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October 31, 2001

U.S. Nuclear Regulatory Commission
11555 Rockville Pike
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Attn: Document Control Desk

Subject: Submittal of the Revised Nonproprietary Version of the NAC International High Burnup Fuel Topical Report (790-TR-001-NP, Revision 1)

Docket No. 72-1015 and 72-1025

References:
1. Submittal of NAC International Topical Report 790-TR-001, Revision 0, "Requirements for Dry Storage of High Burnup Fuel," (Proprietary Version), NAC International, July 25, 2001
2. Submittal of Nonproprietary Version of the NAC International High Burnup Fuel Topical Report (790-TR-001-NP, Revision 0), NAC International, August 15, 2001
3. Public Disclosure Determination for NAC High Burnup Topical, U.S. Nuclear Regulatory Commission, September 13, 2001
4. Conference Calls - License to Disclose Bibliography in Appendix, NRC and NAC, October 12, 2001
5. Submittal of Revised Nonproprietary Version of the NAC International High Burnup Fuel Topical Report (790-TR-001-NP, Revision 0), NAC International, October 12, 2001
6. Submittal of NAC International Topical Report 790-TR-001, Revision 1, "Requirements for Dry Storage of High Burnup Fuel," (Proprietary Version), NAC International, October 31, 2001

NAC International (NAC) herewith submits five copies of the revised nonproprietary version of the NAC Topical Report, 790-TR-001-NP, Revision 1, "Requirements for Dry Storage of High Burnup Fuel." This submittal is the nonproprietary version of the Reference 6 Topical Report. The Appendix A, Bibliography, has been deleted and the Table of Contents and Page 1-2 have been revised to delete any reference to the Appendix to be consistent with the Reference 6 submittal.

If you have any comments or questions, please contact me on my direct line at (678) 328-1321.

Sincerely,

TC Thompson

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Director, Licensing
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Enclosures

NMSS01Public

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Cover, revision control sheet, pages 3 and 1-2	1	István Frankl Lester Goldstein Dr. John E. Harbottle October 2001	James Ballowe October 2001	Deleted Appendix A and all references to Appendix A in the main text of the report.

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EXECUTIVE SUMMARY

The overall objectives of this report are two-fold. The first objective is to develop and present a technically justified approach and methodology for predicting fuel clad behavior during dry storage. The second objective is to define and propose new technical criteria for the dry storage of commercial PWR and BWR spent nuclear fuel up to 55 GWD/MTU.

In accomplishing these objectives, this report first reviews the underlying limits and metallurgical processes used to establish the current U.S. Nuclear Regulatory Commission (NRC) guidelines (ISG-15) for the transportation and storage of spent fuel with burnups in excess of 45 GWD/MTU. Next, this report documents the pertinent results of an exhaustive effort to research relevant technical literature and experiments relating to the behavior of Zircaloy clad nuclear fuel under dry storage conditions. Finally, based on the conclusions dictated by the data, this report develops and benchmarks a method and computer code for the best estimate prediction of clad creep during dry storage.

Regarding the review of the NRC guidance, the report shows that, based on empirical data obtained and documented herein, the limits and acceptance criteria in ISG-15 are unnecessarily conservative. Consequently, NAC International (NAC) proposes and provides the justification for new limits and acceptance criteria based on actual mechanistic behavior of Zircaloy clad spent nuclear fuel under dry storage conditions. These new limits and acceptance criteria are:

1. The fuel integrity criteria during storage can be satisfied by a creep strain limit of 2.5%. There is no requirement to specify or characterize failure modes because this strain level is demonstrated to be reached without material degradation.

2. The oxide thickness acceptance limit on the fuel cladding for interim dry storage can be safely raised to 120μm. This value is consistent with volume average hydrogen concentrations in the cladding of up to 800 ppm.

3. The maximum cladding temperature during vacuum drying and helium backfill can be safely limited to 450°C. Peak clad temperatures up to this level, during the relatively short period of vacuum drying and helium backfill, are demonstrated to have negligible effects on both hydride reorientation and annealing of radiation hardening.

Creep and creep rupture data of hydrided and irradiated fuel cladding presented in this document clearly demonstrate that Zircaloy can sustain uniform plastic strains in excess of 2.5% without failing or entering the tertiary creep stage. Further, this report provides a sound technical justification to illustrate that diffusion-controlled cavity growth (DCCG), which underlies the current NRC approach, is overly conservative when applied to thermal creep of cladding. DCCG predicts continuous material degradation during creep that is not supported experimentally for irradiated Zircaloy. This report also establishes the foundations for a conservative but practical creep strain limit approach to replace the creep failure philosophy in ISG-15.

This report also presents convincing evidence demonstrating that the current perception of ISG-15 that the hydrogen levels should not exceed ~400 ppm (to avoid affecting the strength and ductility of the cladding) is not supported by the body of evidence and is ultraconservative. This report points out, using relevant experimental data, that the strength of Zircaloy is insensitive to hydrogen concentrations well in excess of 1000 ppm for all temperatures in the range 20-430°C. The report also supports the conclusion that a uniform plastic elongation (ductility) of at least 5% is still retained at hydrogen levels up to at least 800 ppm for temperatures above 150°C. Together with the creep data, this provides a sound justification for raising the hydrogen limit to 800 ppm. This hydrogen limit is equivalent to an oxide thickness in excess of 130μm. Thus, allowing for uncertainties in measurement, a revised acceptance limit on clad oxide thickness of 120μm is justified and proposed.

By implementing the recommended dual limits of 800 ppm of hydrogen and 120μm of oxide, all currently discharged sound fuel in the U.S. would qualify for dry storage without restriction.

Other technical issues reviewed in this report include: oxide spalling (leading to localized hydride concentrations); annealing and recovery effects during vacuum drying of the storage cask and fuel; radial hydride reorientation as a result of annealing and then cooling of the fuel rods; and finally, creep modeling.

The thermal creep methodology developed by NAC and presented herein is centered upon a [redacted]. The selection of this approach was based upon a comprehensive review and comparison of published creep models and supporting data. The [redacted] equation, [redacted] is applicable to [redacted] cladding over a broad range of temperature and hoop stresses. Essentially all Zircaloy cladding on [redacted]. The NAC methodology and computer code, termed DSCREEP, was successfully benchmarked against the measured data used to formulate the semi-empirical equation and its solution technique.

Another reason for [redacted] dry storage creep model was that the researchers also performed a number of *irradiated* creep experiments on [redacted]. These irradiated test specimens had burnups [redacted]

The NAC computer code DSCREEP [redacted] used in a conservative manner to determine the thermal creep characteristics of high burnup fuel [redacted] cladding in dry storage in an NAC-UMS® cask. The analysis considered various prior pool residence time/initial clad temperature combinations. The results show that the proposed 2.5% total strain limit can be met easily [redacted] In fact, the current 1% total strain limit can be met with maximum cladding temperatures [redacted], respectively, [redacted]. These temperatures correspond [redacted]. NAC has factored numerous conservatisms into the DSCREEP methodology. A listing of many of the conservatisms used in the DSCREEP analysis/input development is presented in ***Table EX-1***.

[redacted]

[redacted]

[redacted]

Supplemental analyses indicate that the drying process prior to cask storage can have an effect on cladding hoop stresses in dry storage and the saturated strain. The magnitude of the effect will depend upon the time and temperature levels used in the drying operation.

To ensure no detrimental effects on the fuel cladding during loading operations, NAC proposes that the maximum cladding temperature during vacuum drying and helium backfill should be maintained below 450°C.

This temperature limit should be reflected in cask operating procedures and Technical Specification allowable limits.

Table EX-1: Conservative/Bounding Assumptions Used in DSCREEP Analysis/Input Development

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

1. INTRODUCTION

NRC recently issued ISG-15, "Materials Evaluation" [1]. This document deals with material-related issues related to spent nuclear fuel dry storage systems and transportation packages. ISG-15 incorporates and supersedes the information in ISG-11, Revision 1, "Transportation and Storage of Spent Fuel Having Burnups in Excess of 45 GWD/MTU," and thus sets the current acceptance criteria that must be met for storage of fuel with burnups in excess of 45 GWD/MTU. For long-term dry storage, the criteria are aimed at ensuring fuel rod integrity, i.e., avoidance of gross rupture due to possible materials "degradation" under conditions existing within the cask. The criteria are based on several perceived technological limits and metallurgical processes involving irradiated Zircaloy fuel cladding.

This NAC report reviews the NRC criteria and underlying limits and processes with the objective of demonstrating that they are overly conservative, unrealistic and should be relaxed or replaced with new limits, *which still meet the primary objective of ensuring fuel rod integrity*. In this way, the case is developed and proposed for more practical and realistic acceptance criteria than those described within ISG-15. The report then develops and technically justifies these new proposed criteria and limits.

ISG-15, under Section X.5.4.2, High Burnup Fuel, refers to four aspects of Zircaloy performance and behavior which, in the opinion of the NRC and its consultants at Pacific Northwest National Laboratory (PNNL), are limiting. These four aspects include: uniform creep strain; use of Commercial Spent Fuel Management (CSFM) methodology; hydrogen concentration; and oxide thickness. Each of these aspects is addressed individually in this report. For each, NAC presents recent and/or relevant test, experimental and scientific information demonstrating expected fuel cladding behavior with respect to each of the four aspects. Based on the evidence presented for each aspect, justifiable conclusions are developed identifying acceptance criteria and limits that are adequate, conservative and practical. These aspects are addressed in Section 2 of this report.

In addition, the NRC has identified other near-term, high burnup storage 'issues' requiring further investigation. In this report, we research these issues and provide specific well justified technical approaches fully addressing these issues. This information is presented in Section 3 of this report.

Section 4 of this report describes and explains the creep modeling methodology and the NAC proprietary computer program (DSCREEP) used to evaluate the acceptability of storing high burnup fuel in an NAC-UMS® cask. This section concludes with an NAC-UMS® storage cask specific analysis that considers fuel at a burnup of 55 GWD/MTU.

Section 5 of this report contains a summary of the conclusion and recommendations based on the investigation, evaluation and analysis of the information presented in the preceeding sections.

Section 6 contains a listing of reference material directly supporting information presented in Sections 1 through 5 of the report.

2. CURRENT GUIDELINES

2.1 MAXIMUM UNIFORM DIAMETRAL CREEP STRAIN

NRC guidelines (ISG-15) assume that the maximum increase in rod diameter due to thermal creep of the cladding during storage life should be 1%. This 1% creep strain is considered to be the limit of *uniform* plastic strain of the irradiated clad wall in the presence of not more than 400-500 ppm of hydrogen. The NRC guidelines imply that beyond these strain and hydrogen values that non-uniform straining will occur, leading to a thinned or 'necked' region and perforation of the wall. ISG-15 then argues that cladding perforation is a loss of fuel rod integrity, even though it may be a pinhole leak and not a gross rupture. The theoretical basis for the ISG-15 limits on creep strain is the mechanistic description of diffusion controlled cavity growth (DCCG) within the Commercial Spent Fuel Management (CSFM) methodology for calculating cladding temperature limits [2]. A detailed discussion of DCCG with reference to thermal creep is presented in Section 2.2.

It is understood that the NRC guidance in ISG-15 must, by nature, be conservative. To this end, and in the absence of conflicting information, the NRC adopted known conservative limits and endorsed the 1% strain and 400 ppm hydrogen limits promulgated by PNNL. However, there is now experimental evidence to demonstrate that irradiated and hydrided Zircaloy cladding can sustain uniform diametral thermal creep strains of at least 2.5% without failing or entering a tertiary creep stage. Several examples are presented below to justify this conclusion.

The reader should note that diametral and circumferential strains are equivalent and the terms can be used interchangeably.

French Data [3-4]

Recent French research and testing has contributed significantly to the body of evidence related to Zircaloy cladding creep characteristics. ***Figure 2-1*** is from the joint CEA-EdF-Framatome program [3] [redacted]

[redacted]

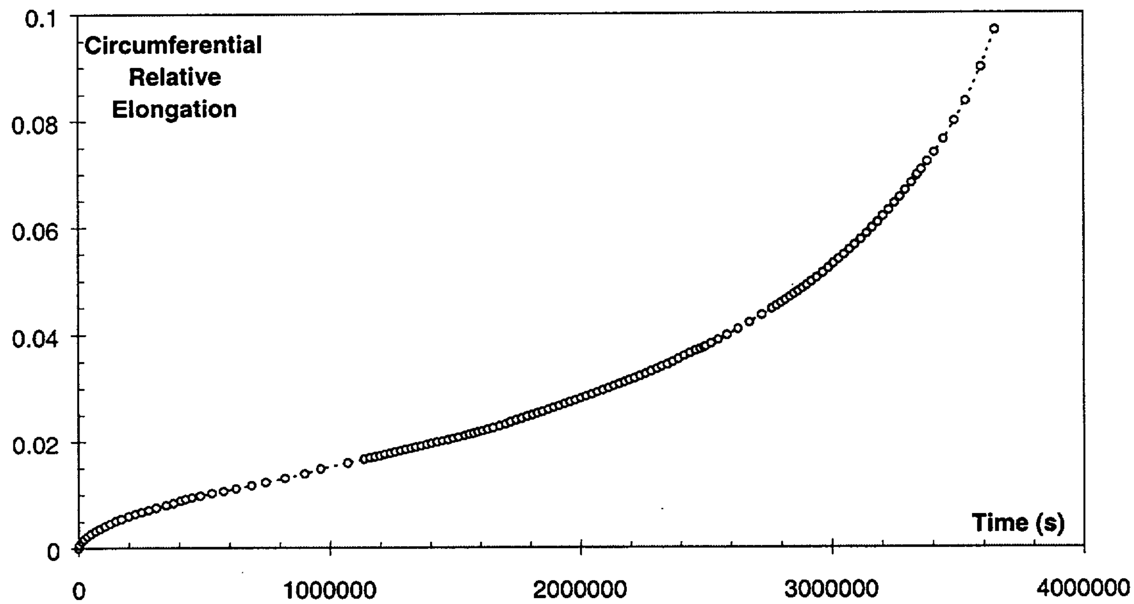


Figure 2-1: Creep rupture test at T = 420°C and Stress = 226 MPa. Test time = 42 days. Irradiated CWSR Zr –4 (4 cycles PWR; BU~47.2 GWD/MTU). H content ~200 ppm. [3]

Other data from the same program on *unirradiated* material support [redacted] This is illustrated in ***Figure 2-2*** and ***Figure 2-3***.

Figure 2-2 depicts time to rupture for Zircaloy cladding with hydrogen concentrations in the range of 0 – 360 ppm and with an axial stress of 350 MPa at a temperature of 400°C.

Figure 2-3 shows biaxial creep time to rupture data for Zircaloy cladding with a hydrogen content of 1040 ppm with a stress of 386 MPa at a temperature of 350°C.

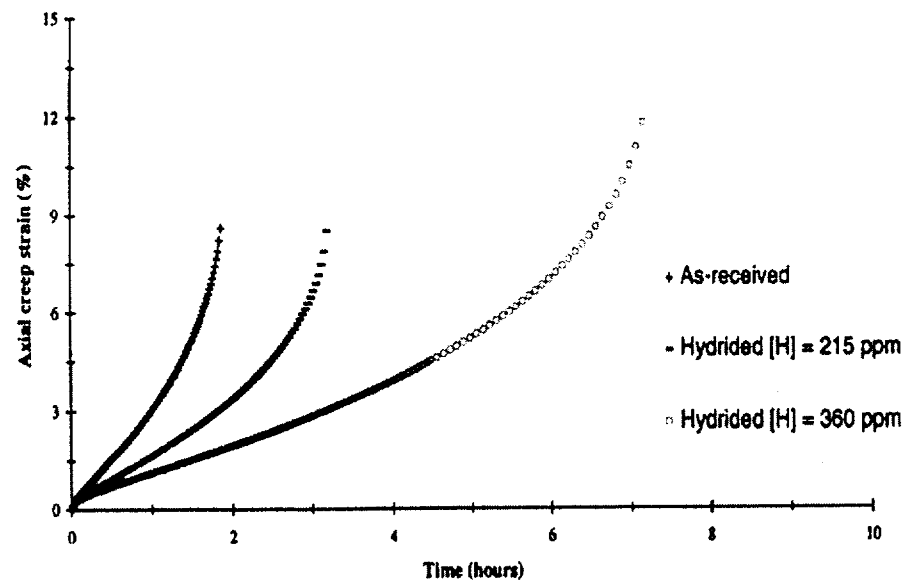
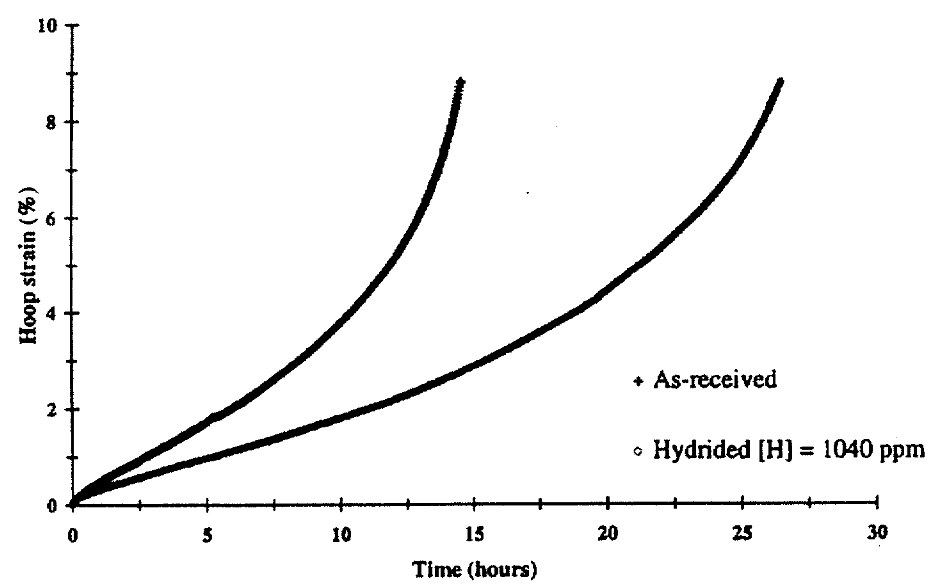


Figure 2-2: **Uniaxial creep to rupture of unirradiated CWSR Zr-4. Hydrided vs. non-hydrided. Stress = 350 MPa. T = 400°C.**



[4]

Figure 2-3: **Biaxial creep to rupture of unirradiated Zr-4. Hydrided vs. non-hydrided. Stress = 386 MPa. T = 350°C. [4]**

The key observations and conclusions that can be reached from ***Figures 2-1*** to ***2-3*** are:

[redacted]

German Data [5]

The results of short time creep rupture and creep ductility tests have recently been reported by German researchers. These tests were designed to assess the strain potential under conditions of dry storage of Zircaloy cladding extracted from high burnup rods [5]. A graph depicting the German creep test program is shown in ***Figure 2-4***. The test samples were partially recrystallized Zr-4, of low tin composition (Sn ~1.29 wt%), taken from fuel rods with burnups between 54-64 GWD/MTU. The peak oxide thickness range was 40 – 100 μm, so the equivalent hydrogen concentrations would have been within the range of 200 – 600 ppm.

The creep tests were carried out in two stages. High temperature tests were performed at 300°C (573K) and 370°C (643K) at hoop stresses of about 400 and 600 MPa and were designed to achieve ~2.5% strain in 3-4 days. These were followed by lower temperature tests at 150°C (423K) and 100 MPa to assess the affect of hydrogen/hydrides on creep ductility and to simulate the cooling stage under dry storage. The results of the high temperature creep tests are shown in ***Figure 2-5***. Uniform plastic hoop strain is plotted against test time (for the nonfailed specimens) or time to failure, for the specimens that failed.

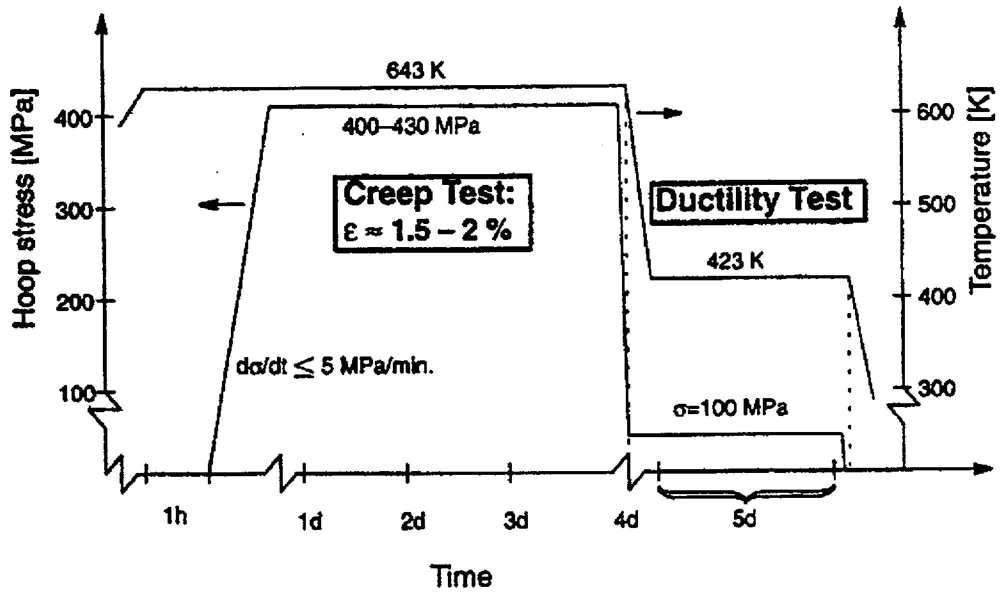


Figure 2-4: **Testing sequence [5]**

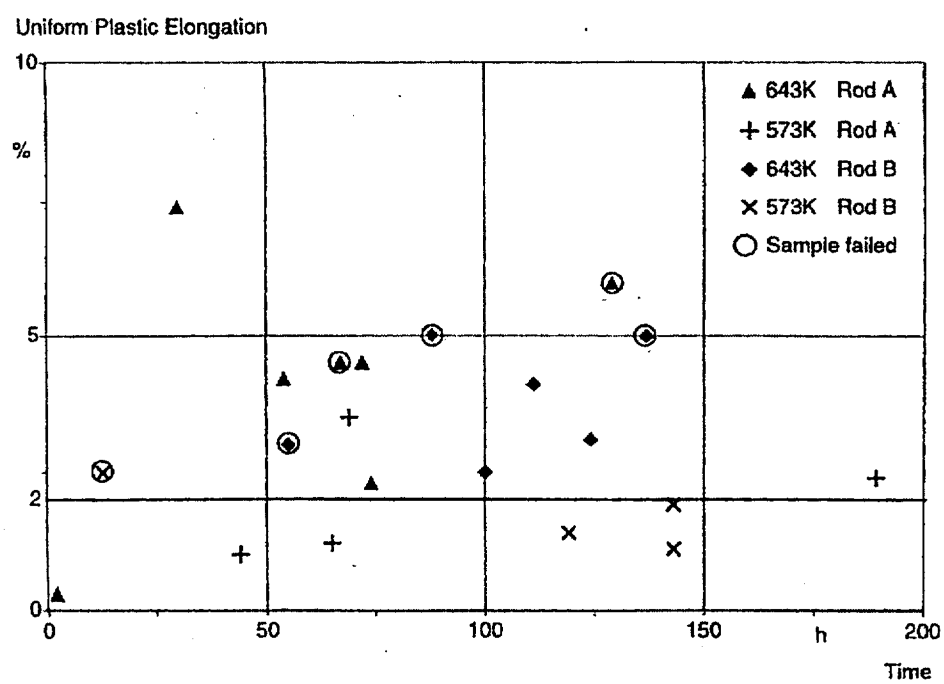


Figure 2-5: **High temperature (300°C and 370°C) creep-time data. Rod A = 54 GWD/MTU and Rod B = 64 GWD/MTU [5]**

The key observations from this program are:

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

Japanese Data [*6*]

Japanese researchers have also performed creep testing on irradiated CWSR Zr-4 and RXA Zr-2 cladding. In these tests, samples were taken from irradiated PWR (BU=46 GWD/MTU) and BWR (BU not given) fuel assemblies, respectively. They were creep-tested over a range of stresses and temperatures in order to determine safe creep strain limits for long-term dry storage. A graph showing results of steady state creep vs. time testing is shown in ***Figure 2-6***. As this graph illustrates, nontertiary creep strains of 2-3% were achieved without failure after 500 hours at a stress of 154 MPA and at a temperature of 420°C. As with the French and German data above, many of the Japanese accelerated creep rupture test conditions represented higher stresses (>200 MPa) than those occurring in a spent fuel rod. The cumulative results of rupture strain vs. hoop stress are presented in ***Figure 2-7***. It should be noted that all strains at failure are at least 3%. The strain at rupture for these tests is conservative when compared with longer-term tests at lower hoop stresses. This is supported by the results at 77 MPa and 600°C (873K) in ***Figure 2-7*** where a strain of more than 15% was reached before failure.

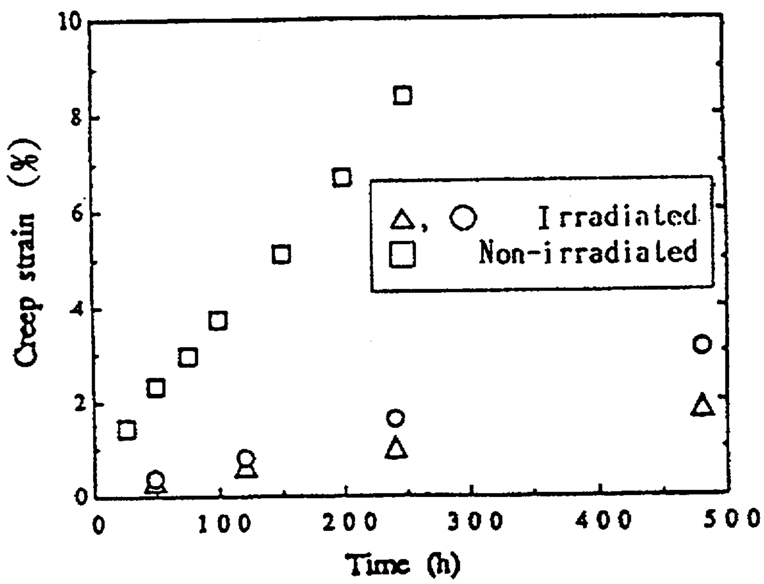


Figure 2-6: Results of creep tests on irradiated and nonirradiated RXA Zr-2 fuel cladding samples. Temperature = 420°C; hoop stress = 154 MPa [6]

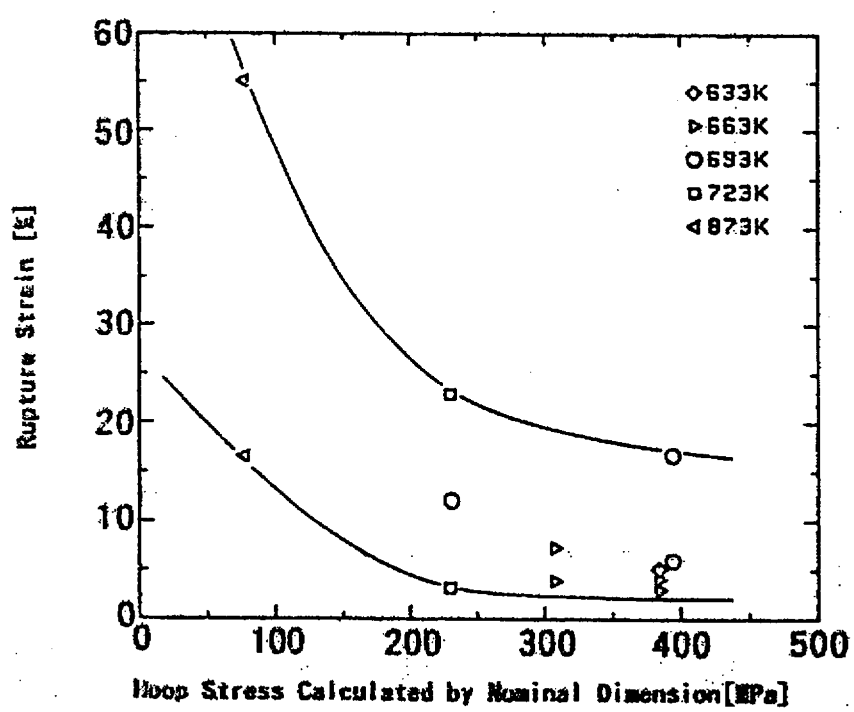


Figure 2-7: Creep rupture results on irradiated CWSR Zr-4 (BU=46 GWD/MTU) [24]

The key observations from this program are:

[redacted]

[redacted]

Conclusions

The data presented above all consistently verify that irradiated and/or hydrided Zircaloy cladding can sustain *uniform* plastic creep strains of >2.5% without failure. This is particularly germane to the confirmation of spent fuel cladding integrity during long-term dry storage. In the experimental creep tests data presented herein, it can be concluded that it does not matter how quickly the creep strain is accumulated, as long as it can be shown that failure *by any mechanism* does not occur for strains less than 2.5%. It follows that the accelerated creep test results in references [3-6], as well as the rupture tests presented above, are valid data for this purpose. In fact, the high stress, high creep rate tests are conservative when compared to long-term creep at lower stresses. In the latter case, Zircaloy cladding material is similar to many other materials in supporting larger *uniform* creep strains before the onset of tertiary creep. Consequently, this supports a creep strain limit of >2.5% without the need to consider the mode of failure.

Based on the data referenced and presented above, and taking into account the data and discussion in Sections 2.2 through 2.4, NAC proposes a technically justifiable and safe life-time creep strain limit of 2.5%. This proposed new strain limit has been demonstrated to be well within the uniform creep capability of irradiated Zr-2 and Zr-4 cladding containing hydrogen up to at least 800 ppm (oxide thickness equivalent =120μm).

2.2 CREEP MODELS, LIMITS AND THE VALIDITY OF CSFM

It is generally accepted that thermal creep of fuel cladding under high internal gas pressure is the limiting governing process for fuel rod integrity during long-term dry storage. However, the specific mechanism by which cladding perforation or rupture finally occurs is not firmly established. Currently the maximum allowable temperatures for dry storage are determined using either the equations developed by Lawrence Livermore National Laboratory (LLNL) [7] or temperature limit curves using a life-fraction rule within the CSFM methodology developed by PNNL [2]. Both methods are founded on the *same* basic assumption that creep failure/rupture of the cladding occurs by DCCG during grain boundary sliding. However, because cavities nucleate and grow through all stages of creep in various metals (but not necessarily Zircaloy), DCCG is *not* a function of creep strain. This would not then be compatible with a creep strain limit, assuming that fracture only occurs during tertiary creep.

However, the DCCG model has been recently updated to include Zircaloy properties [8]. It can be concluded from this recent LLNL work that it is not technically justifiable to expect DCCG to be a cladding failure mechanism for any reasonable dry storage temperatures. In addition, for Zircaloy in the CWSR final state, there are no clearly defined grain boundaries that are physical prerequisites for DCCG to occur. This applies to all Zr-4 used for PWR cladding and for a small fraction of Zr-2 supplied by Siemens Power Corporation (SPC) for BWR fuel mainly in the United States. The remainder of BWR cladding is all recrystallized annealed (RXA) and, therefore, has well-defined grain boundaries. Nevertheless, the creep of even RXA material shows no evidence of DCCG behavior leading to low failure strains.

This information clearly demonstrates that DCCG is not an appropriate material degradation mechanism for Zircaloy cladding in dry storage. Neither can it be justifiably used as the basis for a creep-driven clad deformation model. Consequently, these facts:

[redacted]

[redacted]

Conclusions

The DCCG mechanism has been shown not to apply to the material degradation process during creep. Therefore, it can be concluded that the replacement of DCCG by a *non-failing* creep strain limit approach removes the need to consider failure characterization (i.e., pinhole, narrow axial split or gross rupture) in stored fuel rods at any level of burnup. This conclusion applies to all Zr-clad fuel designs and all discharge burnups. The emphasis can now be placed only on creep and temperature modeling to ensure that a 2.5% lifetime creep strain of the cladding circumference is not exceeded.

2.3 THE EFFECT OF HYDROGEN ON THE STRENGTH AND DUCTILITY OF IRRADIATED ZIRCALOY

One of the factors in ISG-15 underpinning the current hydrogen concentration limit of ~400 ppm is the perception that the strength and ductility of irradiated Zircaloy remains largely unaffected by hydrogen and hydrides *only* at levels below this concentration. Our research demonstrates that higher limits (800 ppm) should be used and that 400 ppm is unduly conservative. This report includes the relevant data to justify this position.

a) Strength

There is considerable data to show that the yield strength (YS) and ultimate tensile strength (UTS) of unirradiated and irradiated Zircaloy remain largely insensitive to hydrogen concentration levels up to at least 1000 ppm. Some recently published data by Wisner and Adamson [9] shown in ***Figure 2-8*** proves the point for RXA Zr-2 tubing, preirradiated in a commercial BWR over a range of equivalent burnups to a maximum of ~52 GWD/MTU. Other measurements [10] of yield stress on unirradiated material (***Figure 2-9***) confirm a similar insensitivity to hydrogen for CWSR Zr-4 tubing used in PWR fuel rods.

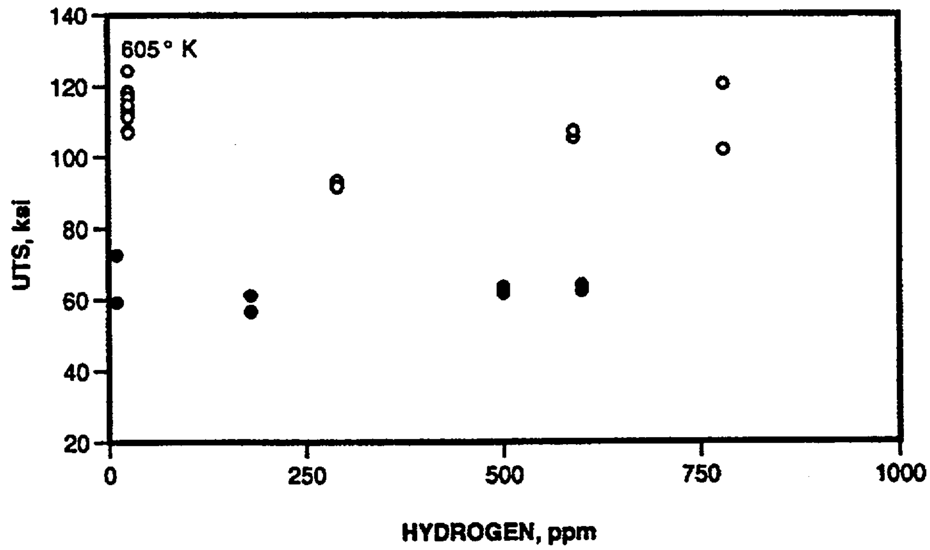


Figure 2-8: Ultimate tensile strength (UTS) vs. hydrogen content for unirradiated and irradiated Zr-2 tested at 332°C (605K). Data points: irradiated (open circles); unirradiated (closed circles) [9].

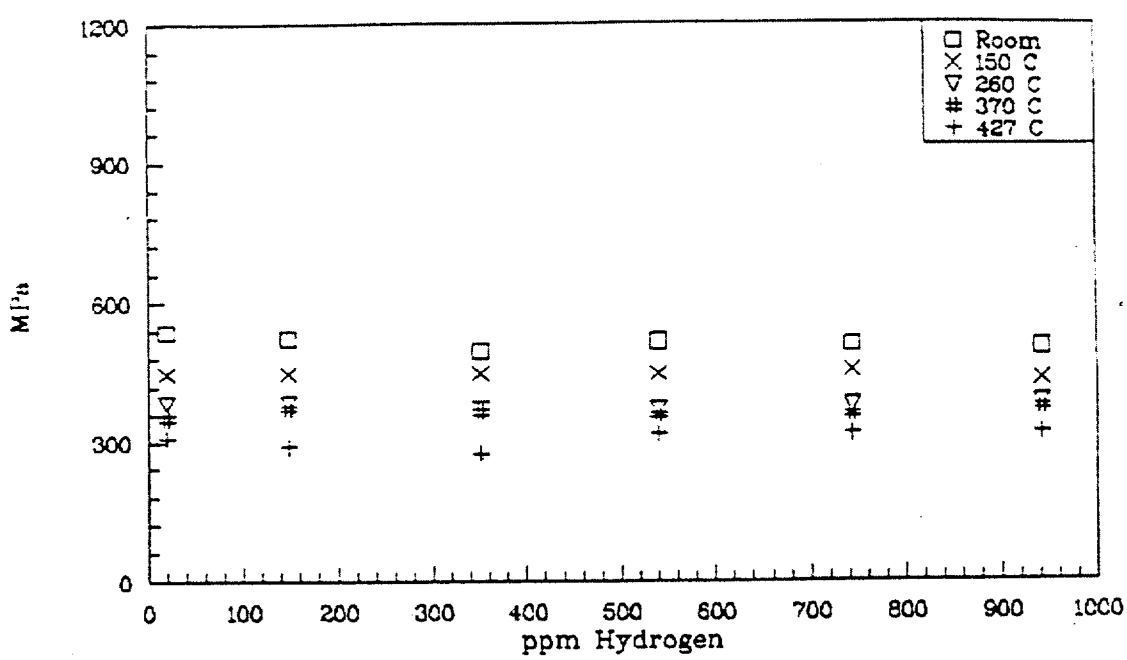


Figure 2-9: Yield strength of CWSR Zr-4 tubing vs. hydrogen content at various test temperatures in the range 20-427°C [10].

b) Ductility

Ductility is generally measured in one of three ways: uniform plastic elongation strain; total elongation strain and reduction in area. All of these are expressed as a fraction or percentage change. Of these three, uniform plastic elongation strain has been demonstrated (see ***Figure 2-10***) to be the most conservative, or limiting, measurement.

In their recent work, Wisner and Adamson [9] used the localized ductility arc specimen technique to provide a realistic comparison between test variables as a function of hydrogen concentration. Measurements at 332°C (605K) compare unirradiated with preirradiated Zr-2 samples, as shown in ***Figure 2-10***. These are the same material conditions as used for the strength measurements in ***Figure 2-8***. After tensile testing to failure, the fracture surfaces of the test specimens were examined metallographically for evidence of ductility or brittleness. The results and conclusions for the tests conducted at 332°C are summarized as follows:

[redacted]

[redacted]

[redacted]

[redacted]

Other ductility data from Sweden [11] on unirradiated Zr-2 shown in ***Figure 2-11*** confirm that the *total* strain to failure is also largely independent of hydrogen content for measurements up to ~1500 ppm. This is valid for the test temperatures of 25-300°C in this study. However, this trend will continue to higher temperatures as the hydrides dissolve into the Zircaloy matrix and overall ductility increases.

Conclusions

The data presented above on the influence of hydrogen on the tensile properties of Zircaloy show convincingly that:

a) There is no distinction between hydrogen influence on Zr-2 and Zr-4. For these alloys, the influence of hydrogen is germane to the zirconium. These results are independent of the fabrication process, final heat temperature and direction of testing.

b) The ductility of irradiated Zircaloy tubing, measured as uniform plastic elongation, is unaffected by hydrogen and precipitated hydrides for concentrations up to at least 800 ppm. The uniform elongation strain at this concentration is ~5%. It should be noted that uniform elongation strain is the most conservative or limiting measurement of ductility.

c) Hydrogen concentrations up to at least 1500 ppm do not markedly affect the ductility of unirradiated Zircaloy.

d) The strength of irradiated and unirradiated Zircaloy tubing is unaffected by hydrogen and precipitated hydrides for concentrations well in excess of 1000 ppm. This observation is valid for a wide range of temperatures from 20 - 430°C.

e) Over the temperature range of practical interest for dry storage, there is no limitation on the yield strength, ultimate tensile strength and ductility of the cladding for volume-average hydrogen levels up to at least 800 ppm.

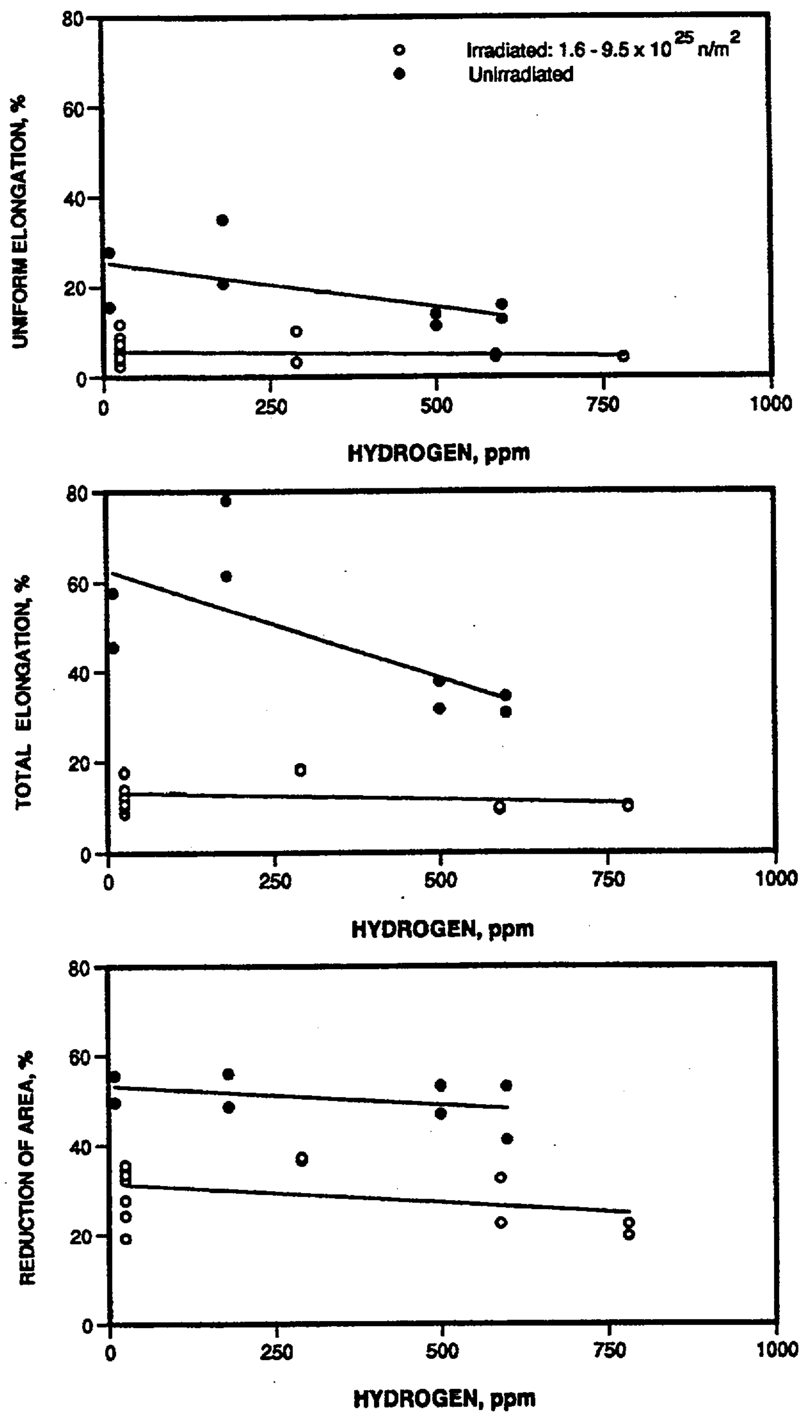


Figure 2-10: Measurements of ductility as a function of hydrogen content for unirradiated and irradiated Zr-2 tested at 332°C (605K). [9].

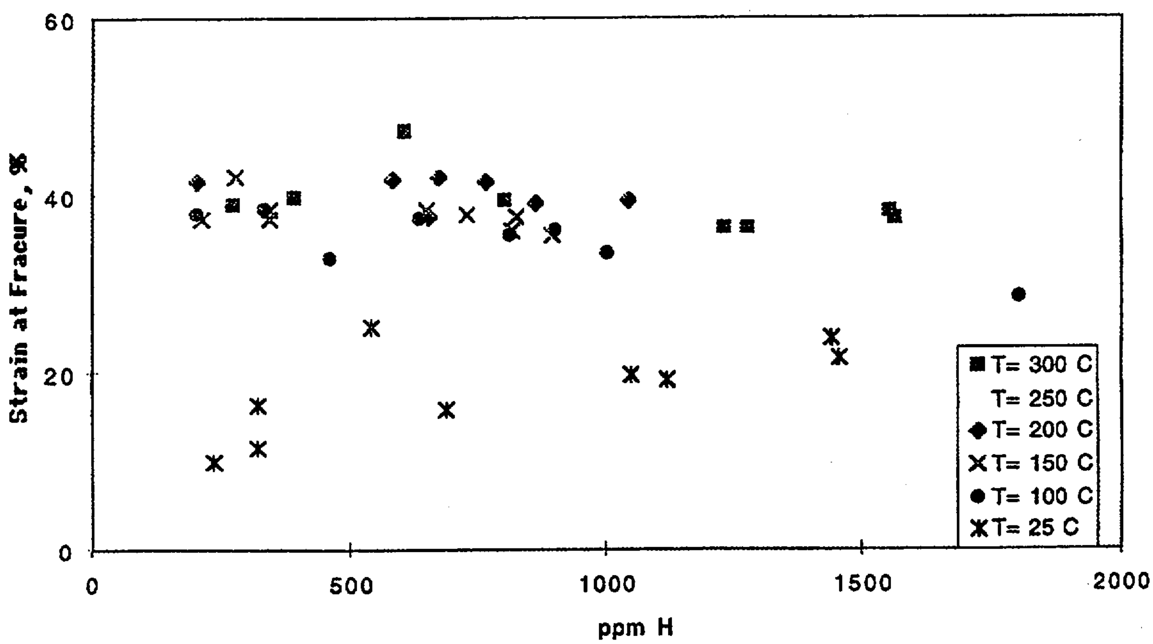


Figure 2-11: **Measurements of total strain ductility as a function of hydrogen content for unirradiated Zr-2 tested at constant temperatures over the range 25 - 300°C. [11]**

2.4 HYDROGEN, OXIDE THICKNESS AND BURNUP LIMITS FOR THE CLADDING

One of the acceptance criteria in ISG-15 states that the majority (97%) of high burnup (>45 GWD/MTU) rods should have a peak oxide thickness <70μm to qualify for long-term dry storage. This conservative peak oxide value effectively sets an unspecified upper limit on burnup. The basic reason for the technological limits underpinning the criterion is the perceived effect of the accumulated hydrogen on the fuel cladding properties.

The hydrogen is generated from the Zr-water corrosion reaction:

$$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2$$

About 15% of the hydrogen is absorbed directly into the cladding [12,13], where it precipitates as hydrides at any concentration above the temperature-dependent solubility limit (see ***Figure 3-2***). Because hydrogen concentrations cannot be measured directly in a fuel rod without destructive techniques, a correlation with oxide thickness is employed, which is measurable.

Previous sections (Sections 2.1 through 2.3) of this report have provided convincing data to demonstrate that hydrogen concentrations up to at least 800 ppm do not adversely affect either the long-term creep or the mechanical properties of irradiated Zircaloy cladding. This level of hydrogen concentration corresponds to a uniform oxide thickness of over 130μm, using the pickup fraction of 15% accepted by the NRC in [1]. On the other hand, the maximum clad oxide thickness for LWR fuel is limited to 100μm by current operational and regulatory practices. Instances where this value has been exceeded are very rare and have generally been restricted to only a few rods in the core. The cause of excessive end-of-life corrosion greater than 100μm is usually linked to nonoptimized water chemistry and/or crud deposition onto the fuel.

Conclusions

The evidence presented in this report clearly demonstrates that:

a) The oxide thickness acceptance limit for interim dry storage can be safely raised to 120μm.

b) Spent PWR fuel is currently available with leading assembly average burnups at discharge of ~55 GWD/MTU. At this burnup, the peak *rod* oxide thickness could be up to ~100μm, equivalent to a volume-averaged hydrogen content of around 600 ppm in the cladding.

c) BWR bundle average burnups tend to be less than those of PWR assemblies and also the cladding temperature is limited to <290°C by surface boiling. Although the maximum oxide thickness for a BWR rod can be up to ~85μm for older design cladding, the data for modern cladding indicates almost half this value at peak rod burnups of ~55 GWD/MTU.

d) With the recommended dual limits of 800 ppm of hydrogen and 120μm of oxide, all sound fuel currently discharged would qualify for dry storage without restriction.

3. OTHER ISSUES

3.1 LOCALIZED HYDRIDE CONCENTRATIONS

Hydrogen enters and migrates throughout the fuel cladding by thermal diffusion. The final hydride distribution is particularly influenced by temperature gradients and, as a result, the highest concentrations are found in the coolest regions. In most fuel rods these are near the outer wall, at pellet-pellet gaps and close to the bottom end plug (mainly BWR rods). They all give rise to nonhomogeneous hydride distributions in both the radial and axial directions. The outer wall effect leads to a "rim" of dense hydrides often observed in the metalographic sections of a high burnup rod. High duty PWR rods are more prone to localized hydride concentrations than BWR rods because of the higher radial temperature gradients arising from the heat flux.

Oxide spalling during reactor operation also creates a relative cool spot on the outer wall of the cladding. It occurs occasionally on PWR assemblies but is not generally seen in BWRs. Free hydrogen from the corrosion reaction may nucleate and grow a very dense hydride "lens", or blister that can penetrate into the cladding wall and consume Zr metal (up to about 50% of the wall thickness in the worst cases). Oxide spalling is relatively rare but is observed occasionally during poolside inspection of PWR assemblies having oxide thickness in excess of ~80-90 μm. In these situations, it is usually exacerbated by high heat fluxes in highly rated PWRs.

The importance of considering localized hydride concentrations is that they may behave as sites of embrittlement, or reduced ductility during creep, and could assist the nucleation of outer clad surface cracks. In this way, localized hydrides could potentially reduce the failure threshold to below the recommended creep strain limit of 2.5%. However, this has been demonstrated to be a low risk, based on the creep and mechanical property tensile test results that have been carried out on ex-reactor fuel cladding. Some of this data is referenced in Sections 2.1 and 2.3. Also, severe oxide spalling leading to hydride lens formation is rarely observed. There is no evidence to date that ex-reactor cladding samples, with non-homogeneous hydride distributions, show inferior creep behavior (see, for example, [3] and [5] and ***Table 4-2***). These samples all had a higher concentration of hydrogen close to the outer cladding surface but did not fail prematurely during creep.

Conclusions

From the information presented in this section, it can be concluded that:

a) Oxide spalling occurs occasionally on PWR assemblies but is not generally seen in BWRs.

b) The presence of localized hydride concentrations has been demonstrated to be a low risk to clad failure, based on the creep and mechanical property tensile test results that have been carried out on ex-reactor fuel cladding.

c) There is no evidence to date that ex-reactor cladding samples, with nonhomogeneous hydride distributions, show inferior creep behavior.

3.2 CASK DRYING AND ANNEALING EFFECTS

In preparation for normal operation, storage and/or transport, spent fuel canisters or casks are typically vacuum dried and then backfilled with inert gas, typically helium. This is normally performed in the transfer cask used to move the spent fuel from the pool area to the dry storage cask. The duration of the combined vacuum drying and gas backfill processes varies [redacted].

During the drying sequence prior to canister placement in the storage cask, the spent fuel reaches temperatures that exceed anticipated temperatures in any other normal storage condition. The maximum temperatures are achieved first because of the vacuum (i.e. a lack of heat conducting medium) and second because of the relatively poor heat removal characteristics of the canister and transfer cask in the vertical orientation. Once the canister is removed from the transfer cask and placed within the storage overpack, the fuel and fuel cladding temperatures immediately decrease because of greatly improved heat removal capability.

This cycling of the fuel temperature during the vacuum drying and helium backfill processes can effectively produce a heat treatment of the fuel cladding. This heat treatment may be sufficient to cause some annealing of irradiation defects and produce re-solution of some hydrides in the cladding. In general, the effects of this higher temperature transient phase (as a consequence of the drying process) are summarized below.

- Irradiation damage, which causes almost all of the hardening at cladding operating temperatures, will begin to anneal out when $T>T_{irrad.}$, where $T_{irrad.}$ is the cladding irradiation temperature (~340°C - 360°C for PWR). Typical annealing behavior of irradiated Zircaloy is shown in ***Figure 3-1*** [14]. Irradiation defects would completely disappear after a short time (<1 hour) at 500°C and the ductility would then increase. [redacted]

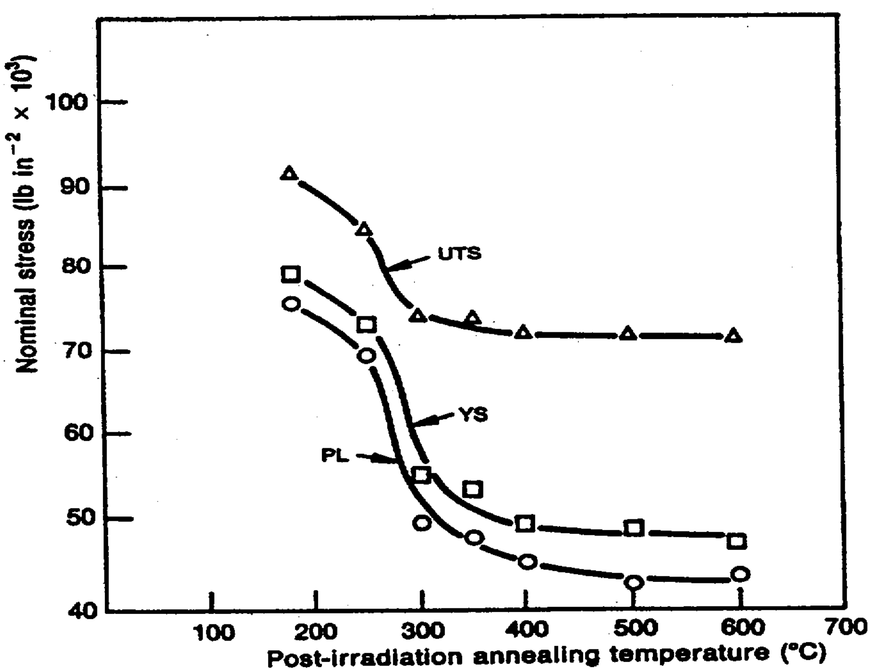


Figure 3-1: **Recovery of damage in Zr-2 after irradiation at 280°C. YS = yield stress; UTS = ultimate tensile strength; PL = plasticity limit [14].** (N.B. 145 lb in^{-2} ≅ 1 MPa)

- Hydrides are distributed nonuniformly in discharged high burnup fuel cladding. There is a general gradient in concentration increasing towards the outer surface, often culminating in a high density "rim" of hydrides directly adjacent to the oxide layer. French experimental evidence [15] indicates [redacted]

  Raising the temperature during the drying process would increase the fraction of hydrides taken back into solution and could have the beneficial effect of re-distributing the nonuniform massive hydrides (rim and lenses) discussed in the previous section. The question of hydride re-orientation is further discussed in Section 2.3.

- Fuel rod internal gas pressure will increase with cladding temperature and will cause some additional thermal creep strain prior to dry storage. [redacted]

- Thermal recovery of the cold worked structure in the form of grain recrystallization begins to occur if the cladding is annealed [redacted]

[redacted]

Conclusions

This report presents clear evidence that:

a) If cladding temperatures exceed 470°C for a short length of time (e.g., ~1 hour) during the drying process, this will anneal out cladding irradiation hardening, resulting in the cladding recovering some of its original ductility. Lower temperatures between ~360°C (cladding irradiation temperature) and 470°C will also produce some annealing, but at slower rates.

b) If cladding temperatures are not allowed to exceed 470°C, there is no significant recrystallization of any previously cold worked state of the Zircaloy. This principally applies to PWR fuel assembly cladding.

3.3 RADIAL HYDRIDES

The usual habit plane for the precipitation of hydride platelets in fuel cladding is parallel, within a few degrees, to the circumferential (i.e., hoop) direction. Geometrically, these circumferential hydrides in the θ-z plane can have only a minimum influence on the tensile mechanical properties during creep, or in a rupture test, because the platelet cross-section is extremely small when compared with the fracture surface area of the cladding wall. Radially oriented hydride platelets that reside in the r-z plane are normal to a hoop stress and present a larger relative cross-sectional area. In this orientation, they can reduce the fracture stress of the cladding if the concentration and spacing are sufficient. However, the radial orientation is not energetically favorable and is rarely observed, even in high burnup fuel cladding, unless the *local* hydrogen concentration is very high, i.e., greater than ~1000-2000 ppm. In these cases, only a mixed network of mainly circumferential and some radial hydrides has been found [33].

Hydride reorientation from circumferential to radial can occur only if three conditions are met. First, it can only occur during hydride re-precipitation. Second, it can only occur in the presence of a sufficiently high tensile stress (the threshold stress). Third, it can only occur after sufficient circumferential hydrides have dissolved in the cladding as a result of an increase in temperature.

Assuming a maximum drying temperature of 450°C during the vacuum drying process prior to dry storage will establish a concentration of dissolved atomic hydrogen in the range of ~ 200 – 400 ppm. This is illustrated in the terminal solubility vs. temperature curve [16], reproduced in ***Figure 3-2***. Hydride reorientation could then occur as the dissolved hydrogen re-precipitates during slow cooling, but this can only occur if the hoop stress in the fuel cladding exceeds the threshold stress.

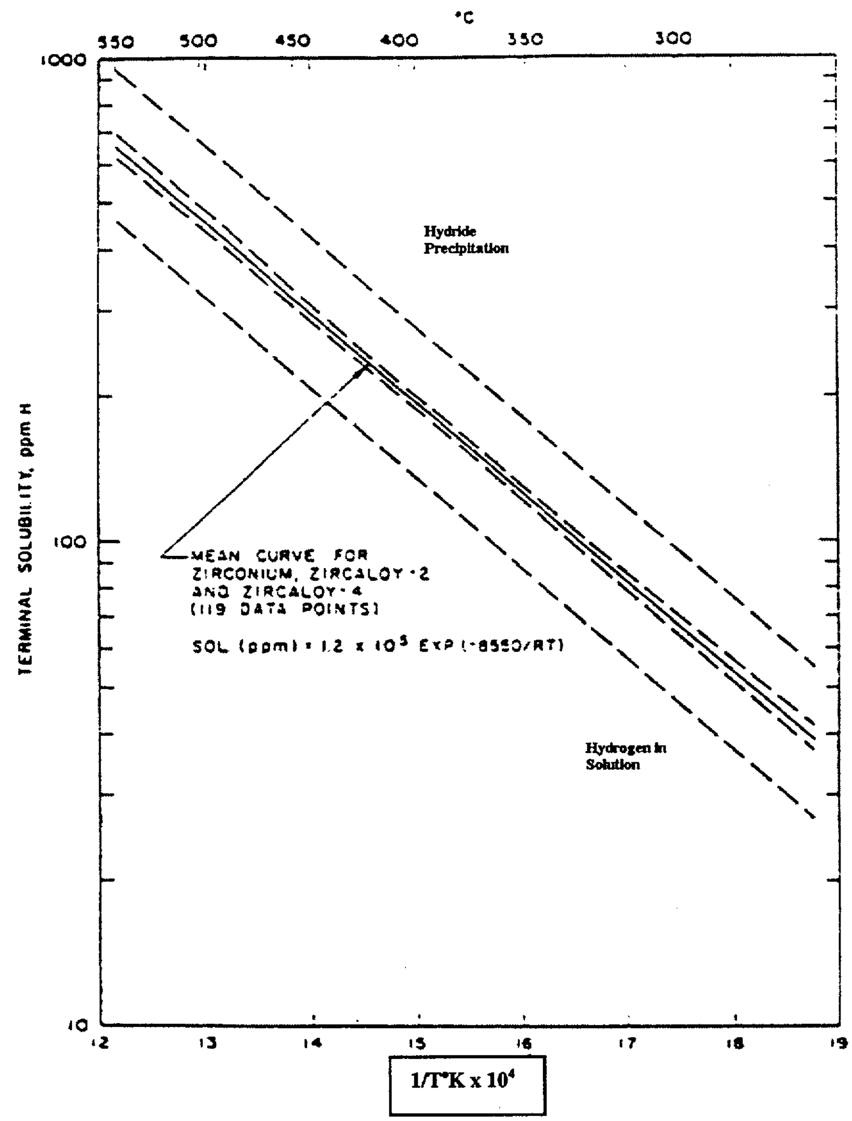


Figure 3-2: **Hydrogen solubility as a function of temperature for Zircaloy. The inner and outer dashed lines represent, respectively, the 95% confidence bands on the fit to the data (mean curve) and the 95/99% tolerance bands for the data [16].**

The measured threshold stress for hydride reorientation is in the range σ_{rad} = 80–200 MPa [11, 15]. Most, if not all, measurements have been made on unirradiated material, so that any influence of irradiation damage on σ_{rad} is not known. Although the presence of radiation-produced defects is unlikely to reduce σ_{rad}, many of these defects will have annealed during vacuum drying so the net effect is likely to be small. [redacted]

[redacted] It is therefore a possibility that some radial hydrides will be formed during cooling of the fuel rod.

However, recent published work from Sweden [11] shows that it does not necessarily follow, even then, that the margin for cladding integrity is eroded. The results of this work generate several important conclusions applicable to dry storage conditions:

[redacted]

[redacted]

[redacted]

The latter point is supported by experimental work performed by EdF [15] and by Goll et al referred to in [34] on the influence of radial hydrides on the uniform and total elongation ductility of Zr-4. In [15], samples containing a total of ~220 ppm of hydrogen were tensile tested as a function of increasing radial hydride concentration at room temperature and 300°C. The results support the conclusion that [redacted]

[redacted] In [34], intact samples from the creep program reviewed in Section 2.1 (German data) were submitted to ductility tests at a stress of 100 MPa at a temperature of 150°C. The samples contained radially oriented hydrides [redacted]. This result is in agreement with other work on crack propagation [35, 36].

Based on the available data, the hoop stress of fuel rods (up to 55 GWD/MTU) prepared for and under dry storage conditions will not be sufficient to cause more than minimal realignment, with a negligible effect on mechanical properties. In support of the latter, the Swedish results show that tensile properties are only impaired if the strain rate is high and that a large fraction of the new hydride network is radial and interconnected. In actual application, the creep strain rate in stored fuel is relatively low and continues to decrease with time and temperature. The maximum hydrogen concentration in discharged, high burnup fuel cladding is expected to be ~600 ppm. This exceeds the hydrogen solubility limits of ~300 ppm (for a drying temperature of 450°C) resulting in only a fraction of the total hydrogen concentration being at risk of nucleating radial hydrides during cooling. In addition, according to the Swedish results, hydrogen concentrations >350 ppm require a larger value of $\sigma_{rad} \cong 200$ MPa to produce reorientation. This value is significantly greater than the NAC calculated maximum hoop stress [redacted] in the cladding at the start of dry storage in an NAC-UMS® cask (see ***Figure 4-29***). Finally, it should be noted that the fracture stress for Zircaloy containing radially-oriented hydrides is in the range of 143-400 MPa [17], which again exceeds any anticipated hoop stress in dry storage.

Conclusions

The evidence presented in this section clearly demonstrates that:

a) Even though a small fraction of radial hydrides may exist in high burnup fuel cladding at the start of dry storage, the maximum operative tensile stress is significantly below the critical value for fracture to occur.

b) The creep strain rate for fuel cladding in dry storage is low and well within the range for ductile and not brittle mechanical behavior in the presence of radial hydrides.

c) Given these conditions, no detrimental effects are predicted from a re-orientation of a small fraction of the hydrides in Zircaloy fuel cladding in the NAC storage system during cask drying and helium backfilling.

4. CREEP MODELING

4.1 GENERAL

There are several requirements and functions of a thermal creep model used for predicting the diametral changes of fuel rod cladding during long-term dry storage. The *ideal* creep equation should be able to:

1. Model the effects of
   - Zircaloy tube fabrication: composition, mechanical reduction and heat treatment
   - irradiation damage
   - hydriding
   - decreasing stress and temperature in the cask.
2. Demonstrate agreement with measured creep strain data for a range of stresses, temperatures and times representative of dry storage conditions.
3. Demonstrate agreement with data to total strains equal to or greater than the proposed strain limit.
4. Allow the extrapolation with reasonable confidence beyond the database, to provide acceptable SNF strain predictions.
5. Permit the calculation of maximum allowable fuel cladding temperatures in the cask to ensure that the cumulative lifetime strain of any SNF rod does not exceed the allowable strain limit.

Justified by the models and data below, we propose that the creep law contain primary (ε_P) and secondary (ε_S) strain components formulated in a constitutive equation of the form:

$$\text{Total hoop strain, } \varepsilon_t = \varepsilon_P + \varepsilon_S \qquad (1)$$

This is consistent with many of the experimental observations concerning the creep behavior of Zircaloy fuel cladding summarized in ***Tables 4-1*** and ***4-2***. An example of typical Zircaloy behavior is shown in ***Figure 4-1***. [redacted]

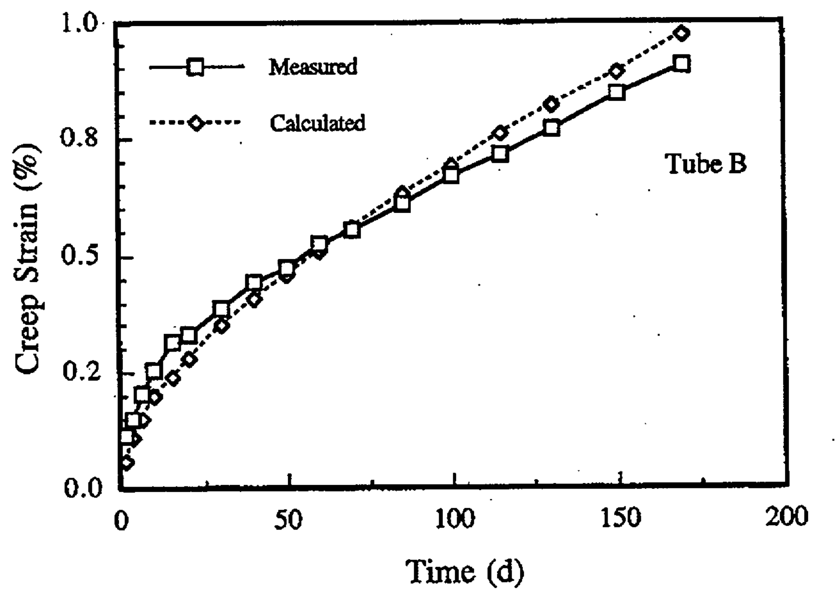


Figure 4-1: **Verification of a generalized creep model in the form of equations (1) and (2) by comparing the calculated and the measured creep strains for Zr-4 at a hoop stress =138 MPa and temperature = 350°C [19]**

4.2 REVIEW OF MODELS AND DATA

NAC has performed a comprehensive review of the current published creep models and creep data, which is summarized in ***Table 4-1*** and ***Table 4-2***. NAC's selection of creep models for evaluation was based [redacted]. Purely mechanistic approaches or those involving DCCG (for the reasons given in Section 2.2) were not included. ***Table 4-1*** covers unirradiated, and ***Table 4-2*** irradiated, materials.

The summary tables and a review of the published references reveal the following general observations:

[redacted]

[redacted]

[redacted]

Some specific observations on several of the referenced works in ***Table 4-1*** and ***Table 4-2*** are pertinent before providing our recommended approach as part of this report.

Spilker [21] has applied an equation to SNF dry cask predictions consisting only of the primary creep process, based on data from unirradiated cladding. This model is characterized by a continually reducing creep rate with time, at a constant stress and temperature. This is independent of the concomitant effects of decreasing pressure and hoop stress and the declining temperature in a dry storage cask that also reduce creep rates. The overall product of the Spilker approach, taking both strain reduction effects into account, predicts a saturated and therefore limiting creep strain of <1%. However, the primary creep model is inconsistent with most long-term creep measurements and also with mechanistic understanding, and may well under-predict cladding strains if extrapolated to long times relevant for dry storage. Evidence for this is shown in ***Figure 4-2***, taken from Spilker's paper [21], where at 10,000 hours, the measured and predicted creep *strains* are about equal at 375°C, but the empirical creep *rate* exceeds that given by the model. Gilbert et al in PNNL's review of the WESFLEX license application [22] pointed this out.

Thermal creep data show that for values of stresses and temperatures typical of SNF conditions, primary creep is transient and saturates at strains less than 1%. Most creep models for Zircaloy are based on an initial primary (transient) process giving way to a steady state secondary stage where the creep progresses at a constant rate, in agreement with empirical data. An example of this behavior is shown in ***Figure 4-1***.

Table 4-1: Biaxial thermal creep tests on cladding materials - Unirradiated

Alloy	Condition	Range of Measurements (max.)				Date	1st Author	Ref.
		T °C	σ_h MPa	ε_d %	Time h			
Zr-4	CWSR	353-420	51-126	0-10	0-8000	1990	Mayuzumi	18
Zr-4	CWSR	390-420	159	0-8	0-1500	1996	Saegusa	6
Zr-4	CWSR & PRX	350-400	80-150	0-5	0-2500	1997	Kim	19
Zr-2 Zr-4	CWSR, PRX, & RXA; CWSR	330-400 385	120 80-160	0-3	0-480; 960	1996	Limbäck	20
Zr-4	CWSR-PRX (x4 types)	250-400	80-150	0-80	0-10,000	1997	Spilker	21
Zr-4	CWSR	325-400 453	50-128 40-120	0-2 0-22	0-9,600	1999	Veselý	23

Table 4-2: Biaxial thermal creep tests on cladding materials - Irradiated

Alloy	Condition	Exposure	Range of Measurements (max.)				Date	1st Author	Ref.
			T °C	σ_h MPa	ε_d %	Time h			
Zr-4	CWSR	46, 47.5 GWD/MTU	390-450	77-320	0-3	0-1800	1993	Mayuzumi	24
Zr-4	PRX	Various	400	100-120	0-80	0-10,000	1997	Spilker	21
Zr-4	CWSR	47 GWD/MTU	380-420	150-250	0-1	0-1368	2000	Limon	3
Zr-4	CWSR	12-49 GWD/MTU	400	130	0-1.6	0-250	2000	Bouffioux	25
Zr-4	CWSR		380-395	86	0.27	0-8000	1985	Kaspar	31

CWSR = cold worked stress relieved; RXA = recrystallized annealed; PRX = partially recrystallized;

T = temperature; σ_h =hoop stress; ε_d = diametral (= circumferential) strain

Primary creep alone generally only affects brittle materials, with low strain capacity, where no ductile straining occurs. All experimental evidence reviewed in this report indicates that this is not the case with Zircaloy, even when the hardening influence of irradiation damage and hydriding is taken into account.

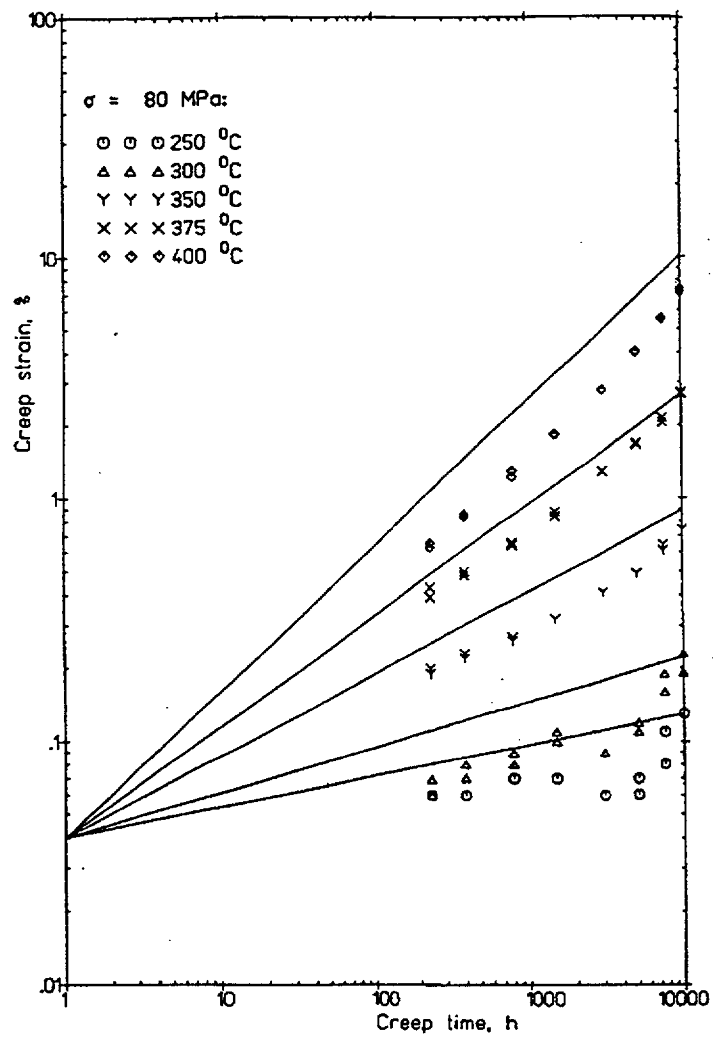


Figure 4-2: **A comparison between experimental creep data (symbols) and calculations (shown as straight lines). Hoop stress = 80 MPa. The equation for calculated creep strain $\varepsilon = At^m$, or, $\log\varepsilon = \log A + m.\log t$, models the strain as arising only from a primary creep process [21].**

One comprehensive approach to creep modeling reviewed by NAC was that being pursued co-operatively by CEA-EdF-Framatome in France for PWR applications [3, 25]. Their basic creep equation is founded on assumed visco-elastic behavior of Zircaloy cladding when subjected to stress at elevated temperatures. EdF and collaborators have studied this mechanism over many years and have accumulated a large database of supporting information. Their strategy for dry storage applications is to establish the basic creep law for as-received, CWSR Zr-4. Next, the hardening effects of hydriding are included (see ***Figure 2-2*** and ***Figure 2-3***, for example). Finally, the combined effects of hydriding and irradiation are measured and modeled. An example, based on data from one, two and four cycle irradiated PWR rods (equivalent to a burnup range of around 12 - 49 GWD/MTU), is shown in ***Figure 4-3***. However, the main problem at present remains the relatively short time range of the creep data (250 hours). During this time frame, the diametral strain rate has probably not reached true steady state (i.e. the secondary creep stage) and this limits its usefulness for modeling and long-term predictive purposes. Also, the thermal recovery of the microstructure, occurring during the vacuum drying stages, and its effect on subsequent creep rates need to be taken into account.

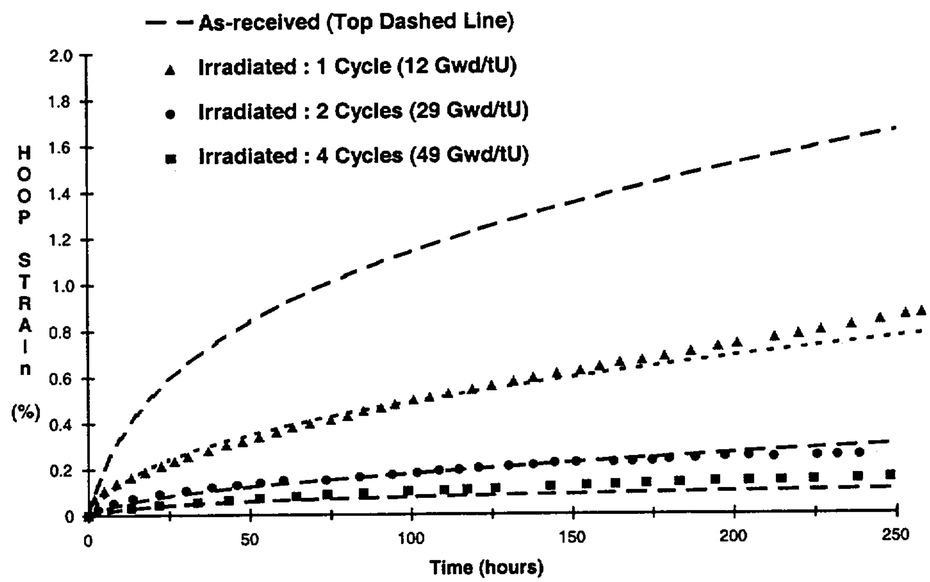


Figure 4-3: **Diametral creep strain (%) vs. time (h) for irradiated CWSR Zr-4. T = 400°C; Hoop stress = 130 MPa; Irradiation 1-4 cycles (annual). Creep law predictions are shown as dashed lines [25]**

Referring to ***Table 4-1***, a similar analytical approach was adopted by references [6, 18-20 and 23] to model their measured creep data. It should be noted that Kim [19] and Limbäck and Andersson [20] do not specifically address their creep data analysis to SNF dry storage. In all cases the total creep strain, ε_t, is of the general form:

[redacted]

The formulation given above produced a good fit to the measured creep data over the range of test variables for each case for references [6, 18-20, 23]. This provides confidence in the analytical approach as well as providing support for a model of the general form shown in equations (1) and (2). Examples of the agreement between model and data are shown in ***Figure 4-4*** and ***Figure 4-5***.

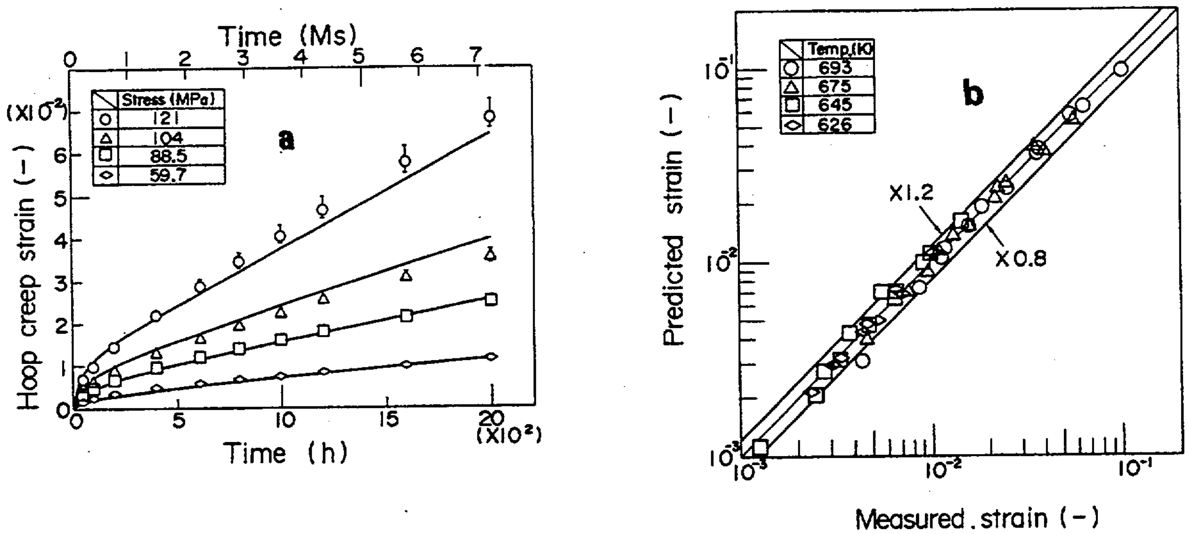


Figure 4-4: (a) Model predictions to fit experimental creep data for various hoop stresses measured at 402°C; (b) Correlation between predicted and measured creep strain values at temperatures in the range 353 - 420°C [18].

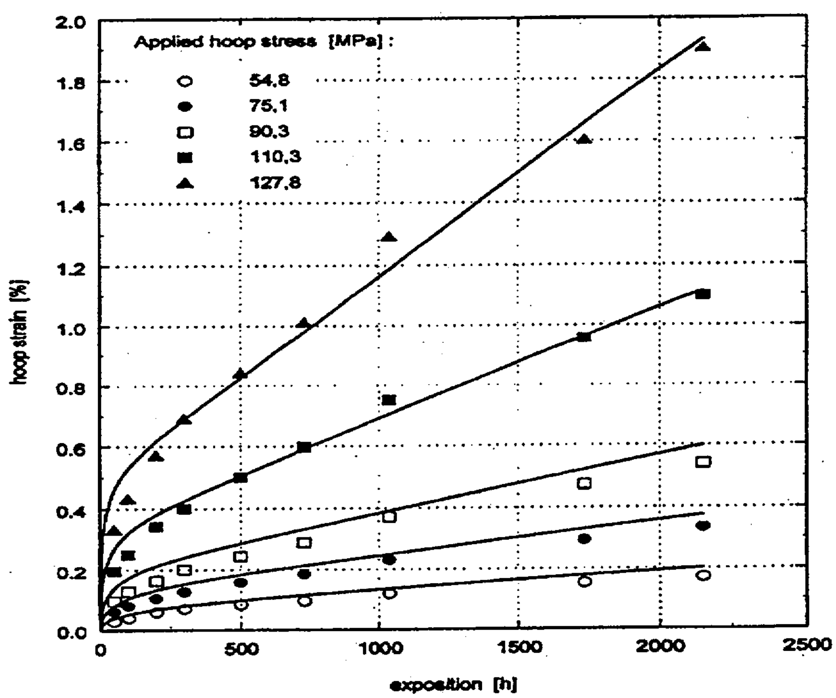


Figure 4-5: Comparison between experimental and predicted creep curves for various hoop stresses at a temperature of 381°C [23]. The solid lines represent the creep predictions.

In the particular case of Ref. [23], the creep data on Zr-4 tubes measured by the Czech group were modeled using the same set of equations as were originally developed and used by Mayuzumi et al [18] and later by Saegusa et al [6]. Good agreement was found for the two key constants, M (stress dependence) and Q (temperature dependence), between the two independent sets of data.

4.3 PROPOSED CREEP MODEL

Following the above review, NAC was satisfied that a thermal creep [redacted] would provide satisfactory yet conservative predictions of the behavior of irradiated spent fuel in long-term dry storage. NAC concluded that the analytical description of the creep strain should be [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

4.4 THE SOLUTION METHOD

The procedure adopted for determining the maximum allowable cladding temperature during dry storage involves solving equation [redacted]. There are two methods that can be employed to accomplish this, [redacted]

[redacted]

[redacted]

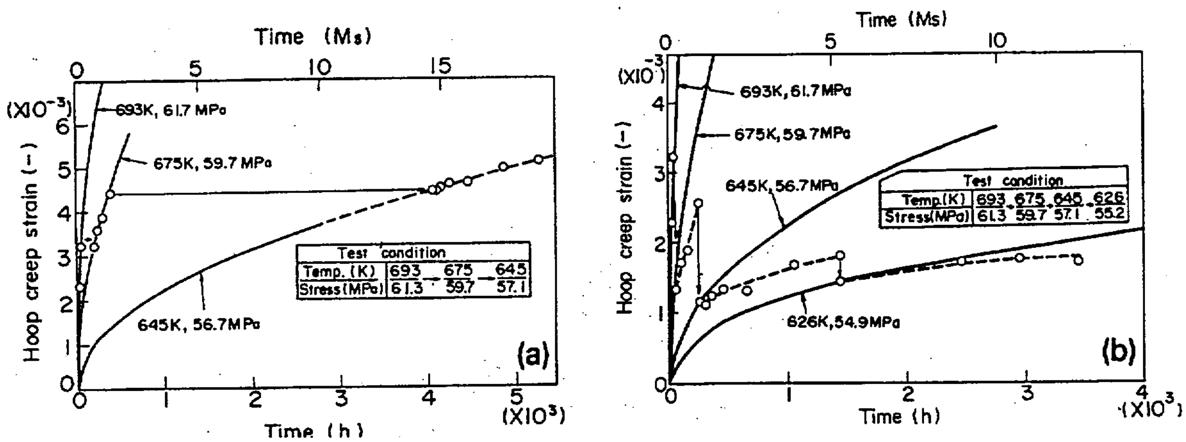


Figure 4-6: **Creep strains during decreasing temperature and stress representative of dry storage cask conditions: (a) assumes strain-hardening and (b) time-hardening is obeyed when compared with constant temperature and stress conditions. The best agreement is obtained using strain-hardening rules [26].**

The inputs to the calculation are the clad initial temperature T_i, the rod internal pressure at a reference temperature, the total free volume in the fuel rod (mostly in the top plenum region), and the time-temperature profiles of the cladding and the fission gas during storage. The temperature profiles are determined independently from a knowledge of the fission gas volume in the limiting fuel rod, the decay heat source term (a function of rod burnup, initial enrichment and pool cooling time) and the cooling characteristics of the cask. [redacted]

4.5 THE CALCULATIONAL MODEL - DSCREEP

NAC has independently developed a proprietary computer program named DSCREEP to fully implement the methodology described above. In addition to solving equation [redacted]

[redacted]

DSCREEP supports the following input options:

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

4.6 DSCREEP BENCHMARKING

DSCREEP has been extensively benchmarked against two sets of data.

- An extensive data set of unirradiated creep measurement data, and
- Available code predictions by others.

The main objectives of this benchmarking were to demonstrate:

[redacted]

The most fundamental way to check the implementation of a creep correlation is to benchmark the code to actual test measurements obtained at constant temperature and pressure (or hoop stress). [redacted]

[redacted]

The first set of benchmarks [redacted] are based on Mayuzumi's own measurement data and calculations [18].

Figure 4-7 shows [redacted]

Figure 4-8 [redacted]

[redacted]

[redacted]

The second set of benchmarks [redacted] is based upon measurements performed by Spilker, et. al. [21]. [redacted]

[redacted]

In addition to the actual measurements reported by Spilker, his paper [21] provides a summary of all previous German creep measurements performed on irradiated and unirradiated Zr-4 claddings in relation to dry storage. This information is presented as Figure 1 in the Spilker paper and is presented herein as ***Figure 4-13***. Unirradiated data are shown as a band near the top of the figure, and irradiated data are shown near the bottom. [redacted]

The third set of key benchmarks for the DSCREEP program is based on a unique series of experiments performed by Mayuzumi et al [26] [redacted]

The code predictions for this benchmark set are shown on ***Figure 4-14***. [redacted]

Mayuzumi et. al. also performed a hypothetical cask calculation using a clad cool-down curve after Peehs [26], assuming that the gas temperature equals the clad temperature (a very conservative assumption), and neglecting clad creep-out effects. The DSCREEP results for this analysis are shown on ***Figure 4-15*** [redacted]

The above benchmark cases demonstrate that DSCREEP successfully reproduces [redacted]. It also demonstrates that DSCREEP can accurately and conservatively predict creep strain for Zircaloy clad spent fuel in anticipated dry storage conditions.

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

4.7 DSCREEP INPUT AND CONSERVATISMS FOR ANALYSIS APPLIED TO NAC-UMS® CASKS

General

The underlying objective to input generation for the creep analyses performed in support of this Topical Report was to develop the inputs in a conservative, but reasonable manner. Since the analyses were to be applied to fuel rods with up to 55 GWD/MTU burnup, all input data were appropriately related to this burnup level.

The following sections discuss important input parameters.

End-of-Life (EOL) Pressure/Temperature State

An EOL (55 GWD/MTU) pressure/temperature state is required so that the initial gas pressure can be determined in dry storage. For this purpose, an ESCORE [38] fuel performance code analysis performed on a Westinghouse CWSR Zr-4 17x17 fuel rod (a North Anna rod) was utilized. The analysis to 45 GWD/MTU was for an EPRI sponsored clad corrosion study. The study included analysis up to 45 GWD/MTU of rods operating up to 6 kW/ft. Based upon [redacted] information available to NAC, at 55 GWD/MTU, 6 kW/ft is a limiting fuel rod.

[redacted]

[redacted]

At burnups greater than about 20 GWD/MTU, under operating conditions, there is pellet-clad contact radially due to pellet swelling. This forces most of the internal rod gas into the cooler upper plenum region. There would be a small difference in the volume of this region between 45 GWD/MTU and modestly higher burnups, e.g., up to 60 GWD/MTU, due to the difference between the irradiation growth rate of the pellet stack and the cladding as irradiation progresses. Since the pellet stack grows at a greater rate than the cladding, the plenum volume decreases with burnup, and calculated data show this decrease to be linear. The plenum volume contraction causes an increase in gas pressure over and beyond the increase from additional fissioning. [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

Fuel Rod Clad Temperature Decay Curve at 55 GWD/MTU

The cladding temperature vs. time is required in the methodology to evaluate its first order effects on creep (see Equations 4, 5 and 6) and its second order effects on mechanical properties of the cladding (see Equation 6) as the temperature falls in dry storage.

Data for temperature vs. time for fuel in dry storage at 55 GWD/MTU were calculated for the NAC-UMS® cask based on [redacted]

[redacted]

Fuel rod temperatures versus time data for both BWR and PWR applications were developed utilizing a series of finite element models, as shown below.

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

Fuel Rod Gas Temperature Decay Curve

Gas temperature vs. time information is required in the methodology to determine the internal pressure of the dry stored rod. The internal pressure is the driving force for cladding creep.

To determine the fuel rod gas temperature vs. time in dry storage, NAC used [redacted]

Fuel Rod Geometry

The most conservative rod geometry is the one that gives the highest total creep strain for a given hoop stress vs. time behavior. Hoop stress, σ_H, is proportional to cladding geometry as follows:

$$\sigma_H \propto [OD-t]/t$$

Where:

OD is the clad outside diameter
t is the clad thickness.

From an initial hoop stress perspective, our review of all PWR fuel designs available in the U.S. indicated that the most conservative rod geometrically would be either a Westinghouse 14x14 or 15x15 rod (same geometry for both). Nevertheless, a series of scoping runs with DSCREEP were performed to verify this rod was the most limiting. [redacted]

[redacted]

Effect of Tin Content on Creep of High Burnup Cladding

Over the years, there has been a distinct trend towards lower tin contents in Zr-4 for PWR applications in an attempt to improve corrosion performance. The lower tin content has the potential to affect the creep behavior of the cladding. However, experimental thermal creep measurements on irradiated (Sn range 1.37 to 1.52 w/o) and hydrided samples of low tin fuel cladding by Limon et al [3] and Bouffioux and Rupa [4], respectively, have demonstrated that the hardening effects on creep of both irradiation damage and hydrogen dominate over the effects of tin. [redacted]

Radiation and Hydrogen Hardening of High Burnup Cladding

As previously noted, the NAC creep methodology is conservative [redacted]

Any decrease in creep strength of irradiated material in either vacuum drying or dry storage would be as a result of annealing of the irradiation damage in the cladding material. [redacted]

There are published data by Mayuzumi [24] for *irradiated* creep of CWSR Zr-4 and also for the creep-hardening factor from irradiated fuel measurements and unirradiated fuel calculations at the same temperatures/pressures/times. This information is published by the same researcher who provided the data on which DSCREEP is based. The burnups of the two fuel rods from which cladding specimens were obtained for these measurements were 46.6 and 47.5

GWD/MTU. [redacted] These conditions correspond approximately to the highest temperatures in the drying sequence prior to placement in NAC-UMS® casks (see ***Figure 4-26***). [redacted]

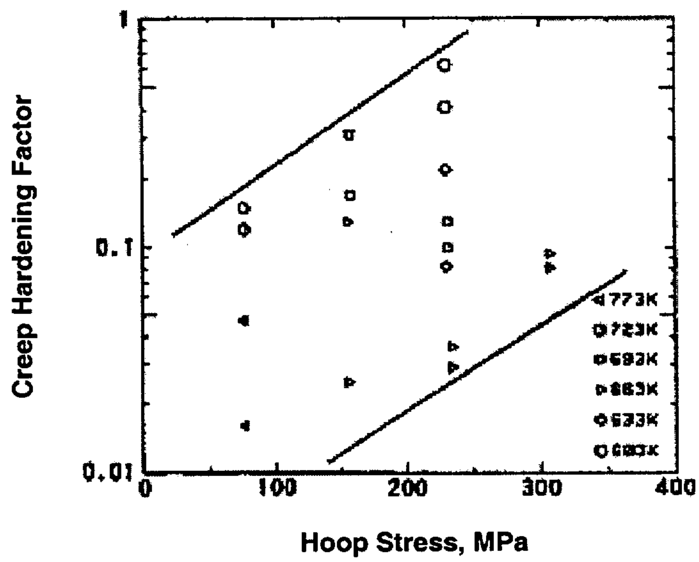


Figure 4-20: Measured Creep Hardening Factor vs. Hoop Stress [24]

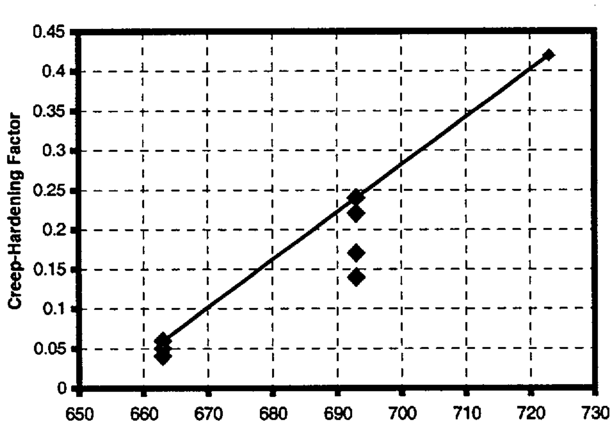


Figure 4-21: Creep-Hardening Factor vs. Temperature @ 154 MPa
(Data from Figs 4-4 & 4-5)

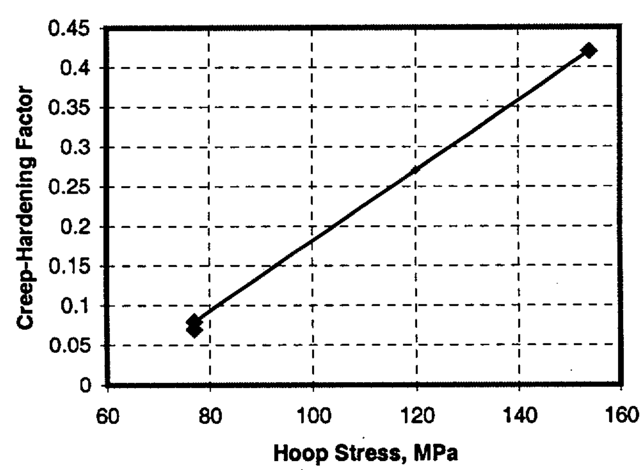


Figure 4-22: Creep-Hardening Factor vs. Hoop Stress @ 450 K
(Data Points from Fig. 4-20 & Table 4-6)

NAC independently evaluated uniform creep [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

As noted, the cladding Mayuzumi tested had a maximum burnup of 47.5 GWD/MTU. This same cladding, if irradiated to 55 GWD/MTU, would contain more hydrogen from waterside corrosion, [redacted]

Figure 4-13 displays a set of German irradiated creep measurements. As noted by the NRC [32], these irradiated data are from the same creep tests reported by Kasper [31] over 10 years earlier, with the only difference in the number of data points and the total time to which the data are presented, i.e., 6000 hours and 8000 hours. The temperatures and hoop stresses used for these measurements are consistent with dry storage conditions. For these measurements, Zr-4 claddings were irradiated in the German FRG-2 fast flux reactor to create irradiation hardening of the specimens. Irradiation times were in the order of 25 days. Creep tests were performed on these irradiated samples for a duration of 6000 hrs (250 days) at 400°C and either 100 or 120 MPa hoop stress. NAC calculated the strain for these samples at the measured conditions using the DSCREEP program [redacted]

[redacted]

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]

[redacted]

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]

[redacted]

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]

[redacted]

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]

Initial Gas Volume Assumed for Creep-Out Pressure Reductions

As discussed previously, NAC's DSCREEP program accurately incorporates the beneficial contribution of the reduction of internal rod pressure due to clad creep-out.

The amount of pressure reduction due to this effect is a function of the initial open volume assumed within the fuel rod. [redacted]

Input Listings

The input listings are given first for the ESCORE fuel performance code that was used to help determine the 55 GWD/MTU EOL rod pressure/temperature conditions ***(Table 4-8)***, and second for the DSCREEP code itself ***(Table 4-9)***.

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

4.8 RESULTS OF DSCREEP ANALYSIS FOR NAC-UMS® CASK

General

The creep analyses performed for the high burnup fuel (55 GWD/MTU) to be loaded into an NAC-UMS® cask utilized the inputs as described in Section 4.7 augmented by the following considerations:

[redacted]

[redacted]

[redacted]

[redacted]

Results

Figures 4-23 and 4-24 provide the DSCREEP results for creep strain vs. time in dry storage. [redacted]

Figure 4-25 gives the calculated correlation between initial clad temperature in dry storage (or pool residence time) and saturated creep strain. [redacted]

[redacted]

The summary data in ***Table 4-10*** were extracted from the correlations derived in ***Figure 4-25***. For pool residence times ranging from [redacted], they show the initial clad temperature and total creep strain for fuel at 55 GWD/MTU, [redacted]

[redacted]

[redacted]		[redacted]		[redacted]		[redacted]	
[redacted]		[redacted]		[redacted]		[redacted]	
[redacted]		[redacted]		[redacted]		[redacted]	
[redacted]		[redacted]		[redacted]		[redacted]	
[redacted]		[redacted]		[redacted]		[redacted]	
[redacted]		[redacted]		[redacted]		[redacted]	

The elevated temperature drying process prior to the placement of the fuel into dry storage will affect the fuel rod hoop stress and the total creep strain. A supplementary analysis was performed to illustrate the direction and change in magnitude of these two variables when some temperature/time level of drying preceded the dry storage. [redacted]

The results of this analysis are shown in ***Figures 4-27*** and ***4-28***. The former gives fuel cladding hoop stress vs. time in dry storage and the latter the creep strain vs. time. [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

5. SUMMARY AND RECOMMENDATIONS

In this report, NAC has researched and addressed each of the current criteria contained in the formal NRC review guidance document (ISG-15) for high burnup fuel.

Regarding the NRC guidance, NAC's research and analyses show that, based on empirical data obtained and documented herein, the limits and acceptance criteria in ISG-15 are unnecessarily conservative. Consequently, NAC proposes and provides the justification for new limits and acceptance criteria based on actual mechanistic behavior of Zircaloy clad under dry storage conditions. These proposed new limits and acceptance criteria are:

1. The fuel integrity criterion during storage can be satisfied by a creep strain limit of 2.5%. There is no requirement to specify or characterize failure modes because this strain level is demonstrated to be reached without material degradation.

2. The oxide thickness acceptance limit on the fuel cladding for interim dry storage can be safely raised to 120μm. This value is consistent with volume-averaged hydrogen concentrations in the cladding of up to 800 ppm. Further, this value is greater than the current U.S. de-facto maximum allowable oxide thickness in operation of 100μm.

3. The maximum cladding temperature during vacuum drying and helium backfill can be safely limited to 450°C. Peak clad temperatures up to this level, during the relatively short period of vacuum drying and helium backfill, are demonstrated to have negligible effects on both hydride reorientation and annealing of radiation hardening.

In this report, we have provided evidence to show that a hydrogen concentration of 800 ppm is below the level that causes any significant loss in strength or ductility for cladding temperatures in excess of 150°C. Even for temperatures below 150°C, the cladding retains ductility of >1%, thus satisfying the requirement that fuel can be handled without fracturing after dry storage. An average hydrogen concentration of 800 ppm would equate to an oxide thickness in excess of

130μm, allowing us to justify a revised acceptance limit of 120μm for the storage of high burnup fuel.

During long-term dry storage, the limiting governing process for fuel integrity is thermal creep deformation of the cladding under the influence of the rod internal pressure. This report has documented extensive experimental evidence to demonstrate that irradiated and hydrided Zircaloy can withstand uniform plastic creep strains >2.5% without failing or entering a tertiary deformation stage. NAC, therefore, recommends a revised lifetime creep strain limit of 2.5%, [redacted]

NAC has proposed a benchmarked thermal creep model based on semiempirical fits to creep test measurements on *unirradiated* Zr-4 cladding. The model is supported by a broad range of data of interest to dry storage analysis, and contains both primary and steady state secondary contributions to the total strain. The fitted equations have been previously used by two independent groups for calculating the maximum allowable cladding temperature to meet the lifetime creep strain limit. The use of [redacted] is justified based on the technical, but realistic, conservative margins provided by [redacted]

Nevertheless, NAC selected [redacted] creep model [redacted] on CWSR Zr-4. The irradiated test specimens had burnups approaching 50 GWD/MTU. [redacted]

[redacted]

NAC has used the thermal creep model proposed in this report as a basis for development of a proprietary dry storage creep analysis computer program. This NAC program, called DSCREEP, has been fully developed and benchmarked. This program has been applied conservatively, [redacted], to CWSR Zr-4 PWR fuel at 55 GWD/MTU and having [redacted] oxide thickness. DSCREEP results show that this fuel will experience a total creep strain of [redacted] (even assuming the fuel experienced a Condition II event in its operating history) at an initial cladding temperature of [redacted] (equivalent to [redacted] years of prior pool residence time). This creep strain value is well below the proposed 2.5% total creep strain limit. In fact, the current 1% total strain limit can be met with maximum cladding temperatures of [redacted], respectively, for Condition I and II. These temperatures [redacted] correspond, respectively, to storing fuel with [redacted] years of pool residence time prior to dry storage.

Supplemental analyses indicate that the drying process prior to cask storage can have an effect on cladding hoop stresses in dry storage and the saturated strain. The magnitude of the effect will depend upon the time and temperature levels used in the drying operation.

To ensure no detrimental effects on the fuel clad during loading operations, NAC proposes that the maximum cladding temperature for vacuum drying and helium backfill be limited to <450°C.

This temperature limit will be reflected in NAC SAR and Technical specification allowable limits for high burnup fuels.

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