement, in part, provides for the establishment of a Joint U.S. - Euratom research and development program which is aimed at reactors to be constructed in Europe under the Joint Program.

The work described in this report represents the Joint U.S. - Euratom effort which is in keeping with the spirit of cooperation in contributing to the common good by sharing of scientific and technical information and minimizing the duplication of effort by the limited pool of technical talent available in Western Europe and the United States.

IMPROVED ZIRCONIUM ALLOYS

ABSTRACT

The objectives of this program are the development of zirconium-base alloys possessing exceptionally good corrosion resistance to 680° F water or 750°-900° F steam and/or improved strength at elevated temperatures with corrosion resistance equivalent to Zircaloy-2. Approximately 100 binary compositions were screened by corrosion testing in 680° F water and 750° F steam; selected compositions were exposed to 900° F steam.

Data obtained from these studies indicated a number of highly promising bases for ternary alloy studies in superheated water or steam. These investigations have been pursued during the current year, and on the basis of 2000-hour tests, alloys have been obtained which initially satisfy the program objectives for 680° F water. These compositions are based on Zr-Nb, Zr-Sn, and Zr-Sb with ternary additions of Te, Ge, or Nb. Initial data for exposure to 750° and 900° F steam indicate very promising corrosion behavior of some ternary alloys.

TABLE OF CONTENTS

LIST OF TABLES

IMPROVED ZIRCONIUM ALLOYS

I. INTRODUCTION

This report summarizes work performed during the period February 1, 1961 to February 28, 1962, on Contract No. AT(11-1)-578, Project Agreement No. 1.

At the present operating temperature of pressurized water reactors (about 600° F), Zircaloy-2 meets most of the requirements for a fuel cladding material. However, at temperatures not much above 600° F, Zircaloy-2 lacks adequate corrosion resistance and strength as an unbonded fuel cladding. With the need for improved zirconium alloys apparent, this program has the objective of developing alloys markedly superior to those currently available. Specifically, the alloys to be developed should possess exceptionally good corrosion resistance to 680° F water and/or 750° -900° F steam with improved or equivalent elevated temperature strength. The development of an alloy having corrosion resistance equivalent to Zircaloy-2 but exhibiting markedly improved strength is also an objective of this program.

The approach to development of such alloys has been corrosion screening a large number of binary materials and preparing ternary compositions based on these results. This procedure has continued to the present position where, on the basis of 2000-hour tests, a number of ternary alloys are available which initially satisfy the program objectives for 680° F water. These compositions are based on Zr-Nb, Zr-Sn, and Zr-Sb with ternary additions of tellurium, germanium, or niobium. Moreover, relatively short time data in 750° and 900° F steam indicate that initially successful alloys will be obtained. Optimization of corrosion and strength properties is now being carried out by minor adjustment of alloy concentration.

II. SUMMARY OF WORK--RESULTS

A detailed discussion of materials used and experimental procedures employed in carrying out this development program will not be presented; a full account of techniques and procedures was given in ARF 2198-13 (Summary Report).[1] For the present discussion all materials were of the highest purity readily obtainable, and close attention was given to insuring alloy concentration accuracy and homogeneity of materials after melting. Fabrication techniques were employed which minimized atmospheric contamination, and standardized procedures[2, 3] were followed for corrosion specimen preparation and autoclave testing.

The program was initiated on approximately 100 binary alloys which were exposed to 680°F water and 750°F steam; selected compositions were evaluated in 900°F steam. A complete list of these alloys was presented previously.[1] It is of interest to mention that the choice of compositions was based on a great deal of personal communication with investigators in zirconium corrosion work as well as published information. An account of these personal contacts has been presented.[4] Table I summarizes the corrosion properties of binary alloys by listing those materials considered promising for further studies in superheated water or steam. (A detailed account of corrosion properties of all binary alloys has previously been presented[1, 5].) Comparison of corrosion properties was made with Zircaloy-2, and the anticipated strengthening effect of a particular alloying element was also considered in selecting materials for further study. Thus, Zr-1Cr showed properties which indicated further development for 680°F water, whereas Zr-3Cr would be considered promising for steam service. For most compositions, the corrosion behavior alone was sufficient information to allow further planning; nevertheless, alloys such as Zr-5Sn and Zr-2.5Sb were also of further interest even though corrosion resistance was inferior to relatively dilute alloys. Potentially, the high solute concentration materials are extremely useful due to the possibility of high strength levels; hopefully, it was felt, a third element might significantly improve corrosion resistance. A similar analysis was applied to the dilute alloys which served as ternary additions; the low concentration binary

TABLE I

SUMMARY OF CORROSION BEHAVIOR OF BINARY ALLOYS SELECTED FOR FURTHER STUDIES

Composition, w/o	Weight Gain, mg/dm^2 680°F Water 2429 hr	750°F Steam 809 hr	900°F Steam 97 hr
Zircaloy-2	39.66	41.57	spalling
Zr-0.5Nb	38.59	42.66	65.05
Zr-0.5Sn	24.70	spalling	--
Zr-5Sn	301.92	spalling	--
Zr-1Cr	30.50	spalling	spalling
Zr-3Cr	spalling at 1043 hr	33.12	51.87
Zr-1Sb	35.92	47.83	spalling
Zr-2.5Sb	spalling at 1902 hr	87.38	spalling
Zr-1V	spalling at 1902 hr	46.32	69.05
Zr-1Fe	spalling at 1902 hr	41.69	spalling
Zr-0.25Cr	22.66	38.76	spalling
Zr-0.25Cu	36.25	40.63	spalling
Zr-0.25Fe	31.60	40.65	spalling
Zr-0.25Mo	26.31	28.06	54.29
Zr-0.25Ge	spalling	spalling	--
Zr-0.25Te	29.60	37.03	--

alloys shown in Table I were selected on the basis of both corrosion resistance and strengthening effect on zirconium.

The ternary alloys prepared at the beginning of the current year together with presently available corrosion data in 680°F water are presented in Table II. Although evaluation of these compositions is presently being continued, the exposure time is sufficient to allow for reliable and meaningful conclusions. Inspection of these data shows that alloys having corrosion resistance superior or equivalent to Zircaloy-2 are the following:

Zr-0.5Sn-0.25Te
Zr-0.5Sn-0.25Cr
Zr-0.5Sn-0.25Ge
Zr-0.5Nb-0.25Fe
Zr-0.5Nb-0.25Cr
Zr-0.5Nb-0.25Te
Zr-1Sb-0.25Ge
Zr-1Cr-0.25Fe

One might note first that the above dilute ternary additions to Zr-0.5Nb and Zr-1Sb enhanced the corrosion resistance of these binary base alloys studied during the first year. Corrosion resistance, per se, is not sufficient to categorize an alloy as useful; without strength at least equivalent to Zircaloy-2 the improved corrosion resistance is of reduced value. In addition, however, a material of enhanced strength and equivalent corrosion resistance could be useful as a cladding alloy or an in-core structural component. In the light of this analysis, three additional compositions seem potentially useful--even though the corrosion resistance is not quite as good as Zircaloy-2:

Zr-1Sb-0.25To
Zr-1Sb-0.5Nb
Zr-1Cr-0.25Te

At this juncture, eleven ternary compositions have been arrived at which represent potentially useful materials. In Table III, the results of cursory tensile tests on these alloys at 680°F are presented; in many cases the strength level appears to be slightly higher than Zircaloy-2. (The inherent scatter in tensile data on single specimens of zirconium alloys allows determination of only a general stress level rather than specific values.)

TABLE II

CORROSION BEHAVIOR OF TERNARY ZIRCONIUM ALLOYS IN 680°F WATER

Composition, w/o	Weight Gain, mg/dm² 336 hr	953 hr	2040 hr
Zircaloy-2	18.99	23.47	29.45
Zr-1Cr-1Sb	17.40	24.56	37.87
Zr-1Cr-1Sb	14.84	23.19	35.24
Zr-1Cr-0.5Nb	15.13	22.18	37.31
Zr-1Cr-0.5Nb	15.81	23.71	38.39
Zr-1Cr-0.5Sn	16.05	21.07	32.10
Zr-1Cr-0.25Cu	20.04	25.06	35.07
Zr-1Cr-0.25Fe	13.47	17.95	30.29
Zr-1Cr-0.25Fe	12.80	17.72	28.56
Zr-1Cr-0.25Mo	24.30	29.16	40.83
Zr-1Cr-0.25Te	16.36	21.48	34.77
Zr-1Cr-0.25Ge	15.40	19.80	34.10
Zr-1Sb-0.5Nb	16.36	22.50	36.81
Zr-1Sb-0.5Nb	15.83	21.10	35.88
Zr-1Sb-0.5Sn	16.04	22.06	37.10
Zr-1Sb-0.25Cr	17.14	21.18	35.29
Zr-1Sb-0.25Cr	17.33	22.43	35.69
Zr-1Sb-0.25Cu	22.08	28.11	42.16
Zr-1Sb-0.25Fe	18.97	26.96	35.94
Zr-1Sb-0.25Fe	19.02	26.03	34.04
Zr-1Sb-0.25Mo	19.33	29.50	90.54
Zr-1Sb-0.25Mo	18.37	29.60	69.40
Zr-1Sb-0.25Te	19.86	26.82	36.75
Zr-1Sb-0.25Ge	15.37	21.14	29.79

TABLE II (continued)

Composition, w/o	Weight Gain, mg/dm^2 336 hr	953 hr	2040 hr
Zr-5Sn-0.5Nb	18.24	25.33	70.93
Zr-5Sn-0.5Nb	19.01	25.02	72.05
Zr-5Sn-1Cr	18.15	29.24	125.03
Zr-5Sn-1Sb	18.98	34.96	95.89
Zr-5Sn-2.5Sb	28.26	84.78	197.81
Zr-5Sn-0.25Cr	17.20	22.26	67.78
Zr-5Sn-0.25Cu	21.31	55.80	152.18
Zr-5Sn-0.25Fe	25.60	43.01	100.34
Zr-5Sn-0.25Fe	24.57	41.28	100.24
Zr-5Sn-0.25Mo	21.93	69.97	173.36
Zr-5Sn-0.25Te	19.05	26.07	92.25
Zr-5Sn-0.25Ge	18.83	29.73	108.00
Zr-2.5Sb-1Cr	21.24	30.34	43.48
Zr-2.5Sb-1Cr	13.07	21.12	37.20
Zr-2.5Sb-0.5Nb	20.49	31.75	49.16
Zr-2.5Sb-0.5Nb	22.39	31.55	46.81
Zr-2.5Sb-0.5Sn	20.15	28.21	41.31
Zr-2.5Sb-1.5Sn	18.75	26.65	59.22
Zr-2.5Sb-1.5Sn	15.89	24.83	51.64
Zr-2.5Sb-0.25Cr	19.77	29.66	39.54
Zr-2.5Sb-0.25Cu	22.67	31.95	46.38
Zr-2.5Sb-0.25Fe	28.51	46.84	78.40
Zr-2.5Sb-0.25Mo	29.04	79.87	181.52
Zr-2.5Sb-0.25Te	24.23	35.34	52.50
Zr-2.5Sb-0.25Ge	20.33	30.49	50.81
Zr-0.5Nb-0.25Cr	14.13	22.21	28.27
Zr-0.5Nb-0.25Cr	15.05	23.08	29.10
Zr-0.5Nb-0.25Cu	19.19	28.28	35.34

TABLE II (continued)

Composition, w/o	Weight Gain, mg/dm^2 336 hr	953 hr	2040 hr
Zr-0.5Nb-0.25Fe	15.40	22.58	27.71
Zr-0.5Nb-0.25Fe	14.89	22.83	27.79
Zr-0.5Nb-0.25Mo	26.99	45.98	61.97
Zr-0.5Nb-0.25Mo	24.01	40.01	53.01
Zr-0.5Nb-0.25Te	14.27	24.46	30.57
Zr-0.5Nb-0.25Ge	17.00	28.00	36.00
Zr-0.5Sn-0.25Cr	14.36	21.53	25.63
Zr-0.5Sn-0.25Cr	14.39	22.14	25.46
Zr-0.5Sn-0.25Cu	20.53	35.61	47.82
Zr-0.5Sn-0.25Fe	19.29	32.49	40.61
Zr-0.5Sn-0.25Mo	16.32	26.52	35.71
Zr-0.5Sn-0.25Te	12.54	19.85	25.08
Zr-0.5Sn-0.25Ge	12.33	21.57	25.68

TABLE III

STRENGTH LEVEL OF TERNARY ZIRCONIUM ALLOYS AT 680° F

Composition	Ultimate Tensile Strength, psi
Zircaloy-2	27,800
Zr-0.5Sn-0.25Te	30,200
Zr-0.5Sn-0.25Cr	28,200
Zr-0.5Sn-0.25Ge	25,100
Zr-0.5Nb-0.25Fe	28,100
Zr-0.5Nb-0.25Cr	29,300
Zr-0.5Nb-0.25Te	30,700
Zr-1Sb-0.25Ge	32,700
Zr-1Cr-0.25Fe	30,700
Zr-1Sb-0.25Te	30,300
Zr-1Cr-0.25Te	32,800

One notes immediately that materials with high alloy concentrations exhibit higher strengths than Zircaloy-2; the Zr-1Sb-0.25Ge alloy is particularly attractive due to good corrosion properties as well as high strength. Additions of tellurium, and perhaps germanium, generally seem to impart strengthening at elevated temperatures. (It is of interest to mention that the present results tend to corroborate the work of Bibb, et al.[6] In a fundamental study of corrosion, tellurium was proposed as an addition which would decrease anion vacancies and thereby improve corrosion resistance; it was further found that tellurium is a powerful solid-solution strengthener.)

For development of materials resistant to superheated steam, an approach identical to that for 680° F water has been used; Table I presents data on the promising binary alloys, obtained at the end of the first year, for service in 750° and 900° F steam. Specifically, those compositions selected for further development are given below:

750° F Steam	900° F Steam
Zr-1Sb	Zr-0.5Nb
Zr-0.5Nb	Zr-3Cr
Zr-3Cr	Zr-1V
Zr-1V	Zr-1Mo
Zr-1Fe	

The selection was again based on both corrosion resistance and strength. Thus, Zr-1Sb and Zr-1V were chosen for their relatively high strength values even though corrosion behavior, in 750° F steam, was slightly inferior to Zircaloy-2. The Zr-3Cr alloy was particularly attractive due to the high alloy content and excellent corrosion resistance as compared to Zircaloy. In 900° F steam, any alloy that did not disintegrate in 97 hours was considered acceptable for further studies even though the weight gains were excessively high for actual reactor use. (Since no zirconium alloy is presently available for service in 900° F steam, or 750° F for that matter, this approach may be justified.)

Based on these results, a series of ternary compositions were conceived for potential use in superheated steam; together with presently available corrosion data, these alloys are presented in Table IV. Although

TABLE IV

CORROSION BEHAVIOR OF TERNARY ZIRCONIUM ALLOYS IN SUPERHEATED STEAM

Composition	Weight Gain, mg/dm^2		
	750°F Steam		900°F Steam
	336 hr	672 hr	73 hr
Zircaloy-2	21.00	30.24	
Zr-1Sb-0.5Nb	21.47	34.36	
Zr-1Sb-0.5Nb	22.61	33.34	
Zr-1Sb-0.25Cr	23.29	44.35	
Zr-1Sb-0.25Cr	24.56	45.77	
Zr-1Sb-0.25Mo	25.27	76.85	
Zr-1Sb-0.25Mo	24.88	72.55	
Zr-1Sb-0.25Te	29.13	38.83	
Zr-0.5Nb-0.25Cr	16.77	27.25	
Zr-0.5Nb-0.25Cr	18.49	28.76	
Zr-0.5Nb-0.25Mo	26.53	42.45	
Zr-0.5Nb-0.25Mo	25.20	38.84	
Zr-0.5Nb-0.25Te	20.29	31.44	
Zr-3Cr-1V	229.02	--	
Zr-3Cr-1V	spalling	--	
Zr-3Cr-1Fe	19.43	28.63	
Zr-3Cr-1Sb	25.96	30.11	
Zr-3Cr-0.5Nb	30.69	53.98	
Zr-3Cr-0.5Nb	28.35	47.25	
Zr-3Cr-0.25Mo	29.92	105.23	
Zr-3Cr-0.25Mo	51.73	--	
Zr-3Cr-0.25Te	18.09	26.60	

TABLE IV (continued)

Composition	Weight Gain, mg/dm² 750°F Steam 336 hr	750°F Steam 672 hr	900°F Steam 73 hr
Zr-1V-1Fe	18.51	28.80	
Zr-1V-1Sb	33.24	--	
Zr-1V-0.5Nb	spalling	--	
Zr-1V-0.5Nb	spalling	--	
Zr-1V-0.25Cr	41.86	--	
Zr-1V-0.25Te	34.77	--	
Zr-1V-0.25Mo	21.64	31.94	
Zr-1V-0.25Mo	23.75	38.21	
Zr-1Fe-1Sb	29.14	41.62	
Zr-1Fe-0.5Nb	26.48	39.72	
Zr-1Fe-0.25Cr	19.30	27.43	
Zr-1Fe-0.25Mo	43.03	--	
Zr-1Fe-0.25Te	25.69	39.04	
Zircaloy-2			762 (spalling)
Zr-0.5Nb-0.25Mo			46.05
Zr-0.5Nb-0.25Mo			46.09
Zr-3Cr-1V			141.81
Zr-3Cr-0.5Nb			55.28
Zr-3Cr-0.25Mo			143.76
Zr-3Cr-1Mo			126.01
Zr-1V-0.5Nb			203.30
Zr-1V-0.5Nb			165.55
Zr-1V-0.25Mo			44.50
Zr-1V-0.25Mo			44.15
Zr-1Mo-1V			460.92
Zr-1Mo-0.5Nb			99.15

exposure times are not sufficiently long to allow definite conclusions, some very encouraging trends and corrosion behavior characteristics are resulting. In 750°F steam, a number of compositions exhibit--so far--corrosion resistance superior or equivalent to Zircaloy-2. Of notable interest are the alloys of Zr-3Cr with 1Sb, 1Fe, or 0.25Te; the corrosion properties are as good as Zircaloy-2, and the strength should be considerably higher. Zr-1V-1Fe can be similarly classified. (Zr-1V exhibited a tensile strength of 40,000 psi at 680°F). The dilute alloy of Nb-Cr and the Zr-1Fe-0.25Cr alloy also exhibit, thus far, corrosion resistance equivalent to or better than Zircaloy-2.

Investigations in 900°F steam have been carried out for only a short exposure time; in 73 hours, however, Zircaloy-2 was badly spalling and is considered useless under these conditions. Although some of the experimental alloys did not spall or disintegrate, the weight gains of those exhibiting the adherent, black oxide corrosion film were still excessively high. These studies are being continued; however, the present results indicate that Zr-0.5Nb-0.25Mo, Zr-3Cr-0.5Nb, and Zr-1V-0.25Mo might be promising compositions for further development.

III. FUTURE WORK

On the basis of both corrosion resistance and strength at 680°F, seven alloys have been chosen for optimization of properties by minor alloy modification. In order of decreasing corrosion resistance, these compositions are as follows:

Zr-0.5Sn-0.25Te
Zr-1Sb-0.25Ge
Zr-0.5Sn-0.25Ge
Zr-0.5Nb 0.25Te
Zr-1Sb-0.25Te
Zr-1Sb-0.5Nb
Zr-1Cr-0.25Te

At present, modified ternary compositions are being prepared for fabrication of corrosion specimens; the specific alloys are listed in Table V. Tellurium and germanium, since they impart strength and corrosion resistance, are being extensively employed in these studies. The general approach is essentially one of increasing the dilute addition of tellurium

TABLE V

TERNARY ALLOYS DESIGNED FOR OPTIMIZATION OF CORROSION PROPERTIES IN WATER BY MINOR ALLOY MODIFICATIONS

Zr-0.4Sn-0.4Te	Zr-0.3Nb-0.5Te
Zr-0.4Sn-0.5Te	Zr-0.4Nb-0.25Te
Zr-0.5Sn-0.3Te	Zr-0.4Nb-0.4Te
Zr-0.5Sn-0.4Te	Zr-0.5Nb-0.3Te
Zr-0.6Sn-0.3Te	Zr-0.5Nb-0.4Te
Zr-0.75Sn-0.25Te	Zr-0.75Nb-0.25Te
Zr-0.75Sn-0.4Te	Zr-0.75Nb-0.4Te
Zr-0.9Sn-0.35Te	
	Zr-0.75Sb-0.2Te
Zr-0.5Sn-0.3Ge	Zr-0.75Sb-0.3Te
Zr-0.5Sn-0.4Ge	Zr-0.75Sb-0.4Te
Zr-0.75Sn-0.25Ge	Zr-1Sb-0.2Te
Zr-0.75Sn-0.4Ge	Zr-1Sb-0.4Te
Zr-0.9Sn-0.35Ge	Zr-1.2Sb-0.15Te
Zr-0.75Sb-0.25Ge	Zr-0.75Sb-0.4Nb
Zr-0.75Sb-0.4Ge	Zr-0.75Sb-0.5Nb
Zr-1Sb-0.2Ge	Zr-1Sb-0.35Nb
Zr-1Sb-0.4Ge	
Zr-1.2Sb-0.25Ge	Zr-0.75Cr-0.4Te
Zr-1.2Sb-0.4Ge	Zr-0.75Cr-0.5Te
Zr-1.5Sb-0.2Ge	Zr-0.85Cr-0.25Te

or germanium (for increased strength) while maintaining or slightly reducing the concentration of major alloy addition (tin, antimony, niobium, and chromium--to maintain or possibly improve corrosion resistance). In addition, a number of compositions contain increased amounts of the major addition with decreased or constant percentages of tellurium or germanium. Utilizing, therefore, the observed past effects of certain elements on corrosion properties and strength of zirconium, these ternaries have been designed, in all cases, for producing alloys better than Zircaloy-2. The chances of successfully meeting one of the program objectives (for 680°F water) appear very good.

Before the end of the current contract, corrosion data on materials for 750° and 900°F steam application will be available. Based on promising materials as indicated in Table IV, a further group of ternary alloys will be prepared designed to optimize the initially successful compositions--as is presently being performed on alloys for 680°F water. For 750°F steam, it is anticipated that the alloys Zr-3Cr-(1Sb, 1Fe, 0.25Te), Zr-1V-1Fe, dilute Zr-Nb-Cr, and Zr-1Fe-0.25Cr will warrant further development effort; for 900°F steam, Zr-0.5Nb-0.25Mo, Zr-3Cr-0.5Nb, and Zr-1V-0.25Mo will probably be of further interest. Moreover, when long-time corrosion data are obtained, mechanical and other properties will be determined. As with the superheated water alloys, corrosion data on modification of the above compositions may suggest additions of a fourth element for further improvement of corrosion (and strength) behavior. (The latter effort could not be initiated for some time--approximately five months.)

In order to obtain more meaningful, and therefore more useful, data, promising and initially successful alloys will be used for studies on pickup of hydrogen during corrosion. When the present corrosion evaluations of ternary alloys in 680°F water and 750°F steam are terminated, these specimens will be employed for this study. Since the exposure time is of considerable duration, the data obtained will be valuable for final recommendations of alloys for reactor use--within the limits of the effect of other variables such as neutron irradiation. The data obtained will also serve, to a great extent, in choice of a quaternary addition; addition of an element known to markedly decrease the corrosion hydrogen pickup might now render an alloy, acceptable but for this property, useful for reactor service.

An exchange of approximately 30 binary zirconium alloys has been accomplished with General Nuclear Engineering Corporation, Dunedin, Florida. The received compositions[7] will be corrosion tested in 680°F water and 750°F steam. Corrosion properties of most of these alloys are not reported in the literature, and perhaps these data might indicate an addition which markedly improves corrosion resistance.

Studies of fabricability, weldability, nuclear properties, and extensive elevated-temperature mechanical properties will commence when final selection of superior materials are made. At that time a relatively large, consumably arc-melted ingot of each alloy would be prepared for extensive investigation. Such work would start late in 1962.

IV. CONCLUSION

A program for development of improved zirconium alloys in superheated water and steam is being pursued. By the approach of corrosion screening a large number of binary compositions, data were obtained which indicated further studies on ternary compositions. Information on corrosion behavior of these alloys is now available, and very promising or initially successful materials have been obtained. In 680°F water, ternary alloys with tin, antimony, niobium, or chromium as the major addition and tellurium, germanium, or niobium as the third element represent materials which initially satisfy the program objectives in terms of corrosion resistance and strength. Although data in 750° and 900°F steam are not complete, some particularly interesting and promising alloys seem to be resulting. These compositions are based on relatively high chromium additions and additions of vanadium or niobium with antimony, iron, or tellurium as the ternary component; molybdenum had a particularly good effect on corrosion resistance in 900°F steam. Future work will involve optimization of properties by minor modification of solute concentrations.

V. LOGBOOKS AND CONTRIBUTING PERSONNEL

The data presented in this report are recorded in ARF Logbooks Nos. C-9748, C-9749, C-9750, C-10612, C-10882, and D-1452.

Personnel contributing to this work are the following:

W. L. Ross	-	Project Technician
F. C. Holtz	-	Group Leader
D. Weinstein	-	Project Engineer

Respectfully submitted,

ARMOUR RESEARCH FOUNDATION OF ILLINOIS INSTITUTE OF TECHNOLOGY

D. Weinstein

D. Weinstein
Associate Metallurgist

F. C. Holtz

F. C. Holtz
Senior Metallurgist

DW/rh

Tech Rev - CRS

REFERENCES

1. D. Weinstein, F. C. Holtz, and R. J. Van Thyne, "Improved Zirconium Alloys", ARF 2198-13, U.S./EURATOM Program, February 28, 1961.

2. S. Kass, "Corrosion Testing of Two Zirconium Alloys", Corrosion, Vol. 16, February, 1960, p. 137.

3. S. Kass, "Corrosion Testing Zircaloy and Hafnium Coupons", WAPD-ZH-25, October, 1960, p. 14; also "Discussion of Corrosion Testing Techniques", ibid, p. 22.

4. D. Weinstein, R. J. Van Thyne, and D. W. Levinson, "Improved Zirconium Alloys--A Summary of Recent Work in This Field", ARF 2198-6, U.S./EURATOM Program, September, 1960.

5. D. Weinstein and F. C. Holtz, "Improved Zirconium Alloys", ARF 2198-18, U.S./EURATOM Program, July, 1961.

6. A. E. Bibb, A. P. Beard, and D. L. Douglass, "Investigation of Binary Zirconium Alloys", KAPL-2162, November 1, 1960

7. D. Weinstein and F. C. Holtz, "Improved Zirconium Alloys", ARF 2198-24, U.S./EURATOM Program, January, 1962.