Dimensional Stability of Zirconium Alloys

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CONTENTS

1   INTRODUCTION (RON ADAMSON)

1.1   APPLICATIONS AND BACKGROUND

One of the most unique aspects of material behaviour in a nuclear power plant is the effect of radiation (mainly neutrons) on the dimensional stability (or instability, i.e., a change in dimensions during service) of the reactor components. In fast breeder reactors the Fe and Ni-based alloys creep and swell, that is, they change dimensions in response to a stress and change their volume in response to radiation damage. In light water reactors, zirconium alloy structural components creep, do not swell, but do change their dimensions through the well-known constant volume process called irradiation growth. Radiation effects are not unexpected since during the lifetime of a typical component every atom is displaced from its normal lattice position at least 20 times! With the possible exception of elastic properties like Young's Modulus, the properties needed for reliable fuel assembly performance are affected by irradiation. A straightforward summary of such effects is given in Adamson R. B., 2000.

Practical effects of dimensional instabilities are well known and it is a rare technical conference in the reactor performance field that does not include discussions on the topic.

- Because of the difference in pressure inside and outside the fuel rod, cladding creeps down on the fuel early in life, and then creeps out again later in life as the fuel begins to swell. A major issue is to have creep strength high enough to resist outward movement of the cladding at a rate higher than the fuel pellet swelling rate if fission gas pressure becomes high at high burnups. Opening of the gap between fuel and cladding could lead to substantial local temperature increases.

- PWR guide tubes can creep downward or laterally due to forces imposed by fuel assembly hold down forces or cross flow hydraulic forces – both leading to assembly bow which can interfere with smooth control rod motion.

- BWR channels can creep out or bulge in response to differential water pressures across the channel wall, again leading toward control blade interference and in difficulties in unloading the channels from the core. Fuel rods, water rods or boxes, guide tubes, and tie rods can lengthen, possibly leading to bowing problems. (For calibration, a recrystallized (RX) Zircaloy water rod or guide tube could lengthen due to irradiation growth more than 2 cm. During service a cold worked/stress relieved (SRA) component could lengthen more than 6 cm).

- Even RX spacer/grids could widen enough due to irradiation growth (if texture or heat treatment was not optimized) to cause uncomfortable interference with the channel.

In addition, corrosion leading to hydrogen absorption in Zircaloy can contribute to component dimensional instability due to the fact that the volume of zirconium hydride is about 16% larger than that of zirconium.

The above discussion leads to the concept that understanding the mechanisms of dimensional instability in the aggressive environment of the nuclear core is important for more than just academic reasons. Reliability of materials and structure performance can depend on such understanding. Table 1-1 illustrates some of the relationships involved in dimensional stability issues.
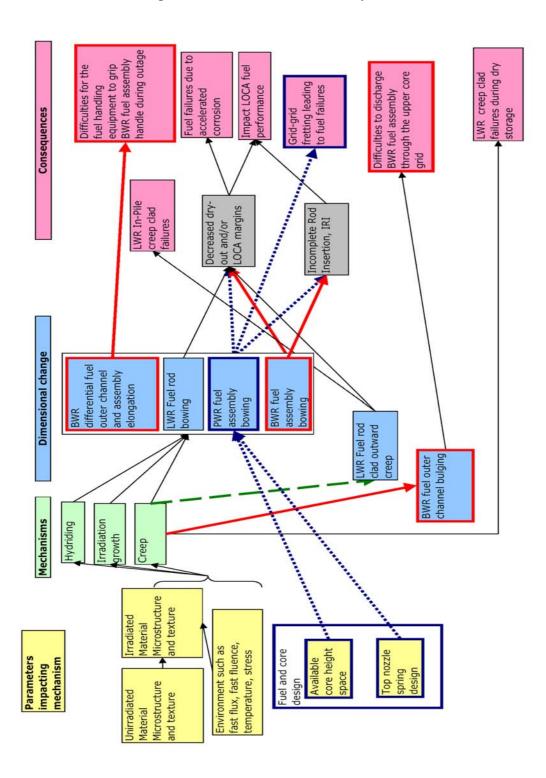
Table 1-1:   Relationships between dimensional stability issues.

In conformance with the NRC Standard Review Plan (SRP), SRP, 1981, dimensional changes of all fuel assembly components such as rod bowing or elongation of fuel rods, control rods, channels and guide tubes must be limited. Accordingly, fuel rod elongation due to irradiation creep and growth, hydriding and pellet-cladding mechanical interaction (PCMI, fuel rods only) must be taken into account in the fuel design to ensure that adequate gaps exist between 1) top end plugs and top tie plate and 2) bottom end plugs and bottom tie plate. Knowing that fuel cladding will elongate, designers must provide space above and below the rods to prevent blockage of the elongation. Such blockage could cause rod bow or snaking, as illustrated in Figure 1-1. This situation could lead (and has led) to fuel failure due to rod-to-rod fretting or dryout.
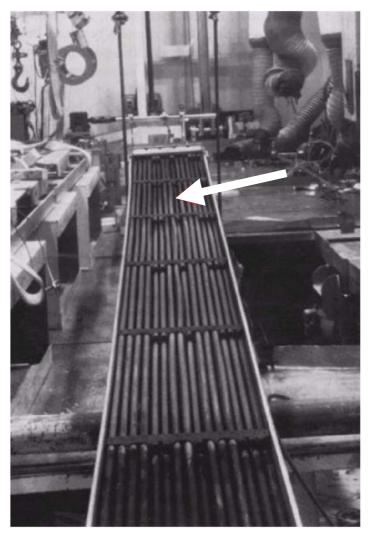
Figure 1-1:   White arrow indicates large fuel rod bowing due to excessive fuel rod growth, Franklin D. G. and Adamson, R. B., 1988.

The SRP also requires that insertability of the control rods must be ensured. This requirement is fulfilled by ensuring that the vertical lift-off forces must not unseat the lower tieplate from the fuel support piece such that the lateral displacement of the fuel assembly occurs. For example, in a PWR this is accomplished by ensuring that the net holding down force from gravitation and hold down springs is larger than the corresponding sum of lifting forces due to buoyancy and the upward coolant flow. However, if the net hold down force becomes much larger than the net lifting force, fuel assembly elastic bowing[*] may occur, Figure 1-2. Lower creep strength of PWR guide tubes will result in larger tendency to transform the elastic bowing stresses into plastic strain, i.e., the fuel assembly keeps its bowed configuration upon unloading. During reloading of the core such bowed fuel assemblies will also result in bowing of adjacent fresh straight fuel assemblies. If the fuel assembly bowing becomes large enough, complete control rod insertion may not be accomplished due to frictional forces between the control rod and the guide tube inner surfaces. Excessive elongation of the guide tubes due to irradiation growth and/or hydriding may result in a larger fuel assembly bowing tendency if this has not been accounted for in the fuel design.

Figure 1-2:   Schematics showing fuel assembly bowing.
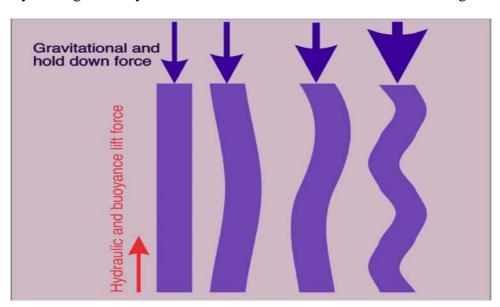
If this bow becomes large enough, the watergaps between the adjacent furl channels may increase significantly, resulting in decreased thermal margins (*dnbr/CPR* and *LOCA*). This situation has resulted in dry-out fuel failures due to excessive channel bowing.

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[*] Elastic bowing means that the fuel assembly will retain its original straight shape upon unloading.

The dimensional changes must also be limited to allow fuel assembly removal and handling during outage.

The choice of cladding material, the fuel assembly design and the reactor operating parameters impact the irradiation growth and creep rates.

Four phenomena may cause zirconium alloy fuel cladding to increase in length:

- Stress-free axial growth due to fast neutron irradiation, as described in Sections 1.3 and 2.2.

- Anisotropic creep (before pellet/cladding contact) from the external reactor system pressure. Because of the tubing texture some axial elongation results from creep down of the cladding diameter. (In a non-textured material, creep down of the cladding would only result in an increase in cladding thickness while the cladding length would remain constant, but almost all zirconium alloy components have a strong texture).

- Creep due to Pellet Cladding Mechanical Interaction, PCMI, later in life after hard fuel-clad contact. As an example the PCMI component was estimated by calculating the difference between the total elongation and the irradiation growth component for some fuel with cold-work stress relieved (CWSRA) and recrystallized (RXA) fuel claddings, Figure 1-3.[*]

- Hydriding of the cladding.

This report proceeds with a perspective overview on the phenomena of creep, irradiation growth, and hydride effects. It is followed by a literature review of the same topics, and then by a more application-related review of dimensional changes of reactor components.

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[*] As a rule of thumb, a fast neutron fluence of 2×10^{25} n/m^2, E>1MeV corresponds to about 10 GWd/MT exposure.
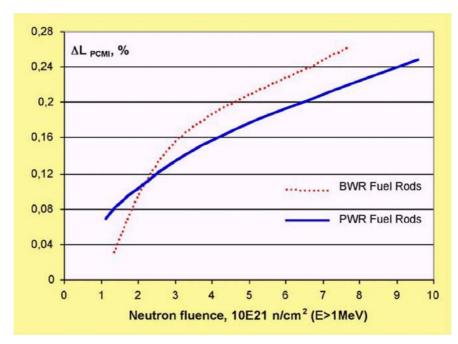
Figure 1-3:   The fuel elongation component in % due to PCMI vs. fast neutron fluence of both PWR SRA-cladding and BWR RXA cladding, modified figure according to Franklin D. G. and Adamson, R. B., 1988.

1.2   PERSPECTIVE ON CREEP

Creep is defined as a time dependent change in dimension of a reactor component (or any material) under a stress, even if that stress is below the yield stress. It is not a property which has much application during fabrication of the components, so it is not often tested as part of fabrication specifications. However, a closely related phenomenon called stress relaxation, defined later, can play some role in the final dimensions of channels or water boxes that are mechanically shaped to final size in the factory. The most important applications of creep are in in-reactor performance of fuel bundle components. Inward creep of cladding early in life and outward creep later in life are very important. Also fuel rods, PWR guide tubes, guide tube assemblies, and BWR channels bow by a combination of creep and growth. BWR channels also creep (bulge) outward due to a pressure differential between the core flow region inside the channel and the core by-pass region outside the channel. These bowing and bulging processes do or can limit bundle burnup by inhibiting control rod movement or by decreasing thermal margins in LOCA or dry out situations. Also unloading of channels through the core grid can be complicated.

Section 1.2 serves as a summary and perspective of key features of irradiation creep. Sections 2.1 and 3 provide a more detailed review of the literature dealing with the same topics.

1.2.1 Creep Phenomena

Creep deformation is plastic strain, so it is a constant volume process. In a reactor component this time dependent deformation usually takes place under conditions of complex loading and operating history. In the laboratory, experiments are generally conducted under simplified conditions, so fuel designers must take the laboratory or, more frequently recently, in-reactor experimental data and apply it to more complex conditions. Figure 1-4 shows a standard creep curve (-strain vs. time-) for a uniaxial test specimen tested in an apparatus as simple as that shown in Figure 1-5. Stage 1 is called primary creep, which is a transient strain that occurs when the stress is applied, or when the direction of magnitude of the stress is changed. Stage II is steady state creep, which can continue for a very long time. Stage III is called tertiary creep and it generally precedes rupture or failure of the test specimen. The rapid increase in strain in stage III is a result of specimen necking. If the applied load were decreased so as to keep the stress constant, the rapid strain increase would not occur. In-reactor stage III creep is rare: reactor components bulge, bow, lengthen, etc., by creep but they do not break by creep alone.
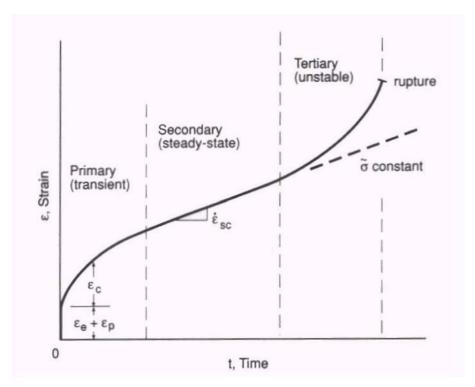
Figure 1-4:   Strain vs. time behaviour during creep under constant load, hence constant engineering stress, and the three stages of creep.
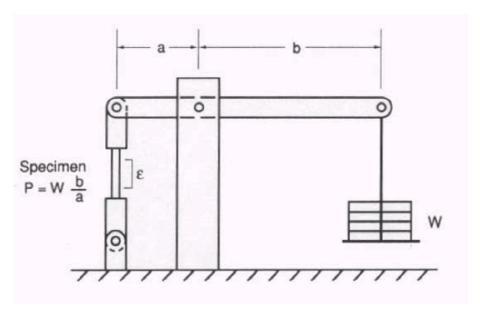
Figure 1-5:   Schematic of a simple creep testing machine.

For application to fuel rods or pressure tubes, biaxial creep tests of pressurized tubes are most useful, and are able to be conducted in reactor. Sealed tubes filled with a gas at known pressure are irradiated under predetermined conditions of temperature and neutron flux. Periodic removal from the reactor allows diameter and length measurements to be made. The data is then used to develop or fix the constants of tube performance models, for example, Gilbon D., et al, 2000.

Stress relaxation is a process that is closely related to creep. In the straightforward case, it can be viewed as follows. A bar is stretched elastically to some length and then held at that constant length. The stress required to maintain that length would be constant if no creep occurred, however, if creep does occur some of the elastic strain in the rod is converted to plastic creep strain, and the stress required to maintain the rod length decreases, or relaxes, Figure 1-6. In reactor components, there are often cases where small residual stresses can be built up, for instance during welding or component straightening. During service, creep can occur which reduces the residual stress to zero, at the expense of causing some local strain or dimensional instability. Relaxation can occur in spacer or grid springs during service, resulting in a significant lowering of the spring-rod contact force, which can create grid-rod fretting issues.
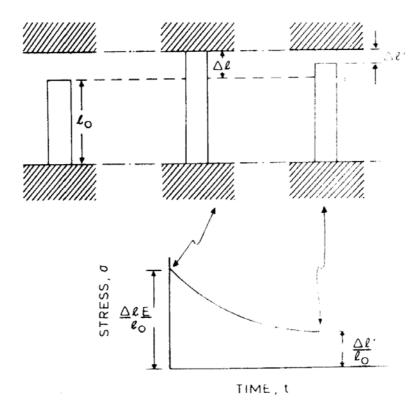
Figure 1-6:   The stress relaxation of a specimen held in tension.

There are three types of creep. The first of these is <u>thermal creep,</u> or creep which occurs due to processes having nothing to do with radiation or reactors. There is a thermal creep component in all creep processes, and it is usually difficult to separate it from radiation-affected creep. The exact nature of thermal creep in-reactor is likely to be different from that which occurs in an unirradiated material tested out-of-reactor. Resistance to thermal creep invariably decreases with increasing temperature. The second type is <u>post-irradiation creep,</u> i.e., creep of irradiated material in the absence of a radiation flux. Because of the large numbers of barriers created by neutron irradiation (the density of <a> type dislocation loops is on the order of $8 \times 10^{14}/m^2$) and the resultant overall strengthening of the material, at temperatures and stresses of interest the resistance to creep of irradiated zirconium alloys is higher than for unirradiated material. That is, the creep strength of irradiated material is higher than the creep strength of unirradiated material. The third type is <u>in-reactor or irradiation-assisted creep</u>. At the low stresses normally of interest in reactor applications (<100 MPa), the steady state creep rate in-reactor is higher than for the unirradiated case. That is, the creep strength is lowered in-reactor. However, there is a transient period early in life before the radiation damage builds up to its saturated value (fluences less than 1×10^{25} n/m^2) where the softer unirradiated material may have the higher creep rate, depending primarily on the magnitude of the applied stress.

1.2.2   Mechanisms

The mechanism of irradiation creep has been thoroughly investigated, but even today is not thoroughly understood. The most extensive reviews have been presented years ago, Franklin D. G., et. al., 1983 and in a compilation of Journal of Nuclear Material articles, Woo, C. H. and McElroy, R. J., 1988. The reader is referred to those documents for details. This section 1.2.2 leans heavily on them and on the individual papers discussed within them, particularly, Fidleris V., 1988.

It is estimated that over 40 mechanisms have been proposed for irradiation creep and it is likely that at least four mechanisms contribute:

- Irradiation growth, which is independent of stress, see Section 2.2.

- Irradiation-produced loop alignment or formation, which is stress dependent.

- Enhanced climb and glide of dislocations, which is stress dependent.

- High stress or strain-rate deformation similar to out-of-reactor deformation.

The first three depend on the fact that irradiation produces point defects (vacant atomic sites called vacancies and extra atomic sites called self-interstitials or just interstitials which diffuse anisotropically to various sinks, such as grain boundaries, dislocations and dislocation loops. Relative contributions of the various mechanisms depends primarily on stress, to be discussed below.

Perhaps the easiest mechanism to visualize is "climb and glide", illustrated in Figure 1-7. The dislocation glides up the barrier and is stopped by it. In the illustrated case, irradiation-produced vacancies diffuse to the dislocation, moving its glide plane up and away from the barrier, allowing it to continue its glide. The creep strain is related to the distance the dislocation has moved, and the creep rate related to the time required. The same process would occur if an interstitial diffused to the dislocation, but in that case the "climb" would be down instead of up. In fact, interstitials are more strongly attracted to dislocations than are vacancies, so the latter process occurs more frequently. The stress dependence of this mechanism depends on the strength of the interaction between the obstacle and the dislocation.
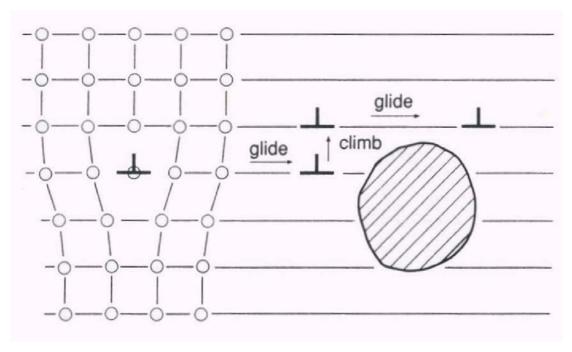
Figure 1-7:   Climb of an edge dislocation, permitting continued glide past an
                obstacle, and enabling deformation to proceed.

The process just described produces true plastic deformation, which is not recoverable
by thermal processes. It is known, however, that post-irradiation thermal annealing of a
creep specimen can in fact recover some of the creep strain. Recrystallized (RXA)
Zircaloy has been observed to recover as much as 68% of the in-reactor creep strain,
Kreyns P. H. and Burkhart M. W., 1968 and Causey A. R., 1974, the amount of
recovery decreasing with increased applied stress. This suggests that irradiation growth
(1) and loop alignment or formation (2) are involved in creep, as it is known that the
irradiation damage involved in both mechanisms can be eliminated by thermal
annealing at temperatures above 673K (400°C). The mechanisms which cause loop
alignment or formation are called stress-induced preferred nucleation (SIPN) or stress
induced preferred absorption (SIPA) and are best examined in the literature,
Matthews J. R. and Finnis M. W., 1988 or Franklin D. G. et. al., 1983. The process is
illustrated in Figure 1-8.
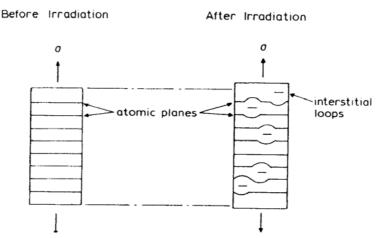
Figure 1-8:   Schematic illustration of material elongation by preferred interstitial loop orientation, Franklin D. G. et. al., 1983.

A schematic summary of creep rates as a function of stress is given in Figure 1-9, Nichols, F. A., 1987. In this figure m = 1/n where n is the exponent on stress to be discussed below. It is seen that various proposed creep mechanism are represented. In practice, stresses in reactor components are usually in the range 50-130 MPa (7-20 ksi). A more quantitative diagram is given in Figure 1-10, where it is seen that "loop alignment" and "climb and glide" are indicated as key mechanisms, Nichols F. A., 1969 in that range.
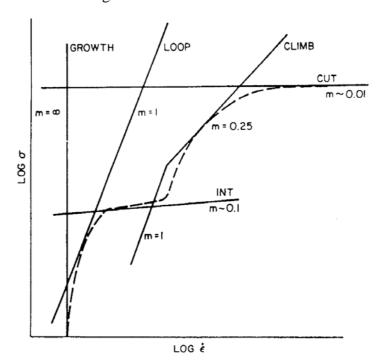
Figure 1-9:   The dashed curve represents in-reactor behaviour, Nichols, F. A., 1987.
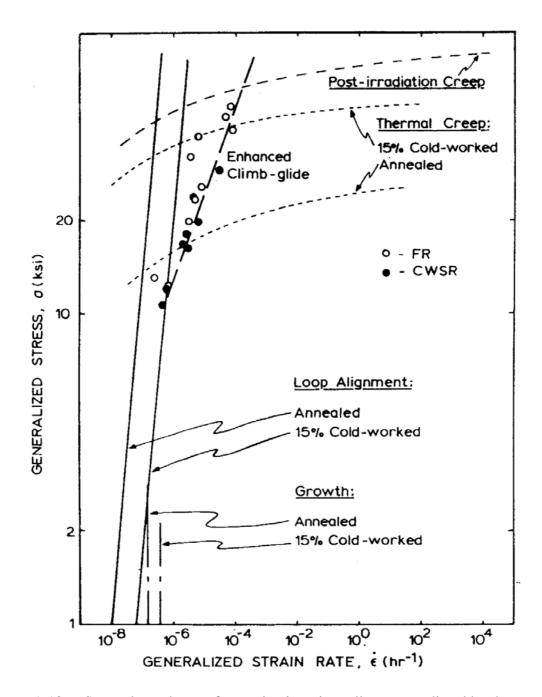
Figure 1-10:   Stress dependence of creep in zirconium alloys as predicted by the multimechanism model, Nichols F. A., 1969.

1.2.3   Variables

The in-reactor strain rate, $\dot{\varepsilon}$, is usually expressed in terms of a number of variables:

$$\dot{\varepsilon} = f\left(\sigma^n, P_m, \frac{1}{B}, \Phi^m, T, f, A\right)$$

where

σ is the applied stress

P_m is the density of mobile dislocations

B is density of barriers to dislocation motion (where B has different strength depending on the nature of the barrier, e.g. dislocations, dislocation loops, solutes, etc.)

Φ is the neutron flux, n/m^2/s

T is irradiation temperature

f is the texture factor

A is a metallurgical factor related to a specific alloy

1.2.3.1   Stress

As discussed above, the stress dependency of in-reactor creep is complicated. In general, a review of literature provided by Franklin D. G. et. al., 1983 shows that

$n \approx 10\text{-}100$ for $400 < \sigma \leq 600$ MPa (58-87 ksi)

$n \approx 4\text{-}10$ for $205 < \sigma \leq 400$ MPa (30-58 ksi)

$n \approx 1$ for $138 < \sigma \leq 205$ MPa (20-58 ksi)

$n \approx 0\text{-}1$ for $\sigma \leq 138$ MPa (0-20 ksi)

This is illustrated again in Figure 1-10 for alloys Zircaloy and Zr2.5Nb.

1.2.3.2   Cold Work

The effect of cold work is expressed through P_m and B, which both increase as the amount of cold work increases. Cold work, which for reactor components means deformation imposed at temperatures below about 373K (100°C), introduces dislocations into the microstructure. In <u>unirradiated</u> material a high density of dislocations introduces many barriers, B, to the climb and glide process, mainly resulting in an increased resistance to creep, or an increased creep strength. At low stresses, however, the high dislocation density appears under some conditions to allow more creep strain than for recrystallized materials. Examples are given in Figure 1-11a and Figure 1-11b where it is seen that in some cases cold work decreases the creep strength.
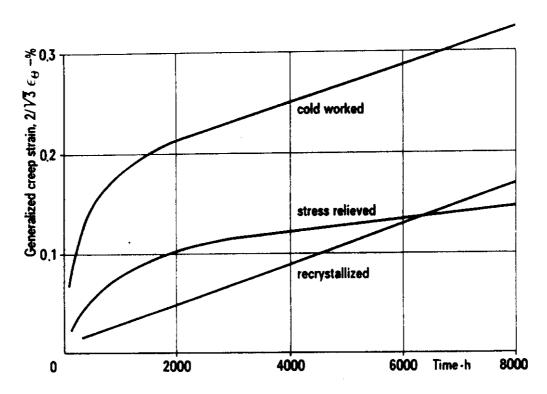

Figure 1-11a: Creep behavior at low stress levels, (internal pressure, closed ends, 113N/mm^2, test temperature, 320°C, unirradiated Zircaloy, Stehle, H., et al., 1977

For cold-worked material in-reactor, however, the dynamic creation of point defects by irradiation provides the driving force for dislocations to climb over barriers (that is, dislocation climb is not rate controlling) and since the number of gliding dislocations is large compared to recrystallized material, the creep strain is also large. Therefore in-reactor the resistance to creep is lower for cold worked material as compared to recrystallized material. The creep strength is higher for recrystallized material. Figure 1-12, Gilbon D., et. al., 2000 nicely illustrates this. The lower curve is for fully recrystallized Zircaloy (and other recrystallized Zr-alloys) and the upper curve is for stress relieved cold work material. The above discussion applies for the low strain rates normally encountered for reactor components. It must be noted that at high strain rates resulting from high stress, the time available for climb of dislocations over barriers may not be sufficient, and cold worked material reverts to having a high creep strength.
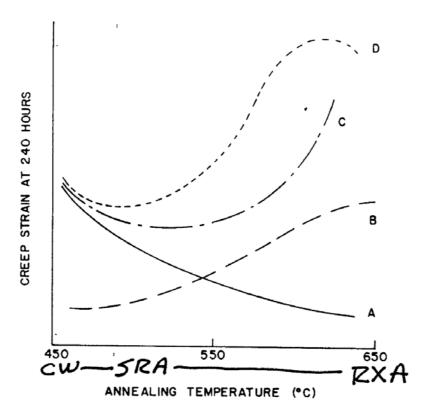
Figure 1-11b: Various observed trends in the dependence on annealing temperature of creep strain in unirradiated Zircaloy at 400°C and 240 h, Franklin D. G. et. al., 1983.
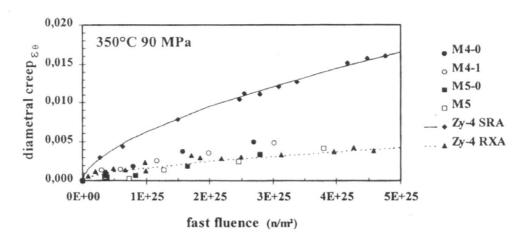
Figure 1-12:   Diametral creep versus fast fluence for Zry-4, M4 and M5 cladding. Upper curve is for SRA material and lower curve is for RXA material, Gilbon D., et. al., 2000.

1.2.3.3   Flux

Figure 1-13 illustrates that neutron flux increases the creep rate at 553K (280°C). The flux dependence is predicted to increase with flux, Dollins C. C., 1971, Figure 1-14, and to decrease with irradiation temperature, becoming negligible near 803K (530°C), Fidleris V., 1988. It should be noted that strain rate sensitivity is most often based on the establishment of a steady state (ie, independent of time) creep rate, which is rarely observed in reactor except perhaps at very long reactor exposure.
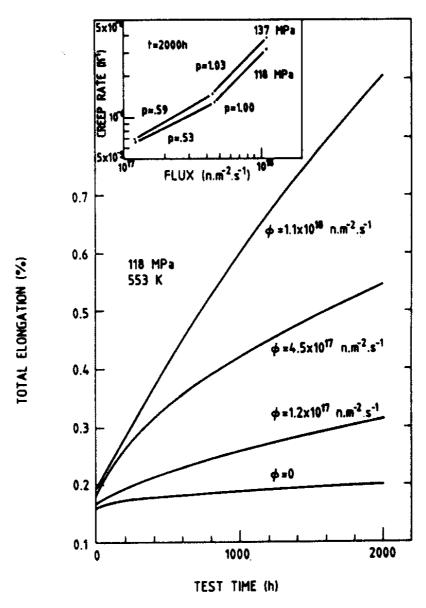
Figure 1-13:   Flux dependence of in-reactor uniaxial creep of cold-worked Zircaloy-2, Tinti F., 1983.
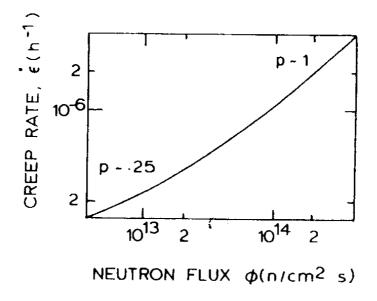
Figure 1-14:   The variation of in-reactor creep rate with neutron flux in zirconium, Dollins C. C., 1971.

1.2.3.4   Temperature

Figure 1-15 (based on compilation of Franklin D. G. et. al., 1983) indicates that creep rate increases with increasing temperature. The correlation is not purely due to irradiation effects, however, as thermal creep increases as the temperature increases.
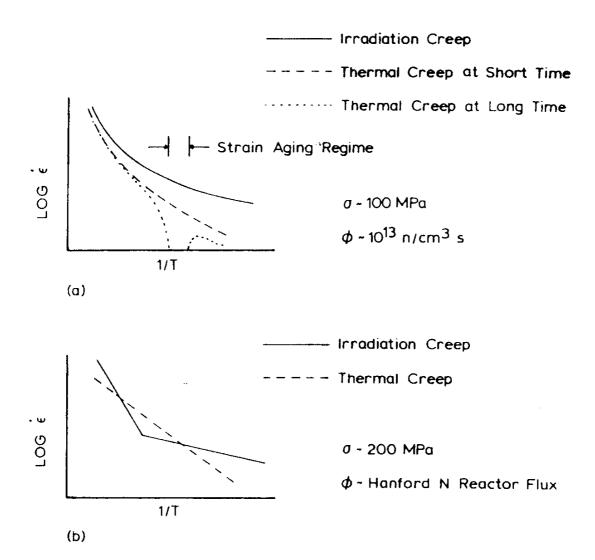
Figure 1-15:   Schematic illustration of the variation of creep rate with temperature in zirconium alloys per a compilation of data in Franklin D. G. et. al., 1983.

1.2.3.5 Texture

During fabrication, zirconium alloys develop a strong texture which causes mechanical properties, irradiation growth, etc., to be highly aniosotropic. Creep properties are also different in different directions, with creep rates higher in the direction of rolling as compared to transverse to rolling direction. This of course has to do with the relative ease of dislocation motions in those directions due to the texture, Adamson R. B., and Rudling P., 2001 for a more complete discussion). However, there appears to be less anisotropy in creep behavior then there is in time independent mechanical properties or irradiation growth. Figure 1-16 illustrates the effect. It is noted that in the temperature ranges of interest to reactor operation the differences in strain rates occur mainly at low fluences where irradiation hardening has not fully developed.
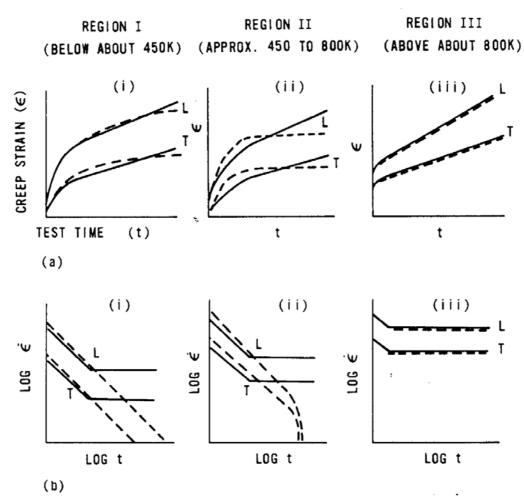
Figure 1-16:   Schematic representation of the effect of flux on the creep of zirconium alloys at constant stress. The solid lines are for irradiation creep, Fidleris V., 1988.

1.2.3.6 Alloys

As discussed in the literature review sections below and in the ZIRAT5 and ZIRAT6 annual reviews, Nb-containing zirconium alloys appear to have higher creep strength than standard Zircaloys. One-to-one comparison, however, is rare.

1.2.3.7 Recent In-Reactor Data

1.2.3.7.1 Zircaloy Pressurized Tubes

It is common to express irradiation creep for a particular material as

$$\dot{\varepsilon} = A(\Phi t)^m \, \sigma^n \, e^{-\frac{Q}{RT}}$$

where the symbols have the standard meanings. A recent experimental program, Soniak A. et al., 2002, produced data in a test reactor using internally pressurized tubes stressed between 50-122 MPa (9-18 ksi) at irradiation temperatures between 593-633K (320-380°C) and neutron fluxes between 1.2-1.9x10^{18}n/m^2/s, E>1 MeV. (For comparison, typical BWR flux is 0.8 and PWR flux 1.0x10^{18} n/m^2/s, E>1MeV). For Zircaloy, they confirmed that creep strains were lower for recrystallized (RXA) as compared to cold worked stress relieved (SRA) materials. They also confirmed that creep strains were higher at higher temperatures. Interestingly they found that for a given time, creep strains were higher for lower fast neutron fluxes, indicating that effects of thermal creep are important in that temperatures range. They also found that steady state creep was not reached until at least 3x10^{25} n/m^2 (E>1MeV)(equivalent to a LWR burnup of about 15 MWd/KgU), if ever. They found that the stress dependency, n, was between 0.8 to 2.0, which appears to be intermediate between what is expected for irradiation creep (n=1) and thermal creep (n>2). It was also found that for both RXA and SRA Zircaloy the flux dependency, m, was between 0.4 to 0.7, which seems appropriate at these higher temperatures. More details of this experiment are given in the literature review sections below.

1.2.3.7.2 Symmetry

It is noteworthy that in Zircaloy there appears to be a difference in creep strength in tension and compression. Such an asymmetry is illustrated schematically in Figure 1-17. That creep strength is lower (creep strain higher) in tension than in compression has been shown by McGrath M. A., 2000 and by Garzarolli F. et. al., 1996 in Figure 1-18. This is somewhat surprising if a climb and glide mechanism is dominating, but is likely related to the existence of different active slip systems in tension and compression.
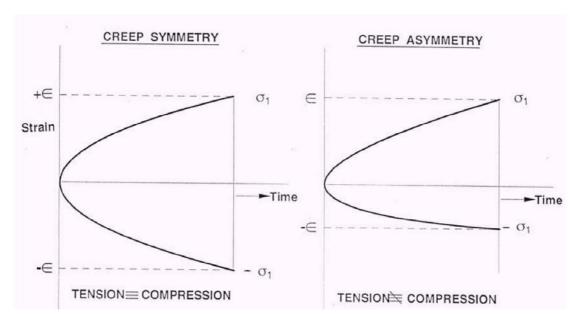
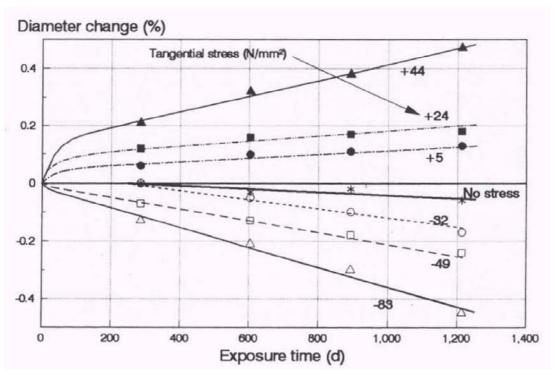Figure 1-17:   Difference between symmetric and asymmetric creep behaviour.

Figure 1-18:   In-reactor diameter change of tubular Zircaloy samples with different membrane stresses. Temperature is near 573K (300°C), Garzarolli F. et. al., 1996.

1.2.3.7.3 Hardening Rules

Hardening rules are a topic which is often discussed when considering empirical modelling of creep strain. Two types of hardening are illustrated in Figure 1-19: strain hardening and time hardening (which in the in-reactor case is also called fluence hardening).

Mathematically strain hardening is expressed

$$\dot{\varepsilon} = f(\sigma, T, \varepsilon)$$

and time hardening is

$$\dot{\varepsilon} = f(\sigma, T, t) \text{ or } \dot{\varepsilon} = f(\sigma, T, \Phi t)$$

where

$\dot{\varepsilon}$ = strain rate

= stress

T = temperature

ε = strain

t = time

Φt = neutron fluence

The rules apply when the applied stress changes during the test or during reactor operation, as in (a) of Figure 1-19. It is assumed that the creep curves for the various individual stresses are known, as in (c). The question is whether the strain response at a new stress will follow the creep curve for that new stress at the same value of time (fluence) or the same value of strain. Evaluation at the same value of strain is shown in (d) and (b). It is compared in (b) to choosing the same value of time. It is seen that the accumulated strains are different for the two different assumptions, and will be unless the creep curves are linear (steady state creep) in the entire range of stresses chosen. For the case of in-reactor irradiation it would seem that time or fluence hardening should be a good approximation at low fluences when the number of barriers to dislocations is still increasing with time or fluence. At fluences where the number of defects is saturated and the creep rate is linear, it would seem that either the time or strain hardening assumptions would be satisfactory. However, it is likely that in-reactor creep is influenced by irradiation-assisted and thermal creep phenomena, and therefore strain hardening might be more appropriate. The one in-reactor experiment that has thoroughly examined this question was conducted at Halden, McGrath M. A., 2000. They found that each stress change induced a primary creep component proportional to the magnitude of the stress increment, suggesting that a form of strain hardening was occurring. Knowing that either approach can be expected to be only a rough approximation, modellers must use best judgment in choosing their modelling approach.

2 LITERATURE (PETER RUDLING)

2.1 *CREEP*

As discussed in section 1.2, creep is a time dependent plastic deformation which occurs at any stress, even below the normal material yield stress. Figure 2-1 gives the familiar creep curve. Creep rate plotted as a function of time is given in Figure 2-2. Here it is seen more clearly that the creep rate changes with time, reaching a minimum, although not necessarily a constant rate in the secondary or steady state region II. In-reactor it is doubtful that a true steady state rate is ever reached. As stated nicely by Franklin D. G. et. al., 1983.

"If the same material is tested under different load and temperature conditions, sets of curves like the ones shown in Figure 2-3 and Figure 2-4 will be obtained. With either increasing temperature or applied stress, the time to rupture decreases, the total creep strain increases, and hence the creep rate increases. At very low temperatures or stresses, creep deformation is so small that it can be generally ignored in design considerations. At very high temperatures or stresses, deformation is so extensive and rupture times so short that behavior in this regime is of little technological interest. Consequently, the creep behavior of most structural materials is examined at intermediate temperatures, 30 to 75% of the absolute melting temperature of the material (T_m), and at stresses below the yield stress".

For nuclear components the temperatures are near 30% of T_m (T_4 and T_3) and the stresses low (σ_5 and σ_4). Figure 2-3 and Figure 2-4.
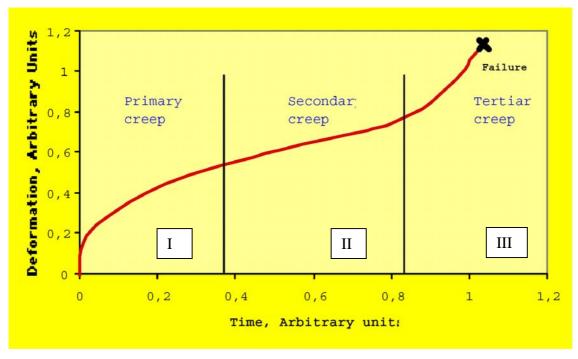
Figure 2-1:   The creep curve: variation of specimen strain with time.
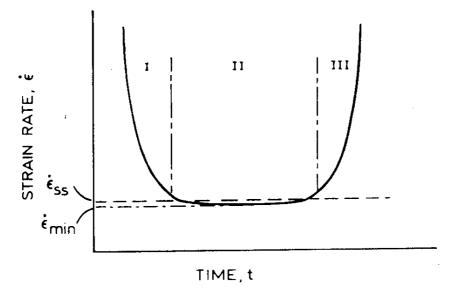
Figure 2-2:   The variation of creep strain rate with time.
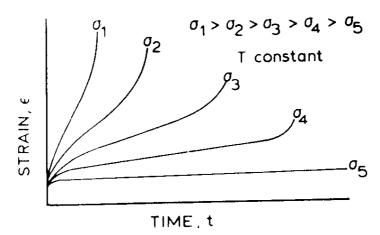
Figure 2-3:   Variation of creep behaviour with stress at constant temperature
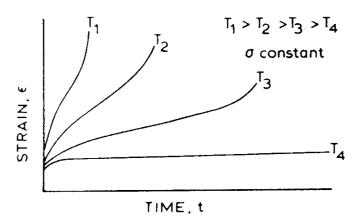
Figure 2-4:   Variation of creep behaviour with temperature at constant stress.

Stress relaxation has been shown to be related to creep in section 1.2. An alternate view of the process is as follows. A bar is stretched elastically to some length and then constrained to retain that length. The stress required would be constant if no creep or relaxation occurred, however, if it does occur some of the elastic strain in the rod is converted to plastic creep strain, and the stress required to maintain the rod length decreases, or relaxes. In reactor components, there are often cases where small residual stresses can be built up, for instance during welding or component straightening. During service, creep can occur which reduces the residual stress to near zero, at the expense of causing some local strain or dimensional instability. The stress relaxation process is schematically shown in Figure 2-5. The top figure on the left represents the initial unstressed material, the spring ("1") and the hydraulic cylinder ("2") represents the material elastic and plastic deformation, respectively.

3 ENGINEERING CORRELATIONS (RON ADAMSON)

3.1 BUNDLE PERFORMANCE

All of the zirconium alloy components of a fuel bundle are subject to dimensional instability. If the applied stress is zero, dimensions will change due to irradiation growth, hydrogen buildup, and corrosion. And if the stress is not zero but low, dimensions will change due to additional mechanisms such as creep and relaxation. If the stress is high, time independent plastic deformation will occur (a topic covered by the ZIRAT Special Topical Report on Mechanical Properties of Zirconium Alloys, Adamson R. B., and Rudling P., 2001. In order, then, to verify and predict bundle performance, measurements must be made of actual bundle components or special tests done on representative test materials. Knowledge of the mechanisms which affect the observed dimensional changes facilitates modeling and prediction of future behavior. This section describes such measurements and briefly reviews mechanisms.

3.1.1 Data Gathering Facilities

Interest in dimensional stability of the various reactor components at high burnup has led to more interest in measurement techniques to gauge the stability. Standard poolside and hot cell techniques still provide the primary data through measure of length, width, diameter, etc. using sophisticated "rulers". Recent open literature reports, however, have described techniques for getting on-line or poolside data of growth and creep. Since hydrides have been shown to influence dimensional stability as well as mechanical performance, efforts have also been directed at accurate means to measure oxide thickness from which hydride content can be estimated, and to measure hydrogen content non-destructively.

Recent reports give results from four test reactor programs. The Halden creep program made continuous diameter measurements on Zircaloy fuelled or non-fuelled tubing in water, and was able to measure strains when the hoop stress was increased, decreased or entirely reversed. The rig is shown in Figure 3-1. Measurement stability is affected by the many changes of power in the Halden reactor, but nevertheless it is believed that model-quality data was obtained. The French, Gilbon D., et. al., 2000 and Canadian, Causey A. R., et. al., 2000 conducted creep and growth experiments in the Siloe and Osiris test reactors in France. Pre- and post-irradiation length and diameter measurements of pre-pressurized tubes irradiated in sodium were made in a hot cell. Measurement errors were estimated to be less than $\pm 0.02\%$ strain, which is sufficient to obtain reliable data. Any many results have been reported from Russia involving the sodium-cooled fast reactor BOR60. Shishov, V. N., et. al., 2001, for example. These four experimental rigs produced data that was very complementary to more standard data from poolside exam of reactor components. The new test reactor data was the first to be gathered in more than a decade.

A fifth technique also operates in a test reactor, and has been tested for low fluences at Studvik's R2 reactor. It is designed to measure relaxation and creep at low stresses typical of guide tube creep. Strips 400 mm long are cut from actual guide tubes and loaded under a constant bending moment, as shown in Figure 3-2. Measurement of the strip curvature is done before and after irradiation, with "excellent consistency and precision", with the detection limit for creep deformation an order of magnitude lower than the total creep deformation. This technique is similar to the stress relaxation experiments conducted in Canada a few years ago, Causey A. R., et. al., 1984.
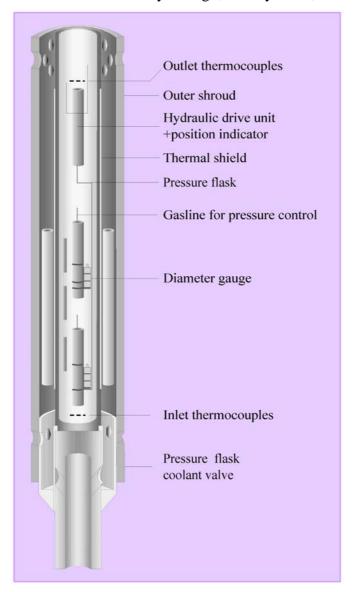

Figure 3-1:   Specially designed rig for creep testing of fuel cladding in a LWR loop under variable stress conditions, McGrath M. A., 2000.
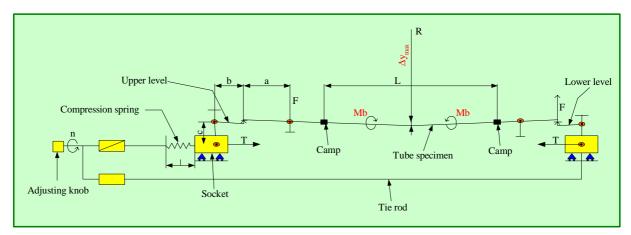
Figure 3-2:   Schematic of the bending creep rig, Tomani H., et. al., 2001.

A sixth new technique is described by Soniak A. et al., 2002, for use in the French Osiris test reactor. It is called Zircimog and is used to test irradiated cladding at high hoop stresses. The device allows irradiation to begin at zero stress, and then transition quickly to high stress by internal gas pressurization. It is used to test cladding under high strain rate power transient conditions. Clad diameter is measured continuously.

The seventh technique is unique in that the test rig operates in the core of a commercial reactor. Material test rods (MTR) are inserted inside control rod guide tubes of a standard PWR fuel assembly. The design is such that irradiation growth, circumferential (tangential) creep or axial creep can be measured, Figure 3-3. A schematic of an axial creep MTR is shown in Figure 3-4. The length of the specimens is measured prior to irradiation and after each cycle during the refuelling shutdown periods. Useful data has been obtained to fluences as high as 10×10^{25} n/m^2 (E>1 MeV).
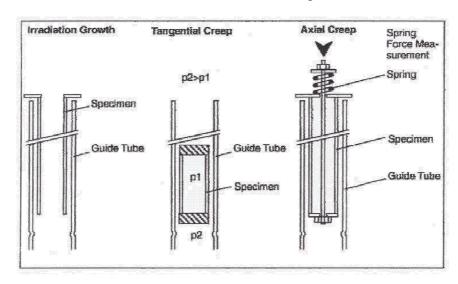

Figure 3-3:   Principle of growth and creep specimens, Hoffmann P. B., et. al., 2001.

4   SUMMARY

The broad topic of dimensional stability of reactor components has been reviewed. The change in dimensions of a component during reactor service has important implications to safety and operational issues. The report starts (Section 1) with a brief review of practical dimensional stability concerns, followed by perspective overview of the phenomenology of the various contributing factors to dimensional stability. It continues (Section 2) with a comprehensive review of the extensive literature available on the topic. The report concludes (Section 3) with an engineering assessment of observed dimensional changes of individual or integral reactor bundle components, based on available literature.

The primary sources of dimensional changes of reactor components (in addition to changes caused by conventional thermal expansion/contraction) are:

- irradiation growth

- irradiation creep

- thermal creep

- stress relaxation (which is a combination of irradiation and thermal creep)

- hydrogen and hydride formation

Irradiation effects are due primarily to the flow of irradiation-produced point defects to sinks such as grain boundaries, deformation-produced dislocations, irradiation-produced dislocations and dislocation loops, and alloying or impurity element complexes. Particularly in zirconium alloys, crystallographic and diffusional anisotropy are key elements in producing the observed dimensional changes.

Hydrogen effects are arguably independent of irradiation, although that is yet to be conclusively shown. Factors which influence corrosion and hydrogen pickup by the zirconium alloy drive the substantial dimensional changes that can be caused by hydrogen or hydrides.

Many factors affect the size and magnitude of dimensional changes, as illustrated in Table 4-1 and discussed in Sections 1, 2 and 3.

Table 4-1.   Parameters which affect dimensional changes caused by various mechanisms.

	Growth	Irradiation Creep	Thermal Creep	Hydride
• stress		√	√	?
• dislocation density, cold work	√	√	√	?
• fluence	√	√	√	
• flux		√		?
• temperature	√	√	√	
• texture	√	√	√	√
• metallurgical factors	√	√	√	√
• heat flux				?

Considerable effort in the past 30 years has resulted in a good empirical description of creep and growth; that is, predictions can be made of dimensional changes expected under various service conditions. Also good progress has been made, particularly in the last decade, in understanding the mechanisms involved. And in the past few years, an enlightened understanding of the effects of hydrides on dimensions has been attained. There are, however, several areas for which more data and understanding is required to assure optimum component performance at the high fluences and burnups currently achieved and proposed for modern fuel designs. These areas include but are certainly not limited to

- factors affecting formation of <c> component dislocations

- effects of alloying elements such as Nb, Fe and Sn and impurities such as C and S

- temperature dependence of growth and creep over the full range of operating temperatures

- long term effects of small residual stresses

- lack of comparable high fluence data for the variety of new alloys being introduced for corrosion resistance

- effects due the influence of heat treatment on microchemistry and texture

Active research and development programs continue in the important area of dimensional stability. Most technical conferences concerning nuclear materials have papers reporting new results. New developments will be followed by ZIRAT. It is anticipated that the current report will provide ample background to facilitate interpretation of future results.

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