HEDL-SA-1794

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MASTER

EXPERIENCE IN THE USE OF FBR CORE COMPONENT STRUCTURAL DESIGN CRITERIA AS APPLIED TO FFTF[1]

S.L. Hecht

ASME 3rd National Congress
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SAN FRANCISCO, CALIFORNIA

HANFORD ENGINEERING DEVELOPMENT LABORATORY
Operated by Westinghouse Hanford Company, a subsidiary of Westinghouse Electric Corporation, under the Department of Energy Contract No. DE-AC14-76FF02170

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EXPERIENCE IN THE USE OF FBR CORE COMPONENT
STRUCTURAL DESIGN CRITERIA AS APPLIED TO FFTF[1]

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ABSTRACT

User gained experience resulting from trial applications of proposed structural design guidelines for Fast Breeder Reactor (FBR) core components is presented. This work was done supporting the design analyses process for consumable core components for the Fast Test Reactor (FTR) of the Fast Flux Test Facility (FFTF). The proposed guidelines were found to be more comprehensive and generally easier to apply than those methods previously used. Component evaluation required a minimum amount of detailed inelastic analysis, primarily through the use of simplified inelastic analysis methods, as given in the guidelines. A major shortcoming of this draft criteria/guidelines is a lack of supporting irradiated material properties. Some areas of guidance given seems ambiguous and may be non-conservative, particularly those related to stress classification unique to FBR environments. Further verification of these areas appears to be in order.

INTRODUCTION

As an integral part of the development plan of the national working group on structural design criteria for Fast Breeder Reactors (FBR) core components, a program of trail applications was undertaken at the various participating organizations.

Since late 1976, Hanford Engineering Development Laboratory (HEDL) Core Engineering, has adopted the use of the working groups' draft criteria, RDT-F9-7, F9-8, F9-9, [1][2] in the design analysis process for new and existing FTR consumable core components. To date these included several new test assemblies and the seismic evaluation of all major core components (e.g., fuel assemblies and control rods).

Previous to the use of RDT F9-7 and F9-8 "Structural Design Guidelines for FBR Core Components," design criteria was essentially based on the ASME Boiler and Pressure Vessel Code, Section III ($T<800°F$) [2], and the high temperature Code Case 1331 [3] and later 1592 [3] rules. Here irradiated material properties were used when available and a limited consideration was given to irradiation induced loss of ductility. This criteria had several shortcomings when applied to irradiated FBR consumable core components, some of which are:

- Inherent design margins could be very small in low ductility irradiated materials when applying stress intensity limits.
- Strain limits do not account for loss of ductility due to irradiation.
- Overly conservative when stresses are low (ductility based life limits).

The "Structural Design Guidelines for FBR Core Components" for the most part, address the effects of high energy neutron irradiation on short life (1-3 year) structural components and integrates these with the rules of the ASME high temperature code cases to provide a comprehensive design criteria for the industry.

[1] Work performed at the Hanford Engineering Development Laboratory, Richland, Washington, operated by Westinghouse Hanford Company, a subsidiary of Westinghouse Electric Corporation, for the U.S. Department of Energy under Contract EY-76-C-14-2170.

[2] Numbers in bracket designate reference at end of paper.

It is the intent of this paper to present the user's "point of view" in respect to applying the rules of F9-7 and the guidance of F9-8 to in-core consumable structural members excluding fuel and absorber pin cladding as there was no criteria application at HEDL in regard to the later. It is expected that early experience gained in FTR application will be similar to those expected industry wide, as loadings and materials are expected to be typical. The content of this paper is based on the 1976 draft criteria.

EXPERIENCE - GENERAL

Application of RDT F9-7 and F9-8 to design analysis at HEDL encompassed most of the rules and guidance set forth in these documents. Components analyzed and the type of analyses performed are given in Table 1. Noteworthy of the fact that for the evaluation of steady-state operation, only stress limits need to be applied. This has several advantages, of which the most obvious is the ease of analysis. Here, only irradiated Yield (S_y) and Ultimate (S_u) strengths need be considered. This is of considerable advantage in case of preliminary design work (steady-state only) where one is considering the use of materials which there is no or limited data on such properties as irradiated stress to rupture (as would be previously required). In many cases, evaluation for off-normal conditions involved the calculation of strains. In most cases, this was done using simplified inelastic analysis methods (SIAM) as outlined in RDT F9-8. Detailed inelastic analysis were rarely required. Because thermal creep in FTR core components is generally negligible, the required damage limits, in many cases, were not evaluated. Damage fractions were calculated in only those cases where the damage fraction due to fatigue (generally low cycle) was significant.

The criteria has been applied extensively to type 304 and 316 (both annealed and cold worked) stainless steel and to a lesser degree to Developmental Alloys of the Solid Solution (Austenitic), ferritic and precipitation strengthened types of metals.

In general, much of the irradiated material properties were not generally available in the official source, the Nuclear Systems Material Handbook [4], and had to be compiled or developed[3] independent of this document. In cases where a limited data base was only available for irradiated material properties, the nominal values were reduced to account for uncertainties.

EVALUATION FOR NORMAL OPERATION

As mentioned earlier, only elastic irradiation creep and swelling analyses needed to be performed for the steady-state operating condition. This is probably typical of the majority of FBR core components, as loadings in FTR are expected to be fairly

[3] Developed in accordance with F9-8.

TABLE I

COMPONENT EVALUATION WITH RDT F9-7 & F9-8 AT HEDL*

COMPONENT	TYPE OF ANALYSIS				OPERATING CONDITIONS		LIMITS EVALUATED								NOTES
							STRESS					STRAIN		DAMAGE	
	Elastic	Elastic-Irradiation Creep and Swelling	Simplified Inelastic Analysis Methods (SIAM)	Inelastic	Steady-State	Transients	P_m	P_L+P_b	$P_L+P_b+Q; \varepsilon_u>1\%$	$P_L+P_b+Q; \varepsilon_u<1\%$	σ_1; R.A.<10%	$\varepsilon_p+\varepsilon_c$ (membrane)	ε_p (max)	D_f+D_c**	
Vibration Open Test Assembly (VOTA)	X	X	X		X	X	X	X	X	X	X	N1 N2	N2	N1 N2	N1 "0" Thermal Creep. N2 Only Needed for Off Normal (Transient) Operation.
Fuels Open Test Assembly (FOTA)				N	N	N						X	X	X	N = Equivalent Loading 1D Cylinder Analysis (Above Core, High Temp. Evaluation).
Multiduct Fuels Test Assembly (MFTA)	X	X		N	X		X	X	X	X	X				Req'd on Non-Std. Sub-Component Only.
Highly Enriched Fuel Assembly (HEFA)	X	X			X		X	X	X	X	X				
Core Component Seismic Analysis (Design Basic Earthquake)	X	X	X	N1		X		X	X	X	X	N2 X	X	N2 X	N2 "0" Thermal Creep. N1 Rarely Required.

X = Required

N = Required, See Note

*As of 2/79

**Generally Negligible Thermal Creep; Damage Fraction Generally Only Evaluated When There Is Significant Low Cycle Fatigue representative of commercial plans. One might typically expect only minor design changes to meet stress limits. Listed below are typical loadings and classification of stresses seen for FTR core components at steady-state operation.

TYPICAL LOADINGS AND STRESS CLASSIFICATIONS FOR FTR CORE COMPONENTS STEADY-STATE OPERATION [4]

Stress Origin	Stress Classification
Internal Pressure	P_m
	P_b
Duct Bowing[5,6]	Q_b, P_b
Gradient[5] Through Wall	Q_b
Weight	P_m
Restrained Axial Expansion[5,6]	Q_m, P_m
Interaction With Adjacent Component[5,6]	Q,P
Discontinuity	Q
Stress Consentrations	Peak

Note the dual classification, i.e., primary (P) and secondary (Q), for loading which might normally be considered as secondary for these deformation controlled conditions. This is a unique characteristic associated with the fast neutron environment related to irradiation creep and swelling. Here the behavior of in-core components is differentiated from those of other environments. This has a direct bearing on the classification of stresses used in the evaluation. In many instances, irradiation induced swelling produces a driving potential which is, in many respects, analagous to elastic follow-up. Hence, some stresses generally classified as secondary should be considered as primary. This is illustrated in Figure 1 for the case of duct tube bending stress, showing classification used at HEDL. As shown, classification is primarily dependent on in-core residence time, and the ratio of the swelling to creep strain rates at equilibrium conditions. This ratio is primarily a function of material and the in-core environment (temperature, and fast neutron flux).

---

[4] P denotes Primary Stress.
Q denotes Secondary Stress.
Subscripts m and b denote membrane and bending stresses, respectively.

[5] These are due to temperature and fast neutron flux gradients which give rise to thermal, irradiation creep and swelling strain gradients.

[6] Q if ΔT only.
Q if irradiation creep dominated.
P if irradiation swelling dominated.

Similar cases can be shown for restrained axial expansion, though very rarely are these expected to act as primary. For the case of a temperature and/or flux gradient through the wall, there generally exists a smooth continuum of flux and temperature so that continuous stresses caused by internal constraints are not transferred to other regions of the structure, and hence, the stresses act as true secondary stresses.

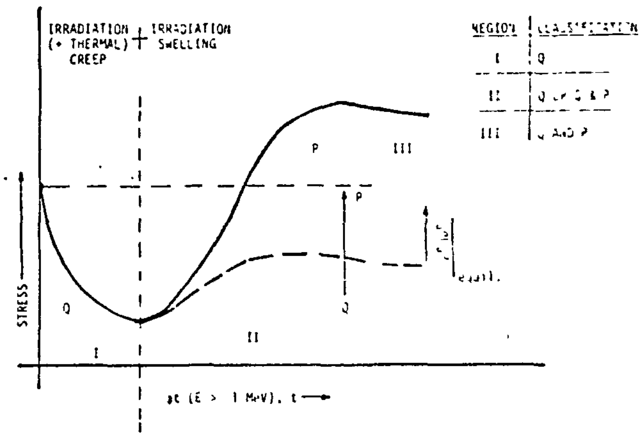

Fig. 1 Classification of Bending Stress Used at HEDL Due to Duct Bowing. Temperature and Neutron Flux Gradient Across Flats.

EVALUATION FOR OFF-NORMAL EVENTS

In the course of evaluating design transients, analytical effort in many cases requires the calculation of inelastic strains where elastically calculated stresses exceeded the allowable limits. In most cases, SIAM were used to calculate strains. Typically for FTR components evaluation, the following can be said:

1. Steady temperatures are low enough or hold times at high temperatures are short enough the thermal creep strains are negligible (in general there is a minimum amount of primary creep).

2. For thermal transients (e.g., scram, loss of electrical power) primary stress intensities were found to be within allowable limits.

3. For thermal transients when primary + secondary stress intensities exceeded the allowable limits, simplified inelastic analysis methods (SIAM) were used to determine strains and the character (i.e., cyclic plasticity or ratchetting).

4. Inelastic analyses were only required for Design Base Earthquate (DBE) seismic events when primary stress intensity limits were exceeded or inelastic deformation needed to be calculated.

5. Inelastic analysis, using equivalent one-dimensional cylinder analysis has been found to be an effective, conservative method for core component analysis.

DETERMINATION OF CRITICAL CROSS SECTION FOR EVALUATION

The FBR environment give rise to a unique situation in the determination of the cross section most subject to failure. Here, one must consider both ductile and brittle modes of failure in the evaluation. In general, for the overall assembly, the critical sections were found to be at welds. This is primarily due to the annealed state of the material (as opposed to the cold worked ducts) and stress/strains concentrations (note that work hardening during operation is not taken into account). Minor redesign of the weld in question is usually sufficient to overcome any potential problems.

Except for welds, the selection of critical cross section for evaluation is not always obvious. For most standard component ducts (channel), the highest stressed location is adjacent to the ACLP (adjacent to top load pad for some FTR head hung components). This, however, is not necessarily the most critical section because irradiated material properties, primarily loss of ductility, effectively reduce the strength of other sections. Here the critical section is usually between the top of the core and the Above Core Load Pad (ACLP) when considering ductile failure mode, and near the top of the core when brittle modes of failure are evaluated.

The effect of temperature and irradiation environment on material tensile properties is illustrated in Figures 2 and 3. This behavior can sometimes exhibit drastic changes over the assembly length and should be looked at in detail so that critical cross sections are not overlooked. Note that minimum properties, irrespective of time, are used in evaluation and could add considerable conservatism in evaluation. Figure 2 illustrates the combined effect of the neutron flux and temperature profiles on yield strength of 20% CW 316 SS. Here S_y increases with irradiation at temperatures below, and decreases with irradiation for temperatures higher than approximately 450°C (842°F). Component evaluation of the type shown graphically in Figures 4 and 5 overcomes the chance of missing this critical section.

EXPERIENCE IN APPLICATION OF CRITERIA LIMITS
(See Appendix for Design Rules)

Primary Membrane Stress Intensities

These stresses P_m are typically low in FTR core component and, consequently, result in high design margins.

Primary Membrane and Bending Stress Intensity

For steady-state normal operation and most design transients, stress intensities were within allowable limits (in some cases these stress limits were exceeded for the Design Basis Earthquakes).

Primary + Secondary Stress Intensities

In evaluating design transients, many cases exceeded the allowable stress limits. A dual evaluation of "low" and "high" ductility failure modes is generally required for each component. No generalization can be made as to which is most limiting. In most cases, simplified inelastic analysis methods could be used to evaluate strains and damage limits.

Evaluation using the modified shakedown diagram, to assure that the stresses are elastic or shakedown to elastic stresses in the first cycle (see Appendix Figure A1), generally reduces the overly conservative approach taken in earlier evaluation methods used based on Code Case 1592.

Maximum Principal Stress (R.A.<10%)

This is generally not as limiting as the $P_L + P_b + Q$ limit for minimum uniform evaluation (ε_u)<1% for normal design configuration. This limit was found to be limiting in the case where there are stress concentration such as holes in duct for instrument attachments.

Strain Limits (optional or required if stress limits are exceeded)

As menitoned earlier, thermal creep strains for FTR components are generally negligible. In the FTR the maximum steady-state structural component temperature rarely exceeds 550°C (1020°F) for a 360°C (680°F) initial core inlet temperature. Because of this, the maximum membrane strain limit is usually based on plastic strain only. The maximum plastic membrane strain is generally due to the accumulation of strain due to ratchetting (generally seismic induced), and the maximum of the "non-accumulative" strains (generally due to the most severe thermal transient).

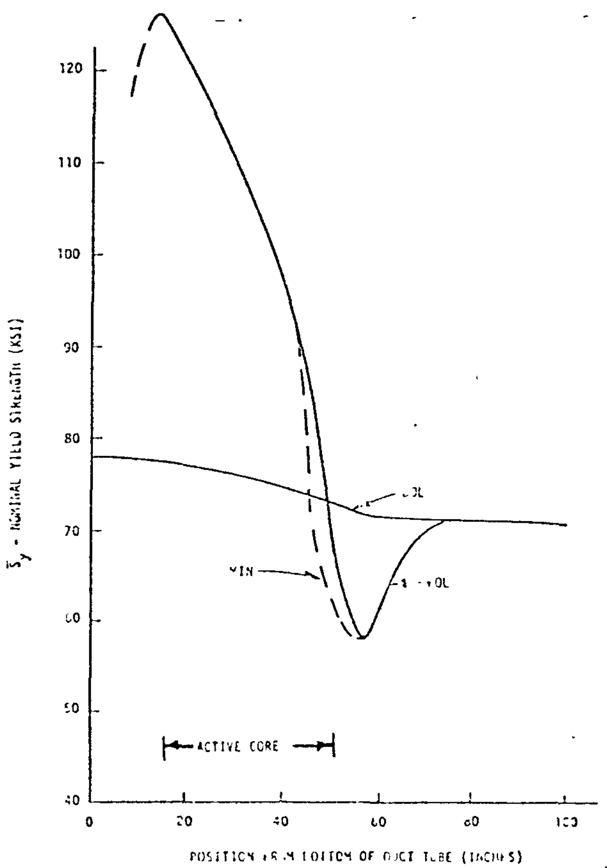

Fig 2 Nominal Yield Strength - Driver Fuel Assembly Duct (Assumes Hottest and Highest Irradiated Assembly; Conversion Factors: (cm.)=(.3937)(inches);(MP_a)=(6.894)(KSI)

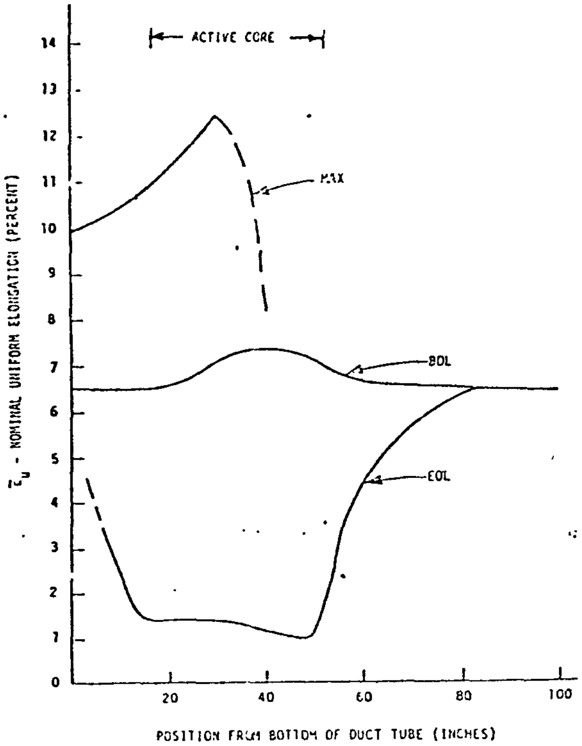

Fig. 3 Nominal Uniform Elongation - Driver Fuel Assembly Duct (Assumes Hottest and Highest Irradiated Assembly). Conversion Factors: (cm.)=(.3937) (inches)

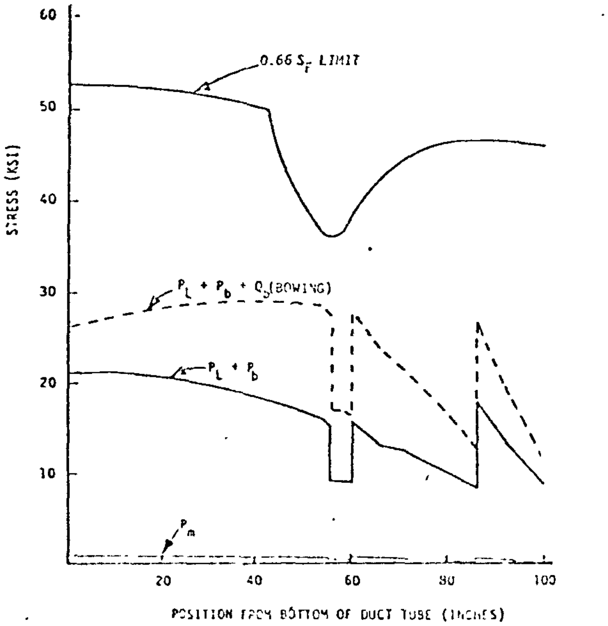

Fig. 4 Driver Fuel Assembly Duct - Primary Membrane + Bending Stress Intensity and Limit for Design Basis Earthquake. Conversion Factors: (cm.)=(.3937) (inches);(MPa)=(6.894)(KSI)

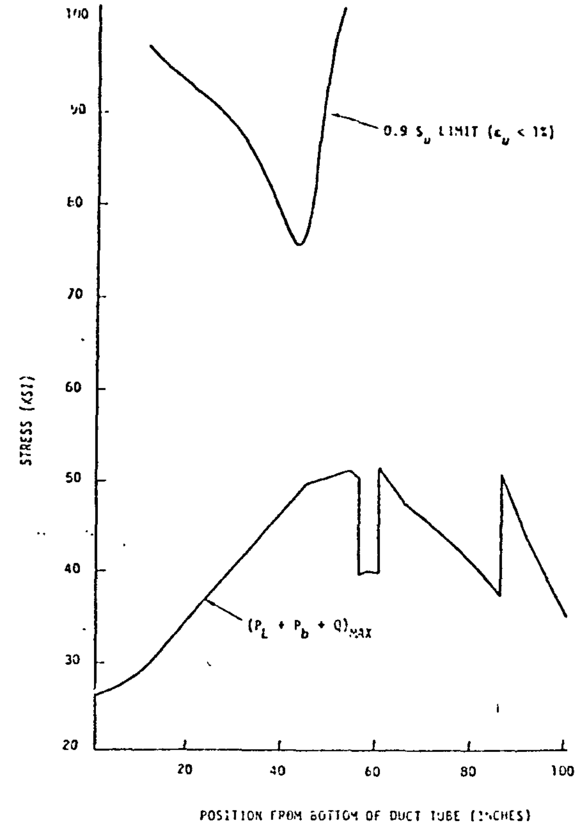

Fig. 5 Driver Fuel Assembly Duct - Primary + Secondary Stress Intensity Limits for Section Where ε_u<1% Design Basis Earthquake. Conversion Factors: (cm.)=(.3937)(inches);(MPa)=(6.894)(KSI)

Some FTR components were evaluated for a higher inlet temperature of 442°C (792°F) for rated core conditions. For this case, thermal creep strains are significant and contributed significantly to both strain limits and creep-fatigue damage limits.

Local peak strain limits are generally not much more limiting than the membrane strain limit in normal FTR components. These do, however, become paramount at regions of stress concentration such as welds and instrument attachments in FTR test assemblies.

Damage Limits (Required)

Since thermal creep is negligible for current FTR operating conditions, damage is primarily due to low cycle fatigue.

Deformation Limits for Functional Adequacy

These limits are currently more life limiting for FTR core components than any of the stress or strain limits imposed. This may change with the use of advanced alloy materials. Duct dilation, duct axial swelling, and duct bow limits are generally evaluated with elastic irradiation creep and swelling analysis, and are usually performed prior to any stress/stain evaluation.

Fracture (Unstable Crack Propogation) and Buckling Limits

These limits are optional and are applied on an "as needed basis".

SUPPORTING IRRADIATED MATERIAL DATA

This is perhaps the most disturbing area a user will face when he applies RDT F9-7. Our experience over the past few years regarding the availability of irradiated material data has been frustrating at the least. Until just recently, the Nuclear System's Material Handbook, the official source of this information, contained no irradiated material properties. Material properties had to be obtained through other means, mostly through communication with HEDL metallurgist or from recent publications. In many cases, the information available is non-user oriented, i.e., had to be further processed to be of direct use. Some of the properties used in evaluation where generated with the guidance of RDT F9-8 and other sources. Figures 6 and 7 shows examples of this for irradiated design fatigue curves.

The user needs minimum material properties when applying F9-7. However, the limited data, at best, represents nominal values. Here the user must reduce the values to account for the uncertainties. Hopefully, this situation will change when expanded data base becomes available.[7] Typical conservative reduction factors applied to "nominal" values are

80-90% on S_y, S_u

50-80% on ϵ_u, R.A., stress to rupture data.

INSUFFICIENT GUIDANCE

There are certain areas of RDT F9-8 where the guidance may not explicitly meet the analysts need or may be in question. These are:

1) *Classification of stress for cases where irradiation swelling can act as a driving potential.*

The current guidance is to classify the stresses as secondary unless there is a large amount of elastic follow-up involved in which case the stresses should be treated as primary. This is somewhat ambiguous, since irradiation swelling, in many cases, can act like elastic follow-up. It is our experience that stresses should be treated as discussed earlier in this paper.

2) *Some of the uses of simplified inelastic analysis, based on analysis by Bree [6] and O'Donnel and Porowski [7] of thin walled cylinder subject to axisymmetric loading, may not be appropriate or non-conservative when applied to certain loading experience by FBR core components. These loadings are:*

a) Short term cyclic primary stress such as seismic disturbance with constant secondary stress. There is no guidance given for this type of loading. Application of SIAM may not be appropriate here.

b) Non-axisymmetric beam bending stresses can, under the Guidelines, be classified as secondary when applying SIAM. This classification, in many cases, have been shown to be non-conservative [5]. Classification as a primary stress, however, has been shown to give ultra-conservative results. Recent work at HEDL suggests that factors applied to stress classified as secondary will give reasonable results.

Verification of the above a & b would appear to be worthwhile.

OTHER EXPERIENCE

1. The use of varying design margins to account for component reliability (classification) and event classification is useful in that it gives the analyst flexibility in treating his unique system.

2. There are no design margins associated with extremely unlikely faulted conditions (1976 draft only). One method of treating this condition is to use design margins assigned to unlikely faulted (emergency) conditions. When these limits are exceeded, additional analyses are performed as needed (e.g, limit analysis) to ensure safe reactor shutdown, maintenance of a coolable geometry, and to ensure that radioactive releases are within acceptable limits.

CONCLUSION

Trial applications at HEDL have shown RDT F9-7 and F9-8 to be a valuable tool in the structural evaluation of FBR core components. The criteria/guideline is generally comprehensive in addressing the unique structural problem of the FBR core environment. Our experience suggests that core component evaluation could be accomplished with a minimum amount of detailed inelastic analysis when applying the criteria/guidelines. It is our experience that deformations related to functional adequacy be evaluated before the detail stress and strain analysis is performed. These deformation limits are generally the life limiting mech-

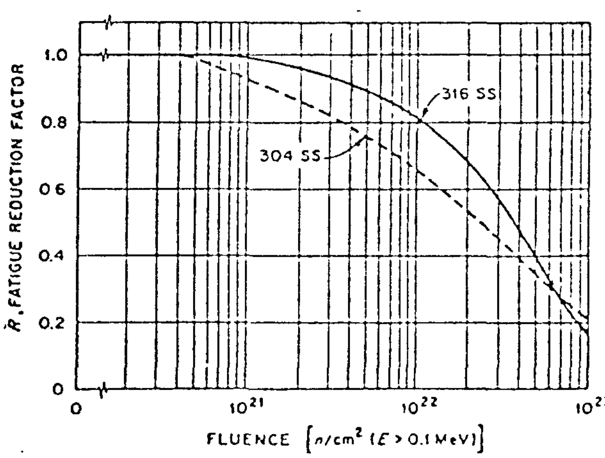

Fig. 6 Fatigue Reduction Factors for Annealed and Irradiated types 304 and 316 Stainless Steel. From Ref. [5].

[7] The Material Science Community at HEDL expects that these reductions used are conservative.

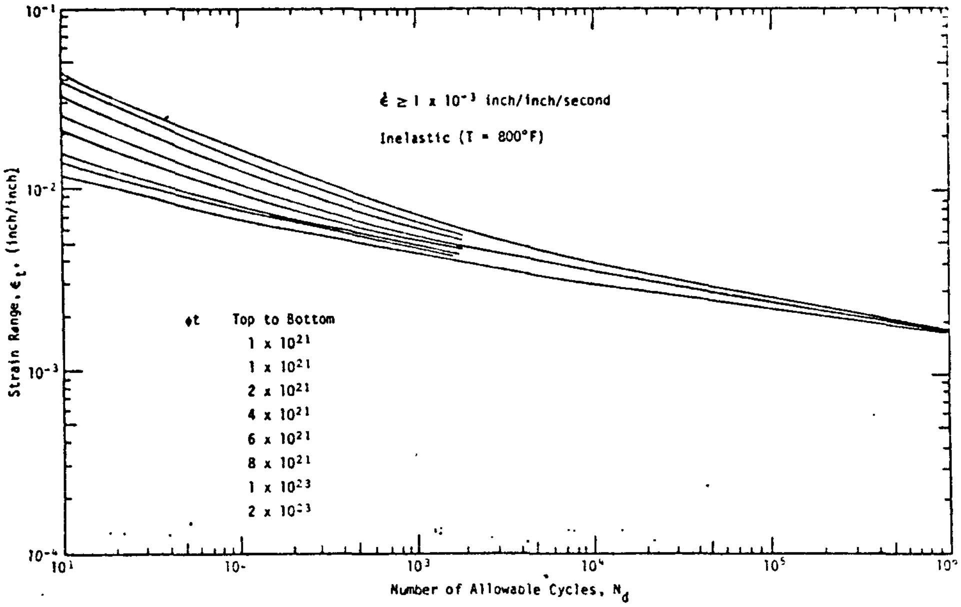

Fig. 7 Inelastic Design Fatigue Strain Range ϵ_t, for Irradiated Type 316 SS (T = 800°F).

anism for present day component design and materials. The current lack of irradiated material properties greatly "handicaps" both the usability and the effectiveness of the proposed criteria, but this will be overcome in time as more in-reactor data becomes available. Several areas of guidance given are questionable and need further verification either through a program of detailed inelastic analysis or in-reactor experiments. The applicability of the criteria as is to materials which behave grossly different than 304 and 316 stainless steel is questionable.

Service experience on FTR core components will be available soon and can provide the needed feedback which will aid in the confirmation of the validity of the criteria/guidelines.

APPENDIX - BASIC DESIGN RULE SUMMARY (1976 DRAFT) [1]

Primary Stress Intensity Limits

The following limits on the primary general membrane stress intensity, P_m, and combined primary (general or local) membrane plus bending stress intensity $(P_L + P_B)$ shall be satisfied:

$$P_m \leq \alpha S_F$$

$$(P_L + P_B) \leq \alpha K\ S_F$$

where: K = 1 if the ratio (S_y/S_u) is greater than 0.6 for the material at the time under consideration.

$K = 1 + (K_t - 1)\ (1 - P_m/S_A)$ if the ratio (S_y/S_u) is lower than 0.6 for the material at the time under consideration, where $S_A = \alpha S_F$.

α = the design margin for the class of component under consideration specified as a function of event condition.

S_F = A time-independent stress intensity limit based on material behavior in short-term loading. S_F shall be taken as the lesser of 1.66 S_y and S_u at the given material temperature for the exposure at the time under consideration.

K_t = A quantity dependent on the geometry of the component used to account for the stress redistribution from the elastically calculated stresses to that in the plastic condition.

Primary Plus Secondary Stress Inensity Limits

The following limits on the combined primary (general or local) membrane plus bending plus secondary stress intensity range shall be satisfied.

When ϵ_u<1%, the combination of $(P_L + P_B/K_t)/S_y$ and $(Q_R)_{max}/S_y$ stress intensities shall be restricted to within 80% of the shakedown boundary based on an elastic-perfectly plastic model of material behavior with the flow stress taken to be S_y. For axisymmetric structures away from discontinuities, the modified shakedown diagram shown in Figure A-1 can be used. Here (Q_R) max is the maximum range of secondary stress intensity.

If the above is not satisfied, a simplified inelastic analysis may be conducted based on the guidelines given in F9-8. The resulting strain contributions from thermal creep and time independent plastic deformation shall then be used to satisfy the strain limits

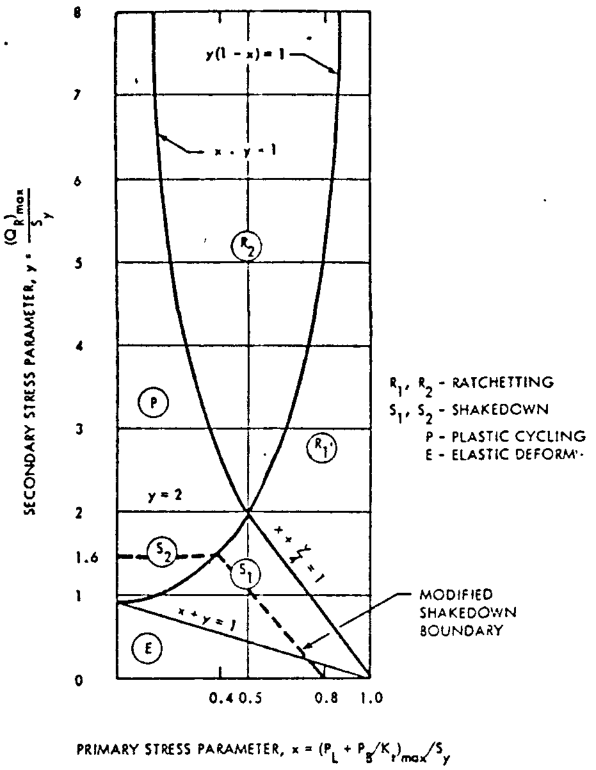

Fig. A-1 Modified Shakedown Diagram used for Primary + Secondary Stress Evaluation. Also Shown are Stress Regimes Based on Bree's Analysis. [6]

When $\varepsilon_u < 1\%$ $(P_L + P_B + Q)_{max}$ < 0.75 S_u for normal and anticipated faulted conditions

< 0.9 S_u for unlikely faulted conditions

Where ε_u = Minimum uniform elongation in a tension test, specified as a function of temperature and fluence.

Peak Stress Limit

The peak principal tensile stress including the effect of stress concentrations shall be limited to be below the minimum ultimate strength for material with a reduction in area of less than 10%, as measured in a tension test.

Deformation Limits for Functional Adequacy

Deformation limits for functional adequacy shall meet the requirements of the Equipment Specification. Distortion in service due to elastic, plastic, thermal creep, irradiation creep and swelling and ratchetting shall be limited so as not to impair functional adequacy of the assembly.

Strain Limits (Required if Stress Limits are not Satisfied)

Membrane Strain Limit. The principal membrane plastic and thermal creep strain increments shall be summed on a normalized basis over the lifetime of the component and limited to:

$$\Sigma\left(\frac{\Delta\varepsilon^P}{\varepsilon_L^P}\right) + \Sigma\left(\frac{\Delta\varepsilon^c}{\varepsilon_L^c}\right) < \beta$$

Where: $\Delta\varepsilon^P$, $\Delta\varepsilon^c$ are principal membrane plastic and thermal creep strain increment averaged across the section, respectively.

ε_L^P : A membrane plastic strain limit, specified as a function of temperature and fluence, which shall be taken as $\varepsilon_u/2$, but not to exceed 0.02.

ε_L^c : A thermal creep strain limit, specified as a function of temperature, fluence, and stress state. The unaxial value of this limit, $(\varepsilon_L^c)_0$ shall be taken as either:

(i) the minimum creep strain at failure when tertiary creep is absent,

or (ii) the minimum creep strain at a point determined by projecting the portion of the creep strain versus time curve at the secondary creep strain rate to the rupture time.

ε_L^c shall be determined from (ε_L^c) as a function of stress state dependent on the value m denotes the exponent in the creep relation.

Where: $\dot{\varepsilon}^c = A\sigma^m$

β: A design margin applied to strains which is a function of component and event classification.

Local Strain Limit. The total accumulated local plastic strain at any point in the structure shall be limited according to:

$$\varepsilon_{max}^P \leq \left(\frac{0.3}{T.F.}\right)(\varepsilon_f)$$

Where: ε_{max}^P = the maximum principal plastic strain.

(ε_f) = the true strain at fracture in a tension test as a function of fluence and temperature.

T.F. = triaxiality factor.

Damage Limits (Required)

The following total damage limit shall be satisfied:

$$\text{Total damage, } D = D^c + D^f < \beta$$

where: D^c is the thermal creep damage

D^f is the fatigue damage

β is the design margin

Thermal Creep Damage D^c

The damage due to thermal creep deformation D^c shall be computed as follows:

$$D^c = \Sigma \frac{\Delta t}{t_L^c} \quad \text{summed over the lifetime of the component}$$

Where Δt = a small time increment over which the applied and environmental variables such as temperature, stress and fluence remain relatively constant.

t_L^c = the design allowable time to rupture in thermal creep based on the maximum principal tensile stress from primary and secondary stress sources for the operating temperature and fluence at the time under consideration.

Fatigue Damage, D^f

The damage to varying cycles of strain shall be computed from the relation:

$$D^f = \sum_j (n_j/N_j)$$

Where: D^f = accumulated fatigue damage

n_j = number of cycles at loading condition "j", corresponding to $(\overline{\Delta\varepsilon})_j$.

$(\overline{\Delta\varepsilon})_j$ = equivalent strain range at loading condition "j".

Other Rules

Protections against unstable crack propogation and buckling and instability limits as given in Reference [1] are not present here since they do not directly support the body of the paper.

ACKNOWLEDGEMENT

Special recognition should be given to Dr. L. J. Julyk of HEDL for his considerable contribution in the application and evaluation of RDT F9-7, F9-8, and F9-9.

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