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FACTORS AFFECTING THE MECHANICAL PROPERTIES OF STAINLESS STEELS AT MPRE OPERATING TEMPERATURES

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ABSTRACT

The mechanical properties pertinent to use in the MPRE for types 304 and 316 stainless steel were reviewed. The strength of the unirradiated wrought materials is adequate, but design will have to consider the lower strength of weld metal. Irradiation has a greater effect on ductility than temperature has; quantitative information on radiation embrittlement in creep is inadequate.

INTRODUCTION

Selection of materials for reactor application involves only a few very basic criteria. One must establish the properties necessary for successful component operation and then select the most economical material that satisfies those requirements. The scope of this investigation is not to establish material requirements for the Medium-Power Reactor Experiment (MPRE) but to point out the major factors affecting the mechanical properties of the prime candidate materials at the MPRE operating conditions and some potential problems with materials. We shall limit ourselves to types 304 and 316 stainless steel and to the tensile, creep, and stress-rupture properties. Another important property, strain fatigue, is discussed in a companion report.[1] At MPRE operating temperatures of 1000 to 1550°F, aging effects (time-temperature), compositional variation, and irradiation-induced property changes for both wrought and weld metal are the factors of primary interest for an operating life of 10,000 hr.

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[1] R. W. Swindeman, Fatigue of Austenitic Stainless Steels in the Low and Intermediate Cycle Range, ORNL-TM-1363 (January 1966).

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STRESS-RUPTURE AND CREEP PROPERTIES OF WROUGHT STAINLESS STEEL

Design of the MPRE hardware for the 1000 to 1550°F temperature range requires knowledge of the creep and stress-rupture properties of the candidate structural materials. Typical stress-rupture properties of type 316 stainless steel are given in Fig. 1.

The maximum allowable stress for types 316 and 304 stainless steel as set forth in the revised ASME Pressure Vessel Code is given in Table 1. The maximum nominal hoop stress in the MPRE pressure vessel is about 700 psi, which is well within the limits implied by Table 1 for the projected 10,000-hr lifetime of the MPRE. The relatively heavy wall thickness (1/4 to 1/2 in.) should reduce the probability of these properties being degraded because of reactions between the vessel wall and the environment.

Fission-gas release from the UO_2 fuel will produce the stresses in the fuel element cladding given by Table 2, taken from a companion report.[2] At the anticipated gas-release rate of about 1%, the element wall stress of 31 psi will present no problems. However, the fuel element cladding is thin (0.020 in.) and thus, in contrast to the pressure vessel, its properties can be influenced significantly by interaction with the liquid-metal environment. Most stress-rupture and creep tests[3] performed in liquid metal containing carbon or oxygen show that the strength is not significantly affected. These data are shown in Fig. 2.

As shown in Fig. 3, aging at temperature produces a structure with poorer stress-rupture properties.[4] Several laboratories have accumulated pertinent data. Data of this type are useful for predicting the lifetime of reactor hardware that is not stressed until sometime after reactor startup. It is important, however, to temper the data in Fig. 3 in using

[2] G. Samuels, Fuel Element Design for Boiling Potassium Reactors, ORNL-TM-1344 (January 1966).

[3] MSA Research Corporation Quarterly Progress Report April-May-June 1965, MSA Report 65-96 (Aug. 6, 1965).

[4] Data taken from D. T. Bourgette, Evaporation of Iron-, Nickel-, and Cobalt-Base Alloys at 760 to 980°C in High Vacuums, ORNL-3677 (November 1964) and D. T. Bourgette and H. E. McCoy, unpublished results.

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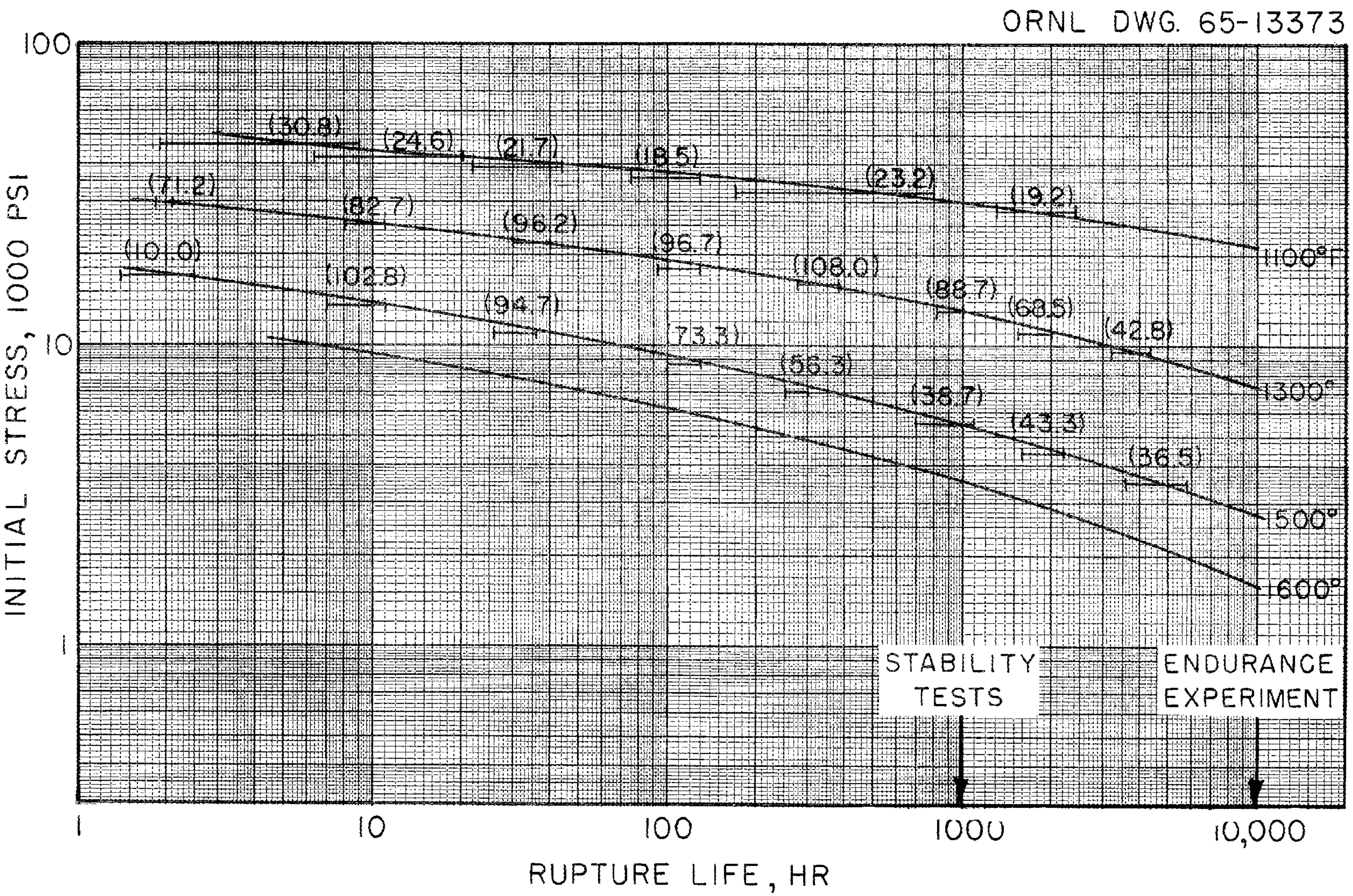

Fig. 1. Stress Rupture of Type 316 Stainless Steel.

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Table 1. Revised ASME Specifications of Maximum Allowable Stress

Material	Maximum Allowable Stress,[a] psi, at Various Temperatures						
	1000°F	1100°F	1200°F	1300°F	1400°F	1500°F	1550°F[b]
Stainless steel type 304	9450	8250	5500	3400	2050	1250	1000
Stainless steel type 304L	7600						
Stainless steel type 316	9800	9450	6950	4000	2000	1150	950
Stainless steel type 316L	8150						

[a]The lesser of the stress to cause creep at a rate of 0.01% per 1000 hr and 60% of the stress to produce rupture in 100,000 hr.

[b]Extrapolated.

Table 2. Effects of Assumed Gas Release Rate from UO_2 on Internal Pressure and Capsule Wall Stress for the MPRE Fuel Element

Assumed Gas Release Rate (%)	Internal Pressure (psia)	Differential Pressure (psi)	Capsule Wall Stress (psi)	Gas Composition (% He)
0	27.7	−1.3	−15	100
1	31.6	2.6	31	88
3	39.4	10.4	125	71
5	47.2	18.2	218	59
10	66.6	37.6	451	42

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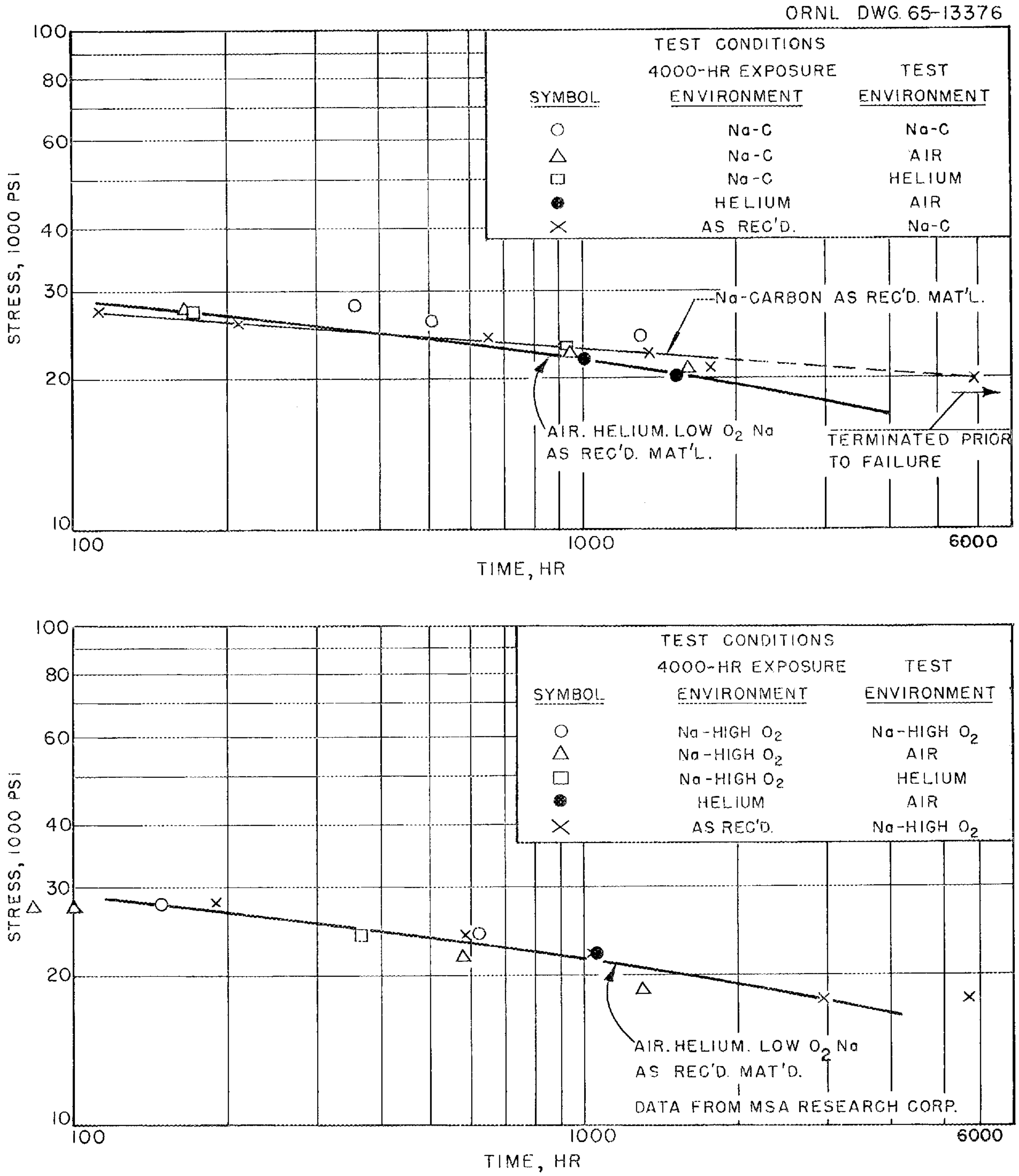

Fig. 2. Effect of Pretest Exposure and Test Environment on Creep Rupture of Type 316 Stainless Steel at 1200°F.

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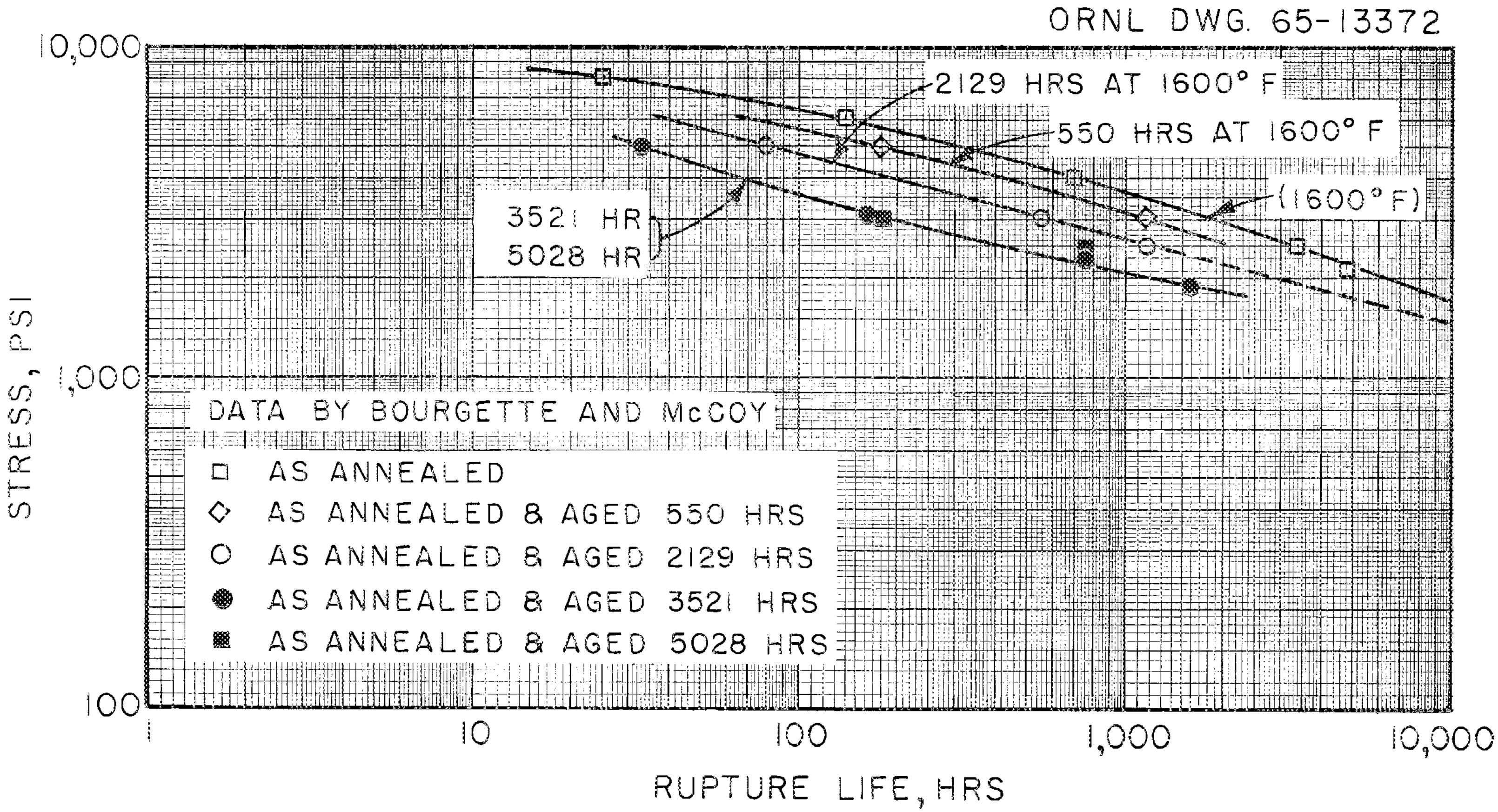

Fig. 3. Effect of Prior Aging at 1600°F on the Stress Rupture Properties of Type 316 Stainless Steel at 1600°F.

it for reactor applications by consideration of both the time to rupture and the aging time to give a better estimate of the lifetime of the equipment. These stress-rupture data for wrought material indicate that type 316 stainless steel should be able to withstand the predicted MPRE stress levels and temperatures, without rupture owing to mechanical stresses, for either the reactor performance or endurance tests.

STRENGTH PROPERTIES OF WELDMENTS

Very little appears to be known about the properties of sound weldments at elevated temperature, particularly as concerns creep and stress rupture. Most designs of hardware circumvent this problem by placing the weldments outside of highly stressed regions so that they do not operate in the creep range. This approach is not always possible.

A review of recent articles[5-7] on the creep and stress-rupture of type 316 stainless steel weldments reveals the following.

1. The stress-rupture properties of weld metal are generally lower than those of the wrought product. This effect becomes substaintial at temperatures above 1100°F, and the difference in strength between wrought material and weld deposits increases with increasing temperature. The reduction in long-time (1000 to 10,000 hr) rupture strength of weldments is associated with poor ductility in the weld bead material (as low as 1%).

2. Over the temperature range 1350 to 1650°F the creep rate of all weldments appears to be quite similar to that of the wrought product.

3. Postweld heat treatment at temperatures above 1650°F, for example 1950°F, improves the rupture strength and ductility at 1500°F for both restrained and unrestrained weldments.

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[5]R. D. Wylie, C. L. Corey, and W. E. Leyda, Trans. Am. Soc. Mech. Engrs. 76, 1094 (1954).

[6]G. H. Rowe and J. R. Stewart, Welding J. (N.Y.) 41, 534-s (1962).

[7]N. T. Williams and G. Willoughby, "The Notch Rupture Behaviour of Some Austenitic Steels," p. 4-11 in Joint International Conference on Creep, Institution of Mechanical Engineers, London, 1963.

Examples of comparative strengths of base metal and weld metal from the work of Wylie, Corey, and Leyda[5] are given in Table 3.

Table 3. Stress Required to Produce Rupture in Wrought and Welded Type 316 Stainless Steel

Temperature (°F)	Rupture Stress, psi, for Time Indicated					
	100 hr		1000 hr		10,000 hr	
	Wrought	Weld	Wrought	Weld	Wrought	Weld
1200	30,000	28,000	20,000	19,500	16,000	10,500
1500	9,000	6,000	5,500	3,000	2,700	2,100

These data should be compared to the weld efficiency data given by Rowe and Stewart.[6] Weld efficiency is a comparison of the relative strength of wrought and weld material. Data for type 316 stainless steel are as follows:

Stress-Rupture Temperature (°F)	Rupture Time (hr)	Weld Efficiency (%)
1350	100	100
	1000	80
	5000	70
1500	100	80
	1000	60
	5000	50
1650	100	80
	1000	70
	5000	60

The influence of postweld heat treatment is best illustrated by the data given in Table 4, again from the work of Rowe and Stewart.[6]

The comparative properties of wrought material and weldments given herein for type 316 stainless steel appear to be generally applicable for the austenitic class of stainless steels. If we then accept these data as applicable for type 304 stainless steel as well, the following conclusions appear in order.

Table 4. Uniaxial Rupture Data for Type 316 Stainless Steel Restrained and Unrestrained Welds Tested at 1350, 1500, and 1650°F

Condition	Stress (psi)	Time at Load (hr)	Reduction in Area (%)	Elongation, % Total	Elongation, % Across Weld	Failure Location	Time to 3rd Stage Creep, hr	Strain at 3rd Stage Creep, %	Minimum Creep Rate (hr^{-1})
1350°F Unrestrained									
As-welded	15,000	179	54	43		Base metal	60	8	1.1×10^{-3}
As-welded	10,000	731	15	10		Weld bead	731	10	1.4×10^{-4}
1600°F anneal	10,000	655	10	6		Weld bead	250	2	6.3×10^{-5}
As-welded	9,000	650[a]		3.4[a]					5.1×10^{-5}
1600°F anneal	8,000	1468	8	6		Weld bead	1150	5	3.9×10^{-5}
1600°F anneal	7,000	2523	3	2		Weld bead	2360	2	7.1×10^{-6}
1500°F Restrained									
As-welded	10,000	57	19	20	7.7	Weld bead	45	5	2.5×10^{-3}
1600°F anneal	10,000	55	21	20	13.4	Weld bead	25	7	2.5×10^{-3}
1950°F anneal	10,000	54	25	29	27.3	Weld bead	20	8	4.0×10^{-3}
As-welded	6,500	234	7	8	2.9	Weld bead	220	5	2.4×10^{-4}
1600°F anneal	6,500	243	4	5	4.9	Weld bead	240	5	1.9×10^{-4}
1950°F anneal	6,500	360	13	11	13.3	Weld bead	310	9	2.8×10^{-4}
As-welded	5,000	467	2	2	6.4	Weld bead	425	2	3.4×10^{-5}
1600°F anneal	5,000	463	0.1	0.9	8.0	Weld bead	463	0.9	2.0×10^{-5}
1950°F anneal	5,000	819	4	4	8.7	Weld bead	725	2	3.2×10^{-5}
1500°F Unrestrained									
As-welded	10,000	74	38	42	15.6	Weld bead	25	7	2.0×10^{-3}
1600°F anneal	10,000	30	17	16	13.9	Weld bead	15	7	4.3×10^{-3}
1950°F anneal	10,000	41	21	26	21.0	Weld bead	15	7	4.6×10^{-3}
As-welded	6,500	225	9	7	2.9	Weld bead	215	6	2.7×10^{-4}
1600°F anneal	6,500	210	8	7	3.4	Weld bead	195	5	2.5×10^{-4}
1950°F anneal	6,500	226	12	12	12.2	Weld bead	210	9	4.6×10^{-4}
As-welded	5,000	373	3	3	4.7	Weld bead	340	2	4.8×10^{-5}
1600°F anneal	5,000	359	3	3		Weld bead	325	2	4.7×10^{-5}
1950°F anneal	5,000	587	4	4	3.6	Weld bead	565	4	9.7×10^{-5}
As-welded	3,500	1285	1	1	5.0	Weld bead	1000	0.3	3.1×10^{-6}
1600°F anneal	3,500	1335	2	2	4.2	Weld bead	850	0.6	6.1×10^{-6}
1950°F anneal	3,500	1819	3	1	0	Weld bead	900	0.5	5.0×10^{-6}
1600°F anneal	2,500	5527	1	2		Weld bead	3250	0.4	1.1×10^{-6}
1650°F Unrestrained									
As-welded	5,000	68	14	10		Weld bead	50	6	1.2×10^{-3}
1950°F anneal	5,000	61	16	17		Weld bead	42	9	
As-welded	3,000	417	3	4		Weld bead	375	3	7.2×10^{-5}
1600°F anneal	3,000	215	3	3		Weld bead	170	1	6.1×10^{-5}
1950°F anneal	3,000	727	12	12		Weld bead	495	7	1.3×10^{-4}
As-welded	2,000	1991	4	3		Weld bead	660	0.5	8.0×10^{-6}
1600°F anneal	2,000	1148	0	0.1		Weld bead	1148	0.1	3.9×10^{-6}
1950°F anneal	2,000	3590	12	12		Weld bead	1100	2	1.4×10^{-5}

[a]Test discontinued prior to failure.

1. Design stresses in or near weldments should not be based on wrought product results, but rather these allowable stresses should be reduced by about 50 or 30% for operating temperatures in the neighborhood of 1500°F and 1300°F, respectively.

2. Postweld heat treatments may improve weld ductility and strength, but these treatments will have to be at a temperature of about 1950°F.

DUCTILITY OF STAINLESS STEELS

The application of stainless steels in the temperature range of 1000 to 1550°F results in the precipitation of carbides and the formation of sigma and chi phases. The morphology of the carbides and the carbon solubility vary greatly over the temperature range cited. Therefore, heat treatment for times of 1000 to 10,000 hr will produce vastly different structures. The relative importance of carbides versus the chi or sigma phases is difficult to determine. It is generally believed, perhaps without justification, that all of these reactions significantly affect the strength and ductility of the alloys. Our interest must be focused on mechanical properties both at elevated temperature and at room temperature. At the higher temperatures one should consider short-time tensile, creep, stress-rupture, and fatigue properties.

Elevated Temperature Ductility

Some typical ductility data obtained in connection with stress-rupture tests are given in Fig. 4. The lowest ductility was found in tests at 1100°F. At the higher temperatures of 1200, 1300, and 1500°F, the ductility is markedly improved even for times of 5000 hr. The type 316 alloy after aging has been evaluated both in short-time tensile tests and stress rupture. The short-time tensile test results for type 316 stainless steel aged 1000 hr at 1475°F and 1650°F are given in Table 5. The aging treatment did not change the elevated-temperature ductility significantly. Figure 4 includes the ductilities associated with the stress-rupture data given in Fig. 3 for aged samples.

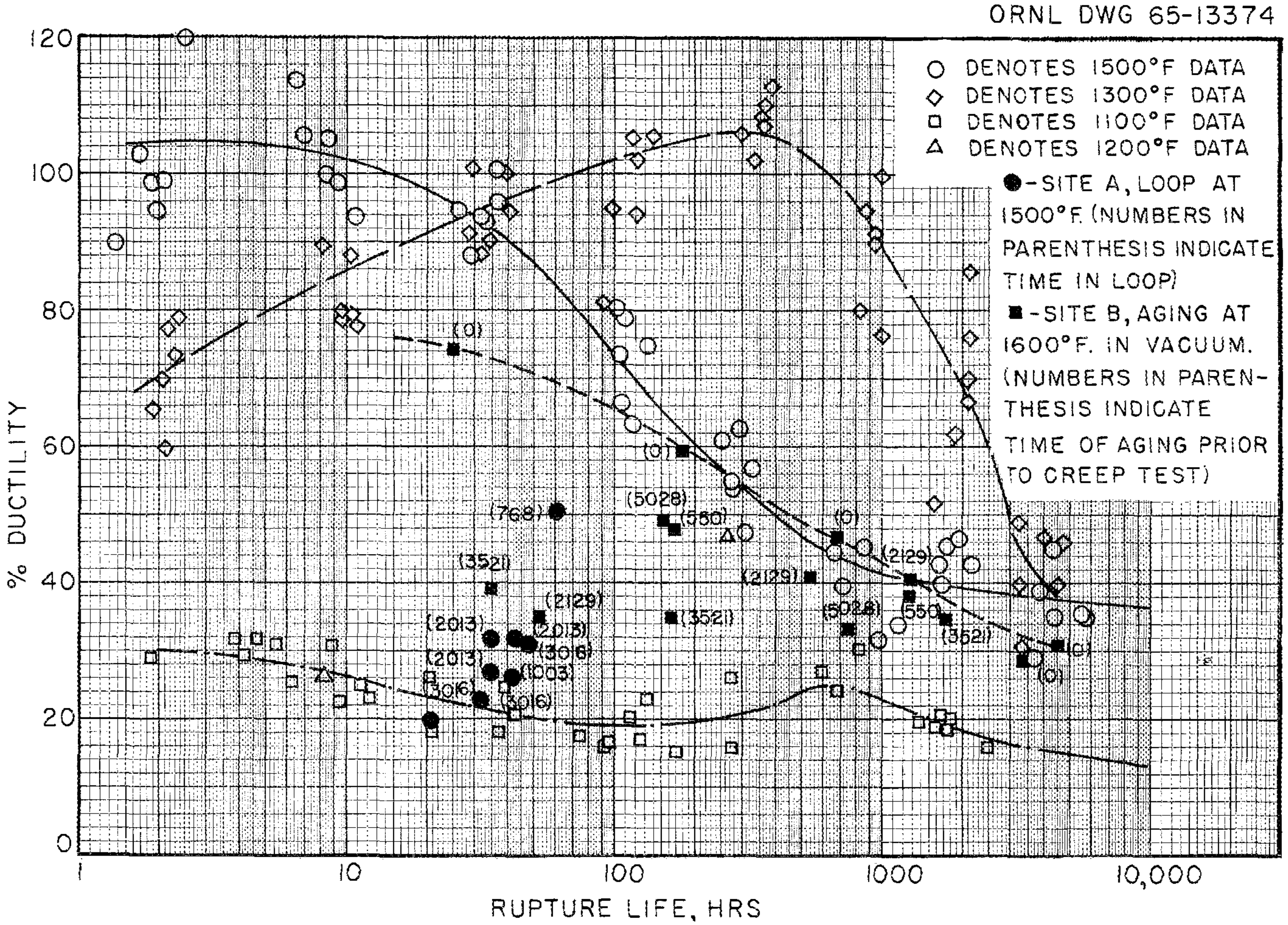

Fig. 4. Effect of Temperature on the Ductility of Type 316 Stainless Steel in Creep-Rupture Tests.

[redacted]

Table 5. Short-Time Tensile Ductility and Yield Strength of Unirradiated Type 316 Stainless Steel

Deformation Temperature (°F)	Aging Conditions[a]	Type 316L (0.06% C)		Type 316H (0.14% C)	
		Total Elongation (%)	Yield Stress (psi)	Total Elongation (%)	Yield Stress (psi)
			x 10^3		x 10^3
1300	As annealed	46.0	19.7	54.5	22.3
1550	As annealed	44.6	12.5	83.5	10.7
1475	1000 hr at 1475°F	38.2	13.4	52.4	14.6
	1000 hr at 1650°F	37.0	14.0	58.4	13.0
1650	1000 hr at 1475°F	36.9	8.5	92.1	7.0
	1000 hr at 1650°F	30.5	7.6	75.8	6.1

[a]Alloy solution annealed at 1900°F for 1 hr prior to aging.

The creep ductility decreased as a result of the aging treatment, but the ductility observed is greater than that observed for unaged samples in the temperature range where the alloy is commonly used (1100°F). From these data one concludes that aging can either increase or decrease the elevated-temperature ductility of type 316 stainless steel. However, for the 1000 to 10,000 hr lifetime the ductility of the unirradiated alloy in all cases is above 20% and therefore is adequate.

Room Temperature Properties After Aging at Elevated Temperature

When the reactor is cooled down, one must be assured that the impact properties of the structural materials are not so low as to present a risk of failure during maintenance or examination. Most of the other room-temperature mechanical properties, such as the tensile strength, are probably of little importance. The data given in Table 6 show the impact values[8] for both types 304 and 316 stainless steels after aging

[8]P. K. Menard, Sigma Phase in Austenitic Stainless Steels, Wood Works Plant of Irvin Works, U.S. Steel, Report P-12-121 (May 14, 1952).

[redacted]

[redacted]

Table 6. Effect of Aging at Elevated Temperatures on Room-Temperature Impact Strength of Austenitic Stainless Steels

Aging Time (hr)	Impact Strength, ft-lb, after Aging at			
	1200°F	1350°F	1500°F	1650°F
		Type 304		
0	86	86	86	86
10	92	92	80[a]	111[a]
100	90	89[a]	76[a]	100[a]
1,000	85[a]	73[a]	82[a]	85[a]
10,000	69[a]	67[a]	74[a]	
		Type 316		
0	83	83	83	83
10	116	67	61[a]	94[a]
100	88	70[a]	68[a]	81[a]
1,000	65[a]	60[a]	79[a]	79[a]
10,000	44[a]	34[a]	56[a]	80[a]

[a]Sigma phase present.

for up to 10,000 hr at 1200, 1350, 1500, and 1650°F. These data show that the impact strength of type 304 stainless steel is better than that of type 316 and the minimum absolute value for either material after aging is at 1350°F. The minimum value is, however, well above the 15 ft-lb normally accepted as sufficient for carbon-steel pressure vessels.

We believe that there are two areas of special note when considering the room temperature properties. One is that the impact values for aged samples may be greatly affected by pretest aging treatments. The data in Table 7 obtained by Timkin Roller Bearing Company[9] show this effect to be more significant for type 316 stainless steel than for type 304. As these data show, the aging temperature for minimum impact energy for type 316 stainless steel can be in the 1600°F range if the preaging heat treatment is at about 2000°F. During the brazing of the MPRE fuel elements the cladding will be exposed to the 1900 to 2000°F temperature range.

[9]Resume of High Temperature Investigation Conducted During 1948–50, The Timkin Roller Bearing Company, Canton, Ohio, 1950, pp. 111–12.

[redacted]

Table 7. Characteristics of Stainless Steels after Various Heat Treatments

Conditions		Sigma Phase[a]	Grain Size	Key-hole Charpy Values
Temperature, °F	Time, hr			
Type 304 -- Prior Treatment 1700°F Normalize				
As normalized		No	6/8	87-76-83
1350	25	No	6/8	72-66-60
1350	500	No	6/8	59-62-60
1350	1000	No	6/8	62-60-64
1500	25	No	6/8	65-65-72
1500	500	No	6/8	66-64-66
1500	1000	No	6/8	64-62-60
1600	25	No	6/8(5)	69-67-69
1600	500	No	6/8(5)	71-68-68
1600	1000	No	6/8	64-68-72
Type 304 -- Prior Treatment 2000°F Water Quench				
As quenched		No	2/4	155-155-156
1350	25	No	2/4	84-84-92
1350	500	No	3/4(2)	49-44-53
1350	1000	No	3/4(2)	31-30-36
1500	25	No	2/4	72-77-71
1500	500	No	3/4(2)	46-49-39
1500	1000	No	3/4(2)	49-53-42
1600	25	No	2/4	61-66-61
1600	500	No	2/4	62-64-62
1600	1000	No	2/4	54-54-53
Type 316 — Prior Treatment 1700°F Normalize				
As normalized		No	6/8	58-56-54
1350	25	No	7/8(6)	38-38-38
1350	500	No	7/8	39-36-39
1350	1000	No	7/8	35-34-33
1500	25	No	7/8	45-44-43
1500	500	No	7/8	43-43-41
1500	1000	No	7/8	40-41-38
1600	25	No	8/7	42-41-43
1600	500	No	7/8	44-44-46
1600	1000	No	7/8	41-43-43
Type 316 — Prior Treatment 1900°F Water Quench				
As quenched		No	3/4	157-163-152
1350	25	No	4/3	53-54-55
1350	500	No	4/3	35-35-33
1350	1000	SNC	3/4(5)	35-36-32
1500	25	SNC	4/3(5)	30-30-31
1500	500	SNC	4/3(5)	20-18-24
1500	1000	PNC	3/5	22-22-20
1600	25	SNC	4/3	25-25-23
1600	500	SNC	4/3	22-23-25
1600	1000	SNC	3/5	20-20-21

[a] SNC, slight amount of needle-like constituent present; PNC, pronounced amount of needle-like constituent present.

[redacted]

Although these impact values are lower than those given by Menard, we do not believe them to be too low.

The second area of special note concerns the room-temperature impact properties of welds after aging at elevated temperatures. If the reduced impact properties previously noted for the wrought product are primarily due to sigma phase, we then believe that weld metal containing 4 to 10% ferrite would be more responsive to the aging treatment. We are unable to find data to clarify this problem and suggest that it might be worthy of further consideration.

Irradiation Effects at Elevated Temperature

The irradiation of stainless steels at temperatures in the range of 1100°F and above results in an embrittlement of the alloy that is significantly different in character from the neutron displacement damage that is of principal importance at temperatures below 1100°F. This embrittlement at elevated temperatures does not necessarily affect the strength of an alloy, but it can be severe, and ductilities less than 1% have been observed for creep conditions.

Irradiation affects the ability of the alloy to resist intergranular fracture, and most experiments point to the principal cause as one related to the production of helium. This may come from two sources, the $^{10}B(n,\alpha)$ reaction induced by thermal neutrons and reactions between the major alloy constituents and fast neutrons having energies in the 3-Mev range.

The neutron doses anticipated for the MPRE after 10,000 hr are given in Table 8. Note that the maximum thermal neutron dose is observed for the pressure vessel rather than the core. The calculated maximum helium concentration for stainless steel in the core and vessel are plotted on Fig. 5 as a function of reactor operating time.

Elevated temperature irradiation of stainless steel can result in one or both of the following phenomena:

1. Embrittlement of the alloy at elevated temperatures.
2. Reduction in stress-rupture strength because of the embrittlement.

[redacted]

[redacted]

Table 8. Peak Neutron Exposure after 10,000 hr in the MPRE

Average Range	Neutron Dose, neutrons/cm^2, in		
	Reactor Core	Pressure Vessel	Expansion Tank
Thermal	1.8×10^{19}	2×10^{20}	1.6×10^{19}
3 to 10 Mev	4.6×10^{20}	1×10^{20}	5.4×10^{19}

The thermal neutron reaction affects the short-time tensile properties of type 316 stainless steel. Typical data for 1×10^{20} and 4.5×10^{20} neutrons/cm^2 are given in Table 9 for deformation test temperatures of 1300 and 1550°F. Note that the ductility in these short-time tests is as low as 7%, a value considerably lower than any observed in the creep-aging studies of unirradiated material. The effect on stress-rupture properties is less easily predicted.

In-reactor tube burst tests on other austenitic stainless steels show no reduction in stress-rupture properties. These same tests show embrittlement. However, in-reactor and postirradiation uniaxial stress-rupture tests of type 304 stainless steel at 1200 and 1475°F do show both embrittlement and reduction in stress-rupture properties at neutron exposures of about 10^{20} neutrons/cm^2. Very possibly the rate of deformation in tube burst tests is such that the reduction in ductility does not produce significant reduction in the rupture life. Therefore it is entirely possible that a reduction of stress rupture properties depends upon the stress state and means by which the member is stressed.

The effect of irradiation on both ductility and stress-rupture properties appears to be the most important factor reviewed in this memorandum. The ductility of the irradiated reactor pressure vessel after 10,000 hr evidently will be in the range of 5 to 10%. The stress-rupture properties of the irradiated alloy are not known.

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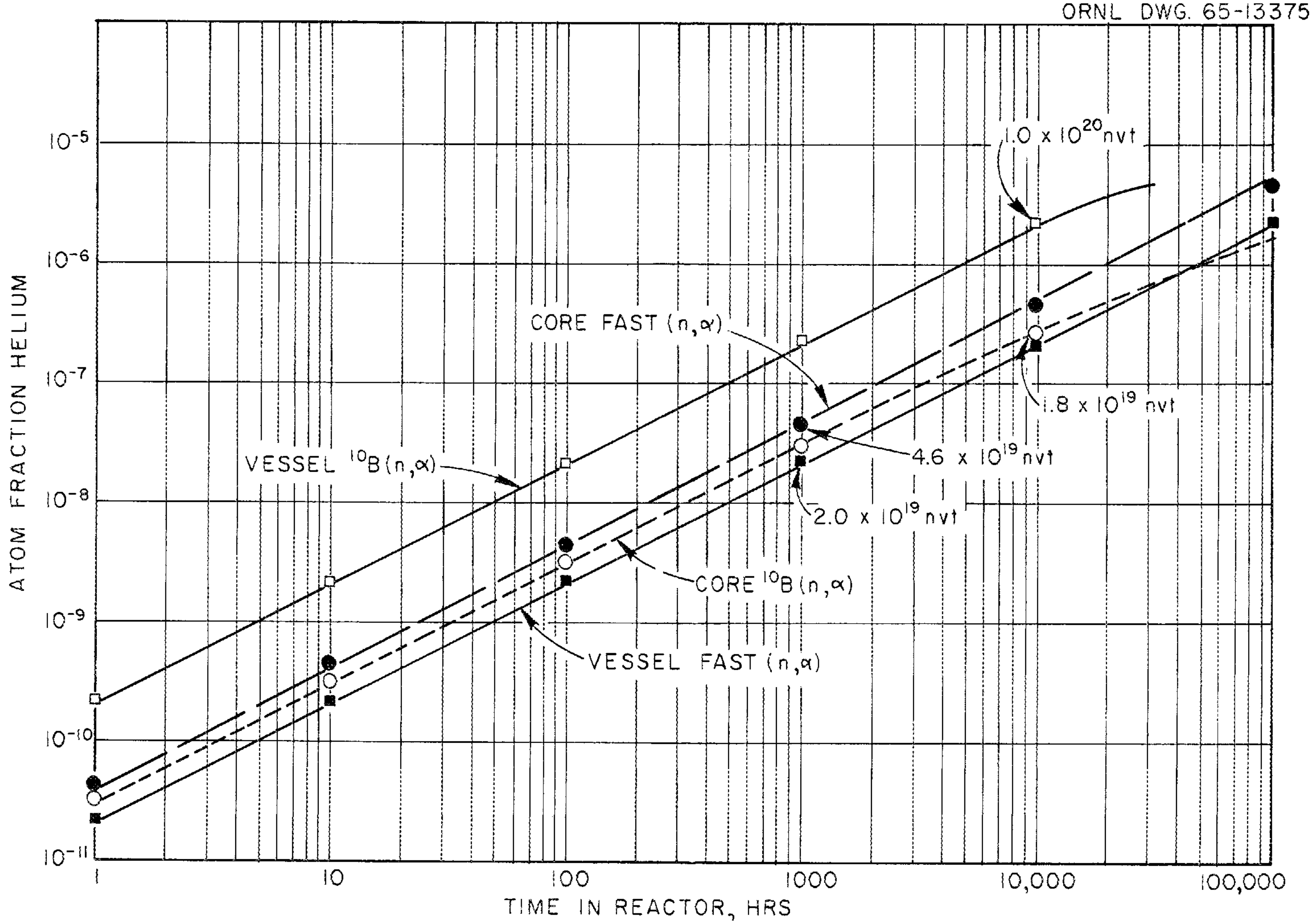

Fig. 5. Calculated Helium Concentrations in Stainless Steel after MPRE Operation.

Table 9. Postirradiation Ductility of Type 316 Stainless Steel[a]

Tensile Test Temperature (°F)	Irradiation Conditions	Ductility, %		
		True Uniform Strain	True Fracture Strain	Total Elongation
1300	Unirradiated	20.0	97.0	41.4
	ORR 126[b]	18.3	36.4	22.7
	ORR 126	18.2	33.1	22.5
	ORR 127[c]	6.9	40.8	21.8
1550	Unirradiated	10.4	124.0	53.6
	ORR 126	6.2	16.7	11.0
	ORR 126	5.6	16.7	11.5
	ORR 127	1.2	15.4	6.9

[a] Annealed 1 hr at 1036°C in argon before irradiation. Analysis (%): C, 0.069; Mn, 1.95; P, 0.024; S, 0.012; Si, 0.51; Ni, 12.57; Cr, 17.3; Mo, 2.21; B, 3.6 ppm.

[b] Irradiated at 120°F in ORR facility P.B.(P-5) to neutron doses of 5×10^{18} (fast) and 1×10^{20} neutrons/cm^2 (thermal).

[c] Irradiated at 1300°F in ORR facility B-8 to neutron doses of 3.5×10^{20} (fast) and 4.5×10^{20} neutrons/cm^2 (thermal).

CONCLUSIONS

1. The elevated-temperature strength of unirradiated wrought stainless steel is sufficient to withstand the predicted MPRE stress levels for the time periods up to 10,000 hr.

2. The creep strength of unirradiated weld bead material can be significantly less than that of the wrought alloy and should be so considered in the design of the reactor plant.

3. The elevated-temperature ductility of unirradiated stainless steel is sufficient for temperatures and times of MPRE operation. The room temperature properties such as impact strength after reactor operation are reduced. Maintenance and examination of the reactor after operation should be performed with care to prevent a needless fracture and failure.

4. Based on thermal neutron exposure, irradiation will have a greater effect on ductility than temperature will. The degree of embrittlement in creep and its subsequent effect on stress-rupture are not known quantitatively.

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