

IMPROVEMENT OF ASME NH

FOR GRADE 91 NEGLIGIBLE CREEP
AND CREEP-FATIGUE



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STP-NU-013

IMPROVEMENT OF ASME NH FOR GRADE 91 NEGLIGIBLE CREEP AND CREEP FATIGUE

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Date of Issuance: September 3, 2008

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ASME Standards Technology, LLC
Three Park Avenue, New York, NY 10016-5990

ISBN No. 978-0-7918-3165-6

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TABLE OF CONTENTS

Foreword	x
Abstract	xi
PART 1 IMPROVEMENT OF ASME NH FOR GRADE 91 (NEGLIGIBLE CREEP)	1
1 INTRODUCTION	2
2 NEGLIGIBLE CREEP CRITERIA	3
2.1 ASME	3
2.2 RCC-MR	4
2.3 Japanese Development of Structural Design Standard	5
3 APPLICATION TO MOD. 9CR-1MO STEEL	6
3.1 Reference Stress for Negligible Creep Curve Using a Criterion Based on Stress to Rupture	6
3.2 Negligible Creep Curve Based on RCC-MR Creep Strain Criterion	8
3.2.1 RCC-MR Creep Strain Law	8
3.2.2 ORNL Creep Strain Law	8
3.2.3 Japanese Creep Strain Law of Reference [6]	9
3.2.4 Applicability of Negligible Creep Curve Based on RCC-MR Creep Strain Criterion	9
3.3 Negligible Creep Curve Based on RCC-MR Stress Relaxation Criteria	10
3.4 Negligible Creep Curve Based on ASME 0.2 % Creep Strain Criterion	10
3.5 Negligible Creep Curve Based on Japanese Creep Strain Criterion	11
4 TENTATIVE IMPROVEMENT IN THE CORRELATION FOR MOD. 9CR-1MO AT MODERATE TEMPERATURES	14
4.1 Minimum Creep Rate	14
4.1.1 Correlation of Minimum Creep Rate and Time to Rupture	14
4.1.2 Minimum Creep Rate Against Stress	14
4.2 Creep Strain Equations	20
4.2.1 Creep Strain Law in RCC-MR	20
4.2.2 Comparison with other Creep Strain Laws	20
4.2.3 New Fit Using RCC-MR Creep Strain Law	21
4.3 Creep Stress to Rupture	44
4.3.1 RCC-MR Stress to Rupture Data for Modified 9Cr-1Mo	44
4.3.2 Average Stress to Rupture of Modified 9Cr-1Mo Steel Using ORNL Data	44
4.3.3 Average Stress to Rupture of Modified 9Cr-1Mo Steel Using Japanese Data	44
4.3.4 Average Stress to Rupture of Modified 9Cr-1Mo Steel Using Minimum Commitment Method	44
4.3.5 Average Stress to Rupture of Modified 9Cr-1Mo Steel at Moderate Temperatures ..	45
4.3.6 Application of Stress to Rupture at Moderate Temperatures to Negligible Creep	46
5 DISCUSSION	54
6 CONCLUSIONS	57
References	58
PART 2 IMPROVEMENT OF ASME NH FOR GRADE 91 (CREEP-FATIGUE)	59

1	INTRODUCTION.....	60
2	MOD 9CR-1MO STEEL.....	61
3	CREEP-FATIGUE PROCEDURES IN THE NUCLEAR CODES	62
	3.1 ASME Procedure	65
	3.2 RCC-MR Procedure.....	66
	3.3 DDS Procedure	67
	3.4 Comparison Between ASME and RCC-MR Design Data for Creep Fatigue.....	68
	3.4.1 Fatigue Design Curves.....	68
	3.4.2 Stress to Rupture Data	68
	3.4.3 Creep and Relaxation Behavior	70
	3.5 Comparison between ASME and RCC-MR Procedures.....	70
4	PRESENTATION OF THE CREEP-FATIGUE TESTS AVAILABLE ON MOD 9CR-1MO ...	72
5	DISTINCTIVE FEATURES OF CREEP FATIGUE OF MOD 9Cr-1 MO	80
	5.1 Effect of Mean Stress.....	80
	5.2 Effect of Air Environment	80
	5.3 Creep Fatigue Tests in Vacuum.....	81
	5.4 Cyclic Softening of Mod 9Cr-1Mo.....	81
	5.5 Effects of Prior Aging.....	83
	5.6 Effect of Cyclic Softening on Creep Damage in Creep Fatigue Tests.....	83
	5.7 Results of Cyclic Creep Tests	84
6	EVALUATION OF EXISTING PROCEDURES.....	85
	6.1 Evaluation of ASME Procedure.....	85
	6.2 Evaluation of RCC-MR Procedure	89
	6.3 Comparison of Calculated Stresses at the Beginning of Hold Time.....	98
	6.4 Comparison of Relaxation Procedures.....	100
	6.5 Comparison of Safety Factors.....	101
	6.6 Creep-Fatigue Damage Envelope	102
	6.7 Example of Application of Proposed Modifications.....	103
	6.8 Combination of Primary and Secondary Stresses	105
7	CONCLUSIONS.....	108
	References.....	109
	PART 3 PROPOSED TEST PROGRAM TO ASSESS NEGLIGIBLE CREEP CONDITIONS OF MODIFIED 9CR-1MO	111
1	INTRODUCTION.....	112
2	MATERIAL	113
3	CREEP TESTS.....	114
	3.1 Improvement of Creep Strain Database	114
	3.2 Creep Strain Rate	114
	3.3 Stress to Rupture at Moderate Temperature	114
	3.4 Test Specimens and Number of Tests.....	114

4	CREEP FATIGUE TESTS	117
5	CONCLUSIONS	118
	References	119
PART 4 PROPOSED TEST PROGRAM TO VALIDATE CREEP-FATIGUE PROCEDURES		
	FOR MODIFIED 9CR-1MO	120
1	INTRODUCTION	121
2	MATERIAL	122
3	SUMMARY OF CREEP-FATIGUE TESTS RESULTS	123
	3.1 Creep-Fatigue Tests in Air	123
	3.2 Preliminary Results of Recent Creep-Fatigue Tests in Vacuum	123
4	PURPOSE OF FUTURE TEST PROGRAM	124
	4.1 Knowledge of Stress-Strain Behavior	124
	4.2 Extension of the Relation of Number of Cycles to Failure versus Viscous Plastic Strain	124
	4.3 Characterization of Softened Material	124
	4.4 Review of Creep-Fatigue Interaction Diagram	124
	4.5 Evaluation of Environmental Effect	125
5	PROPOSED TEST PROGRAM	126
	5.1 Tests in Air at 500°C or 525°C	126
	5.2 Long Term Tests in Air at 550°C	126
	5.3 Tests on Softened Material	127
	5.4 Tests on Aged Material	127
	5.5 Tests in Reactor Environment	127
	5.6 Tests on Post Weld Heat Treated Material	127
	5.7 Creep Fatigue of Welded Joints	127
6	CONCLUSIONS	128
	References	129
	Acknowledgements	130
	Abbreviations And Acronyms	131

LIST OF FIGURE

Figure 1 - Cyclic Stress Strain Behavior of Mod. 9Cr-1Mo at 500°C	6
Figure 2 - Negligible Creep Curve from Ref. [3]	13
Figure 3 - Minimum Strain Rate and Time to Rupture Relationship	16
Figure 4 - Minimum Strain Rate versus Stress at 550°C	17
Figure 5 - Minimum Strain Rate versus Stress at 500°C	18
Figure 6 - Minimum Strain Rate versus Stress at 450°C	19
Figure 7 - Stress for 0.1% Creep Strain at 550°C	24
Figure 8 - Stress for 0.2% Creep Strain at 550°C	25

Figure 9 - Stress for 0.5% Creep Strain at 550°C	26
Figure 10 - Stress for- 1% Creep Strain at 550°C	27
Figure 11 - Stress for 0.1% Creep Strain at 500°C	28
Figure 12 - Stress for 0.2% Creep Strain at 500°C	29
Figure 13 - Stress for 0.5% Creep Strain at 500°C	30
Figure 14 - Stress for 1% Creep Strain at 500°C	31
Figure 15 - Stress for 0.1% Creep Strain at 482°C	32
Figure 16 - Stress for 0.2% Creep Strain at 482°C	32
Figure 17 - Stress for 0.5% Creep Strain at 482°C	33
Figure 18 - Stress for 1% Creep Strain at 482°C	35
Figure 19 - Stress for 0.1% Creep Strain at 475°C	36
Figure 20 - Stress for 0.2% Creep Strain at 475°C	37
Figure 21 - Stress for 0.5% Creep Strain at 475°C	38
Figure 22 - Stress for 1% Creep Strain at 475°C	39
Figure 23 - Stress for 0.1% Creep Strain at 450°C	40
Figure 24 - Stress for 0.2% Creep Strain at 450°C	41
Figure 25 - Stress for 0.5% Creep Strain at 450°C	42
Figure 26 - Stress for 1% Creep Strain at 450°C	43
Figure 27 - Average Stress to Rupture and Experimental Data at 550°C	51
Figure 28 - Average Stress to Rupture and Experimental Data at 500°C	52
Figure 29 - Average Stress to Rupture and Experimental Data at 450°C	53
Figure 30 - Negligible Creep Curve for Manufactured 9Cr-1Mo	55
Figure 31 - Interaction between Negligible Creep and Creep-Fatigue	56
Figure 32 - ASME Fatigue Curves at 540°C.....	66
Figure 33 - Comparison of ASME and RCC-MR Fatigue Design Curves	68
Figure 34 - Comparison of RCC-MR Average and ORNL Stress to Rupture.....	69
Figure 35 - Comparison of ASME and RCC-MR Stress to Rupture	70
Figure 36 - Comparison of ASME and RCC-MR Creep Fatigue Damage Envelopes	71
Figure 37 - JAPC-USDOE Joint Study – Fatigue and Creep Fatigue Test Results	72
Figure 38 - INEL Study – Fatigue and Creep Fatigue Test Results.....	74
Figure 39 - CEA Studies – Fatigue and Creep Fatigue Test Results	74
Figure 40 - EPRI/CRIEPI Joint Studies – Fatigue and Creep Fatigue Test Results	78
Figure 41 - IGCAR – Fatigue and Creep Fatigue Test Results.....	78
Figure 42 - Cyclic Softening at 550°C and 600°C	82
Figure 43 - Tests in Tension-Unaged-Creep-Fatigue Damage-Best Fit Approach.....	91

Figure 44 - Comparison of Stresses at the Beginning of Hold Time at 550°C.....	98
Figure 45 - Ratio between the Initial Stresses Calculated with ASME and RCC-MR.....	99
Figure 46 - Relaxation at 550°C.....	100
Figure 47 - Comparison of EPRI/CRIEPI and RCC-MR Relaxation Curves.....	101
Figure 48 - Example of ASME Evaluation using Proposed Modifications Creep-Fatigue Damage – Tests in Tension.....	104
Figure 49 - Extrapolation of Design ASME Creep Stress to Rupture For T=1hr.....	105
Figure 50 - Comparison of Combination of Primary and Secondary Stresses.....	106

LIST OF TABLES

Table 1 - Negligible Creep Criteria.....	3
Table 2 - Temperature Limits in ASME Code.....	3
Table 3 - Tensile Properties of Modified 9Cr-1Mo.....	7
Table 4 - Negligible Creep at 450°C for Modified 9Cr-1Mo Steel Based on ASME Time Fraction Criterion.....	7
Table 5 - Negligible Creep for Mod. 9Cr-1Mo Steel Based on ASME Time Fraction Criterion.....	8
Table 6 - Negligible Creep for Modified 9Cr-1Mo Steel Based on RCC-MR Creep Strain Criterion.....	10
Table 7 - Negligible Creep for Modified 9Cr-1Mo Steel Based on RCC-MR Stress Relaxation Criteria.....	10
Table 8 - Negligible Creep for Modified 9Cr-1Mo Steel Based on ASME Creep Strain Criterion.....	11
Table 9 - Negligible Creep for Modified 9Cr-1Mo Steel Based on Japanese Creep Strain Criterion.....	12
Table 10 - Set of Parameters of the New Fit Creep Strain Law.....	22
Table 11 - Application of Revised Material Data to Negligible Creep of Table 6.....	22
Table 12 - Application of Revised Material Data to Negligible Creep of Table 8.....	23
Table 13 - Application of Revised Material Data to Negligible Creep of Table 9.....	23
Table 14 - Application of Revised Material Data to Negligible Creep of Table 7.....	23
Table 15 - RCC-MR Average Stress to Rupture for Times from 10,000 hours to 300,000 h.....	47
Table 16 - Average Stress to Rupture for Times from 10,000 hours to 300,000 hours Derived from [5].....	47
Table 17 - Average Stress to Rupture for Times from 10,000 hours to 300,000 hours Derived from [6].....	47
Table 18 - Comparison of Average Stress to Rupture for Times from 10,000 hours to 300,000 hours.....	48
Table 19 - Minimum Stress to Rupture for Times from 10,000 hours to 300,000 hours.....	49
Table 20 - Parameters of the Revised Creep Stress to Rupture.....	50

Table 21 - Application of Revised Material Data to Negligible Creep of Table 5	50
Table 22 - Negligible Creep Times for Modified 9Cr-1Mo	55
Table 23 - Creep-Fatigue – Calculation of Equivalent Strain Range.....	62
Table 24 - Creep-Fatigue Damage	64
Table 25 - ORNL- JAPC- USDOE Joint Study - Creep-Fatigue Tests of Modified 9Cr-1 Mo Steel ..	73
Table 26 - ORNL- JAPC- USDOE Joint Study - Creep-Fatigue Tests of Modified 9Cr-1 Mo Steel Heat 30394	73
Table 27 - JNC – Testing Result Data of Creep Fatigue Test (Mod 9Cr-1Mo).....	75
Table 28 - CEA – Results of Creep Fatigue Tests at 550°C	76
Table 29 - CEA - Results of Stress Controlled Creep-Fatigue Tests at 550°C.....	76
Table 30 - EPRI/CRIEPI Joint Studies – Results of Axial Creep Fatigue Tests	77
Table 31 - IGCAR – Results of Creep Fatigue Tests.....	79
Table 32 - University of Connecticut – Results of Cyclic Creep Tests	79
Table 33 - Stress Amplitudes at Mid Fatigue Life at 550°C	82
Table 34 - ASME Creep Fatigue Evaluation	85
Table 35 - ORNL-JAPC-USDOE Joint Study – ASME Evaluation	86
Table 36 - JNC – ASME Evaluation.....	87
Table 37 - CEA Creep Fatigue Tests – ASME Evaluation	88
Table 38 - Stress Controlled Creep-Fatigue Tests – ASME Evaluation	88
Table 39 EPRI/ CRIEPI Joint Studies – ASME Evaluation	89
Table 40 IGCAR Study – ASME Evaluation	89
Table 41 - RCC-MR Creep-Fatigue Evaluation	90
Table 42 - ORNL- JAPC- USDOE Joint Study – RCC-MR Evaluation	92
Table 43 - JNC – RCC-MR Evaluation.....	93
Table 44 - CEA – Creep Fatigue Tests - RCC-MR Evaluation	94
Table 45 - CEA - Stress Controlled Creep Fatigue Tests - RCC-MR Evaluation	95
Table 46 - EPRI / CRIEPI Joint Studies – RCC-MR Evaluation	96
Table 47 - IGCAR Creep Fatigue Tests Results – RCC-MR Evaluation	97
Table 48 - T-14.3-1	99
Table 49 - Calculated Allowable Life for the Eddystone Pipes (With $K'=0.9$).....	102
Table 50 - Proposal for K' Factor	102
Table 51 - Example of ASME Evaluation using Proposed Modifications	104
Table 52 - Comparison of Combination of Primary and Secondary Stresses.....	107
Table 53 - Proposal of Creep Tests on Mod 9Cr-1 Mo for Assessment of Negligible Creep Conditions	115

Table 54 - Proposal of Creep Tests on Modified 9Cr 1 Mo for Extension of the Database..... 116

Table 55 - Fatigue-Relaxation Tests..... 126

Table 56 - Creep-Fatigue Tests 126

Table 57 - Fatigue-Relaxation Tests at 550°C..... 126

Table 58 - Creep-Fatigue Tests at 550°C 127

FOREWORD

This document is the result of work resulting from Cooperative Agreement DE-FC07-05ID14712 between the US Department of Energy (DOE) and ASME Standards Technology, LLC (ASME ST-LLC) for the Generation IV (Gen IV) Reactor Materials Project. The objective of the project is to provide technical information necessary to update and expand appropriate ASME materials, construction, and design codes for application in future Gen IV nuclear reactor systems that operate at elevated temperatures. The scope of work is divided into specific areas that are tied to the Generation IV Reactors Integrated Materials Technology Program Plan.

ASME ST-LLC has introduced the results of the project into the ASME volunteer standards committees developing new code rules for Generation IV nuclear reactors. The project deliverables are expected to become vital references for the committees and serve as important technical bases for new rules. These new rules will be developed under ASME's voluntary consensus process, which requires balance of interest, openness, consensus, and due process. Through the course of the project ASME ST-LLC has involved key stakeholders from industry and government to help ensure that the technical direction of the research supports the anticipated codes and standards needs. This directed approach and early stakeholder involvement is expected to result in consensus building that will ultimately expedite the standards development process as well as commercialization of the technology.

ASME has been involved in nuclear codes and standards since 1956. The Society created Section III of the Boiler and Pressure Vessel Code, which addresses nuclear reactor technology, in 1963. ASME Standards promote safety, reliability, and component interchangeability in mechanical systems.

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ABSTRACT

This report provides recommendations for improvement of ASME NH for Grade 91 in the areas of negligible creep and creep-fatigue.

The report is separated into the following four parts.

Part I Improvement of ASME NH for Grade 91 (Negligible Creep)

Examines the current approaches available to define negligible creep and checks their applicability to Grade 91 steel. The work is based on material data available in France and the U.S.

Part II Improvement of ASME NH for Grade 91 (Creep-Fatigue)

Compares Subsection NH and RCC-MR creep-fatigue procedures. Comparisons are performed on cases defined on the basis of experimental test results available from Japan, France and the U.S. on Grade 91 steel. Particular attention was paid to the definition of safety factors and creep-fatigue damage envelope. Improvements to existing procedures are recommended.

Part III Proposed Test Program to Assess Negligible Creep Conditions of Modified 9Cr-1Mo

Part III is aimed at defining tests necessary to validate negligible creep conditions for Mod 9Cr-1 Mo material.

Part IV Proposed Test Program to Validate Creep-Fatigue Procedures for Modified 9Cr-1Mo

Part IV completes the work performed in Part II which, on the basis of creep-fatigue tests results available from Japan, Europe and the US, compared creep-fatigue procedures of ASME Subsection NH and RCC-MR Subsection RB.

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**PART 1
IMPROVEMENT OF ASME NH
FOR GRADE 91
(NEGLECTIBLE CREEP)**

1 INTRODUCTION

In the frame of the AREVA HTR-VHTR design, it is recommended to operate the Reactor Pressure Vessel (RPV) in the negligible creep regime in order to avoid the implementation of a surveillance program covering the monitoring of the creep damage throughout the whole life of the reactor. Within the two options that are currently under consideration for the RPV material of ANTARES (AREVA New Technology based on Advanced gas cooled Reactor for Energy Supply), the high chromium-alloyed steel known as grade 91 in ASTM SA 336 standard has more creep properties documented and is also expected to allow more severe hot transients. The purpose of this report is to discuss the negligible creep conditions of this steel, also called Mod. 9Cr-1Mo. Mod. 9Cr-1Mo is a ferritic steel and not an austenitic stainless steel. Following ASME Boiler and Pressure Vessel (B&PV) Code, Section III for Class 1 nuclear components, additional rules of Subsection NH which take creep and creep/fatigue interaction effects into account should be used for applications above the limit of 371°C (700°F). Thus, the definition of the negligible creep conditions is of prime importance to enable the use of elevated core inlet temperatures during normal operating conditions (400°C at least) and to accommodate transients of limited duration. In addition, the negligible creep criteria will need to take account of the 60 year design life of the reactor which corresponds to 4.2×10^5 hours of operation (based on 80% availability).