SUBMITTED TO AEC DEC 26 '68

BNWL-930

FAST FLUX
TEST FACILITY
PROJECT REPORT

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EVALUATION OF INCONEL 718 FOR APPLICATION IN THE FAST TEST REACTOR

DECEMBER 1968

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AEC RESEARCH & DEVELOPMENT REPORT

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PRELIMINARY REPORT

This report contains information of a preliminary nature prepared in the course of work under Atomic Energy Commission Contract AT(45-1)-1830. This information is subject to correction or modification upon the collection and evaluation of additional data.

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BNWL-930

EVALUATION OF INCONEL 718 FOR APPLICATION IN THE FAST TEST REACTOR

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December, 1968

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EVALUATION OF INCONEL 718 FOR APPLICATION IN THE FAST TEST REACTOR

R. A. Moen, K. R. Wheeler, and J. E. Irvin

ABSTRACT

Inconel 718, a precipitation hardenable nickel-base alloy, has recently been considered for use in the Fast Test Reactor (FTR). Properties of interest which qualify this alloy for reactor use are excellent wear resistance, fatigue resistance, ease of fabrication, superior high temperature mechanical properties, and structural stability.

Three distinct areas of possible application have thus far been identified:

Flow Duct Contact Pads. Preliminary data have shown Inconel 718 to Inconel 718 to be one of the more favorable material combinations for contact pad use in 1200 °F sodium systems. Additional development work required for contact pads involves perfecting methods of fabrication and determining performance of fabricated parts in a 1200 °F sodium/neutron environment.

Safety Rod Accelerator Springs. This alloy is of interest for spring applications because of its high yield strength at temperature and its superior stress relaxation properties at 1200 °F. Further work must be done to determine its corrosion resistance and stress relaxation properties in high temperature-low velocity sodium.

Fuel Pin Grid Spacer Assemblies. As in the spring application, high yield strength at temperature and resistance to stress relaxation qualify Inconel 718 for this use. No data exist at this time, however, on its resistance to corrosion by high temperature, high velocity sodium, nor of the deterioration of strength and ductility resulting from prolonged high temperature exposure in fast neutron environments. Extensive corrosion and stress relaxation tests must be performed in a high fast fluence, high temperature, and high velocity sodium environment.

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EVALUATION OF INCONEL 718 FOR APPLICATION IN THE FAST TEST REACTOR

R. A. Moen, K. R. Wheeler, and J. E. Irvin

I. INTRODUCTION

Considerable interest has been expressed recently in the use of alloy Inconel 718 in the Fast Test Reactor (FTR). Three distinct areas of application thus far identified are the fuel pin grid-spacer assemblies, safety rod accelerator springs, and flow duct contact pads. Undoubtedly, additional areas of potential application will be identified during preliminary and final design.

The objective of this report is to summarize the information presently available on Inconel 718 as it applies to the FTR. An additional objective is to identify the development effort required to fully qualify this material for use in the FTR in the three applications presently identified.

II. APPLICATIONS

FUEL ELEMENT GRID SPACERS

Inconel 718 has been recommended for use in FTR fuel assemblies as "honeycomb" grid spacers.[1] In the selection of Inconel 718, three other materials (Type 316 stainless steel, Inconel X-750 and TZM) were considered, but ultimately Inconel 718 was selected because of its attractive yield strength and stress relaxation properties.

The honeycomb grid spacers consist of an ensemble of thin (0.010 to 0.015 in.) corrugated Inconel 718 plates brazed together with a brazing alloy such as Nicrobraz-50 or Nicrobraz-30 to form a hexagonal pattern of holes. The spacer height is about 1 in. with spacers being located every 9.5 in. along the length of the fuel pins. Contact with the fuel pin is achieved at two circumferentially located points by dimples formed on the spacer. The third contact point is provided by a dimpled, cantilevered "spring" that provides a resilient contact point to accommodate fuel pin swelling.

The honeycomb grid spacers will reside in the peak flux, peak temperature, and peak sodium flow region of the reactor. The flux above 0.067 MeV (first 8 groups*) will be 9.1×10^{15} n/cm^2-sec at the core midplane.[2] Flux above 0.82 MeV will be 2×10^{15} n/cm^2-sec (first 5 groups*). With the design life of the fuel elements being 10,000 hr, grid spacers will receive fluences of 3.3×10^{23} n/cm^2 ($E > 0.067$ MeV) or 7.2×10^{22} n/cm^2 ($E > 0.82$ MeV). The temperature of the grid spacers will vary within the fuel subassembly with the highest temperatures occurring above core midplane. The actual temperatures may range from slightly above initial coolant inlet temperatures of about 550 to 600 °F to above the ultimate sodium outlet temperature of slightly greater than 1200 °F. Within any one fuel subassembly, however, the temperature difference between the hot and cold portion of any pin may be as high as 350 °F. Sodium flow past the spacers will range from 25 to 35 ft/sec, with sodium purity assumed to be within acceptable limits (i.e., for $T > 1000$ °F, $O_2 < 1$ ppm, $C < 20$ ppm).[3] Actual stress values have not been reported, but Reference 1 indicated that they would be high.

SAFETY ROD ACCELERATOR SPRINGS

The decision to consider Inconel 718 as the prime candidate for the safety rod accelerator spring was made as the result of the noted advantage in stress relaxation properties of Inconel 718 over those of Inconel X-750.[4] Inconel X-750

---

* *These neutron energy groups are used because they bracket the commonly referred to values of $E > 0.1$ MeV or $E > 1$ MeV, respectively.*

originally appeared to be the best choice for accelerator springs because of the use of this alloy in Fermi for a similar application.[5] The safety rod accelerator springs for the FTR have the following physical characteristics, based on the latest conceptual designs:[6]

Spring free length	97.5 in.
Spring wire diameter	0.7 in.
Spring mean diameter	4 in.
Active coils	95
Fiber stress	30,000 psi

The safety rod springs must accelerate the safety rods into the core within a given time interval at any time during the rod life.

The safety rod will be in the "cocked" position above the core, with the bottom of the spring located about 10 ft above the top of the core. These springs will be exposed to the high temperature outlet sodium which will range from about 900 °F initially to an ultimate of 1200 °F. The sodium flow at this point within the reactor will be about 10% of that experienced within fuel subassemblies, in other words, about 2 to 5 ft/sec. Since the lower tip of the spring is nearest the reactor core, flux at this location is controlling and was determined in the same manner as reported for the grid spacers.[2] If a 10,000 hr safety rod life is assumed, fluence at the tip of the spring will be 4.3×10^{16} n/cm^2 ($E > 0.067$ MeV) or 1.2×10^{13} n/cm^2 ($E > 0.82$ MeV). These fluence values will drop rapidly as one moves up the spring. Depending upon the burnup accumulated by the poison contained in the safety rod, the life of the safety rod could be extended considerably beyond a 10,000 hr minimum.

FLOW DUCT CONTACT PADS

Inconel 718 was selected as the prime candidate material for flow duct contact pads (hard facing or wear surfaces) because of its attractive wear properties and relative ease of fabrication. Tests in progress at the Liquid Metal Engineering Center (LMEC) on determining static and sliding contact behavior of Inconel 718 and other materials in elevated temperature sodium showed encouraging results for Inconel 718 to Inconel 718 contact pads.[7]

The contact pads for FTR flow ducts will be in the form of layers or pads of Inconel 718 approximately 1/8 to 1/4 in. thick by about 2 in. wide. Methods for attaching the Inconel 718 to the flow duct are being explored at this time. Major points of contact between flow ducts will probably be located about 1 ft above and 1 ft below the active core zone.

The temperature of the flow ducts to which the contact pads are attached will vary from initial inlet temperatures of 550 to 600 °F to ultimate temperatures of 1150 to 1200 °F. Fluences were calculated for an area 1 ft above the top of the core (assuming a 10,000 hr flow duct life) and found to be 9×10^{21} n/cm^2 (E > 0.067 MeV) or 5.4×10^{20} n/cm^2 (E > 0.82 MeV).[2]

Each of the three potential applications for Inconel 718 requires specific knowledge of certain properties. Table 1 summarizes the material applications and property requirements for the identified uses in the FTR.

III. COMPOSITION

Inconel 718 is a precipitation hardenable nickel-base alloy with relatively high strength at elevated temperatures. The composition is given in Table 2 along with several other age-hardenable nickel-base alloys and Type 304 stainless steel. Also included are Nicrobraz-50 and Nicrobraz-30 which are the most probable brazing alloys for joining the Inconel 718 grid spacers.

TABLE 1. Applications and Property Requirements for Inconel 718 in the FTR

	Grid spacers for fuel elements	Contact pad for flow ducts	Springs for safety rod accelerators
ENVIRONMENT			
Fluence, n/cm^2	(peak within core)	(1 ft above core top)	(10 ft above core top)
E>0.1 MeV (n groups 1-8)	3.3×10^{23}	9×10^{21}	4.3×10^{16}
E>1 MeV (n groups 1-5)	7.2×10^{22}	5.4×10^{20}	1.2×10^{13}
Temperature °F	600-1200	600-1150	900-1200
Residence time, hr	10^4	10^4	10^4 and up
Sodium velocity, ft/sec	25-35	2-5	2-5
Fiber stress, ksi	High		30
PROPERTY REQUIREMENTS			
Corrosion resistance	Yes	(Resist self-welding)	(Resist intergranular attack)
High yield strength	Yes	No	Yes
Low creep rates	Yes	No	Yes
Stress relaxation resistance	Yes	No	Yes
Adequate ductility	Yes	Yes	Yes
Metallurgical stability	Yes	Yes	Yes
Fabricability	Yes	Yes	Yes

TABLE 2. Chemical Composition of Various Alloys

Designation	Nominal chemical composition, %											Melting point °F	Density lb/in.^{3}
	Ni	C	Mn	Fe	S	Si	Cu	Cr	Al	Ti	Others		
INCONEL alloy 718	52.5	0.04	0.20	18.5	0.007	0.30	0.07	18.6	0.40	0.90	Mo 3.1 Cb 5.0	2300-2437	0.296
INCONEL alloy X-750	73.0	0.04	0.70	6.75	0.007	0.30	0.05	15.0	0.80	2.50	Cb 0.85	2540-2600	0.298
Rene'41	53	0.09		2.50				19	1.50	3.10	Mo 9.75 Co 11.00 B 0.005		0.298
NIMONIC 80A	75.0	0.06						20.0	1.4	2.30		2480-2534	0.294
NICROBRAZ-50	Bal							13			P 10	1615-1640	
NICROBRAZ-30	Bal					10.2		19				1975-2075	
Type 304 stainless steel	8.00-11.00	0.08 max	2.00 max	Bal	0.030 max	1.00 max		18.00-20.00				2550-2650	0.290

The composition is important in determining corrosion behavior and metallurgical stability as well as mechanical properties of the alloy. Any deviation from the limiting chemical composition will result in materials properties degradation. The tendency for over-aging during long time service at elevated temperatures is also affected by compositional changes.

IV. MECHANICAL PROPERTIES

Mechanical properties of the age-hardenable alloy Inconel 718 vary considerably with prior annealing temperature and aging treatment, and (as stated previously) with composition. The subjects of the effect of composition and heat treatment on mechanical properties have been covered in considerable detail by Wagner and Hall.[8] The mechanical property data presented in this section will be selected from existing data compilations, such as the Huntington periodicals,[9-11] to best represent the heat treatments that the material will have for its intended service. Heat treatment of Inconel 718 varies somewhat for desired application. Commercial producers[11] recommend one of two basic treatments for optimizations.

1) For rupture life, notch rupture life, and rupture ductility: Anneal at 1700 to 1850 °F, age at 1325 °F for 8 hr, furnace cool to 1150 °F and hold at 1150 °F for total aging time of 18 hr, air cool.
2) For tensile limited applications, impact strength, and low-temperature notch tensile strength: Anneal at 1900 to 1950 °F, age at 1400 °F for 10 hr, furnace cool to 1200 °F and hold at 1200 °F for total aging time of 20 hr, air cool.

Final material conditions can be specified only after a thorough study of the end-of-life property requirements.

TENSILE

Inconel 718 is a material of interest for springs and grid spacers because of its high yield strength at temperature. A compilation of tensile properties of Inconel 718[9] is shown on Table 3. Also shown are properties for the more commonly known Type 304 Stainless Steel.[12] Regardless of the heat treatment given Inconel 718, the short-time tensile strength does not start to decrease appreciably until above 1200 °F. The ductility remains fairly constant to 1300 °F, increasing thereafter.

TABLE 3. Tensile Properties of Inconel 718 and Type 304 Stainless Steel***

Temp °F	0.2% Yield strength, ksi		Ultimate tensile strength, ksi		Total elongation, %	
	Inconel 718	Type 304	Inconel 718	Type 304	Inconel 718	Type 304
RT	144	35	185	85	44	59
200	139	35	180	80	44	54
400	132	35	173	74	44	48
600	128	35	168	66	44	42
800	126	35	166	61	44	36
900	125	35	161	57	44	36
1000	125	34	160	55	44	35
1100	130	33	164	50	44	33
1200	128	30	164	45	44	31
1300	118	26	146	38	46	32
1400	103	22	120	30	52	33
1500	84	18	87	24	64	36

* *Mill annealed sheet, aged 1325 °F/16 hr*

** *Annealed*

The majority of the irradiation effects data at elevated temperature have been generated by BNW, a summary of which is shown in Table 4.[13-15] Additional irradiation effects data have been generated at Karlsruhe, but the tests were made at low irradiation temperatures. Generally, strength is affected only a small amount (approximately ±20%) by neutron irradiations to 1 x 10^{21} n/cm^2 in thermal reactors. Like most other nickel-base alloys, ductility drops to very low values (less than 1%) at the higher test temperatures (about 1200 °F). Table 4 shows that there are no mechanical property data for Inconel 718 beyond about 10^{21} n/cm^2 and that the data presently available were obtained entirely from thermal reactor irradiations.

STRESS RELAXATION

Inconel 718 can be heat treated and aged such that stress relaxations on the order of 35% (maximum) occur in 10,000 hr at 1200 °F (unirradiated), for initial stresses of 60,000 psi. Table 5 shows the comparative stress relaxation behavior of Inconel 718[9] and Inconel X-750.[16] These same data are plotted in Figure 1. The data shown on Table 5 or Figure 1 justify the selection of Inconel 718 over Inconel X-750 for spring applications.

No stress relaxation data on irradiated Inconel 718 have been generated. The only stress relaxation data available for irradiated materials concerns Rene´ 41, Nimonic 80A, and Type 304 stainless steel.

Rene´ 41 springs irradiated in the Battelle Research Reactor at 1050 and 1150 °F (Table 6) showed apparent effects of neutron exposure for fluences ranging from 2 x 10^{17} to 2 x 10^{19} n/cm^2 (E > 0.1 MeV).[17] However, one might conclude by examination of the unirradiated (3800 hr at 1150 °F) data that thermal rather than irradiation effects are of major importance.

TABLE 4. Tensile Properties of Irradiated Inconel 718

Heat treat-ment*	Test temp, °C	Irrad temp, °C	Fluence nvt E > 1 MeV	0.2% Y.S., ksi		U.T.S., ksi		Total elong., %		Ref
				Unirrad	Irrad	Unirrad	Irrad	Unirrad	Irrad	
A	RT	280	$4.5x10^{19}$	158-162	178	190-200	196	12-21	0	13
A	RT	280	$5.9x10^{19}$	158-162	192	190-200	205	12-21	12-14	13
A	RT	280	$7.2x10^{19}$	158-162	160-170	190-200	180-188	12-21	7-11	13
A	RT	280	$1.1x10^{20}$	158-162	135-150	190-200	164-179	12-21	17-25	13
A	RT	280	$1.6x10^{20}$	122.2	130.8	167.0	156.5	26.7	32.7	14
A	RT	280	$3.1x10^{20}$	164.8	146.7	192.5	173.6	14.7	21.3	14
A	RT	280	$5.4x10^{20}$	158-162	145-163	190-200	170-185	12-21	10-20	13
A	RT	280	$1.5x10^{21}$		176.2		191.4		11.3	14
A	RT	>300	$2.4x10^{20}$	78.2	80.4	132.7	111.4	21.5	6.4	14
A	RT	740	$1.9x10^{20}$		102.4		128.8		5.9	14
A	300	>300	$1.5x10^{20}$	65.3	62.3	122.1	106.3	118.6	8.1	14
A	300	740	$9.2x10^{19}$	65.4	71.4	127	118.8	27	12.2	15
A	300	740	$1.9x10^{20}$	71.4	78	128	117.8	18.7	9.9	14
A	300	740	$2.4x10^{20}$	65.3	62.8	122	106.3	18.7	8.1	15
A	650	50	$1.8x10^{20}$	102	122	119	139	12.6	2.6	15
A	650	50	$1x10^{21}$	102	130	119	131	12.6	0.8	15
A	650	280	$1x10^{20}$	121	87.4	138	99.8	8.1	5.9	15
A	650	280	$1.6x10^{20}$	121	95.5	138	108	8.1	5.7	15
A	650	280	$3.2x10^{20}$	121	139	138	140	8.1	0.9	15
A	650	>300	$1.5x10^{20}$	63.1	51.5	106.1	79.2	25.2	5.7	14
A	650	740	$9.2x10^{19}$	64	51.5	104	79.6	25.6	6.0	15
A	650	740	$1.9x10^{20}$	64	62.1	104	89.1	25.6	5.0	15
A	650	740	$2.4x10^{20}$	64	51.5	104	79.2	25.6	5.7	15
A	650	740	$4.0x10^{20}$	64	62.8	104	94.4	25.6	6.7	15
B	704	704	$3.6x10^{20}$	114.5	95.8	125.2	95.8	18.9	0.8	14

* *Heat treatment code:*

A - S.T., + 1325 °F/8 hr, F.C. 100 °F/hr to 1150 °F/8 hr, A.C.
B - 1750 °F/ 1/2 hr, W.Q.

TABLE 5. A Comparison of Stress Relaxation Data for Two Inconel Alloys

	Inconel 718(1)	% Relax-ation	Inconel X-750(2)	% Relax-ation
Temp °F	1000		1000	
Initial stress, ksi	60		60	
Residual stress				
after: 1 hr	60	0	59.5	0.83
10 hr	60	0	59	1.6
100 hr	60	0	58.5	2.5
1000 hr	62	0	(58)(3)	3.3
10000 hr	58	3.3		
Temp °F	1100		1100	
Initial stress, ksi	60		60	
Residual stress				
after: 1 hr	60	0	59	1.6
10 hr	60	0	57.5	4.2
100 hr	59.6	0.67	54.5	9.2
1000 hr	58.2	3.0	(49)(3)	18.3
10000 hr	51.5	14.2		
Temp °F	1200		1200	
Initial stress, ksi	60		57	
Residual stress				
after: 1 hr	59	1.6	55	3.5
10 hr	58	3.3	49.5	13.2
100 hr	56.9	5.2	36	36.8
1000 hr	54.4	9.3	(15)(3)	73.7
10000 hr	39.0	35		

(1) *ST 1800 °F/1 hr W.Q. + 1325 °F/16 hr* (3) *estimated from ∿700 hr data* *Reference 9*

(2) *ST 1625 °F/24 hr, A.C. + 1300 °F/20 hr* *Reference 16*

(3) *Estimated from ∿700 hr data*

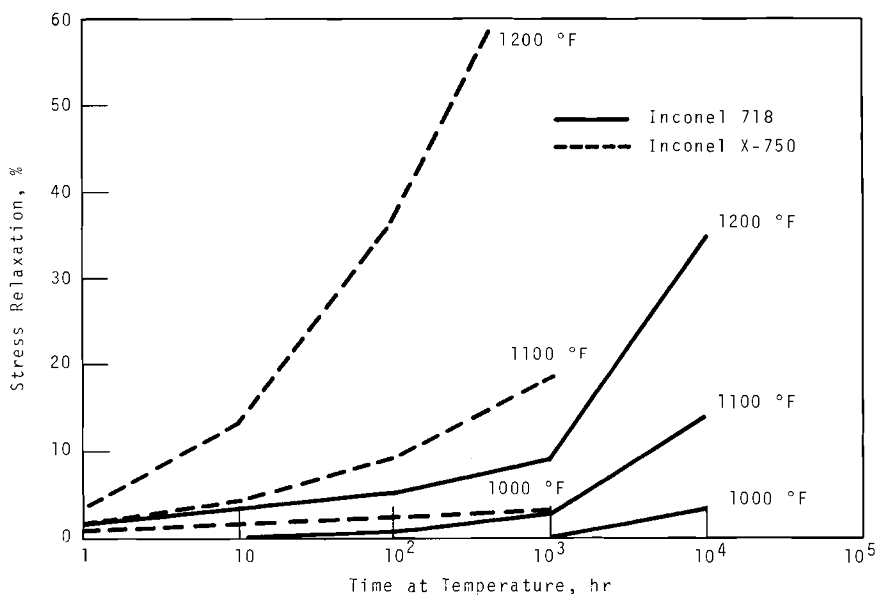

FIGURE 1. Percentage Stress Relaxation of Inconel 718 and Inconel X-750 for Initial Stress from 57 to 60 ksi

The Nimonic 80A data (Figure 2) show that relaxations during neutron irradiations at 615 to 975 °F are considerably larger than the thermal relaxations (30 to 50%) at all temperatures indicating a significant effect of irradiation.[18]

*TABLE 6. Effects of Irradiation on Rene´ 41 Springs**

Temp °F	Fluence, n/cm^2 (E > 0.1 MeV)	Set load, lb		Relaxation (% change)
		Pre-test	Post-test	
1050	2 x 10^{19}	34.0	32.6	- 4.1
1050	0 (3800 hr)	35.0	32.3	- 7.8
1150	1 x 10^{19}	35.4	27.9	-21.2
1150	3.1 x 10^{18}	35.9	29.3	-18.4
1150	2.4 x 10^{17}	35.0	32.6	- 6.9
1150	0 (3800 hr)	34.8	26.1	-25
1150	0 (3800 hr)	34.6	28.3	-18.2

* *Age-hardened 1400 °F for 16 hr*

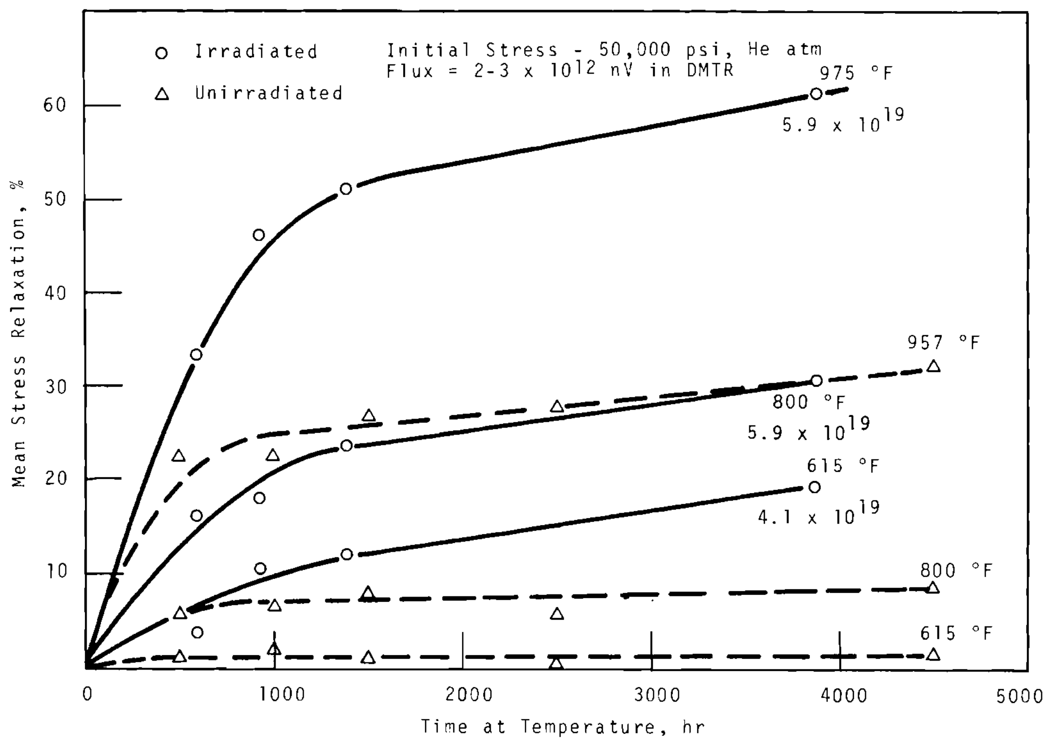

FIGURE 2. Effects of Neutron Irradiation on the Stress-Relaxation Properties of Nimonic 80A

The Type 304 stainless steel data showed that "all of the relaxation experienced by 200 °F in-reactor exposures was the result of irradiation."(19) The stress relaxation in specimens with an initial stress of 20,000 psi saturated at a fluence of 4 x 10^{20} n/cm^2. Saturation effects were not experienced on specimens with higher initial stresses. Figure 3 is a plot of the data obtained in this experiment.

The foregoing irradiation experience with both nickel- and iron-base alloys indicates that combined irradiation and thermally induced stress relaxations in Inconel 718 may be expected in the FTR. The extent of relaxation under simulated operating conditions and its sensitivity to fluence level is yet to be experimentally determined.

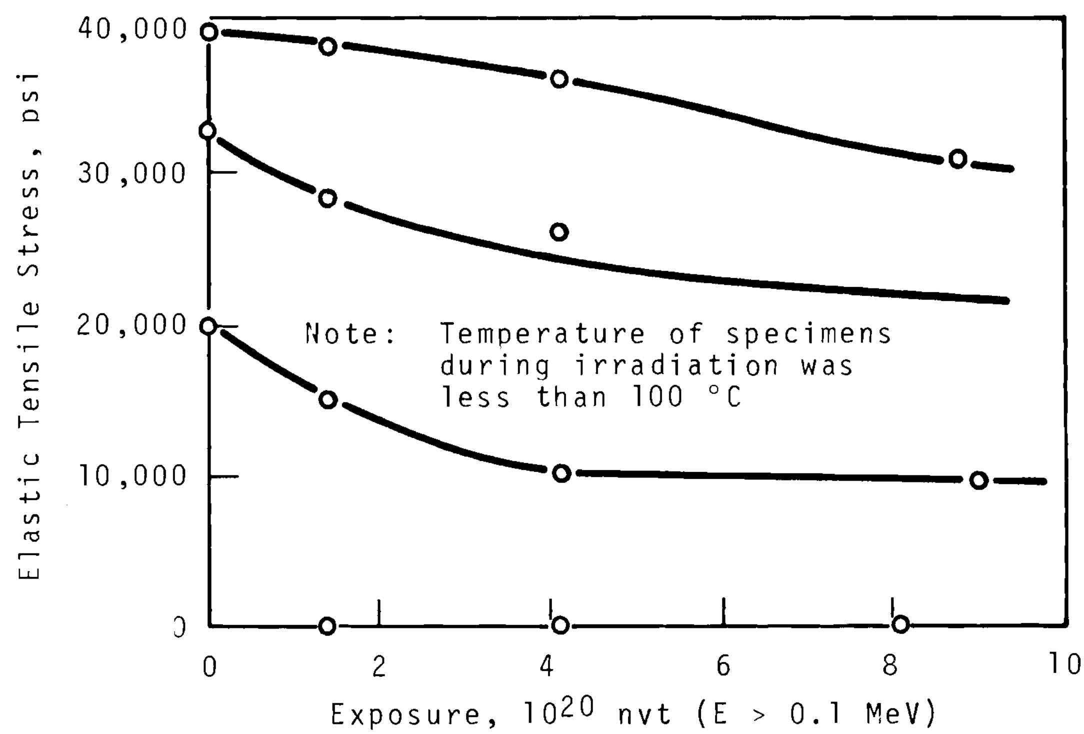

FIGURE 3. Stress Relaxation in Type 304 Stainless Steel Resulting from Irradiation at About 200 °F

CREEP AND STRESS RUPTURE

The creep and stress to rupture properties of Inconel 718 have been well documented by Huntington[9-11] in the temperature range 1000 to 1600 °F and for times to 10,000 hr. The time to rupture and secondary creep rate versus stress curves are shown in Figures 4 and 5, respectively. These curves indicate the strength advantages of Inconel 718 over Type 304 stainless steel for normal nonnuclear applications. Although the slopes of the curves at these temperatures suggest that 304 stainless steel is slightly more stable than Inconel 718, the differences are not significant.

Irradiation effects data for stress rupture and creep properties of Inconel 718 are nonexistent at this time. Data reported for Inconel X-750 indicate that irradiations at 540 °F to fluence levels of 1×10^{20} n/cm^2 (E > 1 MeV) reduce the 1350 °F rupture life (at a stress of 37.5 ksi) by about 70%.[20] This is comparable to what has also been seen in Type 304 stainless steel.[21] It would seem reasonable to expect, in the absence of actual data on Inconel 718, that the time to rupture will be substantially reduced by modest neutron exposures. There is no basis for predicting how creep rates will be affected.

FATIGUE

There is very little difference in the fatigue behavior of Inconel 718 from room temperature to 1200 °F. Figure 6 shows the room temperature (same as 1200 °F) fatigue properties of this material[11] along with low cycle fatigue data for austenitic stainless steels at 1200 °F.[22,23] Again, there are no data on which to base predictions of the effects of irradiation on the fatigue properties of nickel-base alloys, and little or no information on any other materials, including the austenitic stainless steels.

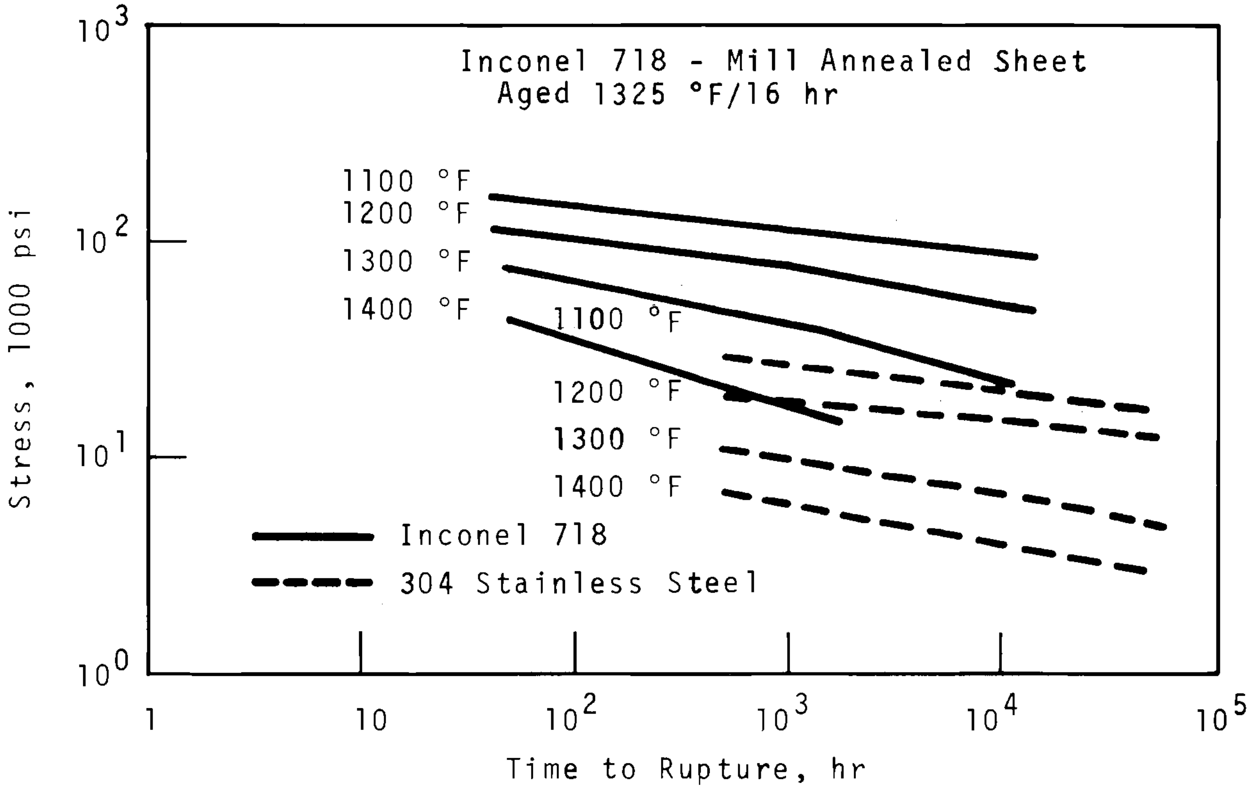

FIGURE 4. Stress Rupture Properties of Inconel 718

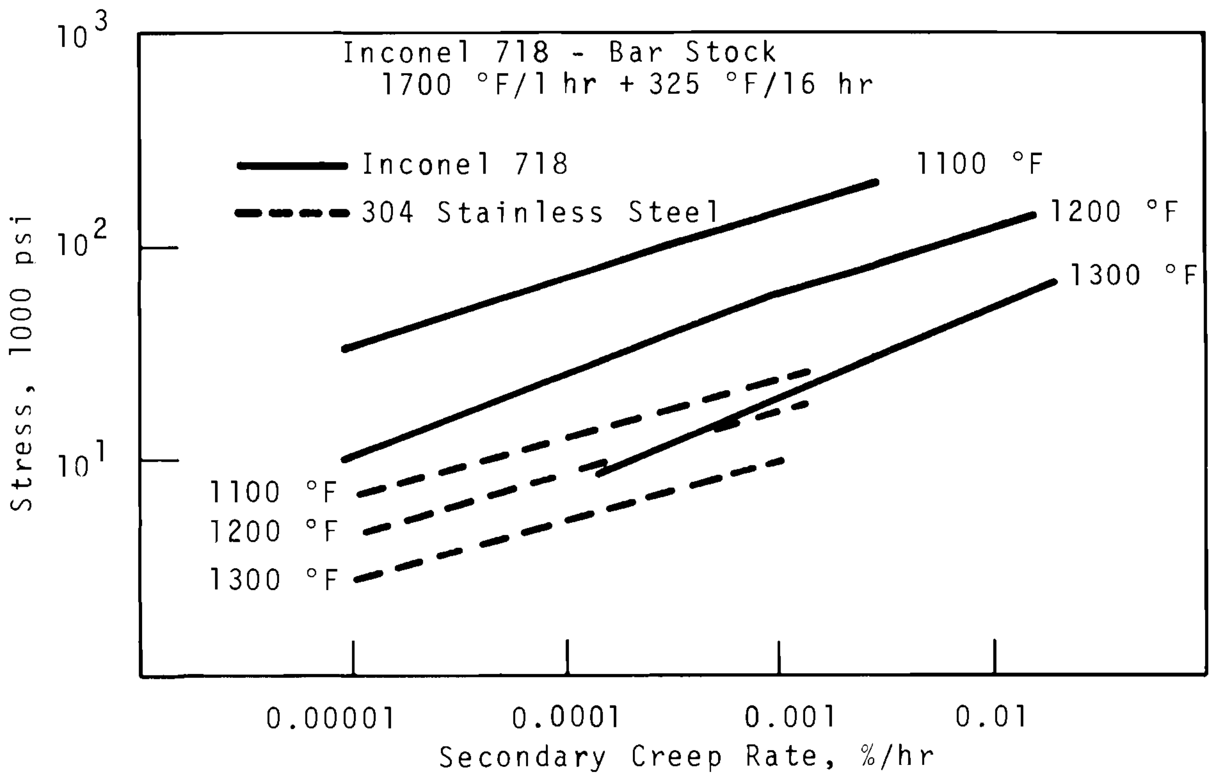

FIGURE 5. Creep Behavior of Inconel 718

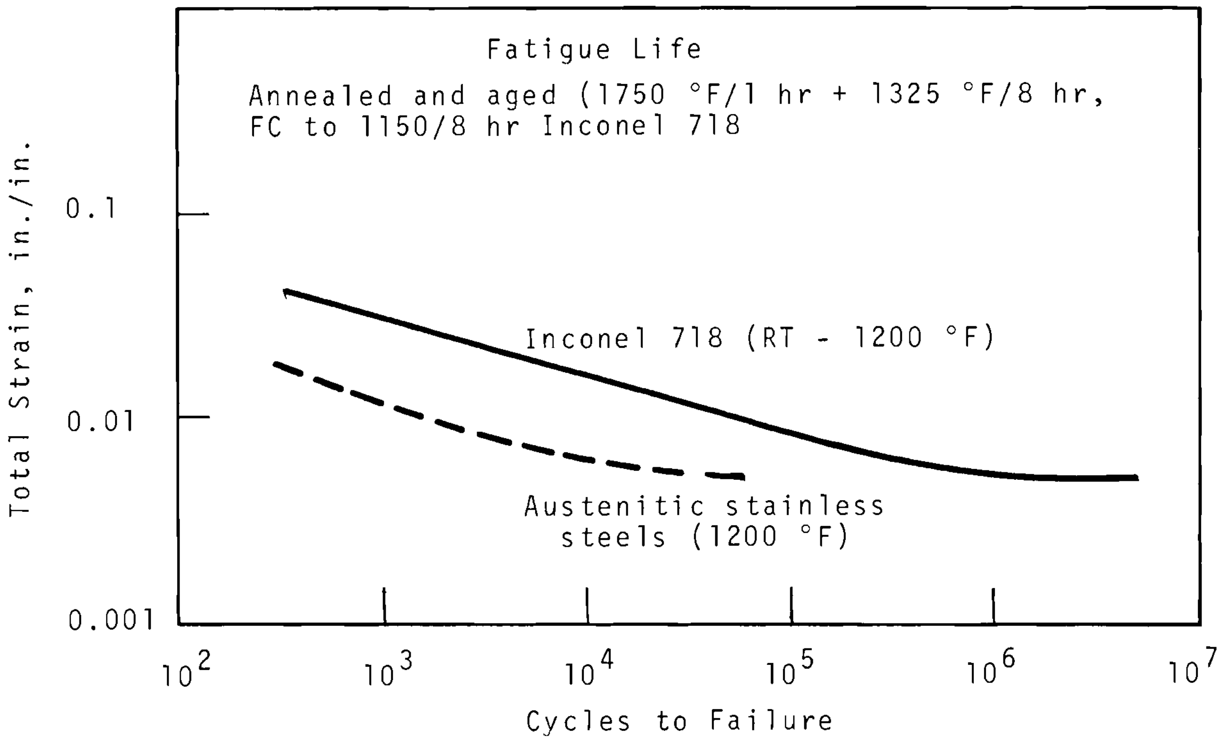

FIGURE 6. Fatigue Behavior of Inconel 718 and Austenitic Stainless Steels

V. CORROSION

Of the three applications discussed for Inconel 718, resistance to sodium corrosion will be an important factor in the final selection of materials for the grid spacer assemblies and safety rod accelerator springs. The primary concern in these two applications is the possibility of a corrosion mechanism by which selective attack of grain boundaries will occur. A corrosion mechanism of this type would be highly undesirable for these components under stress in service. Rapid, uniform corrosion of thin sheet material proposed for use in the grid spacers is also a serious consideration.

Nowhere in the open literature is there specific reference to the behavior of Inconel 718 in sodium. Shannon concluded that, "Inconel is fairly corrosion resistant in an Inconel system, but in a stainless steel system nickel will rapidly dissolve from the Inconel and transfer to stainless steel until both are at equal nickel activities."[24]

Thorley reported that nickel-rich alloys such as Nimonic 80A are insensitive to oxygen changes in sodium, but are markedly sensitive to the velocity of the liquid sodium.[25] From the metallographic appearance of the specimens which he studied, it appeared that the corrosion of this material occurred primarily by the dissolution of nickel from the specimen surface. Sodium velocities in these tests ranged from 5 to 40 ft/sec, with oxygen contents from less than 10 to 25 ppm at 1200 °F. Thorley working with brazing materials found that phosphorous is nearly all removed from Nicrobraz-50 in both static and dynamic 1200 °F sodium. Homogenization of the braze joint before sodium exposure did not prevent or alter the phosphorous removal. Silicon was removed from Nicrobraz-30 to a depth of 0.0025 in. in both dynamic and static sodium tests.

In an earlier report on the generalized use of high nickel content alloys in the FTR, a graph was shown which indicates metal loss as a function of nickel content in various alloys.[26] Based upon a curve of this type, Figure 7, Inconel 718 would be expected to lose 40% more material than austenitic stainless steels under comparable conditions (1200 °F sodium at 30 to 40 ft/sec).

A lot of data were reported in the span of time from 1955 to 1965 on the behavior of Inconel and Inconel X in high temperature alkali metal systems. These data originated from projects such as ANP, MSRE, and SNAP-8. Some of the ANP data were summarized by Oliver.[27] He reported that the creep

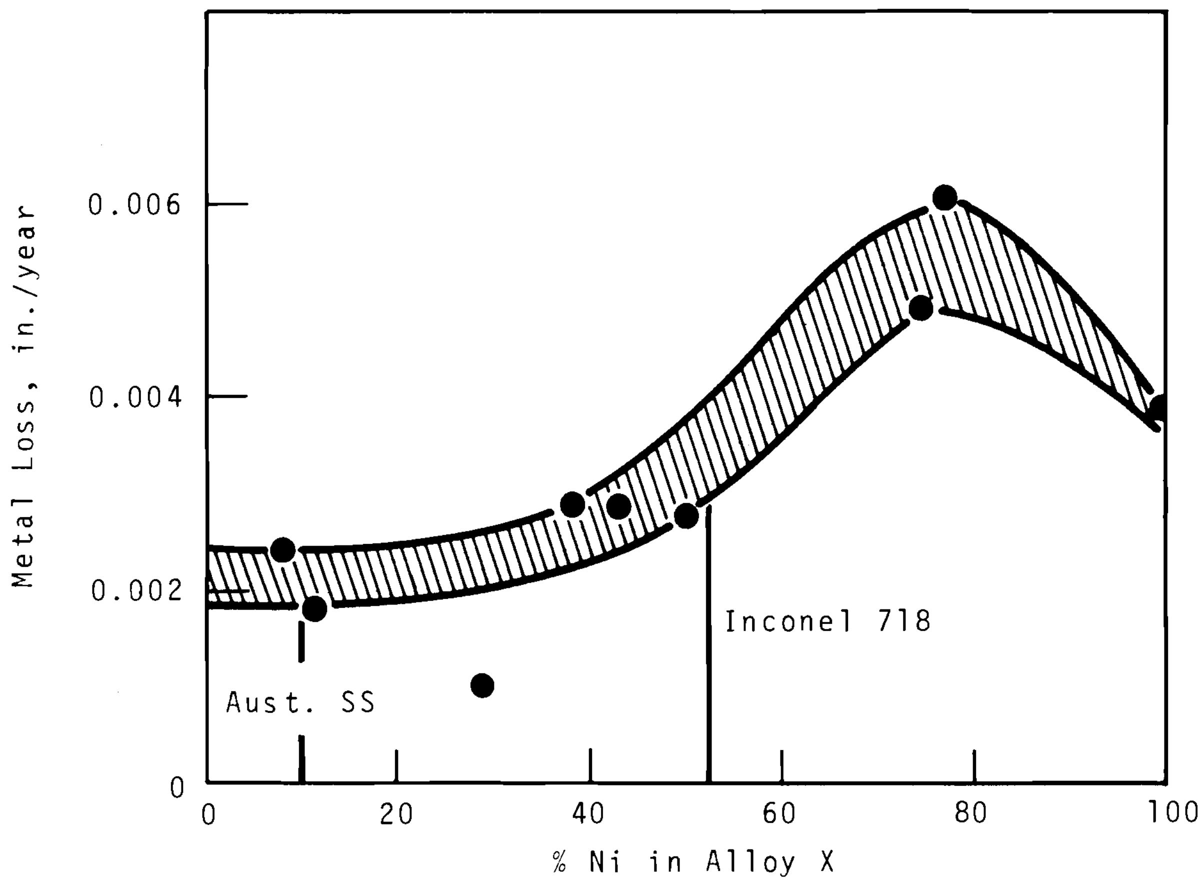

FIGURE 7. Corrosion Behavior of Several Alloys of Varying Nickel Contents

rates of Inconel in 1500 °F sodium at a uniaxial stress of 3500 psi were appreciably greater than creep rates observed in argon or air. The loss of strength in sodium was attributed to decarburization. Microscopic studies, however, revealed no evidence of decarburization. In a later test, Inconel appeared to be compatible with the 1500 °F sodium of high purity; no carburization was observed.

It was possibly from these studies described above that Duffy,[28] in his discussion of materials for Fermi, concluded that Inconel X was highly resistant to irradiation damage in a sodium environment, thus qualifying for use in the Enrico Fermi Reactor.

In summary, there does not appear to be any specific sodium corrosion data on Inconel 718. The general data available on nickel-base alloys, however, indicate that corrosion of the Inconel 718 will not be a design limiting factor, except in the case of thin section components such as grid spacers.

VI. FABRICABILITY

Inconel 718 responds rather slowly at heat-treating temperatures. For this reason, the alloy can be welded and directly aged with less chance of cracking than other nickel-base alloys. Generally then, Inconel 718 should be welded in the solution-annealed condition. The annealing temperature must be consistent with the forming and welding operations and also compatible with the end requirements. The solution-annealing temperature is particularly important for creep-rupture applications, and the temperature should be 1850 °F or lower to avoid notch sensitivity. Where creep-rupture (stress relaxation) strength is not a factor, annealing at 1950 °F following welding and prior to aging results in higher weld metal and heat affected zone ductilities.

Nickel brazing alloys produce joints with adequate strength but with comparatively low ductility. The joint strength approaches that of the base metal. Because of the difficulty in wetting alloys containing aluminum and titanium, these alloys are often nickel-plated prior to brazing. The use of a brazing flux is not recommended on joints that are relatively inaccessible after assembly. The choice of brazing alloy will affect the mechanical properties of the finished joint.

VII. DISCUSSION

In the preceding sections the primary emphasis was placed on reviewing information presently available on the stress relaxation, yield strength, and corrosion behavior of Inconel 718.

These are the properties which will qualify or reject Inconel 718 for use in the FTR. Since these properties are not of equal importance for all three applications, each application must be discussed separately in light of current information.

FLOW DUCT CONTACT PADS

One of the most important requirements of Inconel 718, for applications such as contact pads in high temperature sodium systems, is that it will not undergo self-welding. The flow ducts must be able to be easily withdrawn from the reactor, thus requiring a minimum of sliding contact resistance between ducts. Freede's[(7)] preliminary data indicated that Inconel 718 to Inconel 718 is one of the more favorable material combinations for use in 1200 °F sodium systems. Material removed from the contact pad by uniform corrosion should have little bearing on the overall performance of the reactor. Possibly the most important and unanswered problem at this point is the method for applying or attaching the Inconel 718 to the flow duct (probably austenitic stainless steel) and the overall assurance that the material will not spall off during operation. The point of concern is the difference in the coefficient of thermal expansion between the Inconel 718 and the flow duct material. If the flow duct is constructed of Type 304 stainless steel, the expansion of the stainless steel at 1200 °F will be about 20% greater than the expansion of the Inconel 718. If the Inconel 718 and stainless steel remain ductile throughout their life, localized yielding could compensate for the differences in expansion. However, ductility will be lowered substantially by the neutron fluences involved in this application, thus casting a shadow of doubt on the actual full-term service reliability of these contact pads.

SAFETY ROD ACCELERATOR SPRINGS

There are no unusual fabrication problems associated with the production of the large coil springs of Inconel 718 for safety rod accelerators. The springs will be located in an area of the reactor where neutron exposures will be negligible over the entire length of the spring, as shown in Table 1. The data available indicate that the amount of stress relaxation for unirradiated material at 1200 °F will be within acceptable design limits. The remaining unknown is the corrosion aspect and the effect of high temperature liquid sodium on the stress relaxation behavior of the material. Uniform corrosion of the Inconel 718 spring would present no serious problem from the standpoint of spring performance. Nonuniform corrosion, i.e., intergranular penetration, could however compromise the reliability of the spring, and it is this aspect of corrosion that must be further examined to qualify Inconel 718 for safety rod accelerator springs.

FUEL ELEMENT GRID SPACERS

The thin honeycomb grid spacer assemblies will reside in one of the most adverse environmental regions within the FTR. These assemblies will lie in the peak neutron flux region, the peak sodium flow region, and one of the highest temperature regions. With relatively thin assemblies such as these, uniform corrosion could present serious problems. Furthermore, corrosion problems with nickel-base alloys increase with increasing sodium velocity.

Values for mechanical properties of irradiated Inconel 718 are not available beyond about 10^{21} n/cm^2. If one were to consider thermal reactor data to predict fast reactor behavior, extrapolations over two orders of magnitude would be required. This is not advisable. The acceptable performance of the brazed assembly would also be in question, because of Thorley's[25] sodium corrosion studies on the Nicrobraz alloys. The uncertainties associated with the effects of irradiation on stress relaxation and other mechanical properties of Inconel 718, the effects of high temperature, high velocity sodium, and the behavior of brazed joints in a sodium environment do not favor the use of Inconel 718 honeycomb grid spacers without considerable development efforts.

VIII. CONCLUSIONS

Inconel 718 possesses a number of attractive properties which make that material desirable for certain nuclear and nonnuclear applications. The following conclusions can be drawn in regards to the mechanical, chemical, and fabrication properties of Inconel 718:

- Tensile. Inconel 718 is approximately four times stronger than Type 304 stainless steel over the temperature range of room temperature to 1400 °F, while ductility remains essentially equivalent to that of the stainless steel. There is only a minimal amount of data on the tensile properties of irradiated Inconel 718 to fluences of 10^{21} n/cm^2 (E > 1 MeV).
- Stress Relaxation. Inconel 718 has substantially better stress relaxation properties than Inconel X-750. There are no irradiation effects data per se on Inconel 718, but data on other nickel-base alloys indicate that irradiation will greatly increase the amount of stress relaxation.
- Creep and Stress Rupture. The creep or rupture strength of Inconel 718 is about four times that of Type 304 stainless steel for temperatures of 1100 to 1300 °F. There are no irradiation effects data on these properties for Inconel 718.

- Fatigue. At elevated temperatures (to 1200 °F), the low cycle fatigue life of Inconel 718 is an order of magnitude better than that of austenitic stainless steel, for comparable total strains.
- Corrosion. There are no actual sodium compatibility data for Inconel 718, but data from similar alloys indicate that under comparable conditions of temperature and flow rate, Inconel 718 would experience 40% more metal loss per year than an austenitic stainless steel.
- Fabrication. Inconel 718 is one of the most easily fabricable of the entire family of nickel-base alloys.

A minimal amount of effort will be required to determine if Inconel 718 can qualify for the safety rod accelerator springs and the flow duct contact pads. Development work required for the safety rod springs involves primarily in-sodium stress relaxation tests on scale model springs. Development work required for the contact pads involves perfecting methods of fabrication and determining the performance of fabricated parts, preferably in a sodium/neutron environment at about 1200 °F.

If the reliability for FTR fuel elements is to be determined, the grid spacer assemblies will require extensive corrosion and stress relaxation tests in a prototypic environment: i.e., fast neutron spectrum with high temperature, high velocity sodium.

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