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IMPLICATIONS OF MATERIALS BEHAVIOR ON DESIGN CODES*

by

D. I. Roberts

General Atomic Company
San Diego, California

ABSTRACT

In the U.S., the design of Class 1 elevated-temperature components of reactor systems is governed by the rules of ASME Boiler and Pressure Vessel Cases N47 (design) and N48 (construction). The rules of Case N47, in particular, are sophisticated and complex, and a substantial quantity of materials behavior data is needed to design to these rules. Requirements include a detailed knowledge of creep, rupture, creep-fatigue, etc. In addition, many other factors, including such aspects as the influence on service performance of environment, welds, and fabrication-induced cold work, must be considered in the design. This paper reviews the impact of some recent HTGR materials data on design rules and approaches.

In the construction area, for example, recent data regarding the elevated-temperature properties and behavior of cold-formed austenitic materials such as Alloy 800H have resulted in rule changes. Observed creep-fatigue behavior of Alloy 800H and 2-1/4Cr - 1Mo steel is causing active review of the pertinence of linear damage summation approaches. Consideration of the time-dependent properties of welds and weldments is leading to an examination of the need to incorporate additional margins in design. A significant amount of attention is also being given to the influence of HTGR helium on behavior and the best means of accommodating phenomena such as carburization. Where appropriate, design approaches to each of these considerations are being developed.

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INTRODUCTION

One of the key features that distinguishes high-temperature gas-cooled reactors (HTGRs) from the widely used light water reactors (LWRs) is the much higher operating temperature of the HTGR (700° to 950°C for the HTGR, 350°C for the LWR). Many of the advantages of the HTGR are, of course, derived from this temperature difference. However, the high temperature has disadvantages in that components in an HTGR must operate in the creep regime, whereas LWR components do not. The time-dependency of behavior adds considerably to the complexity of design and analysis of some HTGR components.

One difficulty in design of nuclear reactor components in the creep regime is that, in the U.S., relevant design and analysis rules have less maturity than the rules for design of reactor components at temperatures below the creep range. Section III of the ASME Boiler and Pressure Vessel Code, for example, has the benefit of many years of successful use and application. Thus, while these rules (like any others) are always subject to updating and improvement, they can be considered founded on a strong basis of experience.

Rules governing design of elevated-temperature nuclear components on the other hand, have a far smaller experience base and accordingly these rules rely heavily upon experimental and theoretical considerations. As a result these rules tend to be subject to more rapid change in response to newly developed experimental data and analytical understanding. The degree of change can be appreciated by consideration of the evolution of the current Code rules governing high-temperature nuclear design and construction. Design and construction of Class 1 components, for example, are currently governed by the rules of Code Cases N47 and N48 (Ref. 1). These Cases (particularly Case N47) have undergone major changes since the early Code Case 1331-4, which was utilized, for example, at the time of design of the Fort St. Vrain HTGR. The current version of N47 requires detailed analyses to be performed to assess strain accumulation, ratcheting, and creep fatigue, as well as guarding against failure from the more conventional tensile and rupture modes. The criteria upon which the allowable stresses of Case N47 are based also consider behavior aspects not considered in other Code books. An example is the inclusion of consideration of tertiary creep in the allowable stress criteria.

Because rules such as those of Case N47 are sensitive to experimental data, it is quite common for newly developed materials information to cause reconsideration of current rules. Moreover, even though Case N47 requires detailed analyses, consideration of many other important factors, such as the influence of environment on behavior, is still left to the designers, which requires a great deal of materials information. Thus, overall, new materials data can exercise a strong influence on reactor design and construction codes; the intent of this paper is to highlight a few instances where this has occurred.

FABRICATION-INDUCED COLD WORK

It is often necessary, during fabrication of components such as heat exchangers, to bend materials into required shapes. Where relatively ductile materials, such as 300 series stainless steels and Alloy 800H, are employed, it is common to do this bending cold. The material thus deformed is strain hardened and contains residual stresses that may be significant. If such cold deformed material is of the austenitic type, heat treatment following working is not normally required, and for many construction situations this approach is entirely satisfactory. However, in elevated-temperature nuclear applications, the higher safety demands require that this issue be scrutinized carefully.

In HTGR heat exchangers, it is necessary to form certain Alloy 800H parts and this is, desirably, done cold. In order to assess the acceptability of such cold-deformed Alloy 800H, an evaluation program consisting of the studies listed in Table 1 was performed (Refs. 2,3). As indicated in Table 1, the overall program was too large to review in this paper. However, important conclusions derived from the work were as follows:

1. The expected Bauschinger effect on tensile properties was observed (Fig. 1).

2. An unexpected Bauschinger effect on rupture properties was observed. Specifically, it was noted that, while tensile prestrain increased rupture life significantly, compressive prestrain shortened life compared to comparable annealed material (Fig. 2).

3. Ductilities at rupture of material prestrained in compression were very low, and the material was notch sensitive (also shown in Fig. 2).

4. Residual stress levels in cold formed parts could be relatively high, and patterns were complex.

5. It was noted that, with long exposure time, recrystallization can occur at temperatures within the range of operational interest (Fig. 3).

Overall, it was concluded that the precise modelling of the elevated-temperature behavior of cold-formed parts is so complex, requires such large amounts of materials data, and is subject to so many uncertainties that it is too difficult to grapple with the problem. It was therefore judged prudent to limit the amounts of fabrication-induced cold work permissible without heat treatment in elevated-temperature nuclear components. Such restrictions have now been adopted into Code Case N48 for Alloy 800H and other austenitic alloys that exhibit similar behavior.

CREEP FATIGUE

One of the more complex phenomena which it is necessary to address in elevated-temperature nuclear design is the interaction of creep and fatigue. While an enormous amount of data on this topic has been developed during the last 20 years, it is clear that our understanding of the phenomena involved is still very limited. Code Case N47 requires that creep-fatigue analysis be performed but does not mandate the use of a specific analytical method. The Case does, however, describe one approach to analysis by using the linear damage approach. In this method, the fatigue damage and the creep damage (where the latter is computed by integrating the time spent at different stresses during stress relaxation relative to the monotonic rupture strength capability of the material) are summed linearly and the method assumes that life has been exhausted when the damage sum, D, is unity.

Experimental data generated in the last several years on both Alloy 800H (Ref. 4) and 2-1/4Cr - 1Mo steel have, however, cast some doubt on the universal applicability of the method. An example of a set of data obtained on Alloy 800H at General Atomic is shown in Table 2. The data presented were obtained from tests at relatively low stress ranges approaching those of interest in design. The material tested was a heat for which accurate knowledge of the creep-rupture and continuous cycling fatigue behavior existed. The creep and fatigue damage fractions shown were computed by comparison with the actual properties of the heat under creep-fatigue test. As indicated in the table, the total damage fractions at failure are significantly less than unity in many cases. Since similar observations have been made for 2-1/4Cr - 1Mo steel, there is active consideration of whether other methods (such as strain range partitioning, damage rate theory, etc.) are more appropriate design tools for use with these materials.

WELD PROPERTIES

It has long been recognized that the compositional and microstructural differences between weld region and base metals will cause disparities in the properties. At temperatures below the creep range, it is generally true that these differences are in favor of the weld metal; that is, the weld metal exhibits greater yield and tensile strength than the base metal (Fig. 4). Thus, the weld region can be, to a considerable degree, ignored from a design strength viewpoint. At temperatures in the creep range, on the other hand, it is no longer always true that weld metals are stronger than the base metal. In fact, some of the weld metals commonly used to weld Alloy 800H, for example, show rupture strength inferior to that of the base metal (Fig. 5). Likewise, 2-1/4Cr - 1Mo weld metal can have properties inferior to the annealed base metal, particularly in cases where low-carbon weld filler metal is employed (Fig. 6).

The fact that welds may show inferior strength, and in some cases ductility, was recognized early in the development of Code Case N47, and permissible strain accumulations in welds were halved, relative to the base metal, to account for this. However, as additional data on the strength of welds are obtained, there is an increasing interest in attempting to account more explicitly for the behavior of individual weld materials through the development of strength factors that recognize the available creep and rupture data on weld metals. Factors of this type are currently under active study in the U.S. with the current approach being to derive these factors by ratioing the average rupture strength of the weld and base metals. This is an approach rather similar to that employed in Germany to derive weld factors for liquid metal fast breeder reactor designs.

ENVIRONMENTAL EFFECTS

The coolants employed in any reactor system usually possess characteristics unique to that reactor system. This is certainly the case for the HTGR, where the primary coolant is helium containing levels of impurity gases that will vary as a function of specific reactor design. Codes and regulatory bodies, recognizing the specificity of coolant characteristics, generally do not attempt to define detailed rules on how to handle the potential interactions between coolants and structural materials. Code Case N47, for example, requires the designer to consider the effects on materials of environments specific to the reactor involved, but does not suggest how this should be accomplished. The design organization is, therefore, left with this difficult problem.

The difficulty arises because of the range and complexity of coolant/materials/stress interactions - a situation that is particularly true at elevated temperatures. It is known, for example, that HTGR coolant/material interactions can:

1. Change the bulk and surface chemical composition of materials with accompanying changes in property characteristics.

2. Influence surface-sensitive mechanical properties such as fatigue, creep-fatigue, creep rupture, toughness, and crack propagation.

3. Produce effective loss of metal thickness.

4. Produce surface defects that act as notches.

Since there are usually synergistic interactions between these effects, handling all of them is relatively difficult, and it is usually necessary to make simplifying assumptions. From a design standpoint in the HTGR, these effects are handled at General Atomic as described below.

Internal oxidation from helium/metal interactions can be handled as a metal loss through application of corrosion allowances (Fig. 7).

The influence of environment on mechanical properties can be directly accounted for through the application of factors. For example, if the helium environment reduces creep-rupture life, then allowable stresses can be reduced directly.

Where it is known that the helium environment changes the chemical composition of structural materials, this must be accounted for. Two effects are commonly encountered: carburization and decarburization. To account for both of these, knowledge must exist regarding the kinetics of interactions and the consequences of interactions. In the case of carburization of the wrought and cast austenitic alloys, a body of data exists from which upper bound carburization rates can be derived (Fig. 8). Similarly, since it has been observed that decarburization of 2-1/4Cr - 1Mo in helium exhibits similar kinetics to decarburization in sodium (Fig. 9), decarburization rates can also be estimated. In the case of decarburization of 2-1/4Cr - 1Mo, extensive data currently exist to indicate the correlation between strength loss and carbon loss, and such information can be employed to compute the design consequences of this phenomenon. In the case of carburization of wrought austenitic alloys, it is known that severe embrittlement can be produced (Fig. 10) (Ref. 5). However, an accurate quantification of the effects of carburization under HTGR conditions does not exist, and several programs are currently under way to quantify, in fracture mechanics terms, these effects. Similarly, programs to assess the effects of carburization on other key design properties such as creep, rupture, fatigue, and creep-fatigue are in hand. Once these data are available, it will be possible to compute tolerable carburization levels.

One further aspect of carburization that should be noted in this regard is that a growing body of data exists to indicate that carburization can induce dimensional changes (e.g., cause swelling). If such effects are confirmed and quantified, they will have to be given very careful design consideration, since significant volume changes, particularly if they occur nonuniformly in a structure as is likely if temperature gradients exist, can produce stresses that are significant from a design standpoint.

CONCLUDING COMMENTS

In view of the limited experience underlying elevated-temperature nuclear design and construction, and the large amount of materials research under way in the world in support of gas- and liquid-metal-cooled reactors, it is likely that design rules will continue to evolve in response to new data and experience. In general, it is likely that the changes will be greatest at the highest temperatures of interest, since this is the area that poses the greatest challenge to materials and design engineers. However, presently ongoing programs will provide a good basis for developing the kinds of information needed to permit the ultimate derivation of appropriate design procedures.

REFERENCES

1. ASME Boiler and Pressure Vessel Code Cases, Section III, Division 1: Case N47-17 - Class 1 Components in Elevated-Temperature Service and Case N48 - Fabrication and Installation of Elevated-Temperature Components.

2. Smith, A. B., "The Effect of Tensile and Compressive Prestraining on the Properties of Ni-Fe-Cr Alloy," in Characterization of Materials for Service at Elevated Temperatures, Metal Properties Council Report MPC-7.

3. Lai, G. Y., and A. B. Smith, "Recrystallization Behavior of Alloy 800H," ibid.

4. Kaae, J. L., contribution to "HTGR Generic Technology Program Semi-annual Report for the Period Ending March 31, 1979," DOE Report GA-A15417, General Atomic Company, June 1979.

5. Wenschof, D. E., and J. A. Harris, "The Influence of Carburization on the Mechanical Properties of Wrought Nickel Alloys," Corrosion 34(1), 23 (1978).

TABLE 1
STUDIES PERFORMED ON COLD-FORMED ALLOY 800H

Material study parameters
- Cold work in the range 5% to 20%
- Working direction (i.e., compression versus tension)
- Tests in the temperature range room temperature to 800°C

Materials properties characterized
- Tensile flow and failure
- Creep and rupture
- Stress relaxation
- Time-temperature recrystallization behavior

Structural evaluation
- Residual stress and strain distributions
- Time-temperature deformation behavior of loaded structures

Analytical Studies
- Detailed analysis of behavior of cold-formed components
- Modelling of residual stress distributions

TABLE 2
LOW STRAIN RANGE CREEP-FATIGUE TESTS ON ALLOY 800H

Material Condition[a]	Temp. (°C)	Strain Cycle History (%)	Strain Cycle History Cycles	Strain Cycle History Strain Hold Sequence	Creep Damage at Failure	Fatigue Damage at Failure	Total Damage
SA	760 (1400°F)	0 to 0.2 0 to 0.22 0 to 0.24	0 to 8,520 8,520 to 8,950 8,950 to 11,000	10 min at 0.2% 10 min at 0.22% 10 min at 0.24%	1.45	0.0052	1.455
SA	760	0 to -0.2 0 to -0.24 0 to -0.30	0 to 4,831 4,831 to 12,485 12,485 to 13,409	10 min at -0.2% 10 min at -0.24% 10 min at -0.3%	5.08	0.015	5.1
SA	649 (1200°F)	0 to 0.40	0 to 25,813	1 min at 0.4%	0.45	0.20	0.65
SA	649	0 to -0.40	0 to 10,000	1 min at -0.4%	0.19	0.08	0.27
SA	649	0 to 0.38	0 to 14,090	2.5 min at 0.38%	0.23	0.11	0.34
SA	649	0 to -0.40	0 to 7,986	2.5 min at -0.4%	0.17	0.06	0.23
SA	649	0 to 0.31	0 to 33,680	1 min at 0.31%	0.55	0.025	0.80
SA	649	0 to -0.31	0 to 18,250	1 min at -0.31%	0.0827	0.013	0.0957
CW 10% in tension	649	0 to 0.38	0 to 9,000	1 min at 0.38%	0.67	0.07	0.74
CW 10% in tension	649	0 to -0.38	0 to 10,000	1 min at -0.38%	0.45	0.07	0.52
Aged 8000 h	649	0 to 0.40	0 to 12,300	1 min at 0.4%	0.050	0.07	0.12
Aged 8000 h	649	0 to -0.40	0 to 7,000	1 min at -0.4%	0.038	0.04	0.078
CW 10%, aged 4000 h	649	0 to -0.40	0 to 9,800	1 min at -0.4%	0.066	0.07	0.136
CW 10%, aged 4000 h	649	0 to 0.40	0 to 11,000	1 min at 0.4%	0.075	0.08	0.055
SA	649	0 to 0.40	0 to 5,500	1 min at 0.4% 1 min at 0.0%	0.174	0.055	0.229
SA	649	0 to 0.40	0 to 3,300	2.5 min at 0.4% 2.5 min at 0.0%	0.108	0.033	0.141
SA	649	0 to 0.35	0 to 8,322	1 min at 0.35% 1 min at 0.0%	0.095	0.011	0.106
SA	649	0 to 0.35	0 to 3,606	2.5 min at 0.35% 2.5 min at 0.0%	0.048	0.005	0.053

[a] SA = solution annealed, CW = cold worked.

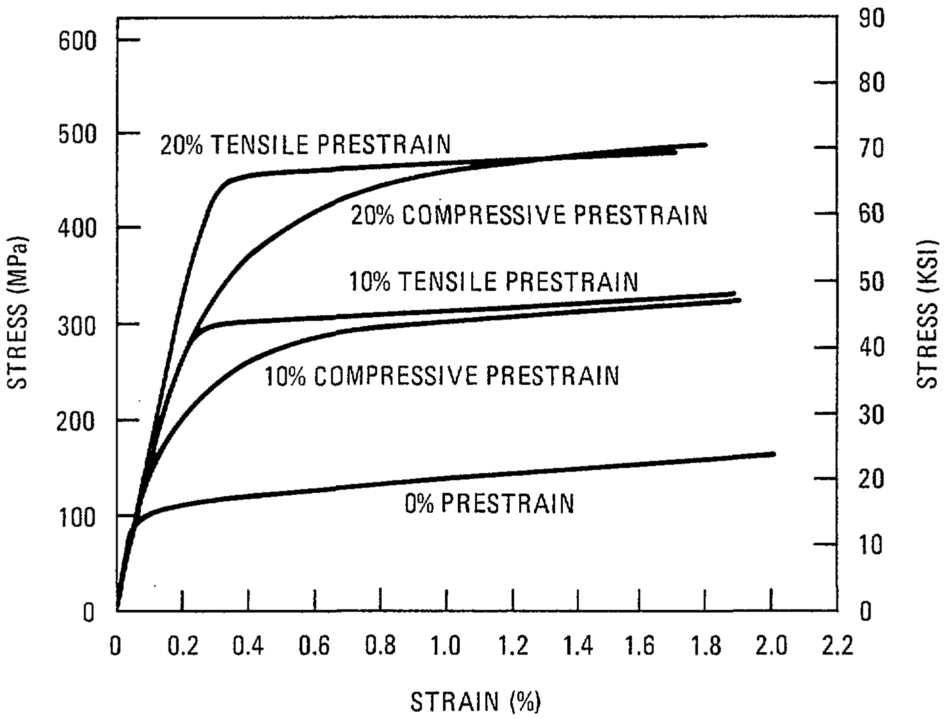

Fig. 1. Effect of prior cold strain on the tensile curves of solution-heat-treated Alloy 800H tested at 538°C (1000°F)

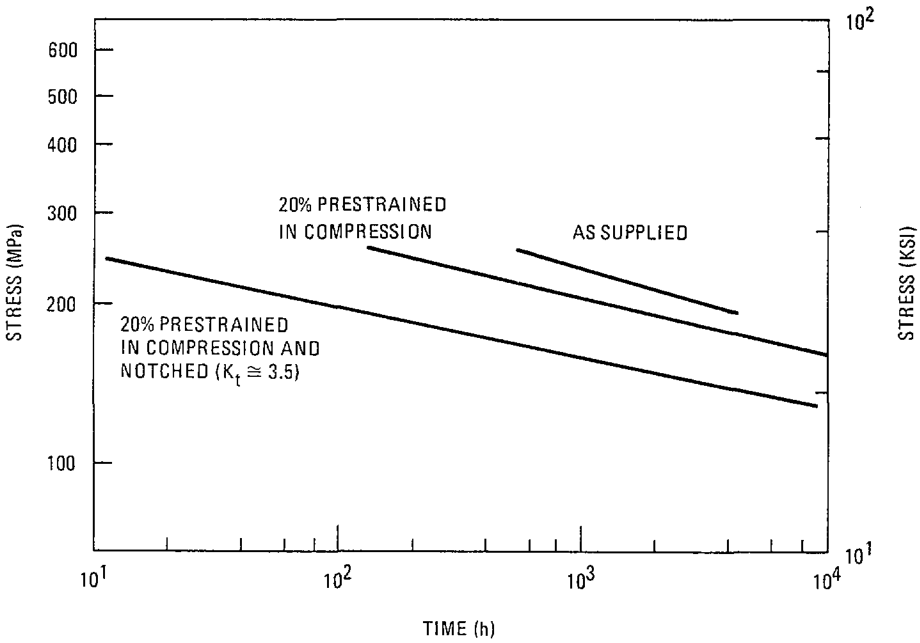

Fig. 2. Rupture behavior of notched and smooth specimens of Alloy 800H in the annealed and 20% prestrained in compression conditions (593°C)

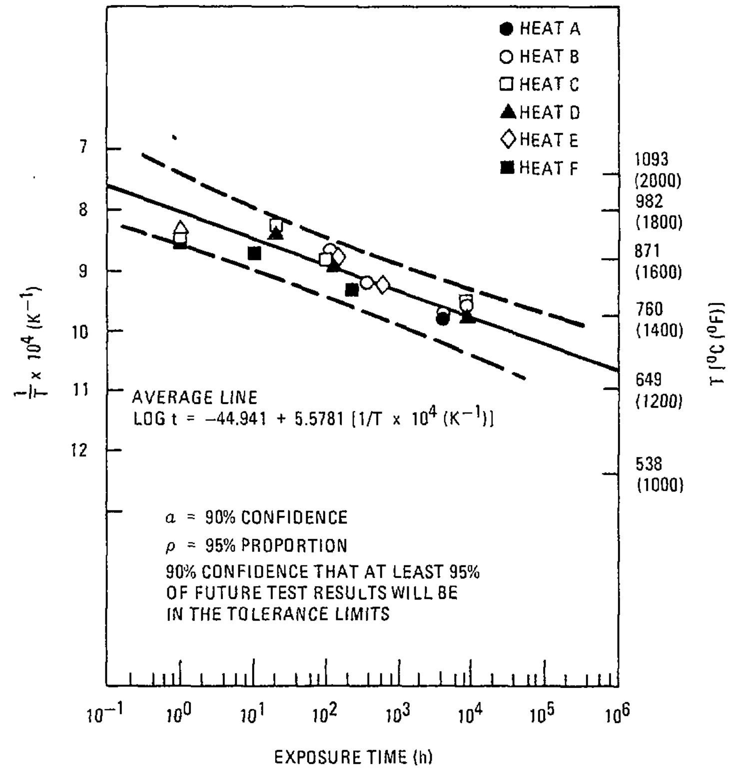

Fig. 3. Temperature-time envelope for the onset of recrystallization in 20% prestrained Alloy 800H

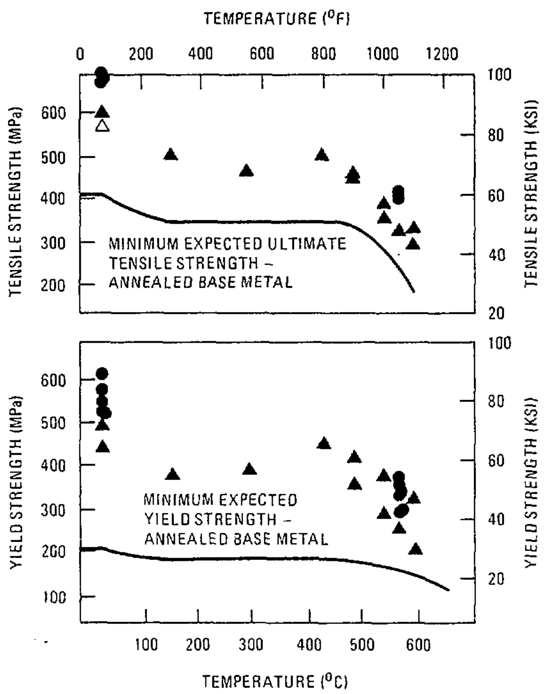

Fig. 4. Comparison of yield and ultimate tensile strength of weld metal and 2-1/4Cr - 1Mo annealed base metal

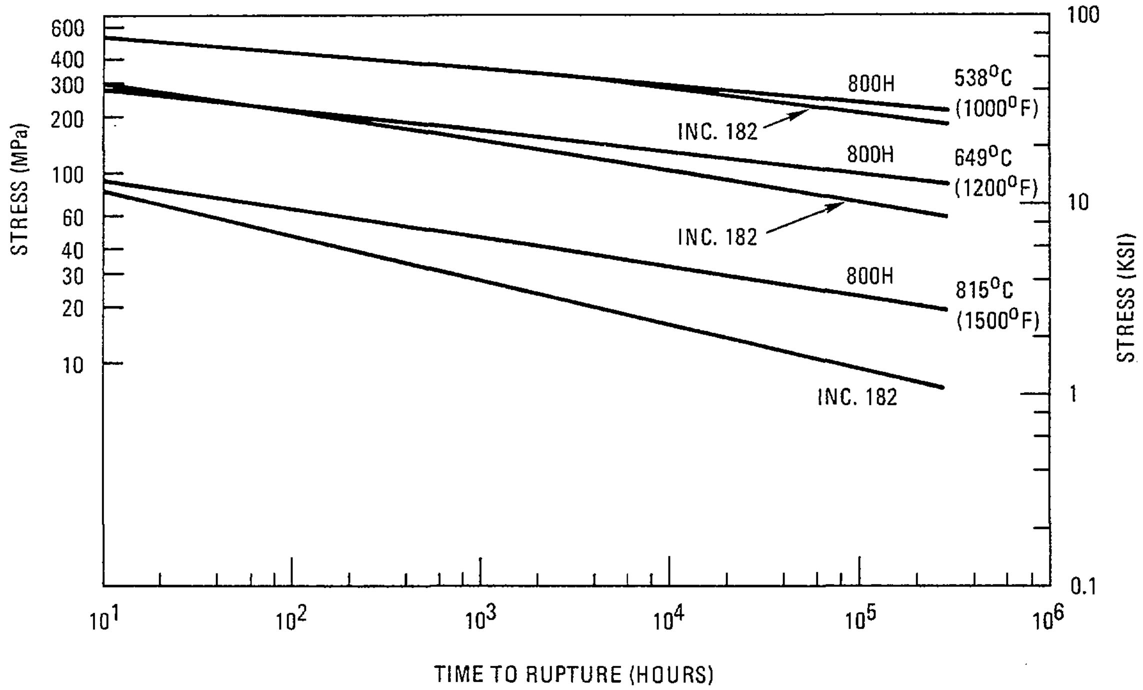

Fig. 5. Stress-rupture comparison between weld metal Inc. 182 and Alloy 800H

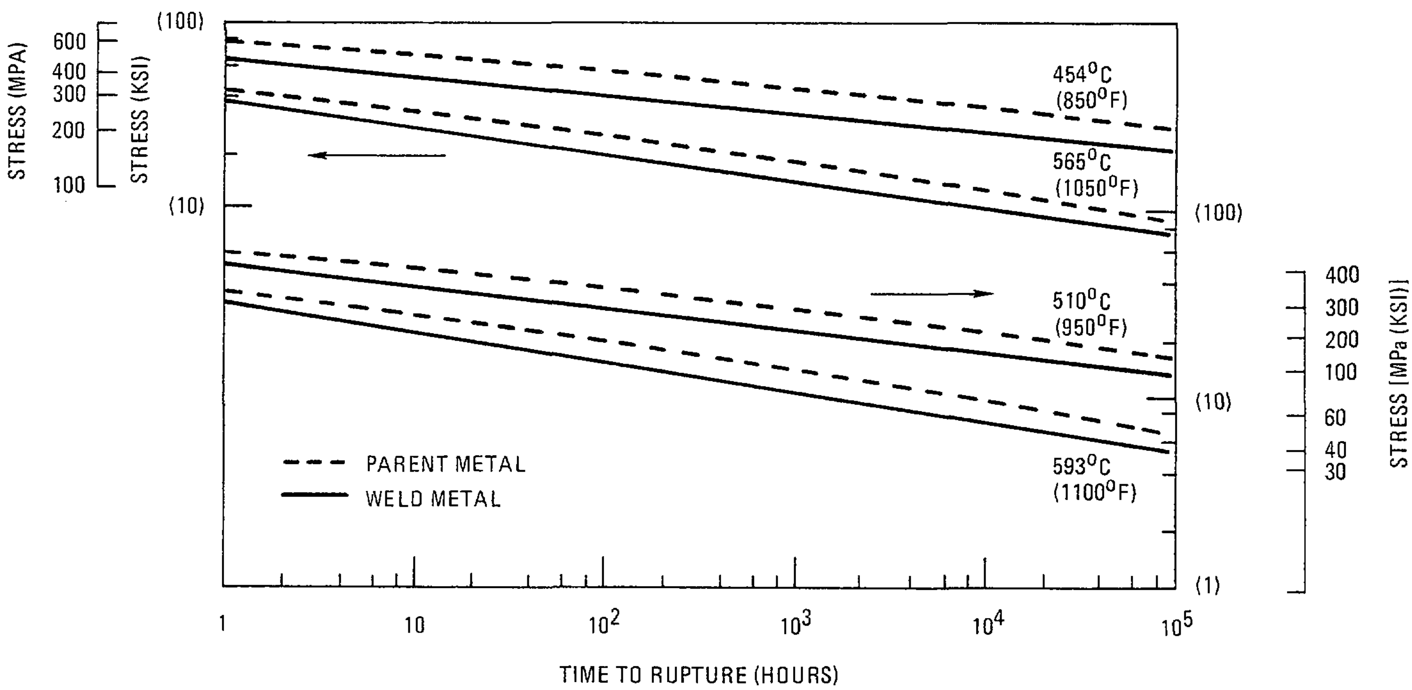

Fig. 6. Stress-rupture comparison between 2-1/4Cr - 1Mo parent metal and low-carbon weld metal

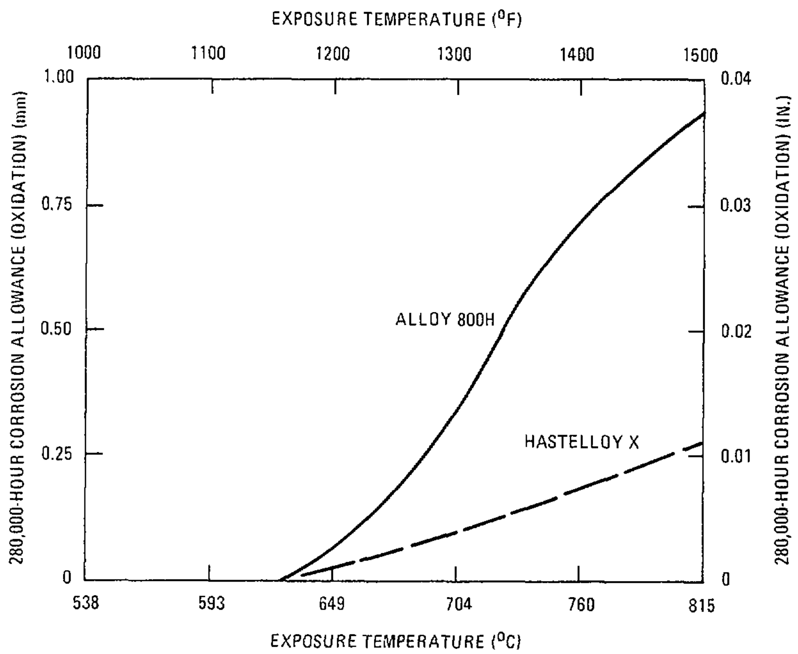

Fig. 7. Typical helium-side oxidation corrosion allowances

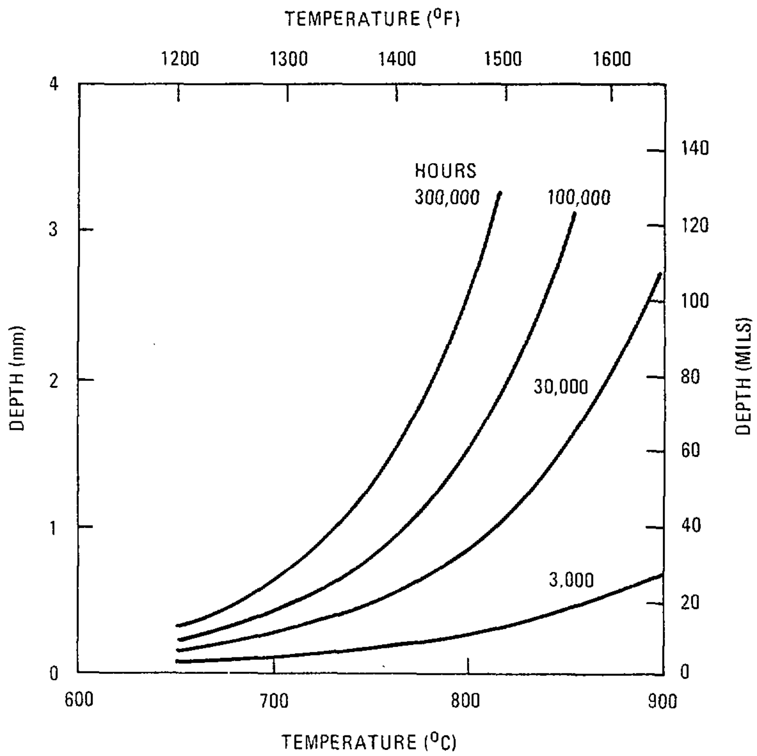

Fig. 8. Predicted carburization depth in Hastelloy X in HTGR helium

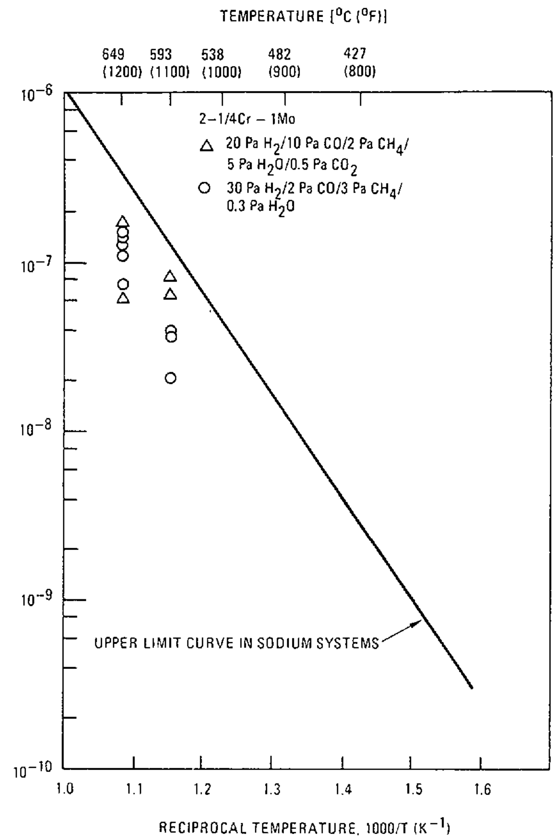

Fig. 9. Decarburization rate constants in HTGR helium compared with those of sodium systems

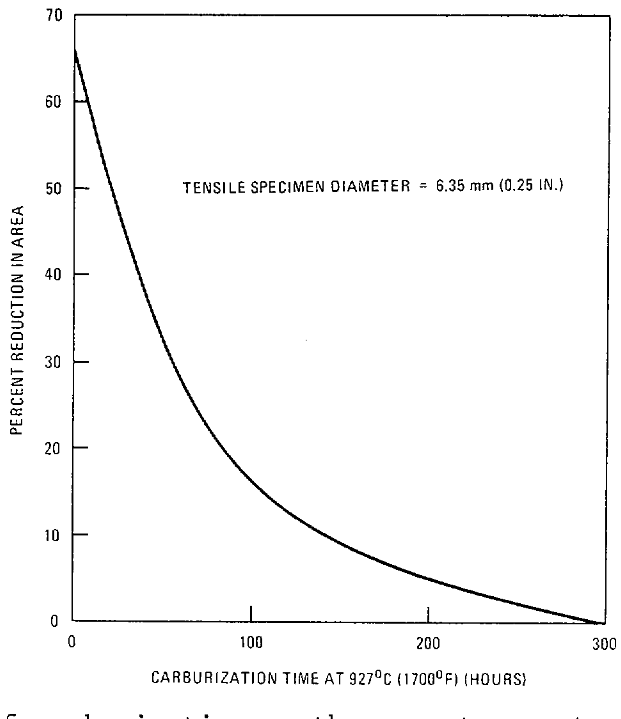

Fig. 10. Effect of carburization on the room-temperature tensile ductility of Alloy 800H (Ref. 5)