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A Survey of Potential Light Water Reactor Fuel Rod Failure Mechanisms and Damage Limits

E. L. Courtright

July 1979

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A SURVEY OF POTENTIAL
LIGHT WATER REACTOR FUEL ROD
FAILURE MECHANISMS AND DAMAGE LIMITS

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ABSTRACT

This report documents the findings and conclusions of a survey to evaluate current information applicable to the development of fuel rod damage and failure limits for light water reactor fuel elements. The survey includes a review of past fuel failures, and identifies potential damage and failure mechanisms for both steady state operating conditions and postulated accident events. Possible relationships between the various damage and failure mechanisms are also proposed. The report identifies limiting criteria where possible, but concludes that sufficient data are not currently available in many important areas.

SUMMARY

This report addresses the principal mechanisms by which fuel rod damage or failure could occur and classifies the mechanisms into two categories: 1) Mechanisms that damage cladding integrity but do not directly cause failure, and; 2) Mechanisms that cause failure when design limits are exceeded. Limiting criteria for steady state operation are summarized in Table 1 together with an assessment of the current data base and "State of the Art" supporting theories. A list of acceptance limits applicable to accident or abnormal conditions are shown in Table 2 and areas where additional data are needed to complement the existing data base have been indicated.

Interactions between mechanisms could alter criteria that are developed from single effects considerations. It may also be necessary to account for path dependent and cumulative effects. Cumulative damage is identified as a factor that will become increasingly more important as fuel duty cycles are extended to higher burnup.

TABLE 1. Summary of Applicable Limits for Fuel Rod Damage and Failure Mechanisms under Steady State Conditions or Operational Transients

Mechanisms	Applicable Limits	Theory Development	Data Base	Data Needs/Requirements
<u>Damage</u>				
Fatigue	177.5 MPa ($\sim 10^7$ cycles)	Adequate	Limited	Need data for $>10^7$ cycles and for low cycle - large strain amplitude
Creep Collapse	Yield Stress	Adequate	Limited	Need better data on yield behavior in compression
Creep Rupture	310 MPa @ 300°C	Adequate	Fair	Should study applicability of (SEAF)* concept
Fretting	None	Poor	Poor	Need to identify fundamental cause/effect relationships
Oxidation	None	Adequate	Adequate	Need data at high burnup
Rod Bow	None	Adequate	Limited	Need to establish first order contributors
Overheating (S.S.)	400°C	Poor	Limited	Confirmation Required
Σ Damage	None	Poor	Poor	Need experimental confirmation of relative effects
<u>Failure</u>				
PCI	Under Development	Fair	Fair	Need to establish general applicability of empirical models or develop proof test
Primary Hydriding	2.0 mg of H ₂ O/cm ³ (void) <20 ppm H ₂ O in fuel	Good	Good	None
Secondary Hydriding	None	Poor	Poor	Need to identify principal sources
Mechanical Fracture	100 ppm H ₂ $K_I = 22 \text{ MPa } \sqrt{\text{m}}$	Adequate	Adequate	Need stress limit
Burst Rupture	Negative pressure differential	Adequate	Limited	Need to evaluate effective stress or SEAF* limit as alternative criteria

*Good - suitable for purpose; little or no additional improvement needed

*Adequate - defines applicable limits

*Fair - development moderately good; could be improved

*Limited - narrow in scope; more data needed

*Poor - lacking in the quality needed for proper application

*Strain Energy Absorption to Failure

TABLE 2. Summary of Potentially Applicable Acceptance Limits
for Fuel Rods Exposed to Accident Conditions

Mechanisms	Applicable Limits	Theory Development	Data Base	Data Needs/Requirements
<u>Loss of Coolant</u>				
Oxidation	$\beta_{\text{eta}} < 0.1 \text{ mm} < 1 \text{ w/o } O_2$	Fair	Adequate	Criteria needs further refinement
H ₂ - Embrittlement	H ₂ /H ₂ O > 0.2	Poor	Fair	Additional confirmation needed
Burst Rupture	Effective stress = $f(\text{temp})$	Adequate	Adequate	Develop functional relationships from current data base
Overheating	1204°C peak cladding temperature	Adequate	Adequate	May not be needed if other requirements are well defined
<u>Reactivity Increases</u>				
Energy Deposition	1.1 J/kg	Fair	Fair	Needed test programs in progress
Oxidation	None	Fair	Fair	Needed data in the 400-800°C range
Overheating	Clad Melting (1852 °C)	Good	Fair	None
Burst Rupture	Effective stress = $f(\text{temp})$	Adequate	Adequate	Verify functional relationships
PCI	None	Poor	Poor	Needs experimental determination for low energy depositions
<u>PCM</u>				
Oxidation	$\beta_{\text{eta}} < 0.1 \text{ mm} < 1 \text{ w/o } O_2$	Fair	Adequate	Criteria needs further refinement
Overheating	$\beta_{\text{eta}} < 0.1 \text{ mm} < 1 \text{ w/o } O_2$	Fair	Adequate	Criteria needs further refinement
Burst Rupture	Effective stress = $f(\text{temp})$	Adequate	Adequate	Develop functional relationships
PCI	None	Poor	Poor	Needs experimental determination

• Good - suitable for purpose; little or no additional improvement needed
 • Adequate - defines applicable limits
 • Fair - development moderately good; could be improved

• Limited - narrow in scope; more data needed
 • Poor - lacking in the quality needed for proper application

INTRODUCTION

The objective of this survey is to evaluate information relevant to the development of fuel damage and/or failure limits for light water reactor (LWR) fuel elements. Most light water reactors use Zircaloy cladding to contain the uranium dioxide (UO_2) fuel and to act as a barrier against the release of radioactive fission products into the primary coolant system. The cladding also acts as the principal heat transfer boundary between fuel and coolant. Each fuel assembly is required to maintain a coolable rod-bundle geometry under all conditions to insure adequate removal of residual heat.

Fuel rod damage and failure criteria are required by the Nuclear Regulatory Commission⁽¹⁾ to assure that fuel system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in safety analyses. Fuel system damage and rod failures are possible during normal operation as well as during abnormal operational occurrences. Modest increases in power following peak load demands may cause fuel rods to perforate as a result of the localized interaction that occurs between the expanding fuel and the Zircaloy cladding. The potential for damage increases if primary system cooling is reduced or lost, since the cladding will heat rapidly above its normal operating temperature causing oxidation and embrittlement, or possibly localized melting. Sudden and excessive increases in reactivity will also cause a rapid deterioration in the structural integrity of the fuel rod, leading to failure and possibly to a loss of coolability.

This survey attempts to identify applicable failure and damage mechanisms since many of the factors that influence fuel coolability are inherent in these mechanisms. Acceptance limits for most of the principal mechanisms were reviewed and areas in need of additional data were identified.

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1.0 REVIEWS OF PAST FUEL FAILURES

Reviews of past water reactor performance^(2,3) suggest that the two major classes of failure have been 1) hydriding of the cladding from fuel moisture, and 2) pellet-clad interaction caused by power oscillations. Cladding collapse arising from fuel densification and fuel rod bowing are additional performance concerns, but neither mechanism appears to have, as yet, resulted in an actual rod failure or in the release of radioactive fission products into the primary system. Hot spots that have developed as a consequence of excessive crud deposition, and failures that are attributable to fretting are two other mechanisms that have produced⁽²⁾ a limited number of boiling water reactor (BWR) rod failures. The problems associated with fretting were corrected by appropriate spacer design modifications, and control of the copper concentration in the coolant has eliminated the deposits that cause hot spots.

Frequencies for either of the two major classes of failures have generally been less than 1% and in some instances may be less than one rod in 10,000.⁽³⁾ Still, concern for public safety provides an incentive to verify the adequacy of safety margins in nuclear power reactors. The ultimate goal is a failure free, hermetically sealed fuel element that would prevent fission products from entering the primary system. Normal operation with small numbers of defective rods is not desirable. Thus, technical specifications on primary coolant activity have resulted in limitations on plant operating maneuverability, with resultant economic penalty. Satisfactory resistance to the propagation of failures under accident conditions must also be demonstrated.

2.0 POTENTIAL DAMAGE MECHANISMS

Several mechanisms are known to adversely affect fuel cladding integrity and may ultimately cause actual fuel rod failure. These mechanisms include strain fatigue, creep collapse, creep rupture, fretting, oxidation, rod bow, and overheating. None have been individually responsible for a LWR fuel rod failure and are not currently considered to be performance limiting. However, damage caused by any one of these mechanisms could lower failure thresholds and reduce what would otherwise be acceptable operating margins. This section addresses the different potential damage mechanisms and defines, where possible, those criteria that could prevent a loss in fuel rod integrity and eventual fuel rod failure.

2.1 STRAIN FATIGUE

One consequence of power cycling is the fatigue that occurs as the cladding strains under stress imposed by the expanding fuel and by compression during creepdown. Fuel rods that are subjected to flow-induced vibrations may be subjected to bending moments with high cyclic frequency and low strain amplitudes.

O'Donnell and Langer's early work⁽⁴⁾ provides a general fatigue design basis for irradiated Zircaloy and suggests an endurance limit of 177.5 MPa (25,730 psi) corrected for maximum mean stress. Irradiation does not seem to significantly alter this limit except to reduce the allowable stress/strain amplitude. The O'Donnell data goes to 10^7 cycles, but many fuel rods could endure up to 10^{10} cycles in their lifetime.⁽⁵⁾ Thus, an uncertainty in failure limits exists at larger exposures.

Few, if any reactor defects have been attributed to high cycle fatigue. It is perhaps the low cycle regime of large plastic strain amplitudes resulting from fuel interaction that should be of primary concern.

Low cycle fatigue in Zircaloy has been found to follow the Coffin-Mason Relationship:

$$\Delta \epsilon_p \cdot N_f^n = C$$

where n and C are constants, N_f is the number of cycles to failure and $\Delta\epsilon_p$ is the dynamic plastic strain increment.

Lee⁽⁶⁾ reports that $n = 0.5$ for most metals and that $C \approx$ approximately 0.4 for Zircaloy. Using the low cycle bending fatigue data of James and Ostrum⁽⁷⁾ for irradiated Zircaloy-2 at 300°C a value of 0.38 can be calculated for C , which is in good agreement with the value reported by Lee.

A realistic upper limit on the dynamic plastic strain component under load cycling conditions⁽⁸⁾ should be on the order of 0.5%. This is assuming that creepdown will occur at low power. If we use the Coffin-Mason relationship and the proposed constants for n and C , failure would be predicted at 6400 cycles.

The cumulative number of strain fatigue cycles experienced by fuel system structural components must of course be less than the design limit. A safety factor of two on the stress amplitude and a factor of 20 on the number of cycles is recommended in the Nuclear Regulatory Commission (NRC) Standard Review Plan.⁽¹⁾ This would mean a limit of 320 cycles which would severely restrict daily load following practices.

2.2 CREEP COLLAPSE

Fuel elements that are subjected to an external pressure that is higher than the internal pressure will likely experience some compressive creepdown. Problems with densification of low density or unstable fuel have accentuated this condition and have caused the cladding to collapse into the gaps which develop between the pellets. The time required for a rod to collapse under compressive creep will decrease as both the fast flux ($>1\text{MeV}$) and the rod ovality increase. The criteria best suited for predicting instantaneous collapse seem to be either a "reduced" modulus⁽⁹⁾ or the yield strength⁽¹⁰⁾. Both compare reasonably well with experimental data, but a yield strength criterion has the advantage of being easier to correlate with manufactured properties, and is, therefore, more useful to design applications.

The minimum yield strengths exhibited by Zircaloy at reactor start up conditions should be enough to prevent instantaneous collapse. Irradiation

exposure then hardens the material even further, but the fast flux component (>1 MeV) of the neutron spectrum offsets this by accelerating creep rates. Fuel rod ovalities gradually increase as a result of creep contribution and can eventually reach a point where collapse is possible.

The effects of irradiation on the in-reactor creep of zirconium alloys have been summarized by Pankaskie.⁽¹¹⁾ Steady-state creep rates generally increase with increasing stress, increasing temperature, and increasing fast flux (>1 MeV) up to about 350°C . At temperatures above 350°C , creep is dominated by thermally activated mechanisms.

Well defined experimental data on in-reactor creep behavior under compressive loading conditions are currently not available. Thus, a degree of uncertainty exists in applying creep data obtained under nontypical stress loading conditions owing to the anisotropic behavior of the Zircaloy. Fortunately, diametral measurements made on fuel rods that have actually experienced various degrees of creepdown, including collapse, are available to use as benchmarks. The rate at which Zircaloy tubing deforms under constant external pressure in the operating temperature range of interest are also currently under investigation.⁽¹²⁾

Data on the effects of preferred crystallographic orientation are limited. Kallstrom⁽¹³⁾ indicates that creep rates increase as the basal pole orientations shift towards the surface normal (zero degrees), which is equivalent to the radial direction in a fuel rod. Rieger and Lee⁽¹⁴⁾ have shown that the yield strength is at a minimum when the basal pole is at an angle of 30° with the radial direction. A material with a zero degree texture should be difficult to deform in the thickness direction because of the lack of active primary slip systems. However, a material with a zero degree texture would be relatively easy to deform in the hoop or circumferential direction because of the preferred $(10\bar{1}0) \langle 11\bar{2}0 \rangle$ orientation. Thus, the direction of the resolved stress components relative to the orientation of preferred deformation systems will influence creep.

Criteria that appear relevant to creep collapse (assuming that the stress conditions from prior densification or other sources can be determined) are:

- creep rate as a function of fast flux (>1 MeV) and temperature
- yield strength
- initial tube ovality
- preferred crystallographic orientation.

2.3 CREEP EXPANSION

Fuel rod failure caused by creep expansion has not been a general concern in the study of LWR fuel performance. This is because safety considerations generally require that external pressures exceed internal pressures over the design life of the fuel rod. However, fuel swelling at high burnups could conceivably result in diametral strains of 1%.

Creep expansion and ultimate rupture pose a major concern for the pressure tubes used in pressurized heavy water reactor (PHWR) systems.⁽¹⁵⁾ Extrapolations of accelerated high temperature test results suggest a creep rupture limit of 20 years at 300°C and 310.3 MPa (45,000 psi).

Reactor fuel, operating at steady state and at heat ratings where pellet-cladding interaction occurs, develops tensile stresses in the cladding wall. If the strains accumulate, they could initiate PCI-type failures. Thus, a strain energy absorption to failure (SEAF) limit may apply in situations where the maximum, allowable time-dependent strain accumulation is of importance. A recent study suggests that the concept of an SEAF limit may also have merit under PCI conditions.⁽¹⁶⁾

Potential criteria applicable to creep expansion are:

- the stress threshold, below which failure will not occur at a defined temperature;
- the maximum allowable time-dependent strain accumulation
- strain energy absorption to failure (SEAF).

2.4 FRETTING

Fuel rod failures resulting from spacer interaction have been identified,⁽²⁾ but seem to have been minimized by bundle design. The elements that lead to failure, such as contact forces, vibration frequencies, and material interactions, have not been established and it is not known whether corrosion or wear is the basic driving mechanism in such failures. Both corrosion and wear may coexist since Zircaloy is susceptible to rapid oxidation (particularly if fresh surfaces are exposed) and to frictional interactions (galling) with other metals as well as with itself. Hardness is an important variable in wear applications; thus, increased resistance to fretting-type wear could be expected in a fuel rod as a function of increasing neutron fluence.

Since criteria for failure caused by fretting have not been established, a danger exists that design changes or improvements that are introduced for other reasons could conceivably reinitiate fretting conditions. Fortunately, fretting is a situation that can be monitored during normal surveillance intervals.

The basic variables associated with fretting are:

- contact force
- frequency of vibration (interaction)
- hardness
- temperature.

2.5 OXIDATION

To date, the only oxidation-related failures that have occurred in Zircaloy-clad fuel rods have occurred in BWRs with copper alloy preheaters.⁽²⁾ These failures were caused by significant accumulations of crud on the oxide surface. High concentrations of copper in the coolant produce a highly adherent crud that significantly reduces thermal conductivity and causes temperatures to rise. The resultant rise in temperature accelerates the oxidation rate. This process can be self sustaining since increased oxidation rates further impede heat conduction and again raise temperatures.

The installation of stainless steel preheaters and better control of feed water purity levels seem to have solved the crud deposition problem.⁽²⁾

Uniform oxidation of fuel cladding in feed water systems with high purity levels does not seem to pose a threat under normal duty cycles. Corrosion rate enhancement due to irradiation is either small or nonexistent in PWRs, but can be as high as a factor of ten in BWRs.⁽¹⁷⁾ The accelerated oxidation kinetics are a function of both the fast flux (>1 MeV) and the amount of oxygen present. The latter influence is apparently related to the availability of nascent oxygen which is a product of radiolytic decomposition. Even the factor-of-ten enhancement in the oxidation rates for BWRs does not appear to influence clad integrity nor does it pose a serious threat to rod failure.

A second form of oxidation known as nodular corrosion has also been observed in BWRs.⁽¹⁸⁾ Nodules of ZrO_2 incubate, grow, then coalesce into patches that eventually spread into a continuous sheet. Extensive spalling has been observed in some BWR flow channels as a result of this process. Wide-spread growth of the nodular oxide has not been reported on fuel rods despite the fact that the surfaces exposed to oxidation are at higher temperatures than the channels. Since nodular growth rates can be as much as 10 times higher than in-reactor uniform corrosion rates, oxide layer thicknesses of up to 150 microns over the life of a fuel element are possible.⁽¹⁹⁾ Cladding strength could eventually deteriorate because of the substantial reduction in cladding wall thickness that accompanies oxidation, and increase the likelihood for mechanical failure. If the coalesced oxide patches do not spall away, reduced heat conductivity may cause the development of local hot spots and further increase the oxidation rates in the patch regions.

Failures caused by accelerated nodular corrosion may become more prevalent at higher burnups as crud layers thicken and the thermal resistance of the rod decreases. The influence of crud on oxidation has not been fully explored either in the steady state operating regime or under departure from nucleate boiling (DNB) conditions where embrittlement could occur faster than would normally be predicted on the basis of ex-reactor oxidation studies.

2.6 ROD BOW

Rod bowing has not been identified as a cause of fuel rod failure in Zircaloy-clad UO_2 fuel. Rod bowing is, however, considered as a critical heat flux limiting condition because coolable geometries could be altered. There appear to be four main causes of bow:

1. differential rod growth
2. differential expansion between cold and hot rods
3. circumferential temperature gradients
4. spacer grid interactions.

Although we are still uncertain of the exact mechanisms that produce bow, the inducement of axial compressive loads in one or more rods seems to provide the principal driving force. Many of the tie rods in BWR fuel assemblies, and the control element guide tubes of more recent PWR designs utilize the same alloy material as the fuel cladding. Thus, creep buckling forces caused by material differences should be small. However, compressive loads resulting from differential irradiation growth can develop if significant differences in crystallographic texture or cold worked substructure exist between rods since both can influence the growth component.⁽²⁰⁾ Irradiation growth and the differential expansion which occurs between cold and hot rods are probably the two main contributors to axial buckling.

Fuel column-cladding interaction as a consequence of fuel relocation and ratchetting is another possible contributor to rod elongation. The degree of interaction will be a function of the design gap and the amount of relocation that occurs during operation. It has been suggested that this mechanism should promote bowing in the inner fuel rods of an assembly where the fast flux is higher, the linear heat generation rates lower, and the propensity for fuel ratchetting increased.⁽⁵⁾ However, assembly bow has been observed to be directed away from the center of the core towards lower power positions,⁽²¹⁾ and towards the bottom of the fuel assembly.

Radial flux variations across a rod can also produce temperature gradients that could induce bowing. These gradients will be intensified to some

degree depending upon the location of the rod relative to a control rod. Fuel pellet eccentricity, cladding wall thickness variations, and ovality also accentuate temperature variations and may play supportive roles.

The effects of irradiation growth, differential expansion between hot and cold rods, or circumferential temperature gradients will be further enhanced by fuel rod spacer grid interaction. If failure finally occurs, it will most likely be a consequence of axial buckling or fretting.

No failure criteria currently exist for fuel rod bow; however, it should be possible to develop analytical models using the methods previously suggested for determining rod collapse.^(9,10)

2.7 OVERHEATING (STEADY STATE)

The steady state operating temperature requirements for Zircaloy fuel cladding generally range from 300° to 345°C. Temperature gradients through the wall section depend primarily upon heat flux, but can be as large as 30°C. This would increase operating temperatures along the inner surface of the fuel cladding to as high as 375°C. Accumulated layers of crud and oxide reduce thermal heat transfer efficiency and will further increase cladding temperatures by several tens of degrees, especially at high burnups. Fuel rods or bundles that are misplaced could also force higher-than-design heat ratings and higher resultant cladding temperatures. Thus, cladding temperatures can vary over a relatively broad range during the life of a fuel rod. Since Zircaloy cladding properties are known to be highly temperature dependent over this range of possible temperatures, an upper limit for steady state operation would seem advisable.

Irradiation damage is one of the factors which significantly alter the mechanical properties of Zircaloy. Strengthening is observed at low temperatures but the damage anneals out rather rapidly above 400°C,⁽²²⁾ and the tensile properties decrease accordingly. Uniform elongations do not seem to follow the conventional inverse relationship observed in unirradiated material, but remain constant with decreasing strength. This means that the

cladding loses strength without a corresponding increase in its ability to strain uniformly. At irradiation temperatures above 350°C, creep rates also increase significantly and appear to be dominated by thermally activated mechanisms.⁽¹¹⁾ A third factor of potential concern is oxide corrosion which can increase by an order of magnitude over the temperature span of interest.⁽¹⁷⁾ This process is potentially self sustaining because increased surface temperatures encourage crud deposition as well as additional oxidation. Both raise thermal resistance and, hence, cladding temperatures.

The considerations outlined above suggest the need for a maximum steady-state-over-heating limit of 400°C to guard against general cladding degradation. Structural reliability could be seriously reduced after extended exposure to temperatures above this level because of increased creep and oxidation.

2.8 CUMULATIVE DAMAGE

So far it has not been established that the previously discussed potential damage mechanisms are in themselves primary causes of failure in Zircaloy-clad LWR fuel rods. Collectively, these mechanisms could result in the accumulation of structural damage to the cladding after prolonged exposure and render the cladding susceptible to crack nucleation or propagation. The term cumulative damage is defined here as a weighting function that accumulates damage increments from all known sources. For example, operating power maneuvers that cause pellet relocation or that induce ratchetting could contribute a damage increment and an eventual lowering of failure thresholds with increasing burnup.

A life fraction rule based on a Larson-Miller parameter approach has been proposed for fast test reactor fuel cladding in an attempt to correlate thermal transient failure data with conventional mechanical property data.⁽²³⁾ A similar attempt at developing a rule for Zircaloy has not been reported and may not be practical since the test programs required to develop the data base would be prohibitive in cost. The best chance of understanding potential damage relationships may be through careful evaluations of commercial fuel rods with well documented power histories and with follow-on post irradiation examinations (PIE).

3.0 POTENTIAL FAILURE MECHANISMS

The two main causes of commercial LWR fuel failures have been ruptures resulting from pellet-clad interaction under power ramp conditions, and internal hydriding. Failures caused by swelling or bursting are postulated under loss of coolant accident (LOCA) conditions but are not expected under steady state operation. This is because internal pressures are normally designed not to exceed the external system pressure over the life span of the fuel. However, supplemental mechanical failure is possible if the Zircaloy cladding becomes brittle (as in the case of secondary hydriding). Criteria applicable to the failure of LWR fuel operating at steady state or exposed to normal operating transients are discussed in the following section.

3.1 PELLET-CLADDING INTERACTION

Fuel failures resulting from interaction between fuel and cladding (PCI) are well documented,^(2,3) and have prompted extensive research.⁽²⁴⁾ It is now generally accepted that a major cause of PCI failures is stress corrosion cracking (SCC) by either iodine as the corrosive species or by liquid metal embrittlement.⁽²⁵⁾

Two types of fractures have been identified:

- an axial split in the cladding that starts as a brittle fracture, but terminates in a ductile rupture; and,
- a pinhole puncture that is characterized by "X" or "V" markings on the outer surface of the cladding.

Postirradiation examinations of failed rods have revealed incipient cracks that penetrate part way through the wall of the cladding.⁽²⁶⁾ These incipient cracks are frequently coincident with radial cracks in the fuel. Chemical deposits containing large concentrations of fission products have also been discovered in the crack regions. These two factors combined have engendered the SCC concept that a highly localized stress field, together with a corrosive agent, is the principal cause of failure.

Measurements of localized strain displacement during deformation and fracture of irradiated Zircaloy have been reported in iodine⁽²⁷⁾ using commercial grade BWR cladding material. Miniature specimens cut from segments of the cladding wall allowed the loading to be oriented circumferentially with respect to the original tube geometry. Tests were performed for both plane strain and plane stress loading conditions. The plane strain specimens exhibited higher yield strengths and larger uniform elongation values compared to similar tests in plane stress. This relationship occurred in both the partially stress-relieved and fully recrystallized cladding. Lee and Adamson⁽²⁸⁾ also reported a lower uniform elongation value for plane stress. Their work suggests a strain-to-failure criterion based on microstructural evidence that iodine-induced cracking occurs in regions of high local strain where deformation bands intersect the surface.

The localized ductility experiments show that stress-relieved specimens will accommodate up to 4 to 5% plastic strain at maximum load under plane stress. These results indicate a higher resistance to localized shear deformation in partially recrystallized material as compared to a more fully annealed structure. More importantly, the high uniform strain values provide evidence that the irradiation-hardened structure is reasonably ductile and that strain localization is responsible for the observed low macroscopic ductilities.

A four-stage process has been proposed to explain SCC-assisted cracking.⁽²⁴⁾ The process is thought to be initiated by oxide penetration, followed by crack formation, crack propagation, and finally by rupture. Related burst testing indicated a 300 MPa failure stress threshold in I_2 at 360°C, which is about 75% of the yield strength for the materials tested. These conclusions were based on unirradiated tests, and as a result, irradiation damage mechanisms such as dislocation channeling were not operative. Also, the stress that exists in a burst specimen is not a good representation of the stress state that exists at a pellet ridge or over a crack. This shortcoming, coupled with the statistical variability inherent in the burst method of testing⁽²⁹⁾ makes the uncertainty in the proposed I_2 SCC stress threshold large.

The effects of preferred crystallographic orientation on stress corrosion susceptibility are difficult to evaluate because of the limited data available.⁽²⁸⁾ A zero degree texture appears to be more resistant to failure and requires a higher strain to initiate yielding, although one investigator discounts any intrinsic relationship between yield strength, texture, and SCC susceptibility.

Wood⁽³⁰⁾ and Coffin⁽³¹⁾ have shown that susceptibility to iodine embrittlement in unirradiated Zircaloy is strain rate dependent. Higher strain energy is required for failure when the strain rate increases. Lower strain rates apparently provide the time required for the corrosion process to advance and achieve the critical strain required for film rupture. Since the basic trends of iodine embrittlement in unirradiated Zircaloy are generally true for irradiated material, a similar strain rate dependency might also be expected. The improved resistance to failure at high strain rates suggests that fuel rods may also be less susceptible to PCI under rapid transient or under accident conditions.

Studies on internal tubing surface defects show a better time-to-failure correlation at constant burst pressure and iodine concentration with the type of flaw than with flaw depth.⁽³²⁾ However, the effects that flaws have on actual rod failures are not well known. Although some flaw-induced ruptures have been reported,⁽³³⁾ all PCI failures can not be related to flaw defects.⁽²⁶⁾ The manufacturing process used to make Zircaloy fuel cladding will likely produce a wide assortment of surface defects and asperities. Flaw detection limits are usually calibrated against artificial defects that are on the order of 10% of the wall thickness. This means that acceptable flaw depths could range from 50 to 75 microns--approximately twice as large as those studied by Tucker.⁽³²⁾ A systematic study of flaw size and geometry on PCI rupture should greatly improve failure predictions.

The revised NRC Standard Review Plan currently list two potential failure criteria for PCI (a uniform strain limit of one percent (1%); and, no fuel melting), but notes that the circumferential strain at failure in most PCI specimens are less than 1%. It is apparent that irradiated Zircaloy can

accommodate considerably higher strains if the stresses are localized, thus a 1% strain limit is somewhat ambiguous. Power coolant mismatch tests provide evidence that fuel melting will also not insure PCI failure,⁽³⁴⁾ perhaps because the plasticity of the fuel at extremely high temperatures, near the melting point, will more than accommodate the stress produced by differential thermal expansion.

In recognizing the problems associated with defining mechanistic criteria that are not ambiguous, efforts have been directed towards evaluating empirical relationships similar to the Canadian FUELOGRAM model.^(16,35) If we assume that the fuel rod population that constitutes the data base is reasonably homogenous, then it should be possible, by careful qualification of data, to define limiting operating criteria. It may also be possible to relate these criteria to mechanistic corollaries.⁽³⁶⁾ However, the potential shortcoming of any empirical model will be its limited application to the data population from which it was originally derived. The large economic incentives to improve fuel power cycling capabilities, coupled with active fuel improvement programs, will eventually generate other statistical populations as well as alternate fuel rod designs.

Perhaps a practical way to establish failure limits, based on mechanistic criteria, would be to devise a proof test. For example, a reasonable simulation of pellet/clad interaction and strain localization can be obtained with an expanded mandrel test similar to that used by Coleman⁽³⁷⁾ and Tomalin et al.⁽³⁸⁾ The mandrel test, originally used to confirm the Gittus⁽³⁹⁾ analysis, is a potential tool for establishing and verifying damage limits, especially if loading conditions can be made to simulate expected power cycles. The test should be able to provide indications of threshold sensitivities for most criteria, and should be able to provide important information on interactive effects. It could also be used to systematically evaluate the effects of prestrain and other cumulative damage contributors that have not, as yet, been investigated. The relevant criteria discussed previously are listed below in suggested order of decreasing importance.

- strain threshold
- strain rate
- chemical potential of aggressive species (I,Cd)
- critical flaw size
- stress state
- stress threshold
- preferred crystallographic orientation.

3.2 INTERNAL HYDRIDING - PRIMARY

Failures resulting from internal hydriding of Zircaloy-clad UD_2 fuel are well documented.^(2,3) These failures are characterized by blisters on the cladding surface that are a result of the 13% volume change that accompanies the $Zr-ZrH_{1.6}$ reaction.⁽⁴⁰⁾ The internal source of hydrogen has generally been attributed to moisture in the fuel, although potential quantities of hydrogenous impurities or residual hydrogen from the sintering operation that is normally performed in H_2 or NH_4 atmospheres are potentially significant. Manufacturing techniques now include rigorous steps to dry the fuel, and there has been a strong decline in the number of reported failures since these precautionary measures were implemented. A practical limit below which the cladding is not susceptible to failure seems to be 2.0-2.5 mg of H_2O/cm^3 .⁽⁴¹⁾

Why the hydrogen is absorbed locally is not clearly understood. It is generally thought that the hydrogen attack occurs preferentially, at flaws or cracks in the protective oxide layer. These flaws or cracks could result from abrasion during the pellet loading operation or from surface defects caused by prior manufacturing operations. Fluorine may act as a catalyst and increase the probability of localized attack by accelerating the deterioration of the protective oxide layer. Residual fluorides are most likely to be found at flaws where the final cleaning operations have failed to penetrate cracks or crevices on the inner tubing surface.

Susceptibility to hydrogen attack may also be expected in heat-affected regions near end plug welds where the protective oxide has been taken into

solution. Gradual hydriding of these colder end regions can occur as the hydrogen migrates down the temperature gradient.

An acceptable moisture level for Zircaloy fuel and poison rods is defined in the NRC-Standard Review Plan as less than 20 ppm. This limit is very conservative and should effectively reduce susceptibility to failure.

3.3 INTERNAL HYDRIDING - SECONDARY

Secondary hydriding is a term that is used to identify hydrogen embrittlement that occurs as a result of a primary failure and the ingress of additional quantities of moisture from the coolant. In heavy water-moderated reactors, detrium can be used to separate secondary effects from primary effects, but in light water reactor systems, it is not always possible to determine the original source of the hydrogen. This shortcoming is of concern since hydride failures could conceal the primary mode of rupture.

The pinhole rupture that results from PCI is a type of primary failure that will be most difficult to detect. The defect is so small that the quantity of radioactive fission products released into the primary coolant may not be of sufficient magnitude to be detected by plant systems. Ingress of water over a period of time could eventually lead to significant accumulations of hydrogen in the fuel rod. This hydrogen would be absorbed in the end plug region because of the lower temperatures and the lack of a protective oxide film. Complete severance of one or both end plugs could ultimately occur during subsequent fuel handling or during cyclic operation. Severed ends could also cause a major dislocation of a fuel rod from its normal position. Such possibilities emphasize the importance of establishing the basic mechanisms and frequencies of pinhole ruptures.

3.4 BURST RUPTURE

A practice of designing fuel rods so that the internal rod pressure does not exceed the external system pressure over the design life of the fuel should prevent burst rupture during normal steady-state operation. This design requirement also provides a margin of safety for abnormal operating

transients. However, a substantial data base now exists on the mechanical properties of irradiated Zircaloy^(11,42) that, coupled with information obtained from studies on Zircaloy-2 pressure tubes,^(43,44) could be used to develop applicable design limits based on either an effective stress or strain energy absorption to failure (SEAF) criteria.

Neither hydrogen embrittlement nor irradiation hardening seem to significantly affect fracture toughness⁽⁴³⁾ at operating temperatures. Thus, a reasonably well behaved ductile response is to be expected under generalized strain (non PCI) conditions.

3.5 MECHANICAL FRACTURE

Little data is currently available to define failure limits under seismic load conditions. A limiting stress criterion would be helpful and should be supplemented with a fracture toughness limit and perhaps a criterion for minimum crack growth. The principal fracture concern is embrittlement from oxidation or the absorption of hydrogen, but as discussed in Section 2.5, embrittlement by oxidation at normal operating temperatures is not a major problem. Hydrogen also has little effect at operating temperatures, but at room temperature, concentrations in excess of 100 ppm can significantly reduce fracture toughness.⁽⁴³⁾ Thus, segments of cladding material that have absorbed significant concentrations of hydrogen during operation (either by corrosion or by secondary hydriding) could be susceptible to fracture following shutdown.

Embrittlement is also a concern where hydrides precipitate radially (with respect to the cladding wall), because the platelets provide pathways that assist the propagation of direct thru-wall cracks. The direction of these precipitates can be influenced by post forming operations (principally straightening) as well as by the sign of an applied stress field imposed during operation. Failures might be possible during a seismic event at shutdown, or during fuel handling operations, if rough treatment is allowed.

Based on existing data, a hydrogen limit of 100 ppm over the design life of the fuel cladding would seem to be a realistic criterion for protection against room temperature embrittlement. A stress limit at some fraction of

the yield strength could also prove useful. A recent review of plane strain fracture toughness data ⁽⁴⁵⁾ indicates that a value of 22 MPa $\sqrt{m}$ (20 ksi $\sqrt{in.}$) is a realistic lower limit and could be used as a criterion where toughness requirements are important.

4.0 POSTULATED ACCIDENTS

Acceptance criteria are used to assure that fuel systems retain functional capabilities during postulated accidents. These are accident events which, if they were to occur, might result in undesirable radiological exposure to the public.

The American Nuclear Society classified potential plant accident conditions into four categories as follows:

- Condition I Normal Operation
- Condition II Incidents of Moderate Frequency
- Condition III Infrequent Incidents
- Condition IV Limiting Faults.

A Condition II event, such as a loss of offsite power, is not expected to result in a fuel rod failure, although some damage may be expected. A Condition III occurrence, such as an insertion of unexplained reactivity, is to be expected on an infrequent basis, and only a small fraction of the fuel elements would fail. Condition IV events, such as a major pipe rupture or control rod ejection, are classified as faults that are not expected to occur. However, if they did occur, significant amounts of radioactivity could be released. These events generally represent the limiting design case.

The NRC Standard Review Plan⁽¹⁾ divides acceptance criteria into three categories: 1) fuel damage, 2) fuel failure, and 3) fuel coolability. Criteria that are most applicable to Condition II, III, and IV events are listed as:

- departure from nucleate boiling (failure criteria)
- fuel melting (failure criteria)
- 280 cal/gm radially averaged peak fuel enthalpy to prevent violent expulsion of fuel (coolability criteria)
- 2200 F (1204°C) peak clad temperature (coolability criteria)
- 17% oxidation of total clad thickness (coolability criteria).

Although accident conditions can arise from a variety of possible scenarios, they can be grouped into three general classifications: 1) loss of cooling, 2) reactivity increases, and 3) power-coolant mismatch (PCM) that combines the effects of 1 and 2 above.

4.1 LOSS OF COOLING

The most severe loss of coolant accident (LOCA) is a result of a postulated pipe rupture and a subsequent blowdown of the core. Acceptable criteria used to bound the event and to prevent complete melting of the core include:

- cladding oxidation less than 17% of the original wall thickness,
- all changes in fuel rod geometry that remain amenable to cooling during the corrective reflood phase, and;
- a maximum cladding temperature not greater than 1204°C.

Experimental investigations^(24,46) are underway to expand the data base needed for reliable calculations and to define the degree of conservatism that may be manifested in the current set of criteria. Oxidation kinetics have been defined in the high temperature beta regime where the maximum peak clad temperatures are expected. The results of these oxidation studies are proving useful in calculating oxidation and in determining the added temperature increment caused by the heat of the reaction.

Work at the Argonne National Laboratory⁽⁴⁷⁾ suggests that the 17% criteria may be overly conservative if the mechanical loads result only from thermal quenching or post accident handling. Embrittlement criteria, based on total absorbed energy, dynamic fracture toughness, or on the fractional thickness of the beta phase with <1 wt% oxygen, seem to show promise as alternate criteria. These limits do not account for the added effect of hydrogen embrittlement, which has been reported to be significant when the volumetric H_2/H_2O ratio is 0.2-0.4.⁽⁴⁸⁾ Near stagnant steam flow conditions would be required to achieve these ratios in a reactor core since the principal

sources of hydrogen are metal-water reaction and radiolytic decomposition. Less than 4% of the available hydrogen generated by the corrosion reaction is absorbed by the metal when the mass flow rates are on the order of 0.45 kg/hr.⁽⁴⁹⁾

The requirement to demonstrate a coolable geometry has been aided by a vastly improved data base that includes the maximum circumferential strain as a function of burst temperature⁽⁵⁰⁾ and the transient deformation properties of Zircaloy under biaxial stress loadings.⁽⁵¹⁾ Functional relationships between effective failure stress and temperature have been developed for both stress relieved and fully recrystallized tubing.⁽⁵²⁾ These relationships can be used to determine uniform dilatation and ballooning strains and assess failure probabilities as functions of fuel pressures or dimensional variations. The effective stress for failure at the maximum peak cladding temperature limit (1204°C) is approximately 6.8 MPa, but increases significantly at temperatures below 927°C.

Localized strains caused by pellet cladding interaction are not expected in a loss of coolant incident. Fuel temperatures drop rapidly following a reactor scram and the redistribution of heat to the cladding, which is not being cooled as efficiently, will generally cause rapid gap development.

With the expanded data base now available, cladding oxidation and deformation can be predicted under temperature ramp conditions. An overheating limit on temperature coupled with emerging embrittlement criteria and knowledge of the temperature-dependent mechanical properties, should provide a sufficient basis for assessing fuel system integrity over the full spectrum of possible path-dependent LOCA transients.

4.2 REACTIVITY INCREASES

A range of reactivity transients that could result in varying degrees of reactor overpower are possible. The most severe are called reactivity insertion accidents (RIA). They can be caused by a control rod ejection in a PWR or a control rod drop in a BWR. Large and rapid depositions of energy in the fuel may cause melting, fragmentation, or even dispersal of fuel. The

mechanical action resulting from fuel dispersal could be sufficient to destroy the cladding or the rod bundle geometry and produce pressure pulses in the primary system. Therefore, the criterion used to avoid fragmentation or dispersal are based on the total energy deposition from the transient. NRC Regulatory Guide 1.77 limits the radially averaged peak enthalpy to 1.17 J/kg (280 cal/g).

Actually, two thresholds can be defined. One for the energy deposition level at which fuel cladding failures begin to occur and one for the energy deposition level at which fuel fragmentation or fuel dispersal can be expected. The threshold for cladding failure results from that combination of cladding temperature and hydrostatic pressure within the fuel rod that will initiate a breach of the cladding. Earlier experiments^(53,54) indicated that pre-irradiated fuel rods fail at lower energy deposition levels than do unirradiated rods. The cladding failure threshold for unirradiated fuel rods was reported to be in the range of 0.94 to 1.13×10^6 J/kg (225 to 270 cal/g), and as low as 0.36 J/kg (85 cal/g) for previously irradiated fuel.

Fuel fragmentation appears to occur between 1.17 and 1.25×10^6 J/kg (280 to 300 cal/g) for unirradiated fuel. The fuel fragmentation failure threshold for irradiated fuel has not been firmly established.

Recent work⁽⁵⁵⁾ at the NSRR reactor in Japan confirms the cladding failure threshold for unirradiated fuel at 1.09 J/kg (260 cal/g), but places the fuel fragmentation failure threshold at 1.59 J/kg (380 cal/g). Cladding failure thresholds were reported to decrease with increasing enrichment and internal rod pressures, but were insensitive to gap width, cladding heat treatment, and to pellet shape. Tests scheduled in the near future at the Power Burst Facility, located at the Idaho National Engineering Laboratory, are expected to yield important new data that will help define failure thresholds for both irradiated and unirradiated fuel.

Transients that are less severe than a control rod ejection or rod drop are also possible and also have a higher probability of occurrence. A modest increase in reactivity (and power) can occur if cooling efficiency is increased or the temperature of the moderator reduced. Increases in fuel

temperatures that accompany the rising power will induce PCI during the early stages of the transient. Whether or not the cladding will be susceptible to SCC assisted rupture under rapid reactivity transient conditions, compared to operational power cycling, is not clear. Zircaloy exhibits a greater resistance to failure at higher strain rates, but this could be counteracted by an increase in cladding temperature. Damage criteria are also needed to define acceptable performance for fuel rods that enter the transient in a defected condition, and additional criteria may be needed to assess subsequent failure probabilities of unbreached rods should steady state operation be resumed.

The importance of oxidation in a reactivity incident will depend largely on the time of the transient and on the maximum temperature achieved. At high energy depositions, local temperatures could be very high, even to the point of melting, and embrittlement from oxidation could be severe. Rapid expansion of the internal gas volume may also cause local ballooning owing to the high temperatures and resultant loss in strength. In lesser transients, moderate temperature increases into the low beta phase region will not produce significant oxidation, making reuse of the fuel possible if PCI rupture does not occur. More data on oxidation kinetics in the 400-800°C range are needed and the effects of post-transient operation should be studied.

4.3 POWER COOLANT MISMATCH (PCM)

A series of power coolant mismatch experiments have been performed on PWR-size rods in the Power Burst Facility. These tests⁽³⁴⁾ indicate that overheating caused by film boiling will not always cause cladding failures. Elapsed times in departures from nucleate boiling (DNB) of up to 210 sec at peak cladding temperatures as high as 1152°C were experienced without failure. When failure finally occurred, it was a consequence of excessive embrittlement by oxidation apparently initiated by the thermal shock following reactor scram, and in one case by post-test handling.

Centerline fuel melting developed in a few tests but did not by itself cause failure. Chemical interaction between the fuel and cladding did contribute to increased internal oxidation.

The data available on fuel rod behavior under film boiling conditions may now be sufficient to develop alternate criteria to the "no departure from nucleate boiling" (DNB) rule currently in force. Studies at Argonne National Laboratory⁽⁴⁷⁾ show that oxidation embrittlement occurs if less than 0.1 mm of beta phase across the cladding wall thickness contains less than 1 wt% oxygen. The deformation and rupture data^(51,52) developed for LOCA calculations should also be applicable to PCM incidents and may be useful in establishing time-dependent mechanical failure limits.

No evidence of PCI-type failures have been reported in the PCM tests performed to date. The fact that some relatively severe and rapid power ramps were imposed following the preconditioning phase of these experiments may be an indication that PCI failure is controlled by different variables during fast transient exposure. Power, delta power, and ramp rate relationships that are known to produce PCI failures following operational power ramps may not be valid during an accident-type transient. The mechanical compliance of the fuel that controls the severity of the PCI interaction will be a function of the integrated energy input and the time-temperature history of the transient. Lee⁽⁵⁶⁾ has shown that strain at failure will depend upon the strain rate that prevails during the fracture process, and in a fast transient power rise, the fuel will not creep or relax very much and can impose a higher strain rate to the cladding. The Zircaloy cladding could also have a higher resistance to SCC assisted failure if the temperature doesn't increase rapidly prior to fuel-cladding separation.

5.0 POTENTIAL DAMAGE/FAILURE RELATIONSHIPS

Potential interactions between damage and failure mechanisms previously discussed are shown in Figure 1. These relationships were intended to represent those that might be possible for either steady-state conditions or operational transients. The suggestion of mechanistic interplay does not obviate unique or fundamental relationships between stress, strain, strain rate, temperature, or irradiation. The diagram simply attempts to relate associated processes, such as the buildup of crud that will enhance oxidation and increase the cladding temperature by some incremental amount. In this case, slightly higher temperatures will in turn contribute to higher creep rates that can speed the process of rod collapse--assuming all other conditions necessary to support the collapse mechanism were present.

The fatigue process involves cyclic or reversed loading. If the processes that cause plastic flow, instability, and failure are not independent of prior strain and strain rate history, then fatigue loadings could hasten other failure processes such as PCI and mechanical fracture.

Damage effects may also accumulate (as discussed previously) and provide the driving force to activate other mechanisms. A pinhole produced from PCI that allows ingress of water, for instance, would increase the chance for massive secondary hydride failures. It may never be feasible, however, to express cumulative damage in terms of extensive parameters that are the summation of all subset contributions. Despite this lack of formalism, it is still useful in an engineering sense to consider the potential cumulative effects resulting from exposure to one or more damage mechanisms, particularly at high burnups.

Similar mechanistic relationships for accident events are suggested in Figure 2. Many time-dependent factors have been eliminated because most accidents are expected to be of a relatively short duration. The exception is oxidation, which is normally considered to be a time-dependent function. Accelerated rates are still expected under LOCA or PCM conditions because the kinetics are dominated by temperature.

Rod collapse or buckling are both possible during a departure from nucleate boiling (DNB) if the system pressure increases relative to the internal rod pressure, and the cladding temperature rises significantly. Mechanical fracture resulting from embrittlement may also occur if the time at temperature is extended. If these events are initiated by a reactivity excursion, then pellet cladding interaction can develop prior to the onset of DNB. The forces induced by fuel and cladding interaction will be magnified if pellet cladding contact exists at the start of the transient or if prior rod collapse has occurred. Annealing out of irradiation-induced defects will be a function of the accident time/temperature events and is assumed here to be inherent in the mechanisms discussed.

FIGURE 1. Potential Damage/Failure Relationships For Zircaloy Clad UO₂ Fuel Under Normal Operating Conditions

FIGURE 2. Potential Damage/Failure Relationships For Zircaloy Clad UO_2 Fuel Under Accident Conditions

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