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Title: Gap Analysis of Material Properties Data for Ferritic/Martensitic HT-9 Steel

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GAP ANALYSIS OF MATERIAL PROPERTIES DATA FOR FERRITIC/MARTENSITIC HT-9 STEEL

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EXECUTIVE SUMMARY

The US Department of Energy (DOE), Office of Nuclear Energy (NE), is supporting the development of an ASME Code Case for adoption of 12Cr-1Mo-VW ferritic/martensitic (F/M) steel, commonly known as HT-9, primarily for use in elevated temperature design of liquid-metal fast reactors (LMFR) and components. In 2011, Los Alamos National Laboratory (LANL) nuclear engineering staff began assisting in the development of a small modular reactor (SMR) design concept, previously known as the Hyperion Module, now called the Gen4 Module. LANL staff immediately proposed HT-9 for the reactor vessel and components, as well as fuel clad and ducting, due to its superior thermal qualities.

Although the ASME material Code Case, for adoption of HT-9 as an approved elevated temperature material for LMFR service, is the ultimate goal of this project, there are several key deliverables that must first be successfully accomplished. The most important key deliverable is the research, accumulation, and documentation of specific material parameters; physical, mechanical, and environmental, which becomes the basis for an ASME Code Case. Time-independent tensile and ductility data and time-dependent creep and creep-rupture behavior are some of the material properties required for a successful ASME Code case. Although this report provides a cursory review of the available data, a much more comprehensive study of open-source data would be necessary.

This report serves three purposes: (a) provides a list of already existing material data information that could ultimately be made available to the ASME Code, (b) determines the HT-9 material properties data missing from available sources that would be required and (c) estimates the necessary material testing required to close the gap. Ultimately, the gap analysis demonstrates that certain material properties testing will be required to fulfill the necessary information package for an ASME Code Case.

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1.0 INTRODUCTION

In support of Los Alamos National Laboratory (LANL) sub-contract #179350, Global Nuclear Network Analysis (GNNA, LLC) has developed a gap analysis between American Society of Mechanical Engineers (ASME) requirements for elevated temperature design material properties and the US Department of Energy (DOE) database titled, *Nuclear Systems Materials Handbook (NSMH)* (1), for 12Cr-1Mo-VW ferritic/martensitic (F/M) steel HT-9. The gap analysis is one of several major milestones supporting the DOE Work Package at LANL.

2.0 BACKGROUND

The Small Modular Reactor (SMR) community has an immediate need for use of ferritic/martensitic (F/M) steel, commonly known as HT-9. The F/M HT-9 material is being considered not only for fuel-cladding but for reactor vessel components and piping.

The US Department of Energy (DOE), Office of Nuclear Energy (NE), is supporting the SMR designs in fostering advanced concepts and materials. Certain SMR designs will be utilizing liquid-metal cooled fast reactor (LMFR) systems that require materials with excellent high-temperature properties and creep resistance. One such design is from Hyperion Power Generation Inc., which has recently changed its name to Gen4 Energy, Inc. Gen4 Module is a liquid lead-bismuth eutectic (LBE) cooled SMR with fuel pellets contained in HT-9 cladding tubes. Along with scientific and engineering support from Los Alamos National Laboratory (LANL) and funding from the DOE/NE, the Gen4 Energy Inc., engineering design staff are pursuing the use of HT-9 for its liquid-metal cooled fast reactor (LMFR).

However, in order to meet Nuclear Regulatory Commission (NRC) compliance for use of specific reactor pressure boundary materials, approval must be obtained from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, Section III, Division 1, "*Rules for Construction of Nuclear Facility Components – Class 1 Components*" (2), Section III, Division 5, "*Rules for Construction of Nuclear Facility Components – High-Temperature Reactors* (3)," and Section II, "*Materials* (4)."

2.1 LANL WORK PACKAGE GOALS

The DOE/LANL Work Package for HT-9 material, *Qualification of HT-9 Structural and Cladding Materials*, ultimately focuses upon the development of a material-specific ASME Code Case to support approval of HT-9 application for elevated temperature service in LMFR. Specifically, the Work Package lists four major milestones:

- Discuss HT-9 Code Case with ASME Sec. III
- Package Submittal for Re-Instatement of ASTM A771
- Gap Analysis Report on HT-9
- Strategic Planning Report on Developing ASME Code Case for HT-9

The first two items have been completed successfully, and this report is provided as fulfillment of the third deliverable, *Gap Analysis for HT-9*. The final Work Package item is slated for delivery in late August, 2012 and will include a strategy for developing an ASME Code Case for HT-9.

2.2 ASTM MATERIAL SPECIFICATION

In 2004, ASTM A771 specification, "*Specification For Seamless Austenitic and Martensitic Stainless Steel Tubing for Liquid Metal-Cooled Reactor Core Components*" (5) was withdrawn by ASTM due to lack of interest and use, but has currently been requested for re-instatement by GNNA and LANL. In mid-April 2012, a submittal package was delivered to ASTM (6) requesting re-instatement of ASTM A771 for F/M steel HT-9. At the May 2012 bi-annual ASTM sub-committee meetings in Phoenix of A01.10, *Stainless and Alloy Steel Tubular Products*, unanimous approval was provided for acceptance and adoption of the re-instated ASTM A771 specification.

For a LMFR system design, several product category specifications are required, where currently there is but only one ASTM Specification for HT-9, that being for a tubular product. In the future, other product form specifications will be required such as, plate, forging, bar, pipe, etc. All product forms base metal and welds must have an appropriate

ASTM specification with minimum specified mechanical and tensile data at room temperature conditions.

2.3 REGULATORY COMPLIANCE

Title 10, Part 50, of the *Code of Federal Regulations* (10 CFR Part 50), *"Domestic Licensing of Production and Utilization Facilities* (7)*,"* Section 50.55a(c), *"Reactor Coolant Pressure Boundary,"* requires, in part, that components of the reactor coolant pressure boundary be designed, fabricated, erected, and tested in accordance with the requirements for Class 1 components of Section III, *"Rules for Construction of Nuclear Power Plant Components* (2)*,"* of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code or equivalent quality standards. For elevated temperature service in the creep regime, design of reactor vessel and components in accordance with Section III, Division 1, Subsection NH is required, and more specifically, the additional requirements from ASME Section III, Division 5 for high-temperature LMFR designs.

Ensuring compliance with rules and regulation of the ASME Code, does not necessarily imply that all NRC regulatory measures are met. Insofar as the ASME Code provides the necessary rules and guidance for design of nuclear components, there are no specific rules for ensuring that environmental effects on the specific material are considered. The ASME Code alludes to the User/Owner being cognizant to consider such effects, yet does not enforce design rules for analyzing these typical material degradation mechanisms. As such, the NRC imposes additional regulatory compliance measures on nuclear component designers for characterization of material degradation mechanisms due to coolant environment, including liquid-metal corrosion, irradiation embrittlement, hydrogen embrittlement, etc.

2.4 ASME CODE CASE

The ultimate goal of the LANL Work Package is to achieve approval and authorization from ASME, Sec. III, Div. 1 (2), Div. 5 (3) and Sec. II (4) for a material Code Case specifically devoted to use of F/M steel HT-9 in elevated temperature applications, primarily with LMFR designs. It is only through formal approval by the ASME Boiler and Pressure Vessel Code that new materials for use in elevated temperature reactor service are allowed.

LANL and GNNA have initiated discussions with ASME, Section III, Division 1 (*Nuclear Reactors*) and Section III, Division 5 (*High-Temperature Reactors*), leading to the development of a material Code Case for adopting HT-9. Wide interest in use of HT-9 was recently voiced by members of the Working Group on Liquid-Metal Reactors (WG-LMR) of Section III, Division 5 at the November 2011 meeting in St Louis, MO.

2.5 NSMH INFORMATION

The Nuclear Systems Materials Handbook (NSMH) (1) was developed in the 1970's for application in the Fast Breeder Reactor Program, and comprises time-dependent and time-independent physical, tensile and creep data from numerous materials characterized. The complete volumes of the NSMH are still considered by the US Department of Energy as *Applied Technology*, and as such imply the information is restricted to US Government use only, and not available to open-source commercial venture. The majority of HT-9 material characterization data contained in the NSMH, and required by the ASME Code, is information with a low probability of being re-classified to allow open-source availability. Furthermore, because some of the NSMH data attributed to HT-9 physical properties is a compilation of several alloys with close proximity to HT-9 (i.e., 410 and 430 stainless steels), these data will more than likely not be accepted by the ASME Code.

A goal of the *Gap Analysis Report*, is to identify (a) the material data required by the ASME Code for elevated temperature service, (b) HT-9 data already existing in the NSMH, (c) additional HT-9 data residing in open-source documentation, and (d) determine the additional material characterization data required.

3.0 ASME CODE DATA REQUIREMENTS

Section III, Division 1, Subsection NB of the ASME Code provides the material data for reactor components subjected to temperatures below 700°F and 800°F for ferritic and austenitic steels, respectively. Reactor vessel components and fuel clad/duct materials whose respective temperatures are below these limits do not require design/analysis for creep, creep-fatigue, or creep-rupture. Below these limiting temperatures of 700°F and 800°F for ferritic and austenitic steels, material behavior is considered *time-independent*, such that prolonged exposure times at these temperatures will not appreciably degrade the material behavior.

For temperatures above 700°F and 800°F for ferritic and austenitic steels, respectively, material behavior is considered *time-dependent*, where degradation mechanisms such as creep and creep-rupture play an important role. The time-dependent material behavior implies that the exposure time at a given elevated temperature promotes creep and creep-rupture under load-controlled conditions and stress relaxation under deformation-controlled conditions. As such, the time-dependent material behavior is critical to the design life of a component.

Dr. Sam Sham, from Oak Ridge National Laboratory (ORNL), developed Appendix Y of ASME Sec. III, Subsection NH, which provides a synopsis of required material testing and properties necessary to conduct an elevated temperature service reactor component design. The information that follows draws heavily from Appendix Y and likewise follows its format. In general, the following set of test data categories as a function of temperature are required for an ASME Sec. III design:

Test Data Category	**Function/Purpose**
Physical	Common physical properties vs temperature.
Tensile	Yield, ultimate, %E, %RA and monotonic curves.
Creep	Determine strength reduction factors; Establish creep regimes and mechanisms; Support design for long-term service; creep rupture.
Stress Relaxation	Strain ratcheting; Supports material modeling.
Cyclic	Fatigue and creep-fatigue; Support design curves for ASME Code. Stress-strain curves; Determine non-isothermal response.
Fracture Toughness	Characterize embrittlement effects due to aging.
Multiaxial	Model component stress states. Verify von Mises assumption.

3.1 PHYSICAL PROPERTIES

Physical properties necessary to conduct an ASME Code, Sec. III, Div. 1 analysis are the following set of parameters, whose data is taken from room temperature to 100°F above maximum use temperature. In LMFR elevated temperature service, such as the Hyperion Power Module *(now called the Gen4 Module designed by Gen4 Energy)*, an upper-bound metal temperature of 600°C (1112°F) is assumed, resulting in limits of 70°F to 1200°F (or higher – current data in ASME shows 1500°F). Again, testing must be accomplished up to 100°F above the maximum-use metal temperature expected in-service. A list of existing material in Sec. III, Div. 1 and Subsection NH, show an upper-limit of 1500°F for physical properties.

- Density, ρ
- Poisson's ratio, ν ,
- Specific heat, c_p
- Thermal diffusivity, α_D ,
- Thermal conductivity, k ,
- Thermal expansion coefficient, α_L

Temperature dependence

where $$\alpha_D = \frac{k}{\rho c_p}$$

3.2 LOAD-CONTROLLED LIMITS

Load-controlled limits, per the ASME Code, are stress or strain measures below which the structure must be maintained to mitigate elastic-plastic collapse from direct applied loads. These are commonly termed *"primary"* stresses. Strength and ductility parameters, such as yield strength, ultimate strength, and percent elongation are a few of such measures. A direct axial load on a bar specimen, or internal pressure in a cylinder resulting in direct hoop stress, are examples of load-controlled situations.

3.2.1 Time-Independent Data

The material properties necessary for the ASME Code Case data package for time-independent behavior are shown below. These data are required for each heat and representative product form (e.g., plate, tube, forging, weld, etc.), and taken from room temperature to 100°F (50°C) above maximum use temperature, at 100°F (50°C) intervals. If there are several different chemical composition ranges for the material, all compositions must be tested.

- Young's modulus, E
- Yield strength, S_y
- Ultimate tensile strength, S_u
- Tensile elongation, $\%Elong$
- Reduction of area, $\%RA$
- Monotonic tensile stress-strain curves, $\sigma - \varepsilon$
- Charpy V-Notch impact, CVN
- Fracture toughness, K_{Ic}
- Fatigue crack-growth rate, da / dN

Temperature dependence

A minimum of 3 commercial heats are required to be tested for each product form. The yield strength and ultimate tensile strength are used to determine the *time-independent* primary stress limits, S_m. Per guidance provided in Appendix Y of Sec. III, Div. 1, Subsection NH:

> "During long-term elevated temperature service, the yield and ultimate tensile strengths of metallic structural materials may be reduced due to metallurgical aging, NH-2160(d). Data on yield strength and ultimate tensile strength are required to demonstrate that a new material is not susceptible to thermal aging over the intended time and temperature range of applications. If the material is susceptible to thermal aging, yield and ultimate tensile strengths data from thermally aged materials are required

to establish *tensile reduction factors for aging* as functions of exposure time and exposure temperature."

As an example of available open-source data, below is a sampling mechanical properties that may prove to have sufficient data to support an ASME Code Case submittal.

Elongation

Klueh et al. (8) measured %Elong. in 12Cr-MoVW steel tempered at two different temperatures 750^{o}C and 780^{o}C. The authors report %Elong. equal to 10.4 and 9.9, respectively, see Table 4 in Klueh et al. (8). Also, total elongation in HT-9 is reported in Fig.10 taken from Klueh and Vitek (9). Further results given for HT-9 (heat HT1) at 550°C show %Elong. of 12.5 for normalized and tempered, aged to 5000h, and after irradiation to 9 and 24 dpa in EBR-II.

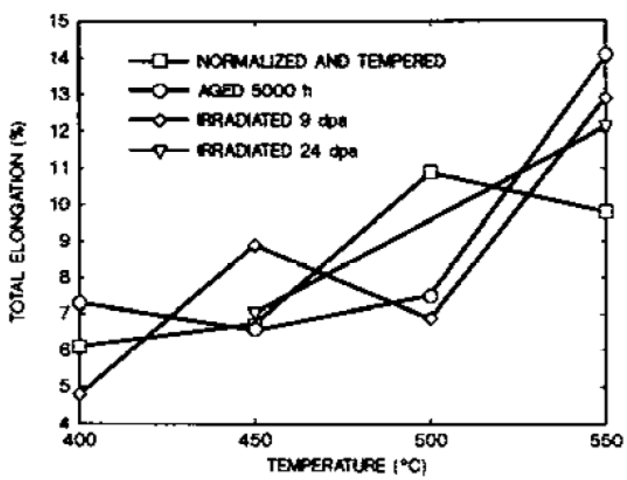

Figure 1 – Elongation results from Klueh for normal and irradiated samples.

Klueh et al. (10) also reported uniform and total elongation for unirradiated, thermally aged and irradiated 12Cr-1MoVW steels (heats HT1 and HT2), see figure below: e.g. HT-1 aged, unirradiated, irradiated (open square, open circle, solid circle), respectively. Total elongation and uniform elongation are also reported as a function of irradiation temperature for room temperature test for the same 12Cr-1MoVW steel (heats HT1 and HT2).

These data seems to be adequate and although there may be 3-heats total from two separate testing periods, this might be significant enough to not warrant further testing. However, the data needs to be reviewed in detail to ensure its validity for ASME Code Case purposes.

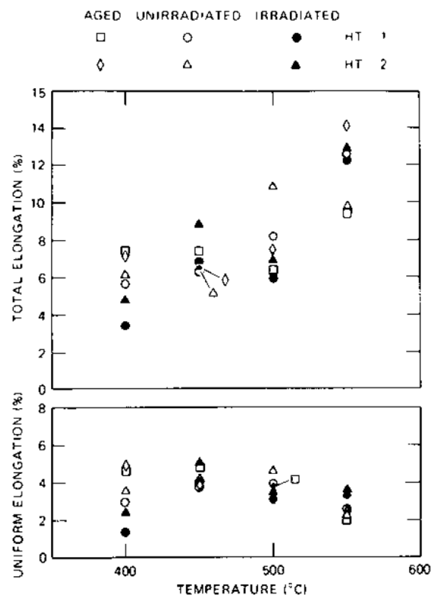

Figure 2 – Aged irradiated and unirradiated samples from Klueh

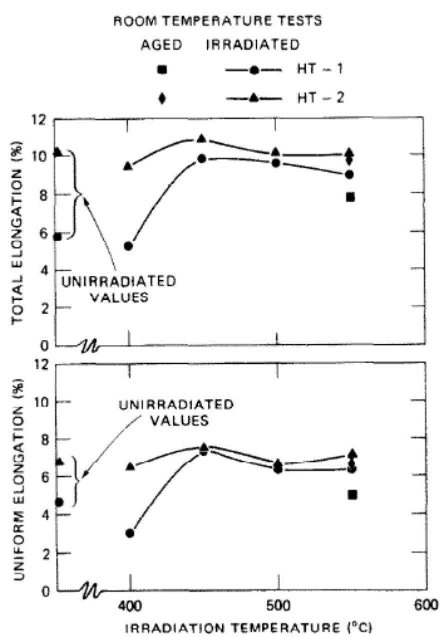

Figure 3 – Room temperature elongation data from Klueh.

Fracture Toughness

Huang and Hamilton (11) reported fracture toughness (J_{Ic}) and tearing modulus (T) for HT-9. The investigation focuses on specimen size effects. In the figure below B and W are the specimen thickness and width, respectively. The toughness of HT-9 decreases from room temperature up to roughly 300°C, increasing back to the room temperature toughness at or near ~ 500°C. The tearing modulus shows a broad maximum where J_{Ic} shows a minimum (reverse behavior). Specimen size seems not to have an effect on J_{Ic}, however, a small dependence on specimen size is observed for the tearing modulus.

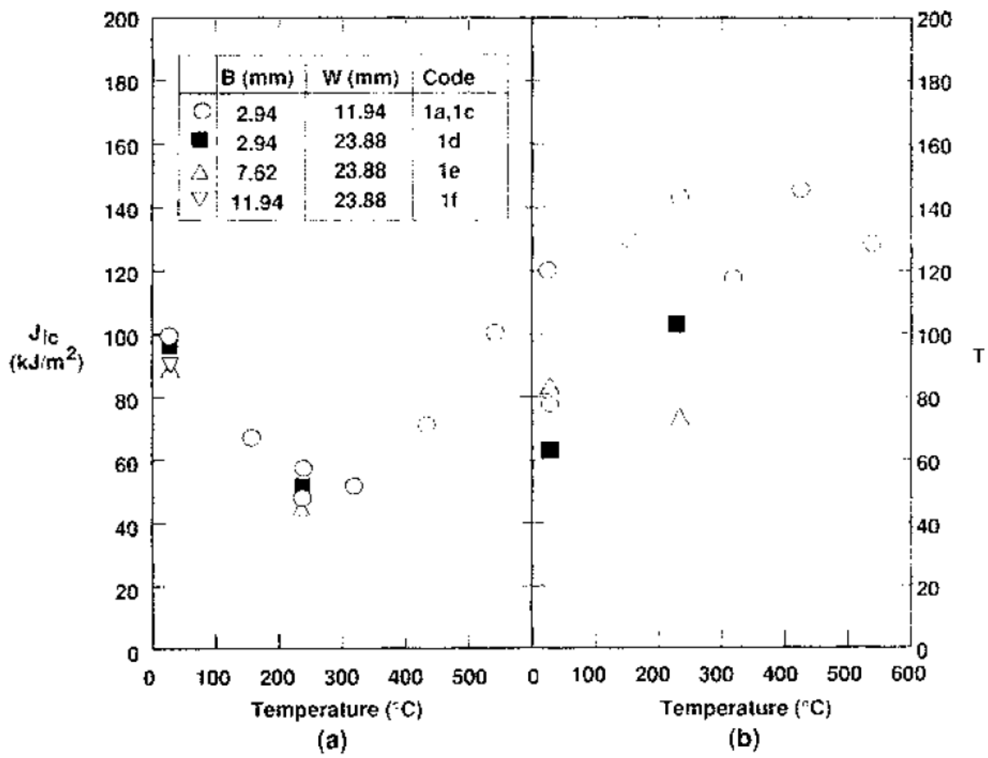

Figure 4 – J_{Ic} vs temperature for different specimen size HT-9.

The effect on J_{Ic} and T of neutron irradiation to different doses in the range from 12 to 108 dpa, and different irradiation temperatures in the range from 290°C to 600°C is displayed in Figure 5 from Huang (11). Irradiations were performed in EBR-II and FFTF fast reactors. The figure also shows the results for the unirradiated condition (open circles). Only minimal variations in toughness are observed after irradiation. However, tearing modulus values vary significantly.

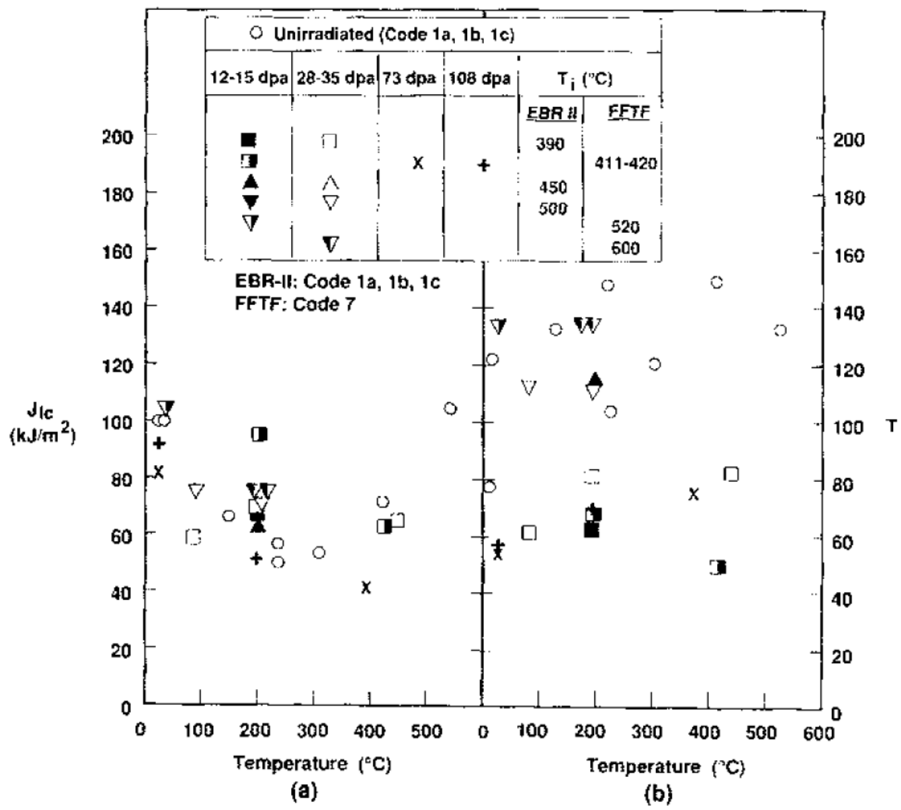

Figure 5 – Irradiated and unirradiated J_{Ic} data vs temperature.

The authors conclude that no additional loss of fracture toughness occurs with irradiation beyond 30 dpa in HT-9 up to doses of ~ 100 dpa. Tearing modulus apparently shows large scatter. Also, Huang (12) reported on fracture behavior of HT-9 (heats 84425 and 91353) irradiated in FFTF to 180 dpa. Temperature dependence of fracture toughness (K_{Ic}) is depicted in Figure 6, for different irradiation conditions in the MOTA experiment, i.e. irradiation temperatures in the range from 410°C to 550°C and fast neutron fluence in the range 13 – 36 x 10^{22} n/cm^2.

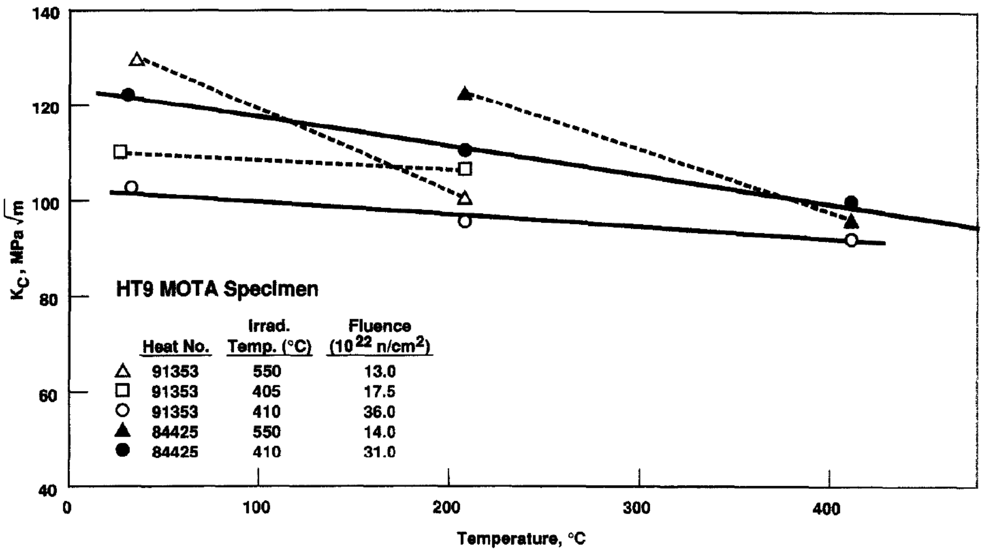

Figure 6 – Irradiated K_{Ic} samples at different temperatures.

The dependence of K_{Ic} on fast neutron fluence is shown in Figure 7. The irradiation was performed in FFTF at an irradiation temperature of 410°C and for a neutron fluence that reached 36 x 10^{22} n/cm^2. K_{Ic} for the non-irradiated condition is also shown. HT-9 fracture toughness seems to saturate with fluence as depicted in Figure 7.

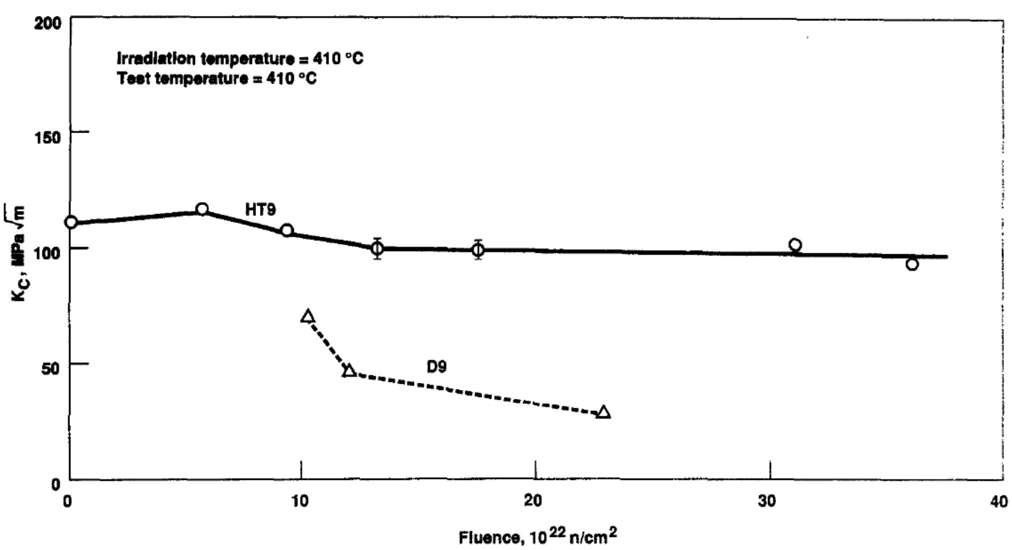

Figure 7 – Irradiated K_{Ic} samples vs fluence level at constant temperature.

3.2.2 Time-Dependent Data

Assuming an LMFR SMR design with a 10-yr core-life, creep testing to a maximum of 100,000 hrs would suffice. Values of *time-dependent* primary stress intensity limit, S_t, as a function of both time and temperature are required per Subsection NH to ensure that gross collapse is mitigated at elevated temperature. The data considered in establishing these values are obtained from long-term, constant-load, uniaxial tests. As an example, Figure 8, shows a family of curves for T91 F/M steel.

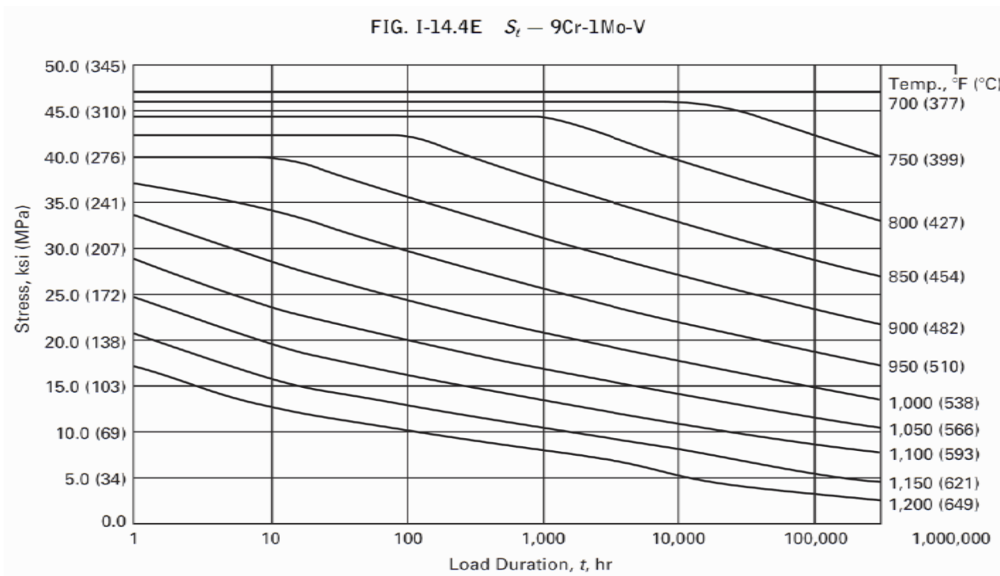

Figure 8 – Primary stress intensity limit, S_t, for T91 F/M steel from ASME NH.

Additional *time-dependent* data required for an ASME Code evaluation are:

(1) average stress to cause 1% total strain
(2) minimum stress to the onset of tertiary creep
(3) minimum stress-to-creep rupture
(4) minimum stress-to-rupture curves (see Figure 9)

The data required to establish the above *time-dependent* allowable stresses are obtained from creep rupture tests at various temperatures and applied primary stress levels. See for example, Figure 9, showing T91 stress rupture data. Since time to 1% total strain and time to onset of tertiary creep are required, it follows that full creep curve data are required from creep rupture tests. Creep rupture data from a minimum of 3 commercial heats, covering the compositional ranges of chemistry, sizes and product forms for the applications are required. Data from 50°F (25°C) below the creep threshold to 100°F (50°C) above the maximum use temperature at 100°F (50°C) intervals and in the time range from 100 hours to beyond 60,000 hours, evenly spaced in log-time, are required. The upper-time limit is dependent upon the intended design life of the component and extrapolation to higher time limits is often recommended.

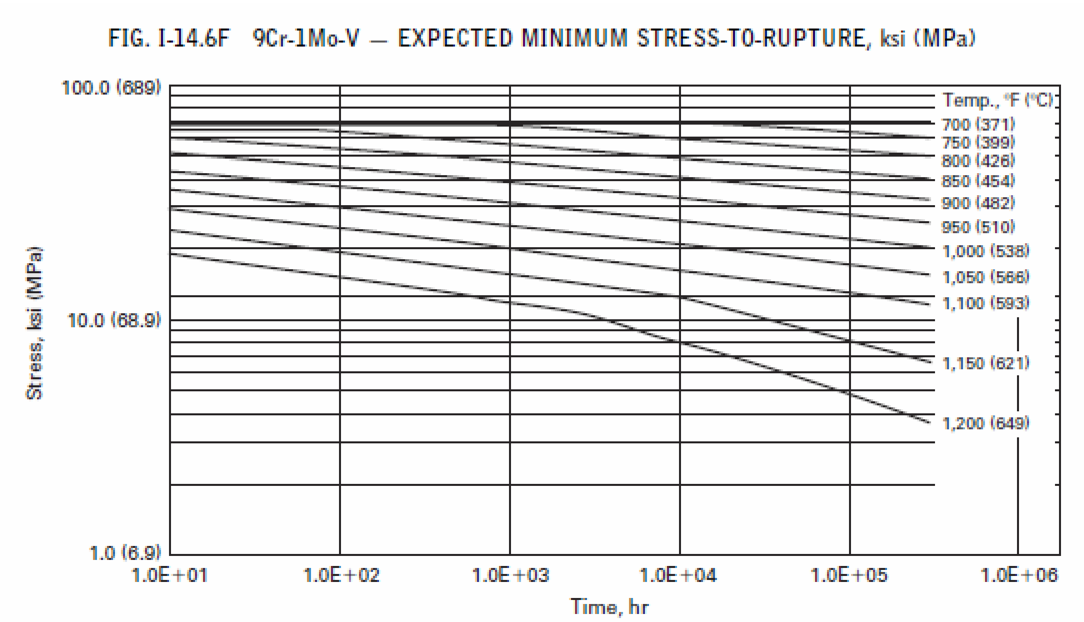

Figure 9 – Typical stress to rupture data for F/M steel T91 from ASME NH.

Therefore, assuming that creep behavior in HT-9 begins at approximately 400°C (752°F) and the maximum use temperature of the RPV component is 600°C (1112°F), then creep testing should be performed from ~700°F to ~1200°F at 100°F intervals. This would require 6 specimen per heat, with 3-heats total, or 18 total specimen.

3.3 DEFORMATION-CONTROLLED LIMITS

The deformation-controlled, or strain-controlled, quantities are stresses, strains, and deformations resulting from compatibility conditions of the structure or of load, deflection and strain. Stresses developed from bending moments at the end-support of a fixed-fixed beam is an example of deformation-controlled measure. The end moment is not required for equilibrium, yet is necessary for compatibility of deformations. Thus, minor plastic deformation will not be detrimental to the overall load-carrying capability of the structure.

3.3.1 Isochronous Stress Strain Curves

Isochronous stress-strain curves are developed from typical creep-strain time histories at constant stress. For analysis purposes, however, the data presentation becomes more convenient to cross-plot creep data on stress-strain axes, producing lines of stress versus strain at constant time (13).

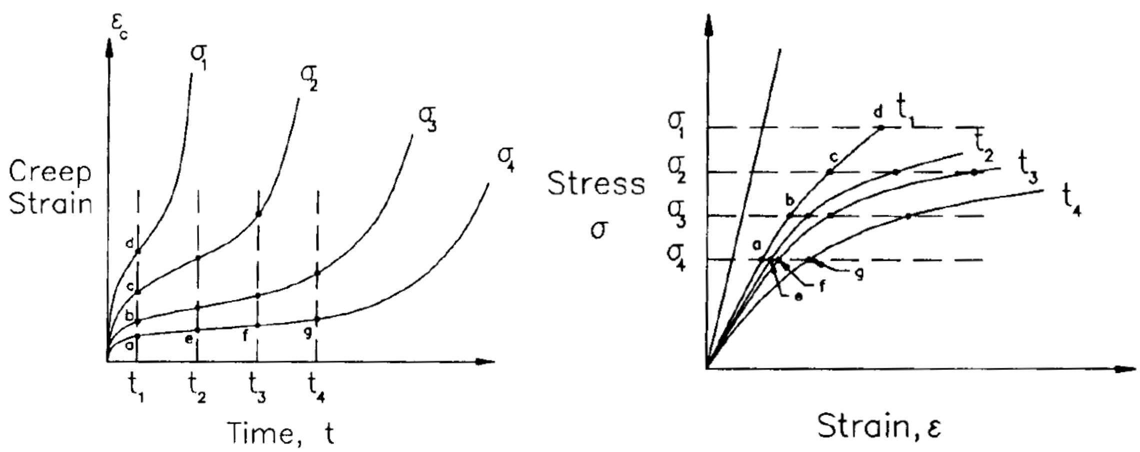

Figure 10 – Isochronous stress-strain curves.

A typical ASME Code isochronous stress-strain curve for 304SS, is shown in Figure 11. In general, there should be a family of curves from 10hrs to 300,000hrs for each temperature tested. Thus, from 700°F to 1200°F, partitioned in 100°F increments, there would be 6 separate plots.

FIG. T-1800-A-4 AVERAGE ISOCHRONOUS STRESS-STRAIN CURVES

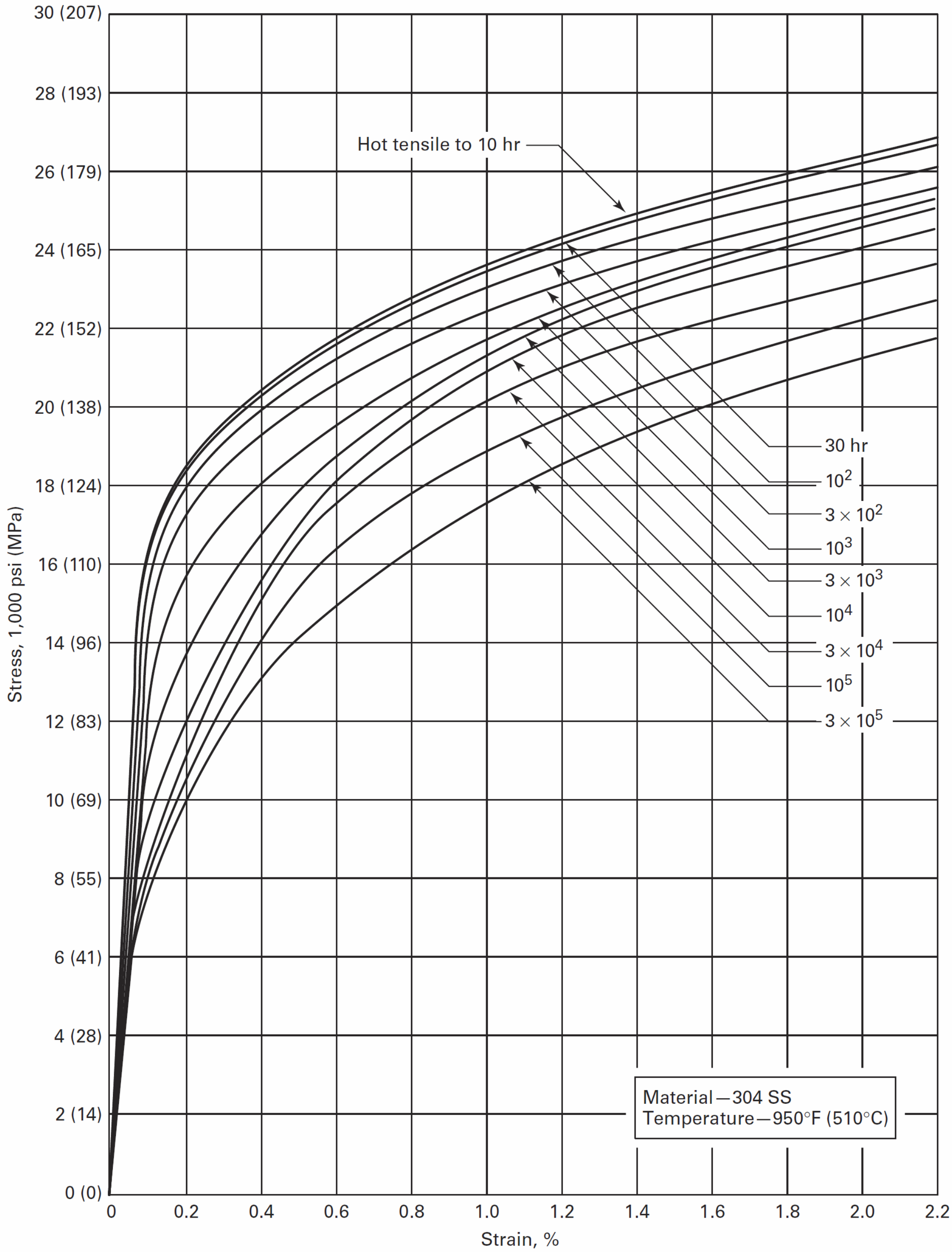

Figure 11 – 304SS isochronous stress-strain curve at 950°F.

3.3.2 Relaxation Strength

Stress relaxation testing may be accomplished with uniaxial tensile specimen as shown in Figure 12, where an axial deformation (i.e., strain) is applied, resulting in a given initial stress level, at a given temperature. The initial stress is then observed to relax as a function of time, as shown in Figure 13. A family of curves, at specific test temperatures, would represent stress versus time throughout a complete life-span.

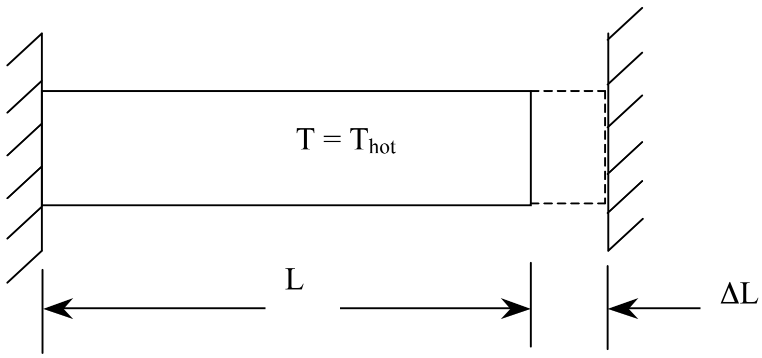

Figure 12 – Stress-relaxation test-specimen.

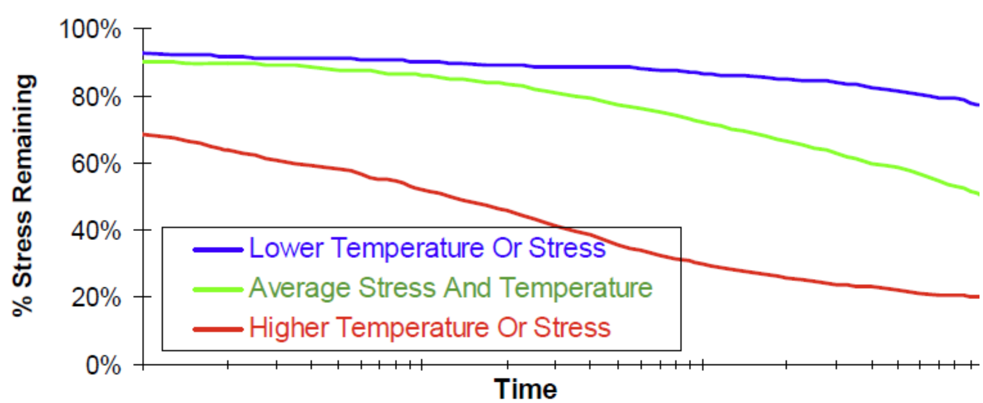

Figure 13 – Relaxation curve.

3.3.3 Creep-Fatigue

Per Sec. III, Div. 1, Subsection NH, creep-fatigue is evaluated through application of a linear summation of cyclic damage and creep damage.

$$\underbrace{\sum_{j=1}^{p}\left(\frac{n}{N_d}\right)_j}_{\boxed{\text{Cyclic damage}}} + \underbrace{\sum_{k=1}^{q}\left(\frac{\Delta t}{T_d}\right)_k}_{\boxed{\text{Creep damage}}} \le D$$

where: $D =$ Total damage from creep-fatigue

The cyclic damage is based on normal fatigue data, except that a hold-time is applied between cycles, which results in a typical S-N curve shifting left-ward due to creep damage accumulation. Creep-fatigue interaction diagram for HT-9 is expected to be similar to the 9Cr-1Mo curve shown in Figure 14.

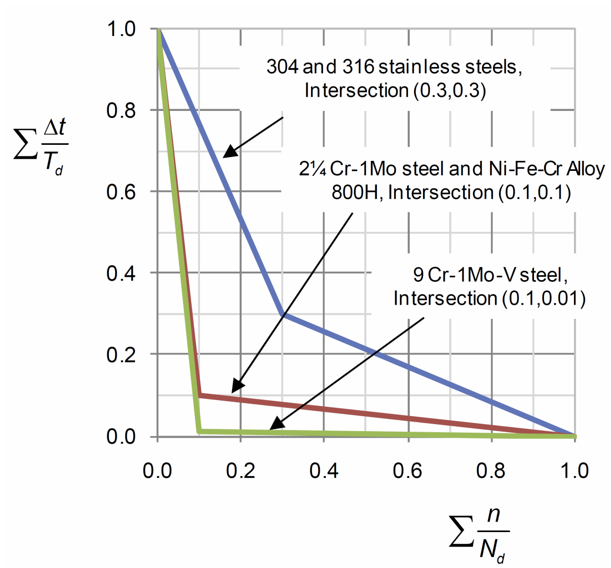

Figure 14 – Creep-fatigue interaction.

3.3.4 Cyclic Stress-Strain Curves

Cyclic stress-strain specimen testing, from room temperature to 100°F above the maximum operating temperature, are required. Testing is accomplished at 100°F intervals throughout the complete temperature range, and is used to assess creep-effect and creep-fatigue interaction.

4.0 NUCLEAR SYSTEMS MATERIALS HANDBOOK

The NSMH contains physical properties and elevated temperature tensile and creep material behavior for HT-9. The NSMH data would be a useful starting point for development of an ASME Code Case, although the data is still considered *Applied Technology*. The NSMH does not have all the necessary data for a complete ASME material Code Case, nonetheless it does provide critical data that might potentially be included in a Code Case.

Unfortunately, some HT-9 physical data contained in NSMH is based not on HT-9 chemistry, but rather on several 400-series chromium steels having a close-proximity in chemistry to HT-9, such as the stainless-steel Ferritic series 400, 430, and 443, and Martensitic series, 410, 420 and 440. Thus, these physical data quantities might not be accepted by ASME, and would need additional testing with an actual HT-9 heat of material.

Although much of the NSMH data is considered good quality, there is a lack of referenced reports where the original data might reside. As an example, under the Biaxial Stress Rupture properties discussion, it states that *"....all tests were performed by M. L. Hamilton, at Westinghouse Hanford Company."* However, there is no further mention of actual test information or report number. The lack of traceability of HT-9 data from the NSMH might make this a moot issue, such that further testing would nevertheless be required for a Code Case, if only to maintain archival documentation. Lastly, the NSMH data for HT-9 is quite sparse in general, and is particular to certain material properties/parameters only. Thus, there would need to be further material test conducted for development of an ASME Code Case. A list of actual material data contained in NSMH for HT-9 is as follows:

- Ultimate tensile strength
- Yield strength
- Total elongation
- Biaxial stress rupture
- Creep, irradiation induced
- Fatigue crack growth, effect of temperature in air
- Density

- Specific heat
- Thermal diffusivity
- Thermal emissivity
- Thermal conductivity
- Thermal expansion coefficient
- Stress-free swelling, irradiation induced

5.0 REGULATORY COMPLIANCE MATERIAL NEEDS

There are specific material data characterization, which are not part of the ASME Code design package requirements, yet are fundamentally required to meet NRC compliance of reactor vessel components under environmental degradation. Although yield and ultimate strength degradation due to long-term operation at elevated temperatures are considered in the ASME Code design requirements, environmental effects on material performance are neglected. Nonetheless, per Title 10, Code of Federal Regulations, Part 50, environmental degradation mechanisms must be considered in RPV and component designs, including but not limited to the following;

- Irradiation embrittlement
- Irradiation-induced creep
    - Coupled thermal and irradiation effects
- Hydrogen embrittlement
- Liquid-metal embrittlement
- Liquid-metal corrosion
- Others

As such, the gap analysis will also consider the regulatory needs for material data.

As an example of available information on specific NRC material needs, LANL material scientist, M. Caro has compiled a literature survey (14) on irradiation embrittlement of alloy HT-9. The document provides an exhaustive survey of all available data ensuring that chemistry from all different sources match, somewhat consistently, the ASTM A771 specification.

6.0 GAP ANALYSIS

The gap analysis merely focuses on answering three questions; (1) what is required, (2) what is available, and (3) what data is missing? Obviously, the required data is governed by the ASME Code and NRC regulations. In answering *"what is available?"* requires a thorough and exhaustive literature survey. Only a cursory open-literature survey has been accomplished herein in an effort to determine the depth and breadth of available information. Thus, the *"missing data"* category might be a misnomer until a fully exhaustive literature survey is conducted.

Section 6.1 provides a matrix of required data and available data.

6.1 ASME, NSMH, AND OPEN-SOURCE CROSS-WALK

Property	ASME Code Required	NRC Required	Available in NSMH	Available Open Source	Ref.	NOTES
Physical						
Density	Y	Per 10CFR50	Y	N		Literature search required.
Poisson's Ratio	Y		Y	N		Literature search required.
Thermal Conductivity	Y		Y	N	(15)	Chu and Ho 1978 data possible.
Thermal Diffusivity	Y		Y	N		Literature search required.
Thermal Emissivity	N		Y	N		Literature search required.
Thermal Expansion	Y		Y	Y	(15)	Chu and Ho 1978 data possible.
Specific Heat	Y		Y	N		Literature search required.
Time-Independent						
Yield Strength	Y	Per 10CFR50	Y	Y	(16)	Monotonic and temperature dependent strength degradation
Ultimate Strength	Y		Y	Y	(16)	
% Elong.	Y		Y	N	(8)(10)(9)	Literature search required.
% Reduc. of Area	Y		Y	N		Literature search required.
Fracture Toughness, K_{Ic}	Y		N	Y	(17)(12)(18)(11)	Literature search may be required.
Impact Toughness, CVN	Y		N	Y	(19)	
Fatigue Curve (S-N)	Y		N	N		Literature search required.
Low-Cycle Fatigue (S-N)	Y		N	Y	(20)	
Fatigue Crack Growth Rate	Y		Y	Y	(21)	Family of da/dN curves for different temperatures
Cyclic stress-strain curve	Y		N	N		Literature search required.
Monotonic stress-strain	Y		N	N		Literature search required.
Young's Modulus	Y		N	N		Literature search required.

Y=Yes N=No Ref.=References in Open Source YELLOW = Need further data.

Property	ASME Code Required	NRC Required	Available in NSMH	Available Open Source	Ref.	NOTES
Time-Dependent						
Creep	Y	Per 10CFR50	N	Y	(22)(23)(24)	NSMH data collected at discrete fluence levels.
Creep Fatigue (with hold time on fatigue cycles)	Y		N	Y	(25)(26)	
Creep Rupture	Y		Y	Y	(24)(26)	
Relaxation strength	Y		N	N	(26)	
Isochronous stress-strain	Y		N	N		Further literature search required.
Bi-Axial Stress Rupture	Y		Y	N	(24)(26)	
Environmental Effects						
Irradiation Embrittlement	N	Y	N	Y	(27)(28)(29)(14)	Caro, M., Irradiation Embrittlement
Hydrogen Embrittlement	N	Y	N	Y	(30)(31)(32)	Further literature search required.
Liquid-Metal Embrittlement	N	Y	N	Y	(33)(34)(29)	LBE specific data required.
Irradiation-induced creep	N	Y	Y	Y	(34)(35)	
Liquid-metal corrosion	N	Y	N	Y	(36)(37)(38)	
Stress-Corrosion Cracking	N	Y	N	Y	(39)(37)	
Yield Strength (Irrad.)	N	Y	Y	Y	(40)(41)(42)	
Ultimate Strength (Irrad.)	N	Y	Y	Y	(40)(41)(42)	
Fracture Toughness (Irrad.)	N	Y	N	Y	(43)	
Impact Toughness (Irrad.)	N	Y	N	Y	(43)(19)	

Y=Yes N=No Ref.=References in Open Source YELLOW = Need further data.

6.2 DISCUSSION

Based on a cursory literature review of open-source data, it appears there is quite a variety of information relative to HT-9 properties, including irradiation-induced creep and liquid-metal degradation effects, tensile and fracture strength, and even corrosion data with liquid-metals. Unfortunately, it is unclear from many of the reports and journal articles, the full chemistry of the material processed for testing, including trace elements. This uncertainty in actual chemistry, and especially material not developed under an approved ASTM Standard, would be suspect from adoption by ASME. As mentioned earlier, the adoption of a material by ASME, Sec. III, would require a fully approved ASTM Standard, to which the material and product form is manufactured and tested.

Furthermore, a number of papers and reports, including the NSMH, discuss the testing of a single-heat of material. Again, this would not be acceptable to the ASME Code since 3-separate heats of the same materials are required to ensure material property repeatability. As such, it appears that although valuable information is contained in the NSMH, as well as the cursory open-source literature survey performed, there is a lack of consistency of chemistry, material heats, and even testing parameters.

Gen4 SMR Design

The Gen4 SMR, currently being designed as a LMFR with LBE coolant, has certain design parameters that have not been solidly embedded. Specifically, the current accepted parameters are:

Fast Neutron Fluence = 4×10^{23} Nvt

DPA = 20 – 30

Operating Temp = 350°-550°C

Gen4 SMR designers are aware of the loss of fracture toughness in HT-9 at operating temperatures between 350°C and 450°C, as discussed by Klueh (44). The toughness loss at these temperatures is associated with an upward temperature-shift and upper-shelf energy reduction (see Figure 15).

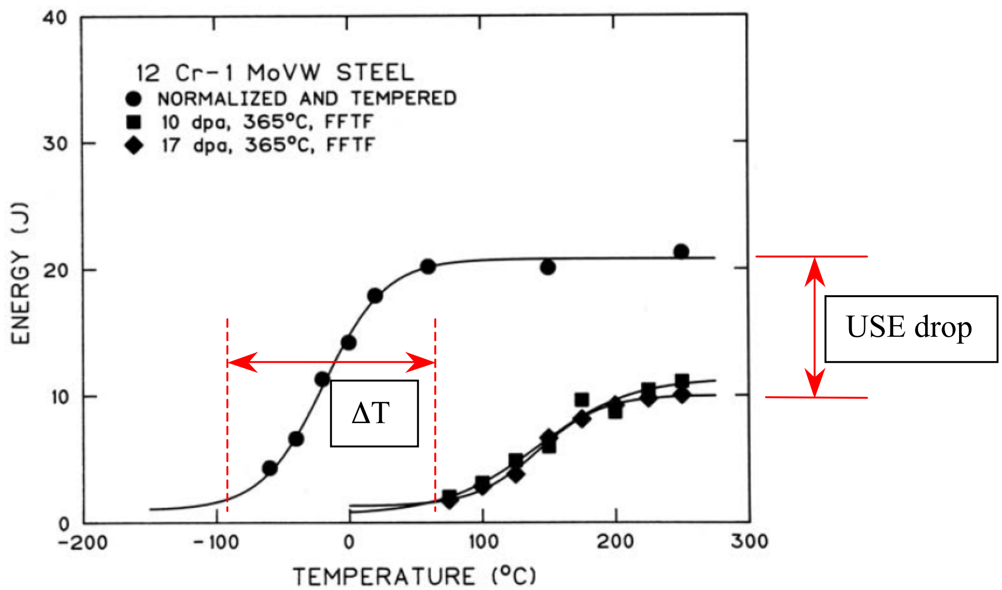

Figure 15 – Charpy V-Notch DBTT shift under irradiation for HT-9.

Because of the potential loss of toughness at temperatures between 350°C and 450°C, the Gen4 designers are modifying design parameters to maintain the operating temperature above the upper-limit of 450°C and thus mitigate a potential problem.

As such, the literature survey will initially focus on obtaining environmental data for material behavior within these operating parameters. From current data gathered, there appears that some information is available within these limits of operation. However, as stated previously, 3-heats of material are required, which implies that additional testing might be required.

6.3 RESULTS

Results of the gap analysis in Section 6.1 show a substantial need to obtain further physical, mechanical, tensile, fracture, and creep properties for HT-9. The *'missing data'* is tentatively presented as a light-yellow colored row. The reasoning for implying that data is *missing* and further testing is necessary, is merely based on the discussion in Section 6.2. That is, if the NSMH has the particular data, but there is no traceability, or the data is pertinent to another material with almost similar chemistry as HT-9, then this material property is said to require further testing. Likewise, if open-source literature is available, but there are questions relative to chemistry or material property repeatability, then further testing is required. In other words, the reasoning is purely subjective on the author's part.

More importantly, when there appears to be very good data in the NSMH and is well documented, then it is assumed that this data is sufficient. The major caveat to this thinking is that the NSMH data, while being *Applied Technology*, is assumed in time would be re-classified to open-source and available to ASME. Again, this process is purely subjective on the author's part. Nevertheless, there are enough questions with published data in the NSMH and open-source journal articles that these data might not be suitable to ASME.

Furthermore, because ASME requires a minimum of 3-heats of materials tested at each temperature, as well as recognizing from open-source information the lack of repeatable data from different heats, it follows that additional material testing will be required for HT-9.

Finally, the NSMH does not cover all necessary data for an ASME Code Case, and as such it becomes incumbent to pursue a full testing phase of HT-9 based on the currently accepted ASTM Standard, A771 (5). A full complement of tests for elevated temperature service, in accordance with ASME Code guidance, is not a trivial matter. As an example of potential material testing required for adoption of an ASME material Code Case, Idaho Engineering Laboratory (INL) produced a program plan for testing Alloy 617, which resulted in development of a report (45) delineating a total number of required tests, total number of specimen, and approximate overall cost. The INL report should be utilized as a template for further consideration of HT-9 testing.

7.0 CONCLUSIONS

A gap-analysis of HT-9 material test data has been accomplished, taking into consideration available data contained in the NSMH and open-source literature. Results demonstrate that further material testing will be necessary to ensure a full complement of accepted ASME material properties with the requisite number of heats reported. However, an exhaustive literature review might initially be performed to determine acceptable open-source material data that could be utilized in the ASME Code Case, and thus reduce the burden of testing.

Next, the development of a test matrix identifying all the test categories, test parameters, and number of specimen required should be completed for all material properties shown in Section 6.1. The test matrix should follow closely with the INL report on Alloy 617, in order to properly estimate an overall cost.

Lastly, a strategy document should be developed to include the above recommendations, proposing a path forward for DOE support of a test program for HT-9. Importantly, material property needs for HT-9 should be championed by those interested parties that are proposing SMR design/development in the near future.

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