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TITLE: DEVELOPMENT OF STRUCTURAL DESIGN CRITERIA FOR HIGHLY IRRADIATED CORE COMPONENTS

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GENERAL ELECTRIC

FAST BREEDER REACTOR DEPARTMENT

SUNNYVALE, CALIFORNIA

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"DEVELOPMENT OF STRUCTURAL DESIGN CRITERIA FOR HIGHLY IRRADIATED CORE COMPONENTS"

D.V. Nelson, M.M. Abo-El-Ata, J.D. Stephen, R.G. Sim

INVITED PAPER FOR 1978 ASME PRESSURE VESSEL & PIPING CONFERENCE

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ABSTRACT

This paper reviews development to date in the U.S. of structural design criteria (Criteria) for fast breeder reactor core components. Since these components operate in an elevated temperature, fast neutron environment, Criteria for them are novel in that they must take into account the significant time-dependent changes in structural material behavior resulting from irradiation. Development of the Criteria is being guided by a national working group sponsored by the Division of Reactor Research and Technology of the Department of Energy. The program objective is to establish a national consensus on design rules, together with supporting rationale and analysis guidelines, with the goal of strengthening design and expediting licensing. After further validation, the Criteria will become an RDT Standard.

Core loading mechanisms, modes of material deformation and damage, and the effect of the operating environment on core structural material behavior are summarized as background to consideration of the Criteria design rules.

Next, the basic approach used in developing the design rules is described, which consists of identifying all known and probable failure mechanisms for core components and preparing rules to provide protection against them. The relation between these rules and the various types of structural analyses conducted (e.g., elastic or elastic-irradiation creep-swelling or detailed inelastic analysis) is also presented.

Selected examples which illustrate in detail how the new design rules take into account the effects of irradiation are given. The examples include: (1) stress intensity limits, based upon a re-evaluation of limit load and shakedown concepts in view of the reduction in ductility with irradiation, (2) plastic and thermal creep strain fraction summation rules for membrane, bending and peak strains, which utilize strain limits that vary to account for time-dependent changes in material ductility, and (3) a combined thermal creep and fatigue damage rule, which considers changes in time-to-rupture and cycles-to-failure due to irradiation. Special considerations in structural analysis resulting from irradiation effects are also briefly described. Finally, there is a brief discussion of what may be required in the way of further test data, analysis and operating experience to validate the design rules and their associated design margins.

NOMENCLATURE

e_u	uniform elongation in a tension test
K_t	bending shape factor
n	number of applied fatigue cycles
N_d	design allowable fatigue life
P	primary stress intensity
P_B	primary bending stress intensity
P_m	primary membrane stress intensity
Q	secondary stress intensity
Q_R	secondary stress intensity range
S_F	stress limit
S_u	ultimate tensile strength
S_y	yield strength (monotonic)
TF	triaxiality factor
t_d	design allowable time-to-rupture
Δt	time increment
α,β,γ	design margins
$\delta\varepsilon$	strain increment
$\Delta\varepsilon$	fatigue strain range
$\delta\varepsilon_m^c$	thermal creep membrane strain increment
$\delta\varepsilon_m^p$	plastic membrane strain increment
$\delta\varepsilon_t^p$	plastic total strain increment
$\Delta\varepsilon$	fatigue strain range
ε_f	true strain at fracture in a tension test
ε_L^c	thermal creep tensile instability strain limit
ε_L^p	plastic tensile instability strain limit
$\sigma_1,\sigma_2,\sigma_3$	principal stresses

INTRODUCTION

The development of design technology is a major goal of the breeder reactor program sponsored by the U.S. Department of Energy, Division of Reactor Research and Technology. Work is in progress on development of consistent sets of analysis methods, material properties and criteria for use in core design. This paper describes the development of nationally recognized guidelines on structural design criteria for the replaceable components of the reactor core. These components are subjected to high energy neutron irradiation at elevated temperature, which causes significant time-dependent changes in structural material behavior. New criteria are needed to extend elevated temperature design-by-analysis to account for these changes.

A major objective of establishing recognized standards for design criteria is to expedite licensing of the reactors by having consensus on probable failure mechanisms, on rules to account for the damage due to these mechanisms, and on design margins to provide the appropriate degree of conservatism for safe operation. Confidence in the structural integrity and functional adequacy of designs is also increased by the use of standards which have received widespread review by many experts in the field. A further objective is to provide a forum for evaluation of the rapidly evolving understanding of core behavior relative to the requirements for design, and to provide a focus for specifying further development and testing needs.

A National Working Group on Core Component Structural Design Guidelines was formed in March 1975 to contribute to these objectives. The Working Group includes representatives from government, national laboratories and industry, as shown in Table 1. In June 1976, a draft "RDT Design Guideline/Criteria for LMFBR Core Components" was completed and approved by the Working Group for use in trial applications, which are in progress at the organizations listed in Table 1. The objective of the trial applications is a thorough review to ensure that the quantities in the criteria can be calculated correctly by the analysis methods and that the criteria are consistent with material properties correlations and uncertainties, and, if not, whether the problem is with the criteria or the correlations. The trial applications are also intended to ensure that the criteria adequately account for the damage mechanisms throughout the range of design, environmental and loading conditions expected in service and at design basis conditions for all components. The results of the trial applications, and of recent test data and methods development, are being continuously evaluated by the Working Group and integrated into updates of the criteria as appropriate. The criteria, although still in draft form, have already had a positive influence in several design areas, including components of the Clinch River Breeder Reactor Plant (CRBRP).

Table 1
National Working Group on Structural Design Guidelines for Core Components of the Breeder Reactor

Argonne National Laboratory
Atomics International
Combustion Engineering
Department of Energy/Division of Reactor Research and Technology
General Atomics
General Electric/Fast Breeder Reactor Department (Chairman)
Hanford Engineering Development Laboratory
Oak Ridge National Laboratory
Westinghouse/Advanced Reactors Division

The program is now moving into the stage of drafting RDT Standards. Three standards have been designated as shown in Table 2. RDT F9-7 will contain the structural design rules and margins for meeting component structural integrity requirements. RDT F9-8 will provide guidance to establish internally consistent procedures for the use of methods, criteria and material properties in design analysis. RDT F9-9 will provide the background and technical basis for the criteria.

Table 2
RDT Standards for Structural Design Guidelines for Breeder Reactor Core Components

F9-7	Structural Design Criteria
F9-8	Guidelines for Analysis
F9-9	Rationale

The structural criteria program influences and is influenced by a number of interfaces. Working relationships are maintained with testing programs. Test data are evaluated for their effect on criteria, and special tests are recommended to support validation of the criteria. Assistance is provided to design projects in the proper application of the criteria, and feedback from the applications is included in refinement of the criteria. Contact is maintained with other high temperature structural criteria programs and activities supporting RDT Standards F9-4, -5, -6 and the ASME Elevated Temperature Code Case 1592. Finally, strong interfaces are maintained with other national programs on reference analysis methods, properties correlations, and other types of criteria (thermal, nuclear, safety, etc.) to ensure that the goal of self-consistent, nationally recognized standards for core design is achieved.

BRIEF DESCRIPTION OF BREEDER CORE, OPERATING ENVIRONMENT AND LOADING MECHANISMS

Core Assemblies

Fast breeder reactor cores, such as that designed for CRBRP, are comprised of a number of different types of removeable assemblies (i.e., fuel, control, blanket and radial shielding) arranged in a hexagonal pattern within a core barrel and resting on a core support plate. See Figure 1. In essence, the assemblies consist of ducts of hexagonal cross section, each containing a bundle of pins, which are thin-walled cylinders (cladding) enclosing pellets of fuel, neutron absorber material, etc., depending on the particular type of assembly. Most of the assembly components are fabricated from 20% Cold Worked (CW) Type 316 Stainless Steel (SS), the current reference core structural material for the U.S. breeder program. The assemblies are separated from each other by load pads, while the pins they enclose are separated either by wire wrapped helically around each pin or by grids of thin metal strips.

Operating Environment

Both the high energy neutron irradiation and elevated temperatures which core components experience vary considerably throughout the core and even within individual pin bundles, thus producing significant temperature and neutron flux gradients. The working fluid, liquid sodium, enters each duct through inlet ports at the core support plate to provide cooling as it is pumped upwards. Typical bulk inlet and outlet sodium temperatures are about 350 and 550 C, respectively,

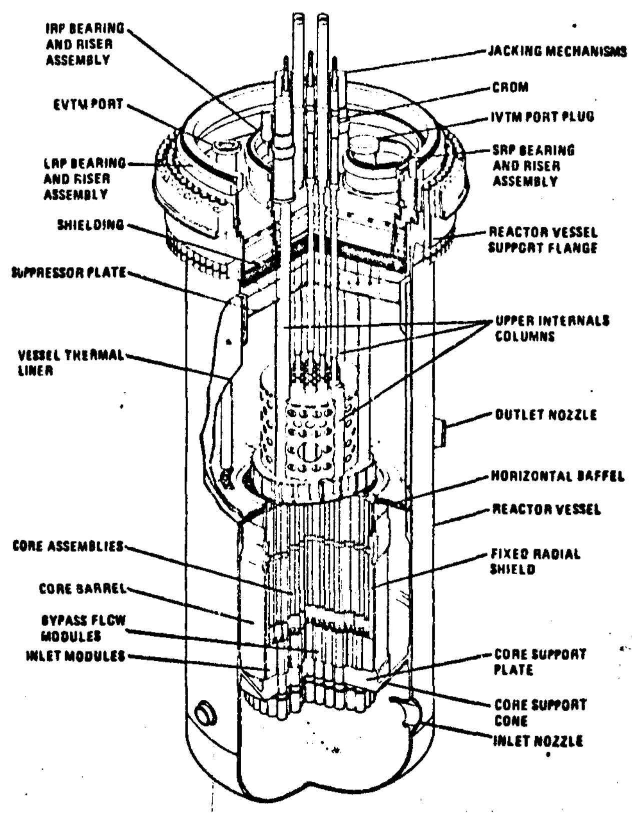

Fig. 1a Schematic View of CRBRP Reactor Assembly

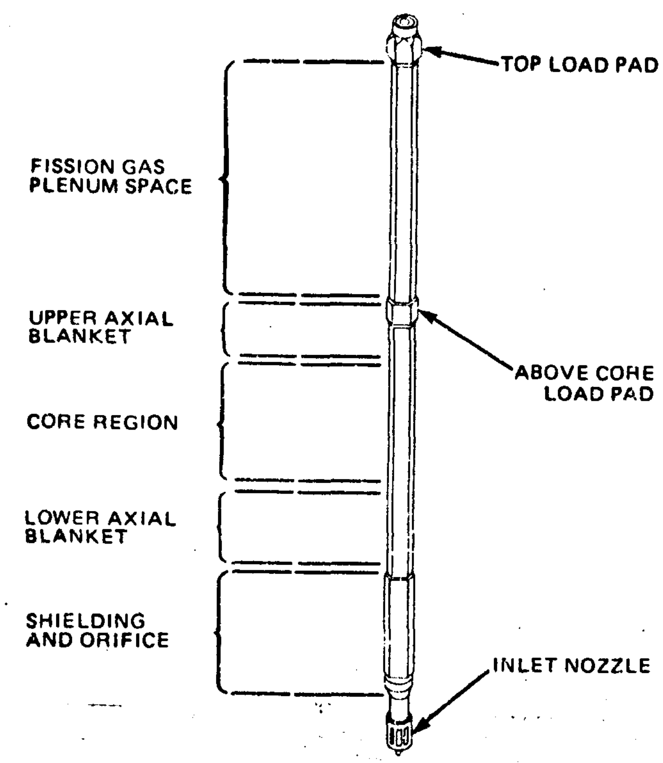

Fig. 1b Schematic Diagram of a Fuel Assembly

although individual components, such as certain fuel pins, may operate at temperatures considerably higher than these and also experience localized "hot spots." Typical design fluences (neutron flux integrated over exposure time) are on the order of 3×10^{23} neutrons/cm^2, for fast neutrons (n) having energies (E) greater than 0.1 Megaelectronvolts (MeV). This corresponds to lives of one to six years, depending on the particular component. The operating environment[1] produces a number of important effects on structural material behavior, as summarized in Table 3. Of these, the effects of irradiation (in particular, the reduction in material ductility) are of greatest concern in the development of structural design criteria and are thus briefly reviewed in a subsequent part of this paper.

[1]The basic features of gas-cooled fast breeder reactor core designs are similar in many respects to liquid metal designs, and the structural criteria described here are equally applicable to both, with the exception of differences in how the effects of coolant on material behavior are treated in design analysis, etc.

Table 3
Environmental Effects on Material Behavior Considered in Development of Structural Design Criteria for Fast Breeder Reactor Core Components

Irradiation
Coolant Corrosion and Interstitial Depletion
Fretting and Wear
Thermal Softening
Cyclic Hardening and Softening
Pin Internal Chemical Interactions
(e.g., fission product attack of cladding)

In addition to the usual deformation modes considered in elevated temperature operation (i.e., elastic, plastic and thermal creep deformation), fast breeder reactor core components are subjected to swelling, a phenomenon in which material dilates due to fast flux irradiation, and to irradiation-induced creep. Since these modes of deformation are functions of temperature and fluence, the variations in neutron flux and temperature in the core cause, for example, components to distort and be loaded by non-uniform swelling. Further consideration of swelling and irradiation creep and their impact on core design criteria and analysis will also be given later.

Loading Mechanisms

In addition to the previously noted environmental effects and deformation modes, development of the criteria has recognized the variety of structural loading mechanisms to which core components are subjected, as summarized in Table 4.

Table 4
Core Component Loading Mechanisms
Considered in Development of Criteria

Mechanism	Examples
Fluid Pressure	• Coolant Pressure on Ducts • Fission Gas Pressure on Fuel Pin Cladding
Temperature Differentials	• Temperature Gradients Across Ducts, Through Cladding • Differential Expansion
Swelling Differentials	• Interaction Loads Between Pins and Duct • Pellet-Cladding Interaction • Duct-to-Duct Interaction
Dynamic	• Seismic Loading • Impact Loading Due to Reactor Scram
Hydrodynamic	• Flow Induced Vibration

EFFECTS OF IRRADIATION ON STRUCTURAL MATERIALS BEHAVIOR

This section provides background information on the effects of fast neutron irradiation on the behavior of structural materials at elevated temperature. It is intended to highlight the effects of the unique environment to which fast breeder reactor core components are exposed and therefore the need for design rules and analysis methods that differ from those used in general elevated temperature design (e.g., ASME Code Case 1592 and RDT F9-4, F9-5 and F9-6). A more detailed review of irradiation effects on structural materials has recently been published by one of the authors [1]. The following discussion applies to austenitic stainless steels, in general, and to 20% CW 316 SS, in particular.

The effects of irradiation on material behavior can be considered in three categories: (i) mechanical properties, (ii) swelling and (iii) irradiation creep.

Irradiation Effects on Mechanical Properties

The parameter used most commonly to correlate irradiation effects is neutron fluence. Irradiation of structural materials in the temperature and fluence ranges of interest in the fast breeder reactor core (350-700 C, 3 x 10^{23} n/cm^2) produces several changes in the microstructure of these materials, which in turn affects their mechanical properties. Neutron bombardment results in a disorder of an otherwise ordered crystal structure by creating interstitials, vacancies, dislocation loops, and, under certain conditions, voids. The consequences of these structural defects are, in general, similar to those of cold work in that they increase the strength and reduce the ductility of the material. While the increase in strength is beneficial, the decrease in ductility is detrimental by making normally ductile alloys more susceptible to fracture.

The effect of fluence on the yield strength of 20% CW Type 316 SS [2] is shown in Figure 2 for temperatures in the range of 370 to 590 C. The effect of irradiation hardening on yield strength is significant only at the lower temperatures (up to $\sim$ 500 C) due to matrix hardening associated with the formation of dislocation loops. The decrease in strength observed at temperatures from 550 to 650 C

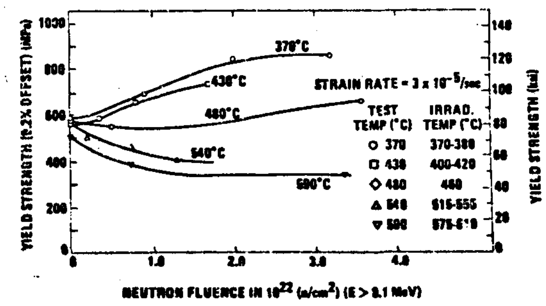

Fig. 2 Effect of Fluence on the Yield Strength of 20% Cold Worked Type 316 Stainless Steel Irradiated Between 370 and 610 C (Ref. 2).

is due to annealing of the cold-worked material. For temperatures above 700 C, fluence has insignificant strengthening effects due to rapid annealing of initial cold work and of the hardening produced by irradiation. The dependence of ultimate strength on fluence at various temperatures is similar to that of the yield strength. Also, as fluence increases, both of these strength properties tend to converge.

The effect of fluence and temperature on ductility [2] is illustrated in Figure 3 for 20% CW 316 SS. Total elongation decreases as the neutron dose increases for all test temperatures. At lower temperatures (e.g., 370 C), reduced ductility is associated with the matrix hardening accompanying the formation of dislocation loops. As temperature increases, ductility is still dominated by matrix hardening, but the density of voids and loops decreases, resulting in higher ductility. As the temperature approaches 540 C, the onset of helium embrittlement occurs and ductility begins to decrease again. Helium is produced by transmutation of the components of stainless steel under fast neutron irradiation. It is insoluble in stainless steel and tends to precipitate into bubbles that segregate at the grain boundaries if the temperature is high enough for its atoms to migrate. Consequently, fracture becomes predominantly intergranular and ductility drops because of helium embrittlement. Other measures of ductility, such as uniform elongation and reduction in area, follow the trend of total elongation relative to fluence and temperature. Stress-strain curves for irradiated 20% CW 316 SS also exhibit progressively reduced strain hardening capability with increasing fluence.

Thermal creep ductility (creep strain at rupture) depends on irradiation temperature and fluence in a manner similar to that for tensile ductility and primarily for the same reasons. Creep rupture life of pre-irradiated and austenitic stainless steels has been found [3] to be substantially lower than the unirradiated life. The deterioration of pre-irradiated rupture life also increases with fluence. In-reactor rupture life may, however, be better than that indicated by tests of pre-irradiated material. This is discussed further in the section on irradiation creep. Low-cycle, elevated temperature fatigue data for irradiated austenitic stainless steels

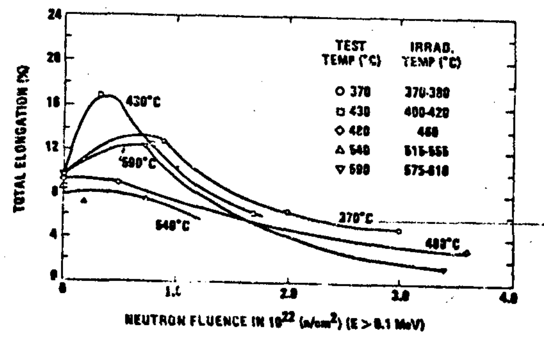

Fig. 3 Effect of Fluence on the Total Elongation of 20% Cold Worked Type 316 Stainless Steel Irradiated Between 370 and 610 C (Ref. 2).

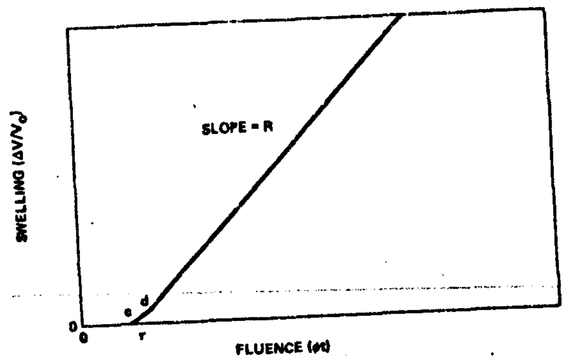

Fig. 4 Schematic of Swelling Dependence on Fluence

also show a reduction in life, the severity of which increases with fluence [4]. This is reasonable in view of the reduction in tensile ductility with irradiation. There are little data on high-cycle fatigue life; however, the effects of irradiation in this case are not expected to be as significant as for low-cycle fatigue.

The fracture toughness of austenitic stainless steel also decreases with fluence [5], which again might be expected from the effects of irradiation on tensile ductility.

Swelling

Swelling has been found to occur in stainless steels subjected to fast neutron irradiation due to the nucleation and growth of voids. These voids were observed by transmission electron microscopy and were found to vary in size from the smallest observable to greater than 1000 Å in diameter. Swelling of a given material depends on many variables. Among these are fluence, temperature, stress and the initial condition (e.g., cold work) of the material. Swelling is usually expressed as a change in volume per unit volume, $\Delta V/V_o$, which is generally obtained from immersion density measurements.

The dependence of the swelling of austenitic stainless steels on fluence is shown schematically in Figure 4. The swelling curve is characterized by three regions: (a) an incubation region, oc, where there is no measurable swelling; (b) a transition region, cd, where the swelling rate increases with fluence; and (c) a steady-state region, beyond point d, where the rate R remains constant with fluence. Experimental evidence shows that void swelling occurs in austenitic stainless steels only in the temperature range of 350 C to 700 C, which coincides with that in which core components operate. The dependence of swelling on irradiation temperature for several different fluences is shown in Figure 5 for annealed Type 316 SS [6]. It is important to note that the swelling reaches a maximum at about 520 C for annealed Type 316 SS, irrespective of the fluence level. The magnitude of this maximum, however, depends on the fluence level, as shown in Figure 5.

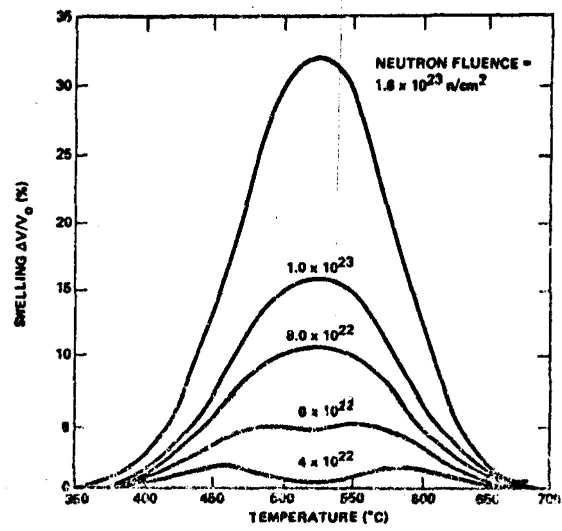

Fig. 5 Prediction of Typical Swelling Equation for Solution Annealed Typed 316 Stainless Steel (Ref. 6)

Finally, it is of interest to note that swelling causes a reduction in the modulus of elasticity [7], another time-dependent effect which complicates the structural analysis of core components.

Irradiation Creep

Irradiation creep is the name given to that part of the time-dependent deformation, under a given load and temperature condition, attributed to neutron irradiation. Time-dependent deformation in the absence of irradiation is, by contrast, called thermal creep. Irradiation creep is manifested in the form of time-dependent deformations at low temperatures where thermal creep is negligible or in the form of increased rates of deformation at temperatures where thermal creep is significant.

Most test data have shown irradiation creep to vary almost linearly with stress and fluence for a given temperature [1]. Very recent experiments, however, have revealed deviations from this linear stress relationship, and thus the exact nature of the relationship is still in a state of evolution.

Irradiation creep, like thermal creep, is due to stress-oriented motions of dislocations, vacancies or interstitials and thus would be expected to increase with higher temperatures. Figure 6 gives the results of an irradiation creep experiment performed on several pressurized tube segments made of 20% CW 316 SS and irradiated at various temperatures. The internal pressure was kept constant to give a hoop stress of 70 MPa. The fluence level reached 3×10^{22} n/cm^2. As can be seen from this figure, the irradiation creep strain more than doubles as the irradiation temperature is increased from 380 C to 580 C.

The relative importance of irradiation creep to thermal creep as components of the overall in-reactor deformation is also temperature dependent. Thermal creep is expected to dominate in-reactor deformation at high temperatures, while irradiation creep will dominate at lower temperatures.

As mentioned previously, the creep rupture lives of pre-irradiated samples are substantially lower than those of unirradiated samples tested under the same stresses and temperatures. Post-irradiation tests cannot portray the effects of irradiation creep on rupture life since these tests are conducted outside the reactor. In order to estimate the in-pile rupture life from available unirradiated creep rupture data, it is important to determine the role of irradiation creep. Irradiation creep occurs by dislocation climb, a mechanism which would tend to relieve concentrated stresses at grain boundaries and triple points. This, in turn, may delay intergranular crack nucleation and prolong rupture life.

DEVELOPMENT OF STRUCTURAL DESIGN CRITERIA

The basic approach used in developing the criteria has been to identify failure modes thought to be potentially significant for core components and to devise rules to provide protection against them. Different sets of rules have been developed for use with different types of structural analysis. This approach follows, in general, that used in producing the elevated temperature criteria in ASME Code Case 1592. The core components criteria, as noted before, furnish an extension of elevated temperature criteria by taking into account the time-dependent changes in material behavior briefly reviewed in the previous section. Examples of how this is done will be given shortly. First, however, the basic types of core component structural analysis will be described.

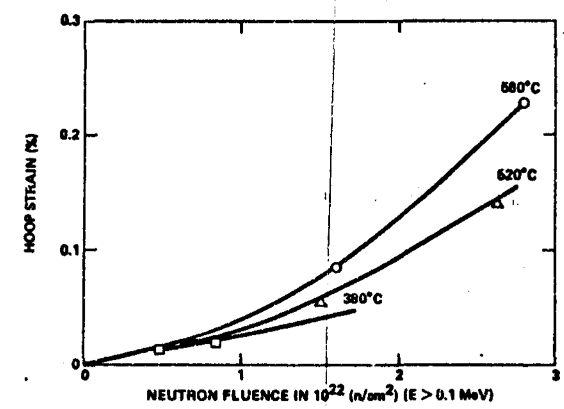

Fig. 6 In-Reactor Creep Less Thermal Creep for 20% Cold Worked Type 316 Stainless Steel Tubes Under Internal Pressure at Various Irradiation Temperatures. Hoop Stress = 70 MPa. (Ref. 8)

Types of Structural Analysis

Three basic types of structural analysis are used in core component design, and thus the criteria have been developed to be compatible with them.

(1) Elastic Analysis

The first type is elastic analysis. As the name implies, this involves only the elastic calculation of stresses, even when yielding occurs. It is often used for components which experience negligible swelling and inelastic deformation or for "first-cut" stress analysis associated with preliminary design. Since only stresses are calculated, design criteria essentially limit the stresses in terms of appropriate strength properties (e.g., ultimate tensile strength).

(2) Elastic-Irradiation Creep-Swelling Analysis

The second type is termed Elastic-Irradiation Creep-Swelling Analysis (EICSA). It also calculates stresses elastically but, in addition, predicts and takes into account the effects of swelling and irradiation creep deformations. It is often used when thermal creep and plastic strains are expected to be small. The same design rules used for elastic analysis also apply to EICSA.

(3) Inelastic Analysis

The third type is inelastic analysis. In this case, thermal creep and plastic strains are calculated in addition to those due to swelling and irradiation creep. Computed stresses reflect the effect of yielding, creep relaxation, etc. Inelastic analysis is intended to provide the best estimate of actual component behavior. Criteria for use with this type of analysis basically limit strains in terms of appropriate measures of ductility (e.g., uniform elongation, true strain at fracture, etc.). So-called Simplified Inelastic Analysis Methods (SIAM) are sometimes used to conservatively estimate plastic and thermal creep strains, based on elastically calculated stresses. The design rules for inelastic analysis are intended to be somewhat less restrictive than those for elastic analysis or EICSA to reflect the higher level of sophistication and better understanding of structural behavior achieved with inelastic analysis.

Finally, rules which provide protection against thermal creep and fatigue damage, crack propagation and buckling, and structural instability must be satisfied when any of the above three types of analysis is performed.

Explanation of Stress (Strain) Nomenclature

Before describing the relation between failure mechanisms and associated design rules, relevant nomenclature used in these criteria, as well as in the ASME Boiler and Pressure Vessel Code, should be defined.

Stress limits are, in general, formulated in terms of stress intensity, i.e., the difference between the largest and smallest principal stresses. This terminology should not be confused with that used in linear elastic fracture mechanics. Primary stress intensities, P, are those required to satisfy the laws of equilibrium with respect to imposed forces and moments. They are not self-limiting. An example is the loading due to fluid pressure. Secondary stress intensities, Q, are those developed by constraint of adjacent material or by [illegible] constraint of the structure. They are self-limiting. An example is the thermal stresses due to a temperature gradient through the wall of a cylinder. Membrane stresses (or strains) are section-averaged values. Bending stresses are the linearized variation across a section of a non-uniform distribution as illustrated in Figure 7. Peak stresses are the additional component of a non-uniform distribution above the membrane and bending components. P_m denotes primary membrane, P_b primary bending, etc., stress intensities.

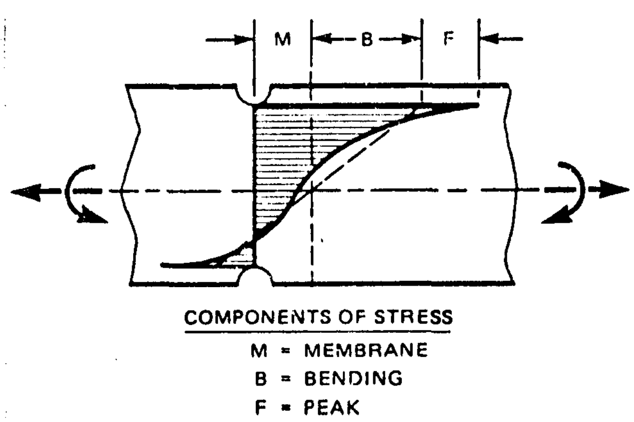

Fig. 7 Illustration of Stress Components in a Non-Uniform Distribution

Failure Mechanisms and Associated Design Criteria Rules

The following failure mechanisms may be potentially important for core components and thus require design rules to protect against them:

(1) Tensile Plastic or Creep Instability
Membrane loadings may cause material to become unstable by necking or localized thinning, leading to rupture. An example is an internally pressurized tube bulging and then bursting. Ratchetting, a process of incremental growth, may also lead to tensile instability.

(2) Structural Instability or Buckling
A structure may become unstable, due to either short-time (plastic) or long-time (creep) loadings, without the necking or thinning associated with tensile instability. Examples include the collapse of a member loaded in bending at a plastic hinge or of a duct being crushed by loading applied across opposite flats. Buckling includes that due to both short- and long-term loadings. Examples are classical Euler column buckling, creep buckling in which a tube collapses under external pressure loading, and local buckling, in which a thin-walled cylinder loaded in compression (or bending) wrinkles locally rather than buckling as a column.

(3) Localized Rupture
Cracking due to short-time loadings may occur at points of strain concentration, such as notches, particularly in highly irradiated material with low ductility and little strain hardening capability. Localized cracking, may, in turn, lead to complete rupture. Bending can also produce strain concentrations leading to localized rupture.

(4) Thermal Creep Rupture
Significant thermal creep damage may occur at various locations in core components with high stresses and temperatures. It can lead to localized creep cracking and the possibility of gross rupture.

(5) Fatigue
Significant fatigue damage may accumulate at locations with large cyclic stresses (strains) and high temperatures and lead to the initiation of cracks. Fatigue and thermal creep damage can interact, accelerating the total damage accumulation.

(6) Unstable Crack Propagation
If crack-like defects are present in a component or develop as a result of service loading, they present the potential for fracture, particularly in highly irradiated material.

(7) Excessive Deformation
The accumulation of inelastic strain may jeopardize the proper functioning of components by producing unacceptably large distortions.

Finally, it should be noted that while swelling and irradiation creep have a profound influence on core design and structural analysis, they are not considered damaging or ductility-limited in themselves, although they may affect the damage due to other mechanisms, as described later.

The relationship between failure mechanisms, corresponding design rules and type of structural analysis is summarized in Table 5. When a given operating condition or event is analyzed, all of the appropriate design rules must be satisfied, as illustrated by the flowchart of Figure 8. Note that if the rules associated with elastic analysis of EICSA cannot be met, then the inelastic analysis may be conducted and its less restrictive set of rules satisfied.

To illustrate how the design rules shown in Table 5 have been developed to account for the time-dependent changes in structural material behavior due to elevated temperature irradiation, the stress intensity limits, strain fraction rules and thermal creep and fatigue damage rules will be described in detail.

Table 5

Summary of Failure Mechanisms and Associated Design Rules

	DESIGN RULE	
POTENTIAL FAILURE MECHANISM	ELASTIC or ELASTIC-IRRADIATION CREEP-SWELLING ANALYSIS	INELASTIC ANALYSIS
TENSILE PLASTIC or CREEP INSTABILITY (INCL. RATCHETING) LEADING TO GROSS RUPTURE STRUCTURAL INSTABILITY/BUCKLING/ GROSS DEFORMATION	STRESS INTENSITY LIMITS ON: P_m $P_m + P_b$ $P_m + P_b + Q$ LIMITS ON LOAD/STRAIN FOR SHORT-TIME AND LONG-TIME INSTABILITY/BUCKLING	MEMBRANE PLASTIC AND THERMAL CREEP STRAIN FRACTION SUMMATION RULE LIMITS ON LOAD/STRAIN FOR SHORT-TIME AND LONG-TIME INSTABILITY/BUCKLING
LOCALIZED RUPTURE (SHORT-TIME)	LIMIT ON MAXIMUM PRINCIPAL (TENSILE) STRESS	TOTAL STRAIN FRACTION SUMMATION RULE
THERMAL CREEP DAMAGE + FATIGUE DAMAGE (INCL. CREEP-FATIGUE INTERACTION)	LIMIT ON: TIME FRACTION SUMMATION + CYCLE FRACTION SUMMATION	LIMIT ON: TIME FRACTION SUMMATION + CYCLE FRACTION SUMMATION
UNSTABLE CRACK PROPAGATION	LIMIT ON LINEAR ELASTIC FRACTURE MECHANICS STRESS INTENSITY	LIMIT ON J-INTEGRAL
EXCESSIVE DEFORMATION	DEFORMATION LIMITS FOR FUNCTIONAL ADEQUACY	DEFORMATION LIMITS FOR FUNCTIONAL ADEQUACY

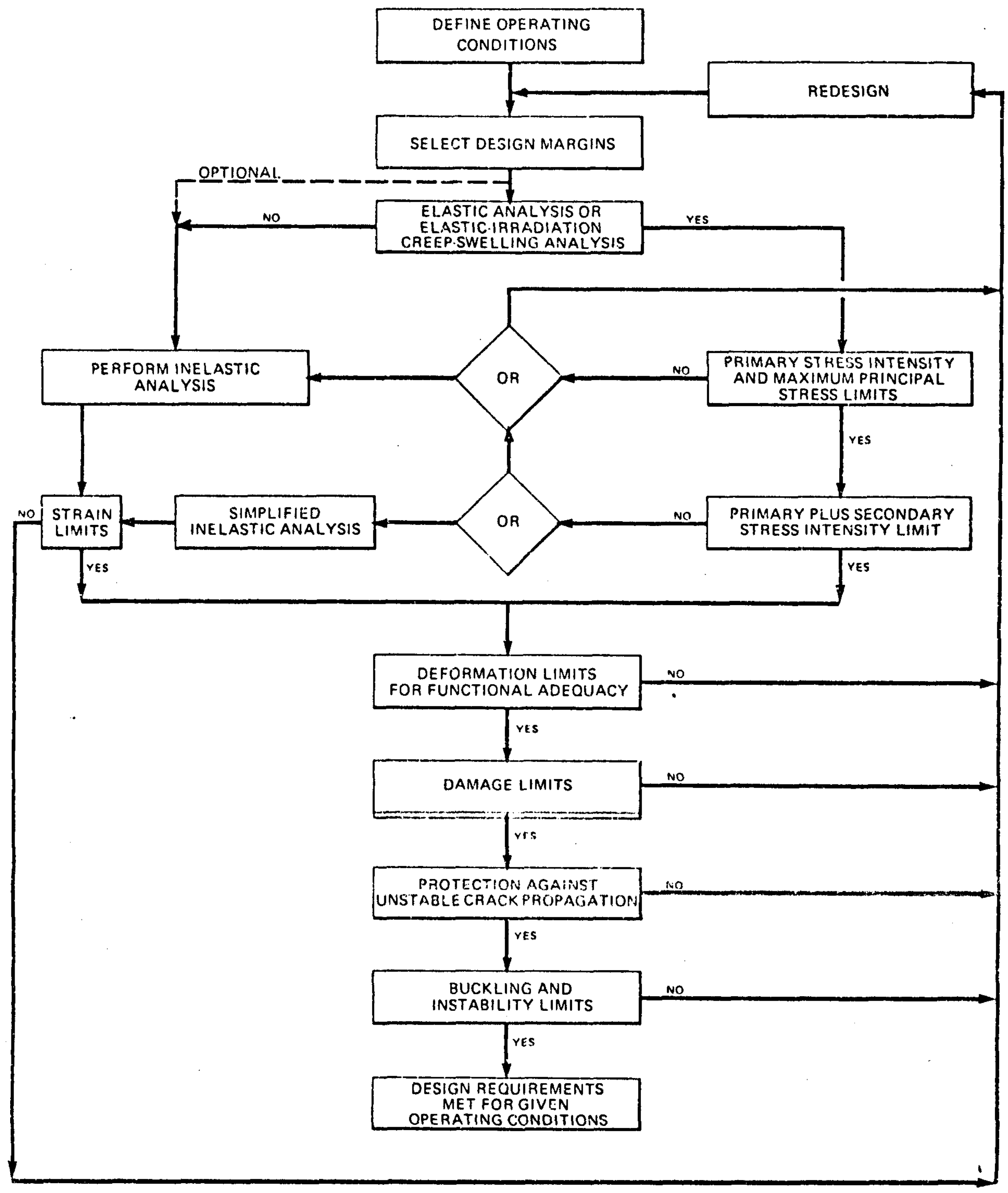

Fig. 8 Flowchart for Satisfying Design Rules for a Given Operating Condition

Primary Membrane Stress Intensity Limit

To protect against gross rupture due to primary membrane loading, the following stress limit is imposed:

$$P_m < \alpha S_F \quad (1)$$

where: P_m = primary membrane stress intensity

α = design margin (fraction <1) depending on component classification and operating condition (typically $\sim$ 0.6 for normal operation)

S_F = stress limit = lesser of 1.66 S_y or S_u

S_y = minimum yield strength as a function of temperature and fluence (i.e., time dependent, including effects of thermal aging)

S_u = minimum ultimate strength as a function of fluence and temperature.

A primary membrane stress limit may be based on S_y when material behavior is ductile, such as early in the life of core components. However, as material loses ductility and strain hardening capability with irradiation, the stress limit shifts to a fraction of S_u to protect against fracture. The transition between a limit based on S_y and one based on S_u is taken when uniform elongation, e_u, drops below 5%. For membrane loading, e_u is the appropriate measure of ductility for protection against tensile plastic instability.

Figure 9 shows e_u for two austenitic stainless steels plotted against the ratio of yield-to-ultimate strength for a wide range of temperatures and fluences [9]. The ratio (S_y/S_u) provides a good normalizing parameter. When minimum e_u is about 5%, (S_y/S_u) is approximately 0.6. Thus the "ductility transition" can be expressed in terms of this strength ratio, which seems more appropriate for use with a limit based on elastically calculated stresses. In other words, when $(S_y/S_u) < 0.6$, $S_F = 1.66\ S_y$. With a design margin of $\alpha = 0.6$, for example, then $P_m < S_y$ for ductile material behavior. When $(S_y/S_u) > 0.6$, as a result of irradiation, then the limit shifts conservatively to $P_m < 0.6\ S_u$.

Primary Membrane Plus Bending Stress Intensity Limit

To protect against gross deformation, collapse or rupture due to a combination of primary membrane plus bending loads, the following limit is prescribed:

$$(P_m + P_B) < \alpha K S_F \quad (2)$$

where: P_B = primary bending stress intensity

$$K = 1 + (K_t - 1)\left(1 - \frac{P_m}{\alpha S_F}\right)$$

$$> 1 \text{ if } (S_y/S_u) < 0.6$$

$$< 1 \text{ if } (S_y/S_u) > 0.6$$

K_t = bending shape factor.

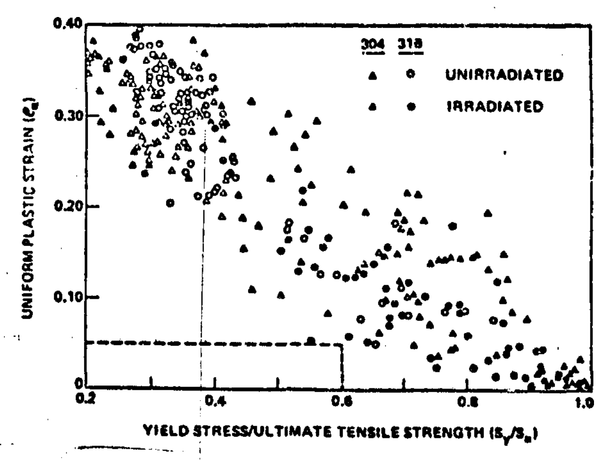

Fig. 9 Uniform Elongation vs. Ratio of Yield to Ultimate Strength (Ref. 9)

For the case of bending stresses only ($P_m = 0$), the elastically calculated outer fiber stress is allowed to be K_t times S_y, based on traditional elastic-perfectly plastic limit load considerations. The factor K_t varies with cross section; for a beam of rectangular section, it is 1.5. When material behavior is ductile, it can accommodate the plastic deformation which accompanies the attainment of a limit load. However, as a material loses ductility with irradiation, fracture may occur at the outer fibers before a limit load is reached. To protect against this, it is again necessary to have a dual stress limit.

In order to account for the time-dependent loss of ductility due to irradiation, K_t should decrease as a function of ductility and strain hardening index. To derive such a relation, the outer fiber strain is conservatively limited to e_u and a bilinear strain hardening model of material assumed, as illustrated in Figure 10. Both e_u and the strain hardening index can be expressed in terms of (S_y/S_u), leading to the relation between K_t and (S_y/S_u) shown in Figure 11. For $(S_y/S_u) >$ about 0.85, K_t decreases to unity at $(S_y/S_u) = 1$. To be conservative and consistent with the primary membrane stress intensity limit, K_t is restricted to unity when $(S_y/S_u) > 0.6$. Thus, when material behavior is ductile, i.e., $(S_y/S_u) < 0.6$, $P_B = 1.5\ S_y$ for a design margin α of 0.6. (The presence of any P_m will reduce the allowable P_B in accordance with the K expression of Equation 2.) When material has been irradiated to the extent that $(S_y/S_u) > 0.6$, the stress limit shifts to $P_B = 0.6 S_u$.

Primary Plus Secondary Stress Intensity Limit

To protect against ratchetting and gross rupture, the following limit is imposed on primary and secondary stresses:

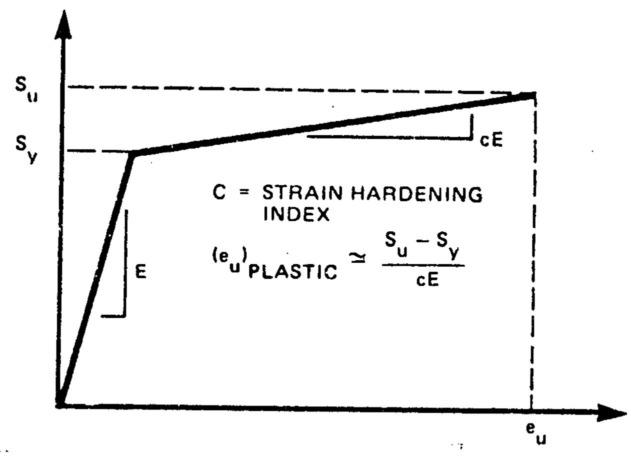

Fig. 10 Bi-Linear Strain Hardening Model

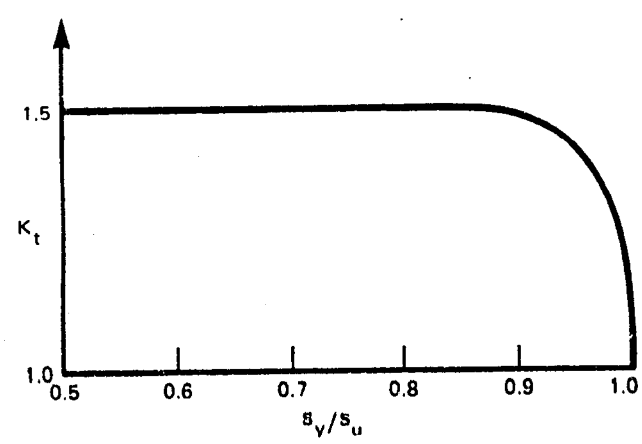

Fig. 11 Bending Shape Factor, K_t, vs. Ratio of Yield to Ultimate Strength for a Beam of Rectangular Cross Section

(a) When $e_u > 1\%$, the combination of $(P_m + P_B/K_t)/S_y$ and $(Q_R)_{max}/S_y$ is restricted to within the modified shakedown boundary shown in Figure 12. $(Q_R)_{max}$ is the maximum range of secondary stress intensity during an operating cycle.

(b) When $e_u < 1\%$, $(P_m + P_B + Q)_{max}$ is restricted to a fraction of S_u. The maximum value of the sum of stress intensities during a cycle is used in this case.

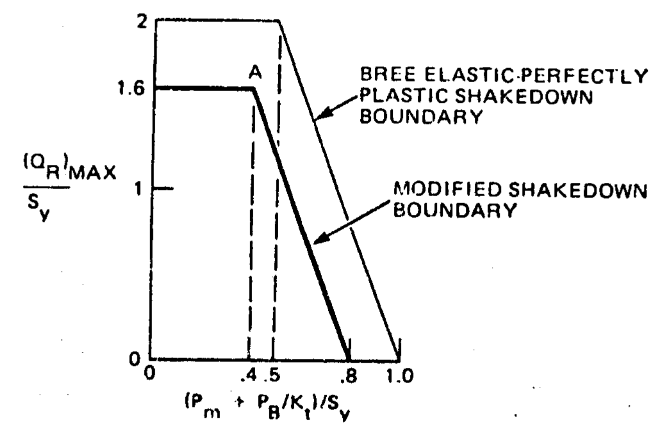

Fig. 12 Shakedown Diagram

Conventional limits on primary plus secondary stresses are based on shakedown considerations which, in turn, presupposes sufficient ductility to accommodate the inelastic straining which precedes shakedown. However, the loss of ductility with irradiation may be great enough to cause fracture before shakedown occurs. Again, a dual limit on primary plus secondary stress intensities is needed, depending on whether material behavior is ductile or not at a given point in the life of a component. Before considering what constitutes an appropriate "ductility transition," the basis for part (a) of the primary plus secondary stress intensity limit will be briefly reviewed.

The Bree elastic-plastic shakedown boundary [10] in Figure 12 is intended to assure shakedown when creep is negligible. Leckie and Ponter [11] have shown that ratchetting strains will be small when creep is significant if loads are restricted to $n/(n+1)$ of the elastic-plastic shakedown load, where n is the exponent in a creep relation of the form $\dot{\varepsilon}^c = A\sigma^n$. For austenitic stainless steels, n varies between about 4 and 10, so that it is conservative to take a modified shakedown boundary as 80% of the Bree boundary (i.e., $n = 4$).

The limit just described may be used safely until the uniform elongation is reduced by irradiation to the point where it may be exceeded by strains corresponding to elastically calculated stresses at point A of Figure 12. The highest S_y for 20% CW 316 SS is about 850 MPa, which would give strains of approximately 1% for the condition at this point. Thus, a "ductility transition" of $e_u \simeq 1\%$ was selected. For lower values of e_u, an alternate limit to prevent fracture is used, i.e., part (b) of the primary plus secondary stress intensity limit. In this case, the potential of secondary stresses to cause fracture is considered equivalent to that of primary stresses for low ductility material, another conservatism.

<u>Membrane Strain Fraction Rule</u>

To protect against tensile plastic and creep instability, membrane strain increments are summed on a normalized basis over the lifetime of a component and limited to:

$$\sum \frac{\delta\epsilon_m^P}{\epsilon_L^P} + \sum \frac{\delta\epsilon_m^c}{\epsilon_L^c} < \beta \qquad (3)$$

where:

$\delta\epsilon_m^P$, $\delta\epsilon_m^c$ = principal membrane true plastic and thermal creep strain increments

ϵ_L^P, ϵ_L^c = plastic and thermal creep strain limits, as a function of temperature, fluence, stress-state and strain rate at the time under consideration

β = a design margin (<1), depending on component classification and operating condition.

(The summation of strain fractions is carried out in each principal direction, with the largest resulting absolute value used in Equation 3.)

To reflect the time-dependent reduction in ductility, a strain fraction summation rule is used rather than a fixed strain limit as in ASME Code Case 1592. To illustrate the reasons for this, consider the first term of Equation 3. Plastic straining early in the life of a core component can be tolerated more readily than later in life. This is taken into account by use of a variable strain limit, ϵ_L^P, which decreases with fluence. The amount of plastic straining allowed at any point in life thus depends on the accumulated fluence and previous plastic strain excursions which have, in effect, "exhausted" available ductility.

The ϵ_L^P limit is taken as $(e_u)_{min}/2$ for protection against tensile plastic instability. Theoretical strain at instability varies as function of component type (e.g., pressurized tube, thin sheet in tension, etc.) and stress-state [12], but $(e_u)/2$ appears to be a conservative lower bound.

Additional conservatism is introduced by restricting the plastic strain increment summation to a design margin, β, less than unity, for normal operation. A variety of data, primarily from tube burst tests of both unirradiated and irradiated material at room and elevated temperatures [13-16], support the ϵ_L^P limit formulation.

The second term of Equation 3 is intended to protect against thermal creep instability. A variable ϵ_L^C limit is used for the same reasons that ϵ_L^P is taken as a function of fluence. Post-irradiation tests of 20% CW 316 SS show little if any tertiary creep. Thus, the ϵ_L^C limit is currently taken as the uniaxial thermal creep strain at the rupture, corrected for the effect of stress-state by a procedure proposed by Manjoine [17].

Another feature of the membrane strain fraction rule is its attempt to account for possible creep-plasticity interaction, that is, the reduction in tensile ductility by prior creep strain, an effect important in annealed, unirradiated austenitic stainless steels [18]. Also, a combined plastic and thermal creep strain summation rule similar to that presented here has produced good predictions for unirradiated Type 304 stainless steel [19].

It should be noted that recent in-pile tests (where creep deformation is both thermally- and irradiation-induced) of 20% CW 316 SS show that substantial strains may be attainable without tensile instability. Furthermore, recent tests on the effect of prior thermal creep exposure on the residual tensile ductility of unirradiated 20% CW 316 SS show that uniform elongation may actually increase after the exposure, probably due to thermal recovery. Thus, for typical in-pile applications of 20% CW 316 SS, it may be possible to neglect the second term of Equation 3.

Again, the membrane strain fraction rule is intended to protect against tensile plastic and creep instability. For more general cases, it is necessary to do a detailed inelastic analysis of a structure over its entire life to determine its stability under both short-time and long-term loadings.

Total Strain Fraction Rule

To protect against localized rupture (cracking) at points of strain concentration due to short-time (plastic) loadings, total (membrane + bending + peak) plastic strains are limited according to:

$$\sum \frac{\delta\epsilon_t^P}{(\epsilon_f/TF)} < \gamma \qquad (4)$$

where:

$\delta\epsilon_t^P$ = total, maximum principal true plastic strain increment (not to be confused with fatigue strain range)

ϵ_f = minimum true strain at fracture in a tension test, as a function of temperature, fluence and strain rate

TF = triaxiality factor

$$= \frac{\sqrt{2}\ (\sigma_1+\sigma_2+\sigma_3)}{[(\sigma_1-\sigma_2)^2 + (\sigma_2-\sigma_3)^2 + (\sigma_3-\sigma_1)^2]^{1/2}}$$

γ = a design margin (<1), depending on component classification and operating event.

As with Equation 3, a normalized strain fraction summation rule is used, again to reflect the progressive reduction in ductility (in this case, true strain at fraction) with irradiation. When material behavior is ductile, strain concentrations such as at nothces or in beams under concentrated loading do not diminish static strength. However, as ductility and strain hardening capability are lost with irradiation, sensitivity to strain concentration increases, and thus it is necessary to limit localized plastic strains to avoid cracking.

True strain at fracture, ϵ_f, was selected as the most appropriate measure of ductility to prevent localized rupture, whether at notches or the outer fibers in bending. Fracture ductility is sensitive to stress-state. To account for the lower ductility observed in certain types of notches (i.e., those with a large restraint on plastic flow), in plane strain bending, etc., ϵ_f from a tension test is corrected for the effect of stress-state by the triaxiality factor, as shown in Equation 4. The initial experimental basis for this rule is shown in Figure 13, for unirradiated metals at room temperature and for a single load application producing a maximum principal plastic strain, ϵ_{max}^P. Note that the more stress-state sensitive materials follow the McClintock model [20], which is based upon the coalescence of postulated voids in a material. In any case, an inverse

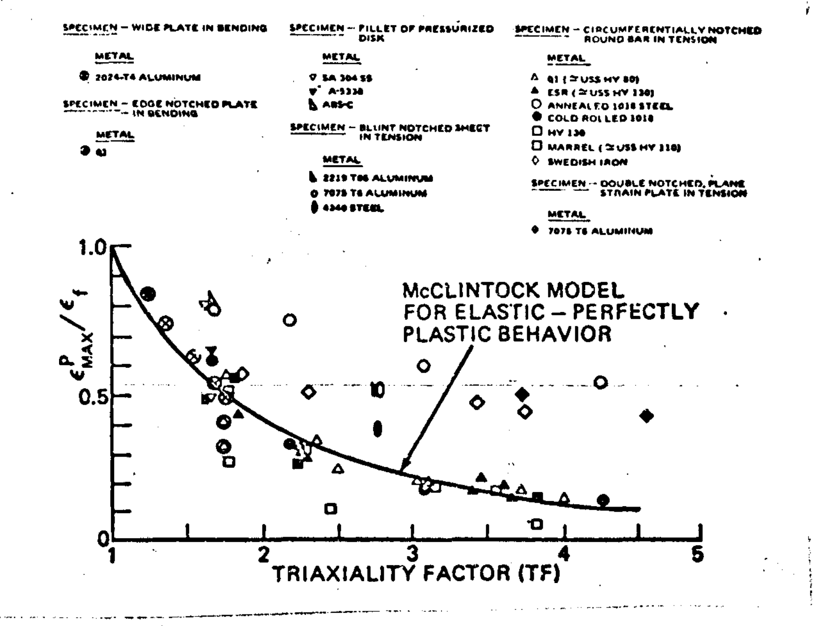

Fig. 13 Effect of Stress-State on Fracture Ductility (Data from Refs. 15, 21-25)

relation between fracture ductility and triaxiality factor is indicated. No design benefit is taken for increases in fracture ductility for stress-states with triaxiality factors less than unity. Evaluation of the results of elevated temperature tests of edge-notched tension specimens [26] and thin beams in plane strain, three-point bending [27] made of highly irradiated austenitic stainless steels tend to support the total strain fraction summation rule.

Thermal Creep and Fatigue Damage

To protect against cracking and rupture due to thermal creep and fatigue damage and their interaction, a combined time fraction and life fraction summation rule is used:

$$\sum \frac{\Delta t}{t_d} + \sum \frac{n}{N_d} < \beta \qquad (5)$$

where:

- Δt = a time increment over which temperature and stress are relatively constant
- t_d = design allowable time-to-rupture, as a function of maximum principal stress, temperature and fluence.
- n = number of cycles with a strain range $\Delta\epsilon$
- N_d = design allowable fatigue cycles at $\Delta\epsilon$, as a function of temperature and fluence
- β = a design margin (<1), depending on component description, operating event

This rule assumes a linear thermal creep-fatigue interaction, and deviations from it are taken into account through a conservative design margin β. To account for the effects of irradiation, the design allowable quantities t_d and N_d decrease with fluence. Thus, a given combination of stress, temperature, strain range, etc., becomes progressively more damaging with time in-reactor.

Equation 5 must be satisfied for all types of analysis. However, the calculation of quantities used in applying the rule differs with the particular type of analysis. For example, when elastic analysis is conducted, peak stresses at discontinuities are conservatively taken at their full elastically calculated values for fatigue damage evaluation, even though they may relax.

Correspondingly, thermal creep damage evaluation uses elastically calculated values of stress unless they exceed S_y, in which case it is used instead. On the other hand, when detailed inelastic analysis is conducted, the effects of stress relaxation are taken into account. Thus strains for fatigue damage evaluations and stresses for thermal creep damage evaluation may be considerably lower than their counterparts calculated elastically. This is another instance of the design rules for use with inelastic analysis being somewhat less restrictive than those for elastic analysis or EICSA.

Guidance on how to treat the effects of strain concentrations, multiaxial loading, mean and residual stresses, hold times, etc., for each type of analysis is also contained in the criteria.

In many core components, there can be large variations in damage due to severe temperature gradients ("hot spots"), local geometrical discontinuities, etc. Very localized damage should not necessarily disqualify components from further, useful service. In such cases, a component is not considered to have failed when the damage limit in a local region is exceeded, if: (a) the locally damaged region is such that a flaw assumed present over the region does not cause unstable crack propagation, (b) the dimensions of this region are less than twenty percent of the nominal thickness of a component, and (c) functional adequacy of a component can be demonstrated with the locally damaged region.

The design rules for fatigue and thermal creep damage, local rupture, etc., are intended to protect against crack initiation. Even when they are satisfied, it is always conceivable that some cracks may develop in service or that small crack-like defects may be present from the time of fabrication which may have escaped detection despite stringent inspection. To protect against fracture originating from flaws, the criteria also contain preliminary fluence-dependent limits on fracture toughness for use with elastic analysis or EICSA and on critical J-integral for use with inelastic analysis, as well as guidelines on how to estimate stable crack growth.

SPECIAL CONSIDERATIONS IN STRUCTURAL ANALYSIS RESULTING FROM IRRADIATION EFFECTS

Irradiation effects on structural materials behavior affect not only the formulation of design rules, but the corresponding structural analysis as well. The purpose of this section is to provide an appreciation of some of the additional considerations introduced into elevated temperature structural analysis by irradiation creep and swelling.

Irradiation creep tends to relax secondary and peak stresses and may redistribute primary stresses even at lower temperatures where thermal creep may be insignificant. When a component is constrained from swelling freely, either externally or internally, swelling-induced stress develops and must be included in stress calculations as an additional loading mechanism. As swelling-induced stress develops, it is relaxed simultaneously by irradiation creep. Since swelling and irradiation creep are two concurrent processes, the swelling-induced stress reaches a plateau, shortly after the incubation dose, and remains approximately constant thereafter. Calculating swelling-induced stress elastically in the same manner as thermal stress could be overly conservative and a procedure similar to that proposed by one of the authors [28] is more appropriate for its evaluation. Swelling is very sensitive to chemical composition of austenitic stainless steels. Consequently, a swelling-induced stress may develop at the junction of two pieces having nominally the same chemical composition. This concentrated swelling-induced stress acts in addition to stress concentrations from geometrical discontinuities that may be present.

Under cyclic loading conditions, both irradiation creep and swelling have a significant impact on the analysis. Irradiation creep relaxes residual stresses resulting from deformation of previous cycles. This relaxation may lead to ratcheting under stress combinations for which shakedown would have otherwise occurred. In the case of strain-controlled cycling, with either a tension or compression hold time, stress relaxation by irradiation creep leads to an increase in the plastic strain component and therefore to a reduction in fatigue life, even though irradiation creep in itself is considered non-damaging to the microstructure. Swelling-induced stress does not cycle in the same manner as thermal stress and therefore may be classified as a mean stress for fatigue damage calculations. Swelling voids act as geometrical discontinuities, possibly giving rise to stress concentration and further reductions in fatigue life.

Thermal creep damage must be evaluated over the entire design life of all core components. The relaxation of peak and secondary stresses by irradiation creep, in addition to thermal creep, would tend to decrease creep damage over what might be expected in the absence of irradiation creep.

Guidance on how these and other similar considerations should be taken into account in core component structural analysis is given in draft RDT F9-8.

DISCUSSION

The design rules illustrated here take into account the effects of irradiation in what is believed to be a conservative manner. In some instances, they may be overly conservative. For example, limiting primary bending stress intensities to a fraction of S_u when (S_y/S_u) exceeds 0.6 may be pessimistic. Considering secondary stresses to have the same potential to cause fracture as primary stresses when $e_u < 1\%$ may also be unduly conservative, and so forth. In addition, many of the rules have been based on an interpretation of the behavior of unirradiated material, for lack of sufficient data on irradiated material. Thus, further tests of irradiated austenitic stainless steels and of 20% CW 316 SS, in particular, would help to validate the rules or suggest improvements in them. A number of tests to do this are either in progress or planned. For instance, bend tests of irradiated 20% CW 316 SS now being conducted will help to check both the primary membrane plus bending stress intensity limit and the total strain fraction rule, as they apply to bending stresses and strains.

Material properties correlations for use in the design rule limit quantities should be improved and expanded. For example, more data on the fatigue and fracture toughness behavior of irradiated core structural materials are needed. Also, the design rules have been based on an understanding of material behavior obtained from tests of pre-irradiated material. Actual in-reactor behavior, such as creep rupture life, may be considerably different than that indicated by post-irradiation tests, and thus correlations for in-reactor behavior will be helpful. To improve the structural analyses used in satisfying the design rules, constitutive relations for irradiated material which take into account the effects of strain rate, thermal aging, cyclic hardening and softening, periodic plastic deformations, etc., are also needed.

Many of the current design margins in the criteria are the same as, or close to, those used in ASME Elevated Temperature Code Case 1592 when material behavior is considered to be ductile. This is conservative since core components are replaceable, whereas Code Case 1592 covers long-life "pressure boundary" components having a greater safety function. With little precedent upon which to rely, the design margins used when material has lost its ductility due to irradiation were selected by the Working Group with the intention of being consistent with the safety and functional requirements of core components, but not so conservative as to unduly penalize core performance. To validate the design margins (and possibly make them less restrictive) will require a continuing evaluation of the uncertainties in material behavior and analysis methods as well as of operating experience gained from both test reactors (e.g., Fast Flux Test Facility) and commercial prototypes in the U.S. and abroad. Increased understanding of material behavior, improved analysis methods, and feedback from operating experience will ultimately establish design margins, as has been the case with the evolution of the ASME Boiler and Pressure Vessel Code over the past several decades.

CONCLUSIONS

(1) Structural design criteria for highly irradiated core components differ from conventional elevated temperature criteria in two basic respects. Traditional design rules must be reformulated to account for low ductility material behavior and associated failure mechanisms. Design limit quantities (e.g., ultimate tensile strength, true strain at fracture, etc.) should be taken as a function of fluence to reflect the significant time-dependent changes in them resulting from irradiation.

(2) Structural analysis of core components is complicated considerably by irradiation creep and swelling, two deformation modes which act in addition to those usually encountered in elevated temperature operation. New analysis methods and guidelines are needed to take them into account.

(3) Validation of and modifications to the current design criteria and its margins will require an integration of: (a) the results of tests to generate basic material properties correlations and to check specific design rules, (b) the development of improved analysis methods, and (c) an evaluation of available operating experience.

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