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### Predicting Creep in an Alloy 617 Pressure Vessel Using Uniaxial Bar Test Data

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### Abstract

Alloy 617 is being investigated as a candidate for application in next generation nuclear plants (NGNPs), including the very high temperature reactor (VHTR). Creep deformation concerns the lifetime and failure of heat exchangers used at the very high operating temperatures and pressures of such plants. In order to meet the American Society of Mechanical Engineers' Boiler and Pressure Vessel code (ASME B&PV), and be certified for use in NGNPs, a thorough understanding of creep must be demonstrated. Current methods of testing creep have limitations and there remains uncertainty and controversy as to the applicability of these tests to more complex designs[1,2]. The aim of this research is to relate data from uniaxial laboratory creep tests to multiaxially loaded service components.

### Disciplines

Materials Science and Engineering

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# Predicting Creep in an Alloy 617 Pressure Vessel Using Uniaxial Bar Test Data

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ITR) design <sup>[1,2]</sup> :											1	
urrent reactor designs while also producing hydrogen										,	C	
es up to 8MPa (~1160psi)												
ulting in deformation at low stresses, severely limits material										Strai		
by, is the leading candidate for use in heat exchangers.										07		
Composition of Alloy 617										- 	Drime	
Со	Мо	Fe	Al	Ti	Mn	Si	C,Cu	I,S,B		Strai		
11.6	8.6	1.6	1.1	0.4	0.1	0.1	≤0	.05		-		
ts												
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strain, with FEA giving a conservative prediction, and the										7	770 Ho	
underestimation.										6		
Summary of Results										uie 4	<b>E</b> 40 11	
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h Kate (%/hr)						0.0079		0.0193				
are $(\%/117)$ ain Rate $(\%/hr)$			0.0042			0.0014			0.0024			

# References

• [1] Natesan, K., Majumdar, S., Shankar, P. S., Shah, V. N., Nuclear, E. D., & Argonne National Laboratory (ANL). (2007). Preliminary materials selection issues for the next generation nuclear plant reactor pressure vessel. United States. [2] Shah, V. N., Majumdar, S., Natesan, K., U.S. Nuclear Regulatory Commission., & Argonne National Laboratory. (2003). *Review and assessment of codes and procedures for HTGR components*. Washington, DC: Division of Engineering Technology, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission.
[3] Carroll, L., Cabet, C., Wright, R., and Idaho National Laboratory (INL). (2010). *The Role of Environment of High Temperature Creep-Fatigue Behavior of Alloy 617*. Washington, D.C: United States. Office of Nuclear Energy, Science, and Technology. [4] Lillo, T., Cole, J., Frary, M., & Schlegel, S. (January 01, 2009). Influence of Grain Boundary Character on Creep Void Formation in Alloy 617. Metallurgical and Materials Transactions A, 40, 12, 2803-2811.



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