~~OFFICIAL USE ONLY~~

~~ENG.20050325.0012~~ 5/17/05

MOL.20050518.0327

DT#14482

QA: N/A

000-30R-MGR0-00700-000-000

March 2005

BECHTEL SAIC COMPANY LLC

Commercial Spent Nuclear Fuel Handling in Air Study

Prepared for:

U.S. Department of Energy
Office of Civilian Radioactive Waste Management
Office of Repository Development
1551 Hillshire Drive
Las Vegas, Nevada 89134-6321

Prepared by:

Bechtel SAIC Company, LLC
1180 Town Center Drive
Las Vegas, Nevada 89144

Under Contract Number
DE-AC28-01RW12101

~~OFFICIAL USE ONLY~~
~~May be exempt from public release under the Freedom of Information Act (5 U.S.C. 552), exemption number and category: 5~~
~~Attorney-Client Privilege~~
~~Department of Energy review required before public release~~
~~Name/Org: Diane Quenell Date: 5/6/05~~

Does Not Contain "Official Use Only" Information
Diane Quenell DOE/ORD
5/06/05

~~OFFICIAL USE ONLY~~

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, nor any of their contractors, subcontractors or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or any third party's use or the results of such use of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof or its contractors or subcontractors. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

000-30R-MGR0-00700-000-000 March 2005

Commercial Spent Nuclear Fuel Handling in Air Study

Originators:

P. McDaniel
Mechanical Handling/Waste Systems Lead — Date: 3/25/05

K.E. Schwartztrauber
Project Engineer — Date: 3-25-05

M. Johnson
Team 1 Lead — Date: 3/25/05

T. Frankert
Team 2 Lead — Date: 3/25/05

N. Kahler
Team 3 Lead — Date: 3/25/05

J. Schulz
Team 4 Lead — Date: 3/25/2005

D. Equilbec, COGEMA ODS
Team 5 Lead — Date: 3/25/2005.

Approved by:

B.E. Rusinko
Engineering Production Manager — Date: 3/25/05

CHANGE HISTORY

Revision Number	Interim Change No.	Date	Description of Change
00		03/25/2005	Initial issue

INTENTIONALLY LEFT BLANK

EXECUTIVE SUMMARY

The current design strategy for the Yucca Mountain Project (YMP) surface waste handling facilities is to have systems for handling commercial spent nuclear fuel (CSNF) assemblies in dry air. In the event that the uranium dioxide fuel (UO_2) contained within the CSNF cladding is exposed to oxygen at the temperatures anticipated in the handling process, the fuel is expected to oxidize. Approximately 96 percent of the CSNF fuel assemblies to be handled have intact cladding and should not experience fuel oxidization while being handled in air. The balance of the assemblies are expected to have varying amounts of cladding damage that could lead to fuel oxidization when the assemblies are handled in air. The majority of fuel with cladding damage will be identified via the reactor records, but a small percentage of assemblies (approximately 0.4 percent or one thousand fuel assemblies) is expected to have unknown or undetected cladding damage that could allow the fuel to oxidize when handled in air. As the UO_2 oxidizes, it changes from a sintered pellet form to U_3O_8 powder, which consists of micron-sized particles. If the U_3O_8 powder is released from the fuel cladding, it will result in high levels of radioactive contamination in the casks, waste packages, and fuel transfer areas of the waste handling facilities.

This CSNF handling study evaluates the handling of CSNF in air and packaging activities in the repository surface facilities. It evaluates the current design, further identifies the fuel performance issues, develops the consequences, and presents the operational considerations associated with the routine handling of CSNF in air. Emphasis is on the process of oxidation of uranium dioxide fuel and additional oxidation-driven failure of fuel assembly cladding. Key issues are identified, and plans to address these issues are proposed. This study provides a basis for ongoing work, future evaluations, decisions, and a path forward to further support design development and the license application (LA).

The study was performed by evaluating five key areas related to handling CSNF in air:

1. Design bases for handling fuel in air
2. Intact fuel cladding performance
3. Failed fuel (breached cladding) performance
4. Contamination and dose consequences associated with handling operations
5. Preventation, mitigation, and recovery actions.

Multidisciplinary teams were formed to focus on each of the five areas. The teams consisted of a team leader and team members from the engineering staffs of U.S. Department of Energy (DOE), Bechtel SAIC Company, LLC (BSC), Management and Technical Services, and Compagnie Générale des Matières Nucléaires (COGEMA). Technical experts from AREVA/COGEMA/Framatome-ANP, Pacific Northwest National Laboratory, Atomic Energy of Canada Limited, Argonne National Laboratory (ANL), and other fuel specialists provided input to the teams.

Design Bases for Handling CSNF in Air

The study concluded that the following preliminary design bases should be used to develop the surface facility design.

- 400°C is the proposed maximum operating temperature for handling CSNF cladding (failed or unfailed) in normal operations.
- 460°C is the proposed maximum temperature for handling CSNF cladding (failed or unfailed) in off-normal operations.
- Failed fuel quantities and types of damage arriving at the repository include: approximately 4 percent of fuel assemblies received are expected to have an average of 2.2 failed fuel rods per assembly; 90 percent of the failed fuel is estimated to be known and identified prior to shipment to the repository; 10 percent of the failed fuel is expected to have pinhole leaks and hairline cracks that will not be identified prior to shipment.

CSNF Oxidation

The current surface waste handling facilities design strategy is to handle CSNF in air. CSNF arrives at the repository at an assumed rate of about 9,000 assemblies per year. During handling operations in the surface facilities, a typical CSNF fuel assembly is expected to be in air for over 100 hours at temperatures up to 400°C. At these times and temperatures, fuel oxidation is expected for failed fuel during normal waste handling operations.

It is expected that about four percent of CSNF assemblies shipped to the repository will include at least one fuel rod classified as failed fuel (CSNF with damaged cladding that allows air to come in contact with the uranium dioxide fuel, UO_2). At temperatures above 250°C, the UO_2 fuel when exposed to air will begin to oxidize and the rate of oxidation depends on time and temperature.

The consequences of fuel oxidation on surface facility operations are uncertain because of the limited amount of information available on fuel oxidation. Specific areas where more information is needed are:

- Parameters affecting oxidation—The study concludes that time and temperature are the primary parameters that affect fuel oxidation. Other variables such as burnup, radiolysis, cladding defect size, and fuel type are estimated to have a secondary effect on fuel oxidation.
- Oxidation and fuel clad unzipping rate—The basis for calculating the rate of fuel clad failure and oxidation needs to be further evaluated and documented.

- Release fractions during oxidation—Release fractions of gaseous fission products, volatile fission products, and oxidized fuel fines need to be determined. The release fractions used in the study may not be conservative and may underestimate the amount of radioactive materials released from oxidized fuel during fuel handling operations.

Contamination and Dose Rates

The contamination levels and dose rates resulting from normal handling of CSNF in air are expected to be much higher than desirable. This is because failed fuel will be handled in air, the failed fuel will oxidize, and fuel fines and volatile radionuclides will be released from the fuel cladding. An estimate of the rate of contamination buildup was made and it was determined that it will take approximately 4 to 40 days of operation to contaminate the fuel transfer cell to a level that may impact radiological safety and require periodic decontamination. The study concludes that this rate of contamination buildup is considered unacceptable.

Material Control and Accounting

As a result of fuel oxidation during handling operations, oxidized material released from fuel rods will be difficult to control and account for, as required under the Yucca Mountain material control and accounting plan.

Criticality

A preliminary criticality analysis concludes that the expected amount of oxidized material released is less than the amount determined to be a criticality concern. However, the uncertainty with oxidation rates and release fractions needs further evaluation to determine if this preliminary analysis is conclusive for normal and accident events.

Recommendations

Recommendations are presented for future work and evaluations to support surface facility design development and the license application. Based on the conclusions and the significant areas of uncertainty related to the conclusions, three options are presented for proceeding with the surface facility design development.

1. Option 1, prevent fuel oxidation. Design features and operating strategies related to this approach are presented in Section 8.

2. Option 2, handle all CSNF in air, as presented in the current design and operational plan. This approach requires that more information on the oxidation and contamination processes be developed so that the consequences and impacts on design and operations can be assessed. Design features to mitigate the consequences of oxidation would be required. Recommendations associated with this approach are presented in Section 8.

3. Option 3, limit handling of CSNF in air to small time durations. This approach is a combination of Options 1 and 2 and would keep CSNF in an inert environment except for short periods when fuel is being transferred between containers, such as between a cask and the waste package or a cask and the staging tubes. The short time that the fuel is exposed to air during transfer is not expected to result in fuel oxidation. Depending on the specific design selected, the design features and operating strategies will be a combination of those presented in Options 1 and 2.

CONTENTS

Page

CHANGE HISTORY V

EXECUTIVE SUMMARY VII

ACRONYMS XV

1. INTRODUCTION 1-1
   1.1 PURPOSE 1-1
   1.2 SCOPE 1-1

2. BACKGROUND 2-1

3. DESIGN BASES FOR HANDLING CSNF IN AIR 3-1
   3.1 CURRENT DESIGN BASES 3-1
   3.2 ANALYSIS 3-2
   3.3 FINDINGS 3-11

4. INTACT FUEL CLADDING PERFORMANCE 4-1
   4.1 FUEL TEMPERATURES 4-1
   4.2 OXIDATION OF ZIRCONIUM CLADDING IN AIR 4-1
   4.3 CLADDING CREEP FOR INTACT FUEL 4-3
   4.4 HYDRIDE RADIAL REORIENTATION 4-7
   4.5 FINDINGS 4-10

5. FAILED FUEL PERFORMANCE 5-1
   5.1 OXIDATION PROCESS 5-1
   5.2 KEY PARAMETERS FOR OXIDATION 5-5
   5.3 FAILED FUEL PERFORMANCE ISSUES 5-8
   5.4 FINDINGS 5-9

6. CONTAMINATION AND DOSE CONSEQUENCES 6-1
   6.1 CURRENT DESIGN AND CONSEQUENCES OF OXIDATION 6-1
   6.2 RELEASE FRACTIONS FROM OXIDATION 6-2
   6.3 HIGH TEMPERATURE FUEL OXIDATION 6-5
   6.4 EXPECTED FAILED FUEL ARRIVAL RATES AT THE REPOSITORY 6-6
   6.5 INPUTS TO CONSEQUENCE ANALYSIS 6-7
   6.6 CONSEQUENCE ANALYSIS 6-8
   6.7 CONTAMINATION IN THE FUEL TRANSFER CELL 6-17
   6.8 REVIEW OF DOSE CONSEQUENCE AND CONTAMINATION ISSUES 6-19
   6.9 FINDINGS 6-22

7. PREVENTION, MITIGATION, AND RECOVERY ACTIONS 7-1
   7.1 DESCRIPTION OF PROCESS FOR HANDLING FUEL IN AIR 7-1
   7.2 PROCESS ANALYSIS 7-5

CONTENTS (Continued)

Page

7.3 POTENTIAL DESIGN FEATURES 7-7
7.4 REVIEW OF THE IDENTIFIED ISSUES 7-12
7.5 FINDINGS 7-16

8. CONCLUSIONS AND RECOMMENDATIONS 8-1
8.1 CONCLUSIONS 8-1
8.2 RECOMMENDATIONS 8-3

9. REFERENCES 9-1
9.1 DOCUMENTS CITED 9-1
9.2 CODES, STANDARDS, REGULATIONS AND PROCEDURES 9-10

APPENDIX A GLOSSARY A-1

APPENDIX B NOV 5, 2004, SAR EXTRACTIONS ON CONTAMINATION REQUIREMENTS B-1

APPENDIX C PRECLOSURE CONSEQUENCE ANALYSIS FOR LICENSE APPLICATION ASSUMPTIONS C-1

APPENDIX D POTENTIAL HAZARDS DURING NORMAL AND OFF-NORMAL OPERATIONS IN FHF D-1

APPENDIX E FUEL CHARACTERIZATION BASED ON FUEL INSPECTION RECORDS E-1

FIGURES

Page

4-1. Stress Distribution for PWR Fuel Rods 4-4
4-2. Rod Failure Rate versus Storage Temperature for 500 Hour Storage 4-5
4-3. Rod Failure Rate versus Storage Times for Storage at 400°C 4-6
4-4. Creep Failures at 570°C as a Function of Storage Time 4-8
4-5. Dissolution and Precipitation Solubility for Hydrogen in Irradiated Zirconium Metal 4-9
5-1. Clad Unzipping Velocity 5-4
7-1. Fuel in Air Study Process 7-6
E-1. Example of End Cap Weld Failure E-9
E-2. Example of Grid Fretting E-10
E-3. Example of Debris Fretting E-11
E-4. Examples of PCI Cracks E-12
E-5. Examples of Corrosion Failures E-13
E-6. Example of Hydride Blister with Cracks E-15
E-7. Example of Axial Crack E-16
E-8. Diagram Showing Secondary Failures after Initial PCI Cracking E-17

TABLES

Page

3-1. Failed Fuel Estimates for BWR and PWR Assemblies 3-4
3-2. Rod Failure Rates for United States PWR and BWR Fuel 3-7
3-3. Comparison of Fuel Reliability from Various Sources 3-8
3-4. Characterization of Fuel Assemblies as Received at YMP 3-10
4-1. Zirconium Oxidation in Air 4-2
4-2. Comparison of High Temperature Creep Tests with the Creep Model 4-7
4-3. Temperatures Necessary to Precipitate Hydrides Dissolved at 400°C 4-9
5-1. Damaged Fuel Parameters at 200°C, 300°C, and 350°C 5-6
5-2. Damaged Fuel Parameters at 400°C, 500°C, and 570°C 5-7
6-1. Normal Operation Dose Consequences—Current Design 6-1
6-2. Yearly Receipt of Damaged Assemblies 6-6
6-3. Normal Annual Operation Dose Consequences 6-9
6-4. Dose Rates Three Feet Above Floor 6-10
6-5. Surface Dose Rate at Each Filter 6-11
6-6. Loss of HVAC TEDE Inhalation Doses 6-12
6-7. Sensitivity Analysis Results 6-14
6-8. Amount of Airborne Material Deposited (kg/yr) Sensitivity Analysis Results 6-14
6-9. Normal Operation Dose Sensitivity Analysis Results—TEDE 6-14
6-10. Worker Dose Rate Sensitivity Analysis Results—3 feet above floor 6-15
6-11. HEPA Filter Dose Rate Sensitivity Analysis Results 6-15
6-12. Loss of HVAC Sensitivity Analysis Dose Results 6-15
6-13. Bounding Amount of Airborne Material Deposited (kg/yr) 6-16
6-14. Bounding Normal Operation TEDE Doses—TEDE 6-16
6-15. Bounding Maintenance Worker Dose Rates 6-16
6-16. Bounding HEPA Filter Dose Rates 6-16
6-17. Bounding Loss of HVAC Inhalation TEDE Doses 6-16
D-1. Temperature Criteria for Cask Components D-2
E-1. Summary of Framatome-ANP UT Inspections through 1998 ..... E-2
E-2. Number of Rods Failed per Assembly from the Survey of 222 Failed Assemblies E-4
E-3. Reconstituted Assemblies from 10/91 through 6/01 ..... E-5
E-4. Recaged Assemblies from 1/98 through 11/96 ..... E-8
E-5. Estimation of Fraction of Failed Assemblies that are Reconstituted or Recaged E-8

ACRONYMS

ALARA	as low as is reasonably achievable
ANL	Argonne National Laboratory
ARF	airborne release fraction
BSC	Bechtel SAIC Company, LLC
BWR	boiling water reactor
CCDF	complimentary cumulative distribution function
COGEMA	Compagnie Générale des Matières Nucléaires
CSNF	commercial spent nuclear fuel
DOE	U.S. Department of Energy
DPC	dual-purpose canister
DR	damage ratio
DTF	Dry Transfer Facility
EPRI	Electrical Power Research Institute
FHF	Fuel Handling Facility
GROA	geologic repository operations area
GWd	gigawatt day
HEPA	high-efficiency particulate air (filter)
HLW	high-level radioactive waste
HVAC	heating, ventilation, and air-conditioning
ISG	interim staff guidance
ITS	important to safety
LA	license application
LCO	limiting condition for operation
LLW	low-level radioactive waste
LPF	leak path factor
MTHM	metric tons of heavy metal
MTU	metric tons of uranium
NRC	U.S. Nuclear Regulatory Commission
PCI	pellet cladding interaction
PCSA	preclosure safety analysis
PWR	pressurized water reactor

ACRONYMS (Continued)

RF	respirable fraction
SAR	safety analysis report
SNF	spent nuclear fuel
SRTC	site rail transfer cart
SSCs	systems, structures, and components
TEDE	total effective dose equivalent
TSPA	total system performance assessment
UT	ultra-sound inspection
USL	upper subcritical limit
YMP	Yucca Mountain Project

1. INTRODUCTION

1.1 PURPOSE

This CSNF study evaluates the handling of fuel in air during processing and packaging activities in the Yucca Mountain surface facilities. It evaluates the current design, further identifies the issues, develops the consequences, and presents the operational considerations associated with the routine processing of CSNF in air. Emphasis is on the implications of oxidation of uranium dioxide fuel and the potential failure of fuel assembly cladding. Key issues are identified, and plans to address these issues are proposed. This study provides a basis for ongoing work, future evaluations, decisions, and a path forward to further support design development and the LA.

1.2 SCOPE

The study was performed by evaluating five technical areas related to handling CSNF in air:

1. Design bases for handling fuel in air
2. Intact fuel cladding performance
3. Failed fuel (breached cladding) performance
4. Contamination and dose consequences associated with handling operations
5. Prevention, mitigation, and recovery actions.

Multidisciplinary teams were formed to focus on each of the five areas. The teams consisted of a team leader and team members from the engineering staffs of DOE, BSC, Management and Technical Services, and COGEMA. Technical experts from AREVA/COGEMA/Framatome-ANP, Pacific Northwest National Laboratory, Atomic Energy of Canada Limited, ANL, and other fuel specialists provided input to the teams and technical reviews of the study results.

This engineering study was developed, reviewed, and issued as a QA: N/A document for information purposes only in accordance with LP-ENG-014-BSC, *Engineering Studies*. This study is considered to be preliminary work. The technical information included in this study should not be used as qualified design input.

INTENTIONALLY LEFT BLANK

2. BACKGROUND

The current Yucca Mountain repository design for handling CSNF in the surface facilities includes handling CSNF in dry air. Typical CSNF handling processes at nuclear power plants are performed wet in spent fuel pools. CSNF exposure to air at utility facilities is limited to cask and canister evacuation and SNF drying prior to dry storage in an inert atmosphere. Handling CSNF in air for extended periods of time (greater than 24 hrs) for operations similar to the repository does not have a clear licensing precedent with the U.S. Nuclear Regulatory Commission (NRC). The nuclear safety design bases and the associated surface facilities CSNF handling design features need to be developed to support a licensable design.

CSNF will arrive at the repository at an assumed rate over 25 years of about 9,000 assemblies per year (BSC 2003 [DIRS 165990], Table 2). It is expected that approximately 4 percent of CSNF assemblies shipped to the repository will include at least one rod classified as failed fuel (known leakers or suspected fuel with pinhole leaks or hairline cracks). Some of this failed fuel will be identified prior to receipt; however, fuel with pinhole leaks or hairline cracks or fuel that fails after loading at the utility sites may not be identified in reactor records. The terminology used in this study is failed fuel, which is fuel with cladding breaches that allow air to come in contact with the uranium dioxide (UO_2) fuel; unfailed fuel is CSNF with intact cladding.

When in an air environment, UO_2 fuel with cladding breaches has been observed to oxidize and generate U_3O_8 powder given sufficient time at an elevated temperature. During the oxidation process, the oxidized fuel swells and may cause further degradation of the CSNF cladding (a process known as clad unzipping). The potential consequence of fuel oxidation is the loss of containment of radioactive materials and the resultant contamination inside the fuel handling facilities or waste container, the potential for unforeseen criticality scenarios, and increased dose rates to the public and workers.

CSNF shipments received at the repository will include:

- Unfailed fuel—Reactor records and transportation cask gas sampling indicate when the CSNF cladding is intact.
- Failed fuel—Reactor records and transportation cask gas sampling indicate when the CSNF cladding is not intact (i.e., pinhole leak, hairline crack, or grossly damaged fuel).

CSNF will be processed through the normal fuel handling process in the repository Dry Transfer Facility (DTF) and Fuel Handling Facility (FHF). This waste stream will contain a small percentage of failed fuel that may be subject to oxidation. The preclosure safety analysis (PCSA) evaluates the consequences of failed fuel oxidizing in the DTF and FHF during processing. The basis for this PCSA evaluation is documented in *Categorization of Event Sequences for License Application* (BSC 2004 [DIRS 171429]).

INTENTIONALLY LEFT BLANK

3. DESIGN BASES FOR HANDLING CSNF IN AIR

This section examines the existing design bases for handling CSNF in air and provides rationale for changes to the design bases. The design bases for handling CSNF in air need to address the following cases:

- Handling Unfailed CSNF—Fuel without a cladding breach (unfailed fuel) in air will be exposed to an oxidizing environment. For Yucca Mountain surface operations, criteria for handling CSNF in air need to be confirmed.

- Handling Failed CSNF (Breached Cladding)—Fuel with a cladding breach (failed fuel) in air will begin to oxidize based on the temperature of the fuel and the time the fuel is exposed to air. The higher the spent fuel temperature, the faster the fuel will oxidize in an air environment. The design criteria for handling failed fuel in air need to be confirmed.

- Expected Quantities of Failed CSNF and Types of Damage—The design basis needs to be confirmed for the types and quantities of failed fuel that will be received at the repository. This will determine the quantity of fuel that is subject to oxidation, and will provide the basis for the safety analysis and design of facility operations.

3.1 CURRENT DESIGN BASES

Project Design Criteria Document (BSC 2004 [DIRS 171599], Section 6.3.2) contains the following criteria for cladding temperature:

- Criteria—Cladding temperature for CSNF handled in the surface facilities shall not exceed 400°C during normal operations and 570°C during off-normal and accident conditions. Cladding temperature for all spent nuclear fuel (SNF) shall not exceed 350°C following the sealing of the waste package.

- Technical Rationale—Cladding temperature is limited to provide a margin to failure by creep rupture. The cladding temperature limits for handling CSNF in the surface facilities are based on Interim Staff Guidance (ISG)-11 (NRC 2003 [DIRS 170332]). The 350°C limit for emplacement and postclosure (following the sealing of the waste package) is based on "Thermal Inputs for Evaluations Supporting TSPA-LA, Supplement" (Williams 2003 [DIRS 162731]).

Information from *Project Requirements Document* (Canori and Leitner 2003 [DIRS 166275]) is consistent with *Project Design Criteria Document* (BSC 2004 [DIRS 171599]):

Req. Number and Title: PRD-014/T-008, Performance Requirements—Cladding

Requirement Text: All CSNF waste forms containing zirconium-based cladding shall be maintained during preclosure and postclosure periods at temperatures that will not accelerate the degradation of the cladding to the point that it affects the performance of the system.

Organization: Design and Engineering

Rationale for Allocation: This requirement mandates system performance capabilities that ensure protection of fuel cladding and justifies any credit taken for the additional barrier provided by the cladding. The thermal loading limit must be considered by the Design and Engineering organization in ensuring, for example, that ventilation provides the cooling to achieve the thermal goals.

Requirement Source: *Civilian Radioactive Waste Management System Requirements Document* (DOE 2004 [DIRS 171945])—3.4.F Zirconium-alloy Cladding Temperature Requirement

There are no existing controlling parameters and ranges of values in the project documents addressing fuel oxidation time and temperature for surface operations. These criteria should be established to define the normal and off-normal source terms if the fuel is allowed to oxidize. Project requirements also need to address the quantity and type of failed fuel assemblies the project will receive since such characteristics will directly influence oxidation source terms.

3.2 ANALYSIS

3.2.1 Handling Unfailed CSNF In Air

Unfailed fuel cladding protects the fuel pellets from the external environment and contains the fission product and fill gasses.

The approach used was to develop the design criteria based on established guidance that best match the YMP operating conditions. Because there is little data available for handling CSNF in air, the NRC staff guidance established in ISG-11 (NRC 2003 [DIRS 170332]) was used for comparison. In addition, other failure modes were addressed to determine if handling fuel in air would cause unfailed fuel to fail.

The presence of an oxidizing atmosphere in the fuel handling facilities requires the consideration of several potential failure modes when considering an overall temperature limit. Electrical Power Research Institute (EPRI) presents a list of potential cladding degradation and failure modes for fuel in dry storage (Pescatore and Cowgill 1994 [DIRS 102066]). This list includes:

- Creep rupture
- Stress corrosion cracking
- Delayed hydride cracking
- Hydride radial reorientation
- Hydrogen redistribution
- Irradiation embrittlement
- Zircaloy oxidation
- Strain rate embrittlement.

Maximum temperatures during normal and off-normal operations and thermal cycling during fuel handling operations were evaluated to address two specific phenomena: creep and hydride reorientation.

Stress corrosion cracking and delayed hydride cracking were judged to be unlikely due to the low stress during storage and insufficient free iodine concentrations. Other studies (including Ferry 2003 [DIRS 172936]) have found that creep strains of up to 7 percent are acceptable for dry storage.

Hydrogen redistribution, irradiation embrittlement, and strain rate embrittlement were also judged to be unlikely to result in failures by themselves, rather they interfere with the ability of the cladding to effectively withstand potentially adverse mechanical interactions. They are not considered further in this study (Section 4.5). Although these five failure modes are not addressed in detail in this study, validation of these conclusions may be needed. The three remaining potential failure modes, creep rupture, hydride radial reorientation, and zircaloy oxidation are discussed in Section 4. Results from Section 4 show that the three credited failure modes of cladding creep rupture, hydride radial reorientation, and cladding oxidation do not appear to cause failure for the proposed normal operating temperature of 400°C for a duration of time up to 1,000 hours.

The off-normal temperature of 570°C presently used for SNF cladding in air, may be too high based on the increased rate of cladding oxidation coupled with the potential increase in creep failure of the cladding. Based on information presented in Section 4, the off-normal/accident temperature design criteria proposed by this study is 460°C for a maximum time duration of 30 days.

3.2.2 Handling Failed CSNF (Breached Cladding) in Air

Handling failed fuel can be done in one of two ways:

- The first approach is to handle all fuel (failed and unfailed) with a temperature limit of 400°C for a normal operation. During normal operations, fuel with a cladding breach (pinhole or hairline crack or greater) is assumed to oxidize based on the expected length of time that the fuel is exposed to air.

- The second approach is to handle unfailed fuel with a temperature limit of 400°C and to handle known failed fuel based on a time and temperature limit that would minimize or prevent fuel oxidation. This approach presents a risk during operations that some unknown failed fuel processed with the unfailed fuel may oxidize.

For the purposes of this study, a normal operation temperature limit of 400°C for handling failed fuel in air is used. Preliminary work has been done on potential time and temperature relationships that could be used to develop operational limitations for fuel known to be failed. However, it will be difficult to apply such limitations to the small percentage of failed fuel that is unknown. Issues in Sections 5 and 6 concerning oxidation rate, airborne release fractions (ARFs), and acceptable contamination levels should be considered in developing operations for failed fuel.

3.2.3 Expected Quantities of Failed CSNF and Types of Damage

CSNF fuel assembly reliability is based on industrial experience reported by DOE and EPRI (Appendix E provides a detailed description based on power plant records). Failed assemblies have been characterized as damaged and leaking. Damaged fuel is defined as fuel with defects greater in size than a pinhole leak or hairline crack (ISG-1, NRC 2002 [DIRS 164018]) that must be placed in damaged fuel cans for transportation. Leaking fuel is defined as fuel with hairline cracks or pinholes. Leaking fuel is often treated as intact fuel (ISG-1) for storage and transportation but are considered failed for repository operations.

Assembly Reliability

From 1968 through 2003, the average assembly-based failure rate for United States CSNF is 3.0 percent with the boiling water reactors (BWRs) having 3.2 percent and pressurized water reactors (PWRs) having 2.8 percent failure rates (Table 3-1). This data is skewed by the 1970s period when BWRs had a high failure rate because of pellet cladding interaction failures. Similarly, PWR fuel performed poorly in the early 1990s. In the last 10 years, the assembly failure rate has been 1.1 percent with the BWRs (0.5 percent failures) outperforming the PWRs (1.9 percent failures). The stainless steel clad fuel included in the above inventory represents 1 percent of the inventory. If separated, it would have an assembly failure rate of about 1.8 percent, close to that of the total inventory.

Table 3-1. Failed Fuel Estimates for BWR and PWR Assemblies

Year	BWR Assemblies Discharged[a]	BWR Assemblies Failed	PWR Assemblies Discharged[a]	PWR Assemblies Failed
1968	5		0	
1969	97	32	0	
1970	29	29	99	0
1971	413	87	113	0
1972	801	68	282	36
1973	564	323	165	4
1974	1,290	671	575	32
1975	1,223	463	797	36
1976	1,666	297	931	33
1977	2,047	108	1,107	16
1978	2,239	119	1,665	16
1979	2,131	124	1,642	42
1980	3,330	112	1,457	9
1981	2,467	42	1,590	34
1982	1,951	59	1,491	40
1983	2,649	26	1,779	85
1984	2,735	81	1,933	35
1985	2,989	99	2,032	35
1986	2,552	41	2,254	94
1987	3,316	24	2,567	106
1988	2,956	64	2,574	108
1989	3,803	57	2,721	204
1990	3,487	15	3,435	104

Table 3-1. Fail Fuel Estimates for BWR and PWR Assemblies (Continued)

Year	BWR Assemblies Discharged[a]	BWR Assemblies Failed	PWR Assemblies Discharged[a]	PWR Assemblies Failed
1991	3,191	24	2,803	111
1992	3,932	12	3,588	122
1993	3,759	15	3,400	104
1994	3,777	15	2,747	55
1995	4425	4	3,741	73
1996	4,690	15	3,536	74
1997	3,849	10	3,414	53
1998	3,867	30	2,166	62
1999	4,586	13	3,637	63
2000	4,361	6	3,177	51
2001	3,904	2	3,019	44
2002	4,274	50	3,854	75
2003	4,198[b]	51	3,171[b]	51
Sum	97,553	3,188	73,462	2,007
% Failed (all years)		3.27		2.73
% Failed (1994–2003)		0.47		1.85

NOTES: [a] Source for the number of discharged assemblies through 2002 is the Energy Information Administration website: www.eia.doe.gov/cneaf/nuclear/spent_fuel/ussnftab3.html/ (EIA 2005 [DIRS 172860]).

[b] Number discharged in 2003 is estimated as an average of the previous 5 years.

[c] Sources for number of damaged assemblies: 1969–1985, Bailey and Wu 1990 [DIRS 109192], Table 30; 1986-1988, BWRs Potts and Proebste 1994 [DIRS 107774], Table 2; PWRs based on average failure rate using 6 years (1983-1985 and 1989-1991); 1989, Yang 1997 [DIRS 102049], Table 2; 1990-2003 Yang et al. 2004 [DIRS 172866].

The trend today in fuel failures is constant or slightly decreasing fuel reliability (Yang et al. 2004 [DIRS 172866]). While the Institute of Nuclear Power Operations has established goals toward reducing failures and reaching zero failure, economic pressure exists at utilities to increase fuel duties. Most utilities try to identify the cause of all fuel rod failures and make the necessary changes in design or operation to preclude future failures. This has led to detailed poolside inspections of failed fuel and close monitoring of the primary water chemistry. Failures from debris fretting have led to filters on the assembly endfittings and a greater effort to clean the primary system after maintenance. Failures from water baffle jetting have led to using more robust fuel grid spacers. Recent pellet-clad interaction failures led to new operating procedures to restrict power changes. Fuel manufacturers have also introduced best practice lines and improved quality assurance programs to reduce manufacturing defects. Counteracting these improvements is a trend toward higher burnups and duty cycles. Many plants have been licensed for stretched power, and some plants have gone to 24-month fuel cycles. Fuel burnup has nearly doubled (30 to 54 GWd/MTHM) over the past decade, but additional burnup increases will be limited by the uranium enrichment limit of 5 percent. Deregulation has also pushed the nuclear utilities to maximize plant availability and capacity factors while decreasing costs. Overall, the historical fuel reliability of 3 to 4 percent should be used to characterize current and future fuels.

The number of known failed assemblies reported by the utilities will differ from the number of failed assemblies received at the repository because of assembly reconstitution and fuel inspections. If the failed assembly was scheduled to go back into the core, the failed rod may be removed and a substitute rod inserted into the assembly. The failed rod is then placed with other failed rods into a damaged fuel can. Some of the early assembly designs did not easily permit

reconstitution. For some last cycle assemblies, the failed rods are removed from the assembly for root cause analysis and then may be placed in a damaged fuel can. Sometimes a failed rod is reinserted into the assembly and placed in the pool. In some cases where the failures are on the outside of the assembly, the rods are not removed from the assembly. Some failed rods are never located.

Rod Reliability

The above data and discussion are based on fuel assemblies. Assemblies have different numbers of rods in them and, therefore, the rod reliability is different than the assembly reliability. PWR assemblies vary in design with the earlier plants using 14 × 14 rods (about 164 rods per assembly since not all locations have fuel rods in them) and the newer designs using 17 × 17 (about 264 rods per assembly). The number of rods in an assembly cannot be changed for a specific plant design. The average for all PWRs is 207 rods per assembly (Yang et al. 1991 [DIRS 125302]). The number of failed rods in a failed assembly varies with time and failure cause. Debris fretting often causes two rods to fail (McDonald and Kaiser 1985 [DIRS 101725], p. 2-5). Baffle jetting or grid fretting might cause many rods along the outer row to fail. In one case of grid fretting, 32 failed rods were identified in one PWR assembly. In that same batch of fuel another fuel assembly contained 25 failed rods.

Manufacturing failures tend to be single-rod failures, although some manufacturing events have led to the failure of many rods in many assemblies. For the early period (before 1986), the number of rods failed per failed assembly averaged 2.2 (EPRI 1997 [DIRS 100444]), but this decreased to 1.4 rods per PWR failed fuel assembly (Yang et al. 1991 [DIRS 125302]). Using the assembly failure rates presented in Table 3-1 and the rod data cited above, the total failure percent on a rod basis for PWR fuel is 0.02 percent (Table 3-2).

BWR assemblies are smaller and have fewer rods in them. Earlier designs were 7 × 7 (around 48 or 49 rods per assembly), but later designs were 10 × 10 (about 96 rods per assembly). The latest designs are 11 × 11. BWRs can change the number of rods per assembly over time. The average for all BWRs is approximately 62 rods/assembly (Yang et al. 1991 [DIRS 125302]). For the early period (before 1986), the number of rods failed per failed assembly averaged 2.2 (EPRI 1997 [DIRS 100444]), but this has decreased to 1.1 rods per BWR assembly (Yang et al. 1991 [DIRS 125302]). Using the assembly failure rate presented in Table 3-1 and the rod data presented in this paragraph, the total failure percent for BWR rods is 0.11 percent (Table 3-2).

For both PWRs and BWRs combined, the total failure percent for the rods is 0.05 percent. The TSPA model for postclosure uses a log uniform distribution for rod failure defined by 0.01 and 1 percent (producing a median failure rate of 0.1 percent) (BSC 2004 [DIRS 172895], Section 4.1). The rod failure rate developed in this report is in the same range used for the TSPA. The stainless steel clad fuel is included in the above data and represents 1 percent of the inventory. If separated, it would have a rod failure rate of about 0.06 percent, close to that of the total inventory.

Table 3-2. Rod Failure Rates for United States PWR and BWR Fuel

Year	BWR Rods Discharged	BWR Rods Failed	PWR Rods Discharged	PWR Rods Failed
1968	310	0	0	0
1969	6,014	70.4	0	0
1970	1,798	63.8	20,493	0
1971	25,606	191.4	23,391	0
1972	49,662	149.6	58,374	79.2
1973	34,968	710.6	34,155	8.8
1974	79,980	1476.2	119,025	70.4
1975	75,826	1018.6	164,979	79.2
1976	103,292	653.4	192,717	72.6
1977	126,914	237.6	229,149	35.2
1978	138,818	261.8	344,655	35.2
1979	132,122	272.8	339,894	92.4
1980	206,460	246.4	301,599	19.8
1981	152,954	92.4	329,130	74.8
1982	120,962	129.8	308,637	88
1983	164,238	57.2	368,253	187
1984	169,570	178.2	400,131	77
1985	185,318	217.8	420,624	77
1986	158,224	45.1	466,578	131.6
1987	205,592	26.4	531,369	148.4
1988	183,272	70.4	532,818	151.2
1989	235,786	62.7	563,247	285.6
1990	216,194	16.5	711,045	145.6
1991	197,842	26.4	580,221	155.4
1992	243,784	13.2	742,716	170.8
1993	233,058	16.5	703,800	145.6
1994	234,174	16.5	568,629	77
1995	274,350	4.4	774,387	102.2
1996	290,780	16.5	731,952	103.6
1997	238,638	11	706,698	74.2
1998	239,754	33	448,362	86.8
1999	284,332	14.3	752,859	88.2
2000	270,382	6.6	657,639	71.4
2001	242,048	2.2	624,933	61.6
2002	264,988	55	797,778	105
2003	260,300	56.1	656,314	71.4
Sum	6,048,311	6,521	15,206,551	3,172
% Failed (all years)		**0.11**		**0.02**
% Failed (all years) PWR & BWR		**0.05 %**		

On average, there are 1.9 failed rods per failed assembly. A conservative average number of failed rods per failed assembly is 2.2. This value should be used for future calculations. However, there will be assemblies received that have a greater number of failures than represented by this average number.

The rod and assembly failure rates reported here are generally consistent with values reported by others for both U.S. and foreign fuels (Table 3-3).

Table 3-3. Comparison of Fuel Reliability from Various Sources

Fuel	Period	Reference	Failure Rate[a], %
BWR	Through 1990	DOE 1992 [DIRS 102812], p. 2.5-4, Table 2.5.2	4.9-(Assembly)
PWR-Westinghouse	Through 1990	DOE 1992 [DIRS 102812], p. 2.5-5, Table 2.5.3	1.6-(Assembly)
PWR-all	Through 1990	DOE 1992 [DIRS 102812], p. 2.5-3, Table 2.5.1	1.5-(Assembly)
PWR-French	1979–1984 1984	Dehon et al. 1985 [DIRS 109197], p. 2-24	0.01 0.005
BWR-Japan PWR-Japan	To 1997	Sasaki and Kuwabara 1997 [DIRS 102074], pp. 13 and 14	0.01 0.002
PWR-Combustion Engineering	To 11/1984	Andrews and Matzie 1985 [DIRS 109190], p. 2-42, Table 2	0.011
All	Through 1984	EPRI 1997 [DIRS 100444], p. 4-1	0.02–0.07
All	After 1984	EPRI 1997 [DIRS 100444], p. 4-2	0.006–0.03
All	To 1986	Sanders et al. 1992 [DIRS 102072], p. I-36	0.10–0.73 0.07–0.48
PWR-Westinghouse	1 core, debris damage after steam generator replacement	McDonald and Kaiser 1985 [DIRS 101725], p. 2-5	0.26
All	1969–1976	Manaktala 1993 [DIRS 101719], pp. 3-2 and 3-3, Figure 3-1	0.01–2+
PWR-Mark-B Babcock & Wilcox	1986–1996	Ravier et al. 1997 [DIRS 102068], p. 34, Figure 4	0–0.055
BWR	2000	Edsinger 2000 [DIRS 154433], p. 162	0.0005
BWR, General Electric	1995–1999	Potts 2000 [DIRS 160783], p. 502, Figure 1	0.00058
PWR, Mitsubishi	1992–1999	Doi et al. 2000 [DIRS 160781], p. 443	0 rod failures
All	All	S. Cohen & Associates 1999 [DIRS 135910]	0.1 range 0.01-1

NOTES: [a]Failure rates are on a rod basis unless noted as assembly-based. The assembly value represents the percentage of assemblies that contain at least one failed rod.

Source: CRWMS M&O 2000 [DIRS 151659], Table 10.

Failure Characterization

The above discussion considers failed rods. This group consists of damaged rods and leaking rods. Damaged fuel is defined as fuel with mechanical damage affecting SNF handling or fuel with defects greater in size than a pinhole leak or hairline crack (ISG-1, NRC 2002 [DIRS 164018]). Damaged fuel is considered failed even though it may not be leaking. Damaged fuel must be placed in damaged fuel cans for shipping. Leaking fuel is defined here as fuel with hairline cracks or pinholes. Leaking fuel is often treated as intact fuel (ISG-1). Some leaking fuel will never be identified as failed. Industry experience has shown that the majority of fuel failures are damaged fuel rather than the leaking fuel (pinhole or hairline category). A brief review of the various fuel failure mechanisms indicates why this observation is expected. Some utilities will take the more conservative characterization approach; if they can identify the assembly as failed, it will be characterized as damaged and the utility will avoid demonstrating that there was only a hairline crack or pinhole.

The size of a defect, or defects, in failed rods discharged from a reactor is dependent upon both the cause of the initial clad penetration and subsequent operation of the rod. A review of the major fuel failure causes indicates only two that would be likely to result in initial penetrations consistent with a pinhole or hairline size, pellet-clad interaction cracks and some manufacturing defects. Other failure types, including debris fretting, grid-to-rod fretting, baffle jetting, primary hydriding, and local and general accelerated corrosion, would be expected to result in a primary defect size in excess of the pinhole or hairline threshold. The distribution of historical fuel failure causes indicates that the latter category of failures is much more likely to occur.

After initial cladding penetration, subsequent operation of the fuel rod can result in further degradation of the initial defect site or formation of a secondary clad penetration at some distance from the initial defect (Edsinger 2000 [DIRS 154433]). Initial operation of a fuel rod forms a hard, relatively impermeable oxide layer on the outside surface of the clad. The secondary clad penetrations are typically the result of hydrogen uptake by the cladding through the relatively unprotected inside diameter of the clad. The rate of formation and extent of these secondary defects are enhanced when the initial defect is small, which limits the quantity of oxygen available to form a protective layer on the clad inside surface. For this reason, even in the cases where the initial defect may meet the pinhole or hairline criterion, the rod may have additional clad degradation on discharge.

Based on the arguments presented above, a team of fuel experts at AREVA/COGEMA estimated that 90 percent of the failed fuel (2.7 to 3.6 percent of total assemblies) would be characterized as damaged, and 10 percent of the failed assemblies (0.3 to 0.4 percent of total assemblies) would be leakers and characterized as intact. In addition, there will be an additional 0.3 to 0.4 percent of total assemblies that are leakers and have not been identified as failed. This breakdown of assembly characteristics is summarized in Table 3-4. Included in the damaged assemblies are damaged fuel cans, which contain only failed rods from reconstitution of assemblies. The above assessment does not address fuel failures during transportation or handling in the surface facilities. They are expected to affect a very small number of assemblies (failure probability $<4 \times 10^{-5}$) (Debes 1999 [DIRS 161193], p. 2).

EPRI (1997 [DIRS 100444], p. 4-1) research indicates that the majority of the failures are of the pinhole or hairline crack variety with a 10 to 20 percent range being failures of cladding larger than pinholes or hairline cracks. If the definition of hairline crack was width of a hair and pinhole was defined as the size of a common pinhole, the EPRI estimate would approximate the AREVA/COGEMA estimate. The fuel experts at AREVA/COGEMA are aware of the EPRI estimate and believe that it did not adequately address the current mix of failure types (probably weighed more heavily toward the earlier manufacturing failures and pellet-clad interaction) or the propensity for secondary failures in the current, more aggressive fuel cycles. The AREVA/COGEMA experts believe that the EPRI estimates for damaged fuel are too low.

Table 3-4. Characterization of Fuel Assemblies as Received at YMP

Assembly Type	Percent of all Assemblies	Packaging
Damaged	2.7–3.6	Damaged fuel can
Known leaker	0.3–0.4	Uncanned but identified
Unknown leaker	0.3–0.4	Uncanned and unidentified
Failed	3.3–4.4	Canned or uncanned
Unfailed	95.6–96.7	Uncanned, no concern for oxidation

Source: Judgment of AREVA/COGEMA fuel experts.

If the assemblies are randomly placed in a 24 assembly PWR transportation cask or 64 assembly BWR transportation cask, then most shipping casks will have one or more assemblies with failed fuel rods. More recent cask designs have significantly higher payloads, increasing the likelihood of a failed fuel rod in a transportation cask. The surface facilities must be designed to handle failed and damaged assemblies during routine operations with CSNF.

The values contained in Table 3-4 are used throughout this study. These failed fuel quantities need to be further reviewed to establish a definitive basis for design and the safety evaluation. The number of rods that might fail in the surface facilities from creep (Section 4.3.2) has not been included in Table 3-4 but will need to be reevaluated and included in future dose and operational consideration. A rod failure rate of 0.5 percent, if applied uniformly, would mean that almost every PWR assembly would have a failed rod. In fact, it is only the extreme rods in terms of burnup and fission gas release that fail from creep, and these would tend to be located in a subgroup of high burnup assemblies, which were exposed to transients that increased the fission gas release.

The review of the Framatome-ANP fuel inspection records (Appendix E) showed an assembly failure rate of about 3.5 percent and the rod failure rate of about 0.026 percent. This is generally consistent with the earlier study of DOE and EPRI data (Siegmann at Framatome-ANP) that showed assembly failure rates for all fuels were from 3.3 to 4.4 percent with the rod failure rate of 0.05 percent. A survey of the Framatome-ANP records showed that the average number of failed rods per failed assembly was in the range of 1.4 to 1.6 rods per assembly. This is consistent with the EPRI (1997 [DIRS 100444]) and Yang et al. (1991 [DIRS 125302]) estimates of 2.2 to 1.1 rods per assembly and shows that the use of the earlier value of 2.2 failed rods per

failed assembly is conservative. In the Framatome survey, over 90 percent of the failed assemblies had one or two failed rods. The highest number of failed rods in an assembly identified in this survey was 8 rods from grid fretting. The survey concluded that about 65 percent of the failed assemblies were reconstituted or recaged, though this number may not be representative of overall industry experience. First and second cycle fuel was reconstituted for reinsertion and some of the discharged fuel was reconstituted because the failed rods were removed for root cause studies. The cause of the failure affects both the size of the initial cladding penetration and the timing of the failure. Two failure mechanisms, manufacturing defects and PCI, tend to produce hairline cracks or pinholes, but these tend to enlarge from secondary failure. The other failure mechanisms tend to cause larger failures that would probably be classified as damaged fuel. The survey of fuel inspection reports and records support the earlier estimate that only 10 percent of the failed fuel would be characterized as intact and not placed into damaged fuel cans.

3.3 FINDINGS

- Normal Temperature for Handling Unfailed Fuel—The three likely failure or degradation modes of cladding creep rupture, hydride radial reorientation, and cladding oxidation, do not appear to cause failure of unfailed fuel for the proposed normal operating temperature limit of 400°C for up to 1,000 hours.

- Off-Normal/Accident Temperature for Handling Unfailed Fuel—The temperature of 570°C for handling CSNF in air may be too high based on the increased rate of cladding oxidation coupled with the potential increase in creep failure of the cladding. Based on information presented in Section 4, the off-normal temperature limit proposed by this study is 460°C with a maximum time duration of 30 days. It may also be shown by future investigations that higher temperatures are acceptable for shorter durations.

- Design Bases for Handling Failed Fuel—Acceptable time and temperature criteria for handling failed fuel are dependent upon the consequence analyses discussed in Section 6. For this study, evaluations were performed based on handling failed fuel at temperatures less than 400°C during normal operations and up to 460°C for off-normal operations, the same as the criteria for handling unfailed fuel.

- Failed Fuel Quantities—For use in this study, the expected quantities of failed CSNF are documented in Table 3-4.

The above findings on unfailed fuel are based on data that contain a high level of uncertainty. The data has been gathered from reactor records over 35 years of operation, including criteria based on conditions of operation that are much different than at the repository, fuel failure modes that can vary significantly from reactor to reactor, storage and handling environment, and extent of fuel clad damage that may be unknown. Additional issues that should be considered in developing the design and operations approach are described below:

- Off-Normal/Accident Design Temperature for Handling Unfailed Fuel—Additional work is required to establish the off-normal/accident scenario temperature(s) and duration(s) (e.g., during a loss of heating, ventilation, and air-conditioning [HVAC]

event). The design criteria need to have a well-defined accident temperature limit and time duration for accident conditions with supporting justifications. In addition, calculations are required to determine the maximum fuel temperature that will result during these off-normal conditions. If the off-normal condition causes the fuel temperature to rise above the design limit, assumptions and calculation conservatisms should be reviewed carefully, then mitigating features may need to be implemented.

- Failed Fuel Quantities—The failed fuel quantities shown in Table 3-4 need to be further reviewed to establish qualified data for design, operations, and safety analysis.

4. INTACT FUEL CLADDING PERFORMANCE

Fuel cladding performance must be evaluated to determine the consequences of handling unfailed, intact fuel in the surface facilities. This section discusses unfailed fuel cladding (SNF without clad breaches) performance through the examination of the following:

- Oxidation of zirconium cladding in air
- Cladding creep for intact fuel
- Hydride radial reorientation.

The vast majority of fuels utilize zirconium cladding or advanced cladding alloys, but approximately 1 percent of CSNF assemblies have stainless steel cladding. This study examines both traditional zirconium-based cladding and advanced zirconium-alloy cladding performance. Stainless steel cladding performance will be addressed at a later date (Section 4.5).

This section discusses unfailed fuel-cladding performance (intact fuel without clad breaches). Fuel oxidation and failed fuel performance are discussed in Section 5.

4.1 FUEL TEMPERATURES

The average PWR reactor assembly is assumed to be 4 percent enrichment and 48 gigawatt days (GWd)/metric tons of heavy metal (MTHM) with 25 years of cooling time (BSC 2004 [DIRS 167441], Section 6.1.1.1.1). The bounding PWR assembly is assumed to be 5 percent enrichment and 80 GWd/MTHM with 5 years cooling time (BSC 2004 [DIRS 169061], Section 5.3). The average BWR assembly is 3.5 percent enrichment and 40 GWd/MTHM with 25 years of cooling, and the bounding BWR assembly is assumed to be 5 percent enrichment and 75 GWd/MTHM with 5 years of cooling time (BSC 2003 [DIRS 164364], Section 5.5.3). The maximum fuel burn-up presently expected at YMP is 60 GWd/MTHM, but provisions are made for receipt of fuel up to 80 GWd/MTHM.

As discussed in Section 3.2.1, the proposed maximum SNF clad temperature for normal, dry transfer operations is 400°C and the suggested maximum allowable temperature for off-normal accident conditions is 460°C for 30 days. During transportation, the typical maximum cladding temperature is expected to be approximately 372°C (BSC 2005 [DIRS 172739], Figure 5-1). The maximum cladding temperature during operations in the DTF and FHF within the transfer cell area or cask preparation area is expected to be 371°C, and the maximum off-normal temperature is estimated to be 451°C for a 30-day loss of HVAC (BSC 2005 [DIRS 172739], Figure 5-1). The maximum allowable clad temperature limits of 400°C and 460°C are greater than the range of expected operating temperatures at the repository.

4.2 OXIDATION OF ZIRCONIUM CLADDING IN AIR

Oxidation of CSNF cladding in air decreases the wall thickness, causing the clad to weaken. Fuel cladding performance will, in part, be dependent on the oxidation of the zirconium cladding in air. An initial evaluation was performed to determine if the cladding can tolerate being at 570°C for 30 days. Table 4-1 gives the measured oxide thicknesses for the corrosion tests (Natesan and Soppet 2004 [DIRS 172868]). Also in this table are the thicknesses predicted using

the equation developed in *The Corrosion of Zircaloy-Clad Fuel Assemblies in a Geologic Repository Environment* (Hillner et al. 1998 [DIRS 100455]). The equation was developed for water corrosion, but the corrosion rates in water, steam, and air are expected to be similar (Rothman 1984 [DIRS 100417], p. 8). The calculated thicknesses are in good agreement or conservative with respect to the tests. Using this equation, it is predicted that only 20 μm of additional cladding would be lost for fuel held at 400°C for 100 days. This represents about four percent of the initial cladding thickness (approximately 500 μm) after reactor operation during which an oxide layer of approximately 50 μm forms for a typical PWR fuel (CRWMS M&O 2000 [DIRS 151659]). However, a much larger oxide thickness may form for higher burnup fuels. For a proposed off-normal case of 571°C for 30 days, about 180 μm, or about 30 percent, of the cladding would be oxidized.

Table 4-1. Zirconium Oxidation in Air

Temperature °C	Exposure Time Hours	ZrO_2 Thickness, Measured μm	Change In Thickness, Measured (unirradiated) μm	Change In Thickness, Calculated (irradiated) μm	% Error	Metal Loss μm
Baseline[a]	Baseline[a]	23.7	0			
300	973	25.8	2.1	0.7	67	
400	326	27.9	4.2	4.6	9	
400	636	28.9	5.2	8.9	71	
500	262	37.2	13.5	33.1	145	
500	500	49.6	25.9	63.2	144	
600	40	50.1	26.4	27.6	5	
600	150	137.4	113.7	103.5	9	
600	315	269.2	245.5	217.3	11	
400	2400			34		20
571	720			316		180

NOTE: [a]Baseline = maximum expected.

Source: Nateson and Soppet 2004 [DIRS 172868].

Table 4-1 shows, in the column titled Change in Thickness, Calculated μm, the thickness changes predicted using the equation developed in *The Corrosion of Zircaloy-Clad Fuel Assemblies in a Geologic Repository Environment* (Hillner et al. 1998 [DIRS 100455]). Though the equation was developed for steam corrosion, the corrosion rates in steam and air are expected to be similar. The calculated thicknesses are in good agreement with the test data. Hillner et al. (1998 [DIRS 100455) also discuss the effects of irradiation on corrosion rates. Irradiated material initially corrodes faster than unirradiated material, but the effect is short lived because it appears to be a surface effect. Table 4-1 compares Hillner's model for irradiated material with the test results for unirradiated material. Based on these results, the effect of irradiation on corrosion rates is determined to be not significant.

The data demonstrate that for temperatures below the current limit of 400°C, the rate of cladding oxidation in air for 636 hours is less than 6μm or about 2 percent of the initial cladding

thickness. At these temperatures, oxidation should not be a limiting failure mode. At temperatures between 400°C and 500°C, there appears to be a slight increase in oxidation rate, though handling times well in excess of 1,000 hours at these temperatures should be supportable. At temperatures between 500°C and 600°C, there appears to be a significant increase in the corrosion rate.

4.3 CLADDING CREEP FOR INTACT FUEL

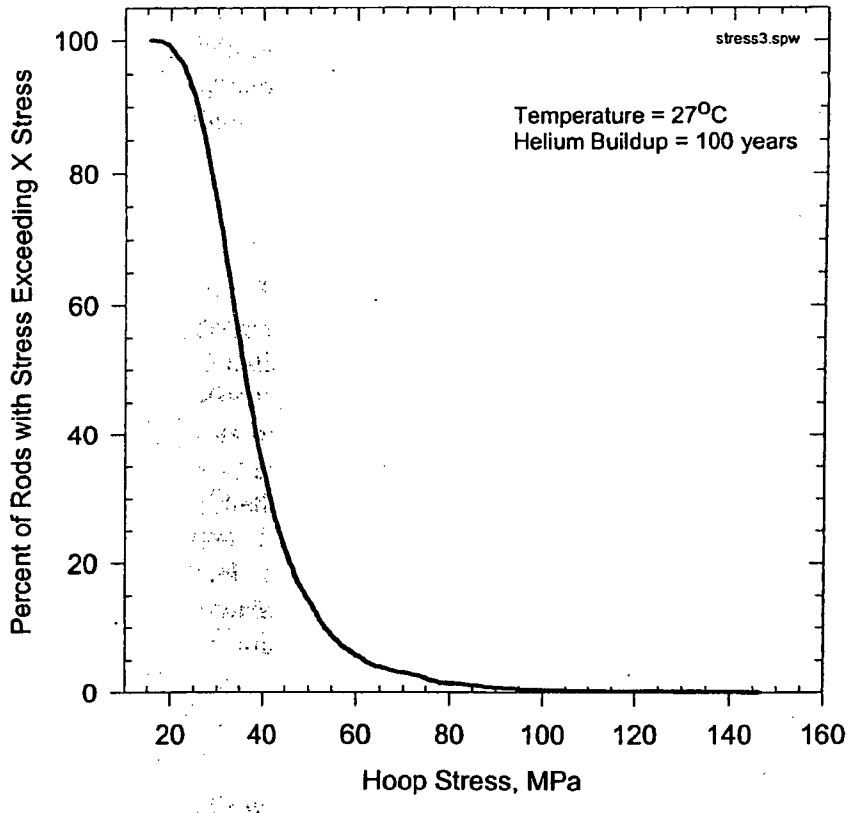
A statistical model was used to evaluate the creep of the cladding at both 400°C and 570°C. The 400°C limit represents the current design for normal operation and the 570°C limit represents the current design for off-normal/accident operations. As discussed in Section 4.2, an examination of surface oxidation concluded that an additional 20 μm of cladding would be oxidized for fuel held at 400°C for 100 days. This represents about three percent of the cladding thickness of a standard 17 × 17 PWR fuel rod. For the off-normal/accident case of 570°C for 30 days, 180 μm, or about 30 percent, of the cladding is predicted to oxidize. This raises the question of whether the fuel would fail from creep during this period since stress would increase by 30 percent. The creep calculations presented here do not address cladding surface oxidation and the corresponding metal loss. For the 400°C case, it is not important; but for the 570°C case, surface oxidation would greatly increase stress with time and increase rod failure rates. Cladding oxidation will be addressed in more detail as the design progresses.

4.3.1 Initial Cladding Stress Distribution

The stress distribution (CRWMS M&O 2000 [DIRS 151659]) was developed for PWR rods with burnup to 75 GWd/MTHM and burnup dependent probability distributions for rod free volume, fission gas release, and oxide thickness. This produced a room temperature stress distribution with the following characteristics:

- Minimum = 16 MPa
- 95 percent = 23 MPa
- Median = 36 MPa
- Mean = 38 MPa
- 5 percent = 62 MPa
- 2 percent = 72 MPa
- 1 percent = 82 MPa
- 0.5 percent = 92 MPa
- Maximum = 146 Mpa.

The distribution, as shown in Figure 4-1, has a large maximum stress tail because of very conservative modeling of fission gas release (up to 50 percent) and oxide thickness (up to 120 μm), which produces a long, low probability tail. Only 1 percent of the fuel has stresses over 82 MPa and 0.5 percent exceed 92 MPa. The maximum stress is 146 MPa. BWR fuels operate with lower stress; therefore, the PWR fuel can be used as the limiting case.



Source: CRWMS M&O 2000 [DIRS 151659], Figure 18.

Figure 4-1. Stress Distribution for PWR Fuel Rods

4.3.2 Creep at 400°C

In ISG-11 (NRC 2003 [DIRS 170332]), the NRC acknowledges that while the peak cladding temperatures are maintained at or below 400°C, the creep caused by hoop stress would be of little concern. Although this staff guidance is for handling fuel in an inert environment, the effects of temperature should not impact clad performance when the fuel is in an inert or air environment as long as oxidation of the clad remains within prescribed limits. A statistical creep model was developed for YMP (CRWMS M&O 2001 [DIRS 151662]) that looks at a statistical distribution of creep rates and failure distributions to calculate the fraction of rods that might fail from creep starting with the stress distribution discussed above. This model was modified to look at creep for a constant temperature (not a spatial and time dependent temperature). When this model is applied to the storage of fuel with temperatures ranging from 375 to 500°C, it shows that failures become more probable when temperatures exceed 440°C (Figure 4-2) and storage times are greater than 500 hours. The figure shows three different failure curves generated with three different failure limits or failure criteria. A one percent strain limit is used in some German dry storage calculations and is a very conservative limit. The six percent strain limit is based on cladding failures during dry oxidation experiments. The complimentary cumulative distribution function (CCDF) failure limit is based on observed failures during creep tests.

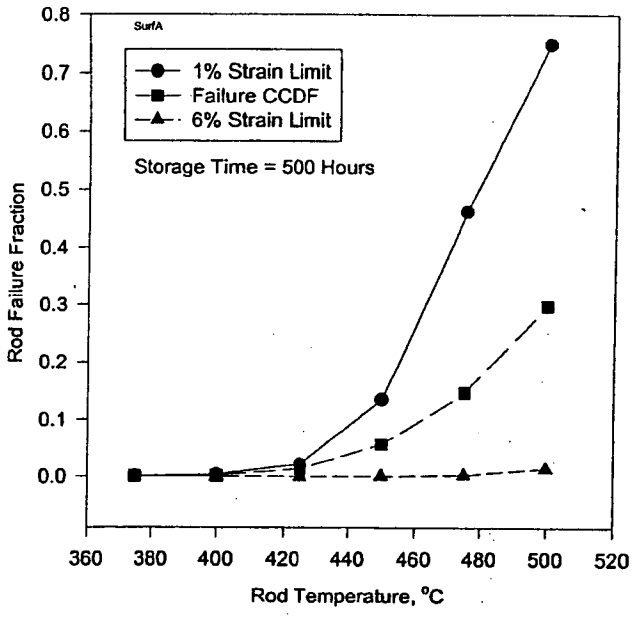


Figure 4-2. Rod Failure Rate versus Storage Temperature for 500 Hour Storage

Figure 4-3 shows that the number of rods that fail at 400°C increases with time. Even at very short times, some rods are expected to fail. These rods have stresses at the extreme end of the statistical distribution and also undergo more severe creep or have lower failure criteria (CCDF failure criteria). After about 1,000 hours of storage, the fraction of rods that fail becomes more important and approaches 0.5 percent, the failure guideline developed by the NRC for dry storage in *Standard Review Plan for Dry Cask Storage Systems* (NRC 1997 [DIRS 101903], p. 4-2). Future evaluations should address the failure of a small number of rods during normal operations (Section 4.5).

A rod failure rate of 0.5 percent, if applied uniformly, would mean that almost every PWR assembly would have one or more failed rods. In fact, it is only the extreme rods in terms of burnup and fission gas release that fail from creep, and these would tend to be located in a subgroup of high burnup assemblies, which were exposed to transients that increased the fission gas release. The stress distribution, creep behavior, and failure criteria will need to be verified and if the number of rod creep failures is in the 0.5 percent range, then the operational procedures and dose calculations will have to address this failure rate (Section 4.5).

A sensitivity case was analyzed to see the effect of the high stress tail to the distribution (Figure 4-1). The distribution was truncated at stresses of 92 MPa (room temperature), eliminating the extreme 0.5 percent tail. For the case of storage at 400°C for 500 hours, the failure fraction was reduced from 0.17 percent (0.33 to 0 percent range) to 0.058 percent (0.075 to 0 percent range). This sensitivity analysis highlights the importance of validating the stress distribution given in Section 4.3.1.

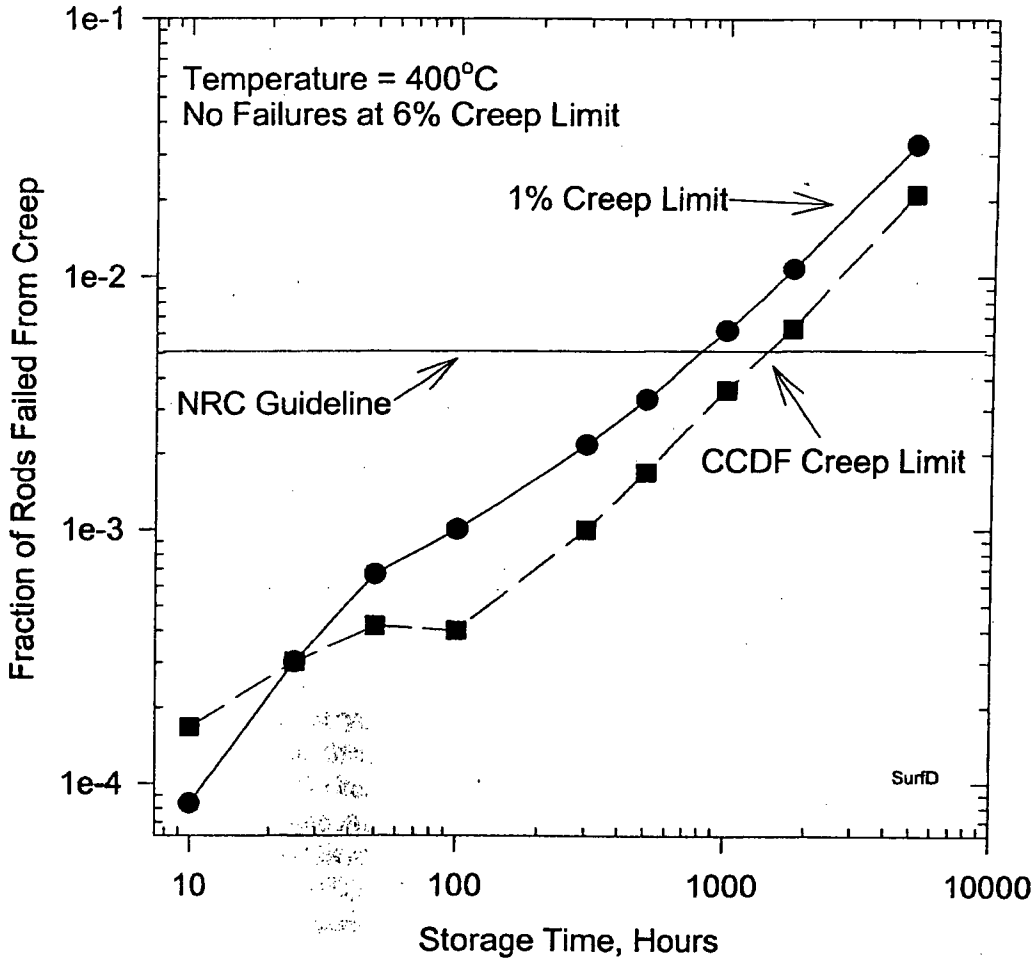


Figure 4-3. Rod Failure Rate versus Storage Times for Storage at 400°C

4.3.3 Creep at 570°C

The creep model was developed using irradiated creep tests, which are usually performed at lower temperatures than 570°C. The actual temperature for off-normal events cited in *Interim Staff Guidance - 11, Revision 3. Cladding Considerations for the Transportation and Storage of Spent Fuel* (NRC 2003 [DIRS 170332]) is similar to the temperatures used in a series of creep experiments documented in *High Temperature Postirradiation Materials Performance of Spent Pressurized Water Reactor Fuel Rods Under Dry Storage Conditions* (Einziger et al. 1982 [DIRS 101604]). These tests were performed on irradiated rods in temperatures ranging from 482 to 571°C and times ranging from 740 to 7,680 hours. No rods failed, although significant creep was observed.

Table 4-2 contains data from *High Temperature Postirradiation Materials Performance of Spent Pressurized Water Reactor Fuel Rods Under Dry Storage Conditions* (Einziger et al. 1982 [DIRS 101604], Table IV). Also shown in the last column of this table are the calculated strains.

The model does a good job in predicting five of the six test results. In general, the model underestimates the strains slightly. The model does not predict the last test well. This test was on a low stress rod (25.2 MPa at 571°C or 9 MPa at room temperature). This test had a similar strain as the 75.7 MPa rod test (fourth test on Table 4-2), although the stress in the fourth test was three times higher. This suggests that the sixth test might have had significantly more annealing than the other tests.

Table 4-2. Comparison of High Temperature Creep Tests with the Creep Model

Temperature °C	Time Hrs	Initial Stress Mpa	Measured Strain[a] %	Calculated Strain %
482	4652	49.5	1.7 ± 0.1	1.4 ± 1.2
510	7680	51.3	3.4 ± 1	3.1 ± 2.5
571	740	55.3	5 ± 0.5	4.6 ± 3.7
571	740	75.7	7 ± 1	6.9 ± 5.5
571	740	39.8	5 ± 0.7	3.1 ± 2.5
571	1000	25.2	7 ± 1	1.9 ± 1.5

Source: [a]Einziger et al. 1982 [DIRS 101604]

The creep model does require a different failure limit or criteria for this temperature range because of annealing. The tests by Einziger et al. (1982 [DIRS 101604]) illustrated in Figure 4-3 showed cladding strains up to about 12 percent without rod failure. A failure limit or criteria of 10 and 14 percent was used in the calculations since this brackets the maximum strain observed by Einziger et al. without failure.

Figure 4-4 shows the fraction of rods that would fail from creep as a function of storage duration. For most times in storage, the fraction of fuel that fails from creep at 570°C exceeds the NRC guideline of 0.5 percent (0.005 fraction). The early failures represent the high stress tail of the distribution shown in Figure 4-1. It must be confirmed that the high stress in the tail of the distribution is real. If so, then either justification for failure of a small fraction of fuel in the off-normal event will need to be made or off-normal temperatures of 570°C will need to be reduced. The fraction of rods that fail increases with storage duration.

A calculation shows that if the HVAC system is lost for 30 days, the fuel will slowly heat up to 451°C (BSC 2005 [DIRS 172739], Figure 5-1). An analysis was performed assuming that the rods quickly heated to 460°C and stayed there for 30 days (720 hours). It was also assumed that no annealing occurred and the nominal (one percent, CCDF, and six percent) failure criteria apply. During this event, 11 percent (range 33 to 0.1 percent) of the fuel would fail from creep. Creep failure, producing small holes in the cladding during accidents, will need to be evaluated (Section 4.5).

4.4 HYDRIDE RADIAL REORIENTATION

ISG-11 (NRC 2003 [DIRS 170332]) indicates that the formation of radial hydrides due to reorientation of existing hydrides within the clad is a strong function of cladding hoop stress. Testing irradiated cladding specimens indicated a threshold for this phenomenon at 120 MPa.

Testing unirradiated samples indicated a lower threshold of 90 MPa. The lower stress was chosen as a conservative value.

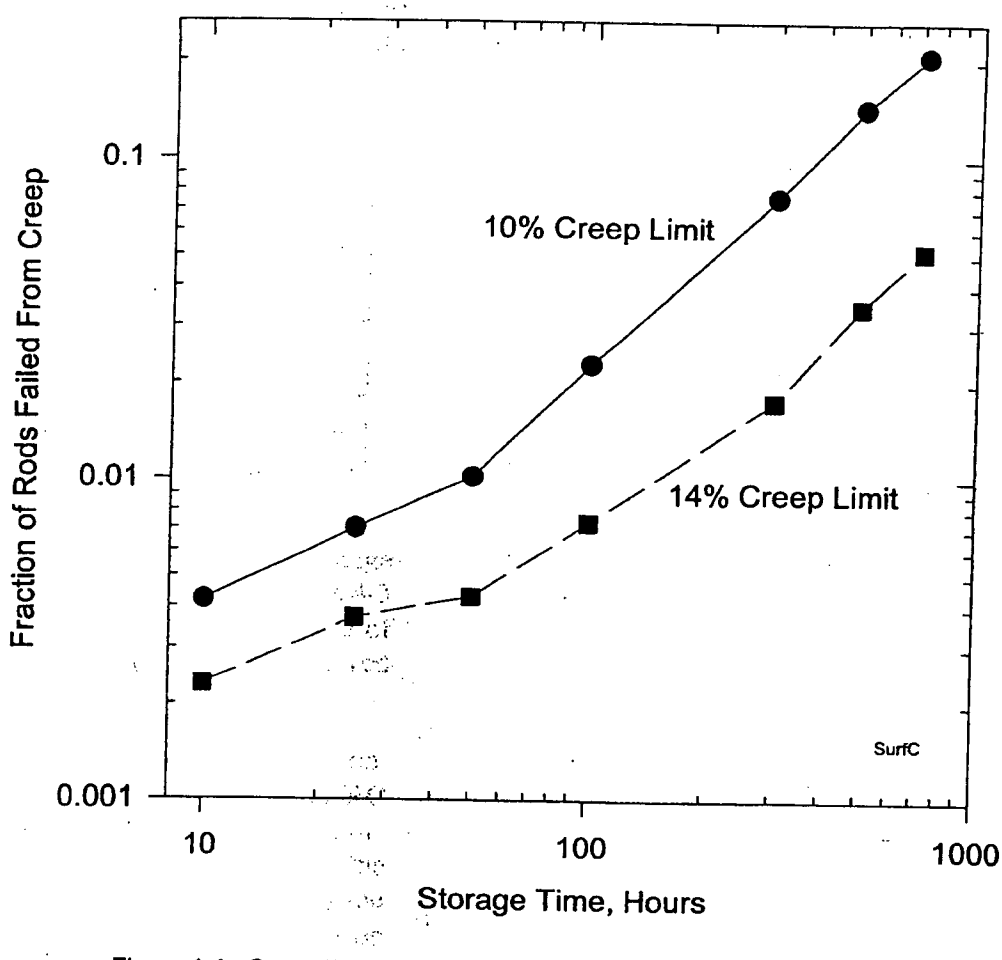
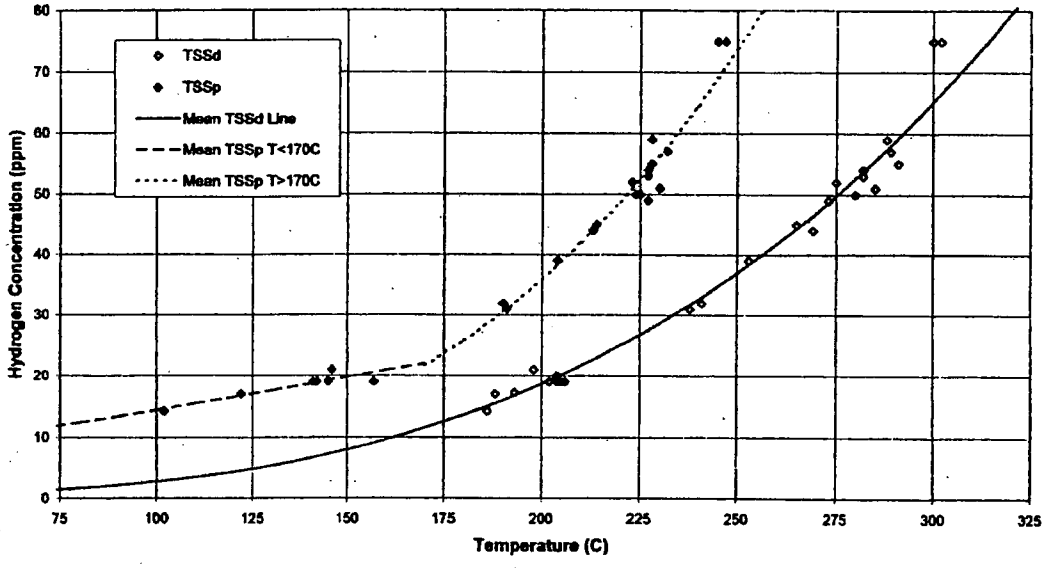


Figure 4-4. Creep Failures at 570°C as a Function of Storage Time

The ISG-11 (NRC 2003 [DIRS 170332]) 400°C temperature limit was established as a means of simplifying calculations while maintaining cladding hoop stresses, due to rod internal pressure, below the 90 MPa limit. That is, the temperature only impacts hydride reorientation through its effect on rod internal pressure and resulting cladding stress.

An additional factor considered by ISG-11 (NRC 2003 [DIRS 170332]) is thermal cycling. Completing over 10 thermal cycles of 65°C or more is assumed to enhance the potential for hydride radial reorientation. This limit is based on Kammenzind 2000 [DIRS 172877]. The Kammenzind 2000 study was designed to consider long-range hydrogen migration within zircaloy components due to tensile and compressive stress gradients. Several factors in this study make its direct relevance to fuel rod performance somewhat limited, including the use of fully annealed rolled plate as the test material.

Thermal cycling has been examined. There is a difference in solubility of hydrogen in zirconium between precipitation and dissolution of hydrogen (Figure 4-5). The difference can be considered supersaturation because it takes energy to generate the additional volume to precipitate the hydrogen as zirconium hydride. Table 4-3 gives the dissolution solubility for 400°C. When irradiated cladding is heated to this temperature, it will have about 157 ppm of hydrogen in solution. If the metal is cooled from 400°C to 335°C, it will not precipitate any of the hydrogen because the solubility is 191 ppm, higher than the 400°C dissolution value. If the metal is cooled to 315°C, hydrides will start to precipitate. A cooling cycle of 100°C could move up to 24 ppm per cycle (reprecipitation at the same locations would not cause movement). It is possible to move 240 ppm in 10 cycles from 400°C to 300°C. Since only two or three large temperature cycles are likely to occur during surface facility handling operations, it is unlikely that significant amounts of hydrogen will be moved. The expected number of thermal cycles and temperature changes will need to be evaluated as the design progresses (Section 4.5).



NOTE: TSSd = Terminal Solid Solubility dissolution; TSSp = Terminal Solid Solubility precipitation.

Source: McMinn et al. 2000 [DIRS 112149]

Figure 4-5. Dissolution and Precipitation Solubility for Hydrogen in Irradiated Zirconium Metal

Table 4-3. Temperatures Necessary to Precipitate Hydrides Dissolved at 400°C

Temperature Drop	Temperature, °C	Solubility, Precipitation, ppm	Solubility, Dissolution, ppm	PPM Move/cycle
	400		157	
65	335	191		0
85	315	156		0.6
100	300	134		24

Recent work by EPRI (2003 [DIRS 172861]) shows that when cladding is cooled from 400°C to room temperature and the initial stress is less than 150 MPa (higher than expected with commercial fuels), about 20 ppm of hydrides will reorient. This is a small fraction of the total hydride and will have a negligible effect on the material strength.

Yagnik et al. (2004 [DIRS 172865]) shows in a series of experiments at temperatures of 300°C that radial hydrides will have little effect in weakening the material. At room temperatures, where the hydrides are more brittle, a 20-ppm concentration of radial hydrides will also have negligible effect on the strength of the material.

In order for radial hydride formation to occur to an extent that fuel rod mechanical performance may be impacted, two conditions must be met. The stresses in the clad must be high enough to allow formation of hydrides in the radial direction, and the temperature of the clad must be below the hydride brittle-ductile transition temperature. In the FHF and DTF, these two criteria are mutually exclusive most of the time. The only mechanism available to produce the stresses necessary to initiate the formation of radial hydrides (120 to 150 MPa) is increased fission gas pressure due to high fuel rod temperature (above 400°C). This temperature is significantly higher than the hydride brittle-ductile transition temperature (below 300°C). Normal operating temperatures will be less than 400°C; therefore, mechanical performance should not be impacted.

It appears that there is a significant amount of conservatism built into the ISG-11 (NRC 2003 [DIRS 170332]) limits for hydride radial reorientation. Should temperatures greater than 400°C or thermal cycling beyond ten 65°C cycles occur, additional studies in this area would be expected to show margin is available to support higher limits.

Another factor for consideration is the nature of the phenomenon itself. Hydride radial reorientation is not a failure mechanism; rather, it degrades the ability of the cladding to withstand mechanical loads. Since the potential challenges to the cladding and consequences of fuel rod failure are different for the DTF and FHF compared to long-term storage and shipping, where accident events, including cask tipover and cask drop scenarios, must be endured without rod failure, it may be appropriate to re-examine the applicability of hydride radial reorientation as a limiting factor in the DTF and FHF (Section 4.5).

4.5 FINDINGS

Fuel cladding conditions have been analyzed to assess zirconium cladding performance during dry handling conditions. These preliminary analyses were performed for this study using computer models that have not been qualified for use in repository design. Therefore, the results should be used for comparison purposes only and should not be used for design. Accident conditions have been bounded by a worst-case situation of 570°C. The following conditions limit the applicability of the findings that have been reached for intact cladding performance during dry handling:

- CSNF Temperatures—At 400°C and for storage times less than 1,000 hours, the fraction of fuel that fails from creep is below the NRC guidelines of 0.5 percent (NRC 1997 [DIRS 101903]). At temperatures above 460°C, creep failures become more probable. At 570°C, the amount of fuel that fails from creep exceeds 0.5 percent for storage times

exceeding 10 hours; such failures should be considered failed fuel as discussed in Section 5. Therefore, clad temperatures should be limited to 460°C for durations approaching 30 days. The stress distribution used to develop these results needs verification. A computer model, created in accordance with Yucca Mountain quality requirements, is needed to substantiate the above results and should be used as a basis for design and the safety analysis.

- Thermal Cycles—The expected number of thermal cycles and temperature changes during normal and off-normal operations need to be evaluated for impact on cladding integrity.

- Impact of Modern Fuel Designs and Fuel Management—The data presented are based on fuel rod samples that do not bound the majority of the spent fuel being discharged today. For example, fuel rod samples with burnups in the 35.7 GWd/MTHM range are used in Einziger et al. (2003 [DIRS 172526]). This is about half of the discharge burnup of modern fuel. In addition, the cladding material considered in the creep analyses is Zircaloy-4. PWRs have transitioned to advanced alloys, including M5 and ZIRLO. Those that have not are employing optimized zircaloy material chemistries that may also impact behavior. Further evaluations should be performed to confirm cladding performance for these advanced alloy materials.

- Impact of Cladding Material Degradation—Current operational strategies, including both burnup increases and changes in reactor primary system chemistry, have resulted in fuel rods and assemblies being discharged with fairly high clad hydride concentrations and thick oxide layers. In addition, many have tenacious crud layers that retain other chemicals at the rod surface (i.e., lithium, noble metals, zinc). ISG-1 (NRC 2002 [DIRS 164018]) requires that all fuel with degraded mechanical and material properties be treated as damaged and stored within a damaged fuel can. This means that some unfailed fuel may be characterized as damaged.

- Clad Degradation—A future study to support of the preliminary study conclusions may be warranted to present a complete evaluation. The failure modes listed below were only briefly discussed in this study:

  - Stress corrosion cracking
  - Delayed hydride cracking
  - Hydrogen redistribution
  - Irradiation embrittlement
  - Strain rate embrittlement.

  When in a water environment, failed CSNF cladding may allow moisture to penetrate into the fuel. Residual moisture may not be removed during fuel drying and may be trapped in the rods. The presence of water can create additional reactions inside the cladding and can provide a source of hydrogen due to radiolysis.

- Stainless Steel Cladding—Zircaloy cladding was evaluated in this study. Calculations and analyses should also be performed to substantiate the expected performance of

stainless steel cladding. An EPRI study (EPRI 1996 [DIRS 160968], p. iii) of the long-term dry storage of stainless steel clad fuel concluded that the storage systems now employed for zircaloy clad fuel could safely accommodate the colder stainless steel clad SNF inventory with similar integrity.

- Hydride Issues—Hydride reorientation and pumping should be evaluated further for cladding integrity.

5. FAILED FUEL PERFORMANCE

This section summarizes the expected performance of failed CSNF that will be handled in air at the repository. Damaged commercial fuel will be shipped to the repository in damaged fuel cans in a transportation cask. ISG-1 (NRC 2002 [DIRS 164018]), however, allows fuel with pinhole or hairline cracks to be categorized as intact. Intact fuel, damaged fuel, damaged fuel in canisters, and fuel with pinhole or hairline cracks will be transferred in air from the transportation cask to staging, an aging cask, or a waste package in fuel transfer cells in the FHF and DTFs during normal operations. Exposing the failed fuel to air while in the FHF or DTFs could cause oxidation of the uranium dioxide fuel pellets due to cladding breaches.

The first part of this section reviews current methods to calculate UO_2 oxidation rates. The next part describes the key fuel parameters for oxidation including crack propagation rates and other failed fuel performance issues. The section ends with a discussion of the findings, issues, and future steps in the design development.

5.1 OXIDATION PROCESS

Hanson (1998 [DIRS 101672], p. iii) describes the oxidation of spent fuel in air as a two-step process of the form $UO_2 \rightarrow UO_{2.4} \rightarrow U_3O_8$. The transition from $UO_2 \rightarrow UO_{2.4}$ does not result in appreciable fuel pellet density changes. However, the transition from $UO_{2.4} \rightarrow U_3O_8$ results in a volume expansion of greater than 36 percent. The increase in volume as spent fuel oxidizes to U_3O_8 places stress on portions of unfailed cladding, which may split as a result. The oxidation process first progresses by the $UO_2 \rightarrow UO_{2.4}$ reaction. Once the spent fuel oxidizes to $UO_{2.4}$, a plateau is reached where the fuel resists oxidation to higher oxides. Following this plateau for temperatures greater than 250°C, oxidation resumes with time until the U_3O_8 phase is reached. No oxidation to U_3O_8 has been observed for temperatures below 250°C. Hanson (1998 [DIRS 101672], p. iii) found that the $UO_{2.4} \rightarrow U_3O_8$ reaction is strongly dependent on both fuel temperature and burnup.

5.1.1 Incubation Time

The extent of oxidation is dependent on both the temperature of the fuel and the time that the exposed fuel is exposed to air. As discussed above, fuel first oxidizes to $UO_{2.4}$, and then to U_3O_8. The volume of U_3O_8 increases and the contact between U_3O_8 and the cladding places stress on the cladding that can lead to clad unzipping. The time between the onset of oxidation to the time of U_3O_8 formation is defined as the incubation time, as defined in Equation 5-2. *Clad Degradation – Dry Unzipping* (CRWMS M&O 2000 [DIRS 149230], Section 6.5) indicates that the incubation time is a function of the pellet/cladding gap at the onset of oxidation, the time for the fuel in any local region to oxidize to $UO_{2.4}$, the time for a fraction of fuel to react to U_3O_8, and the strain necessary to induce cladding crack propagation. The time to oxidize to $UO_{2.4}$ is defined by CRWMS M&O (2000 [DIRS 149230], Equation 26) as:

$$t_{2.4} = k_{2.4} \cdot \exp\left(\frac{Q_{2.4}}{RT}\right) \text{ hours} \qquad \text{(Eq. 5-1)}$$

Where $k_{2.4}$ is 1.40×10^{-8} hours for the nominal case or 2.93×10^{-9} hours for the bounding case (CRWMS M&O 2000 [DIRS 149230], p. 46). For the purposes of evaluating the normal operation consequences of oxidation, the nominal case value is used for $k_{2.4}$. $Q_{2.4}$ is the activation energy, 105 kJ mol^{-1} (CRWMS M&O 2000 [DIRS 149230], Equation 26). R is the universal gas constant, 8.314 J mol^{-1} K^{-1}. T is the absolute temperature (K). It is proper to use the nominal case (not bounding) because hydrated phases would decompose at these temperatures (even transportation temperatures).

The time for incubation at any fixed temperature is given by (CRWMS M&O 2000 [DIRS 149230], Equation 33):

$$t_{inc} = t_{2.4} + \lambda_{inc} k_{7.5} \cdot \exp\left(\frac{Q_{7.5} + \alpha B}{RT}\right) \text{ hours} \qquad \text{(Eq. 5-2)}$$

Where $t_{2.4}$ is defined in Equation 5-1, $k_{7.5}$ is 4.84×10^{-14} hours for the nominal case or 1.48×10^{-14} hours for the bounding case (CRWMS M&O 2000 [DIRS 149230], p. 47). For the purposes of evaluating the normal operation consequences of oxidation, the nominal case value is used for $k_{7.5}$. $Q_{7.5}$ is the activation energy, 150 kJ mol^{-1} (CRWMS M&O 2000 [DIRS 149230], Equation 33). α is 1.0 kJ mol^{-1} per GWd/MTHM, and B is the burnup in GWd/MTHM.

The correction term λ_{inc} is defined (CRWMS M&O 2000 [DIRS 149230], Equation 32) as:

$$\lambda_{inc} = 1 - \frac{r_{11}}{r_{01}} \qquad \text{(Eq. 5-3)}$$

With r_{11}/r_{01} defined (CRWMS M&O 2000 [DIRS 149230], Equation 33) as:

$$\frac{r_{11}}{r_{01}} = \left[\frac{(1+x)^3(1+s)^3 - z_1 z_2}{z_1 - z_1 z_2}\right]^{1/3} \qquad \text{(Eq. 5-4)}$$

Where x is the percentage of the initial fuel pellet gap to fuel pellet radius ratio, s is the percent strain necessary to initiate splitting, z_1 is the ratio of volume for $UO_{2.4}$ to U_3O_8, and z_2 is the ratio of volume for U_3O_8 to $UO_{2.4}$.

The fuel used in the tests reported by EPRI (1986 [DIRS 127313]) was characterized and the fuel to cladding gap was measured. Due to cladding creep, the measured gap ranged from 0.25 mm at the rod ends to 0.03 mm at the rod center (EPRI 1986 [DIRS 127313], p. 2-10). The rod diameter was also measured as a function of rod length. The EPRI report (1986 [DIRS 127313] Figure 2-2) states that a rod diameter of 10.64 mm is representative. Using the cladding thickness of 0.62 mm and a minimum gap thickness of 0.03 mm, a value of 0.64 percent (0.03 mm/[(10.62 mm/2) – (0.62 mm + 0.03 mm)] * 100) is determined for x.

CRWMS M&O (2000 [DIRS 149230], p. 46) observes that, in a review of the literature, the percent strain s necessary to initiate splitting varies from about one to 6.5 percent. For the purposes of determining the consequences from normal operations, a strain of one percent is used.

From CRWMS M&O (2000 [DIRS 149230], Table 3), z_1 is 0.9929 and z_2 is 1.3709/0.9929 = 1.3807.

Using Equation 5-1 through Equation 5-4 and the parameters defined above, the total incubation time at 360°C for the average BWR fuel burnup of 40 GWd/MTHM is 18.7 hours, and for the average PWR fuel burnup of 48 GWd/MTHM, it is 62.3 hours (BSC 2004 [DIRS 167441], Section 6.1.1.1.1).

5.1.2 Crack Propagation

The primary source of data for crack propagation is from EPRI (1986 [DIRS 127313], Table 3-3). Additional data are obtained from Novak et al. (1983 [DIRS 125697], p. 263), Boase and Vandergraaf (1977 [DIRS 117977], Fig. 14), Kohli et al. (1985 [DIRS 126191], p. 197), and Einziger and Cook (1985 [DIRS 126202], p. 69). All of the data, except for the EPRI data (1986 [DIRS 127313], Table 3-3), are for oxide formation front velocity or clad unzipping velocity in one direction only. Since the cladding unzipping will travel in both directions from a cladding defect, the total unzipping velocity is determined by multiplying these data by two. The oxide front velocities presented by Kohli et al. (1985 [DIRS 126191]) are about an order of magnitude lower than those from Novak et al. (1983 [DIRS 125697]) and Boase and Vandergraaf (1977 [DIRS 117977]). Kohli et al. (1985 [DIRS 126191], p. 190) state that the total air introduced into the experimental capsule was estimated to be enough to oxidize only about 6 percent of the fuel. Since at the repository the flowing air will continually replenish the depleted oxygen, the lower oxide front velocity from Kohli et al. (1985 [DIRS 126191]) is not used in the correlation. A correlation of the data yields the following expression for clad unzipping velocity, V:

$$V = 2{,}217 \cdot \exp\left(\frac{-70.87\,kJ\,mol^{-1}}{RT}\right) \text{ cm/min} \qquad \text{(Eq. 5-5)}$$

The correlation coefficient (r^2) for the data is 0.8518, with R the universal gas constant (8.314 J mol^{-1} K^{-1}) and T the absolute temperature in °K. This correlation, along with the data, is plotted on Figure 5-1.

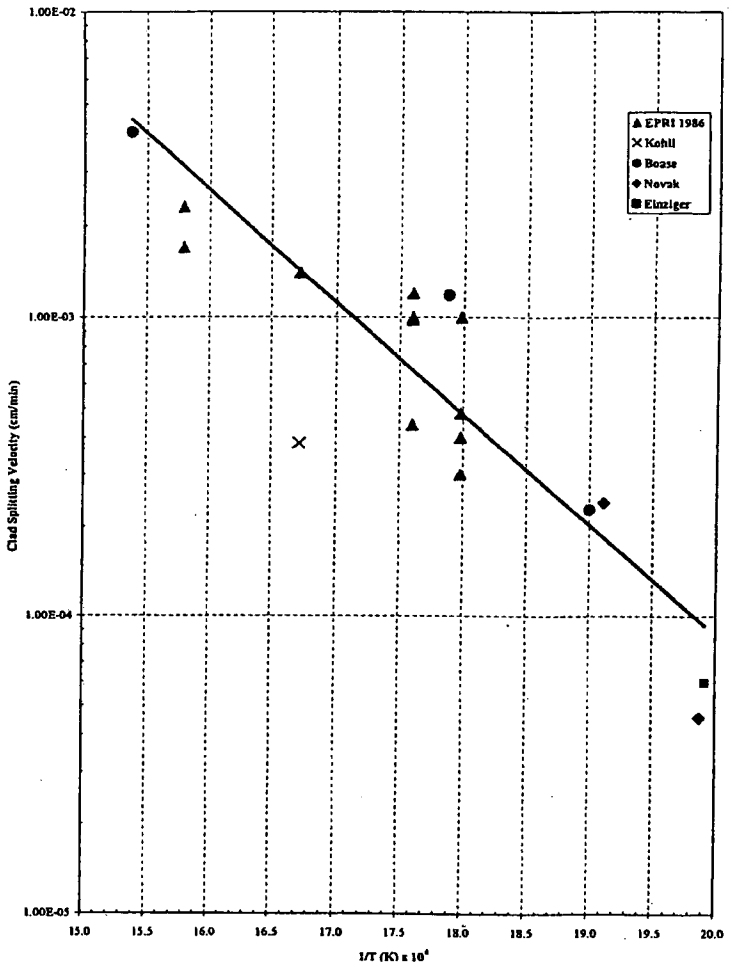


Figure 5-1. Clad Unzipping Velocity

Using this correlation, the crack propagation velocity at 360°C is 3.2×10^{-3} cm/min, which is a factor of 1.4 higher than the EPRI (1986 [DIRS 127313]) measured propagation rate. Assuming that a fuel rod will be in air for 100 hours, the time available for unzipping after the incubation period is 81.3 hours for a BWR rod and 37.7 hours for a PWR fuel rod. During this time, 15.5 cm of the BWR rod may unzip and 7.2 cm of the PWR rod may unzip, or 4.3 percent and 2.2 percent of the rod lengths, respectively. For lower burnup fuel (in the range of 10 to 30 GWd/MTHM) the fraction of the cladding that is unzipped for a PWR assembly is between 5.3 percent and 5.4 percent (slightly less for a BWR assembly). It should be noted that the 100-hr assumption is the normally expected operating time for handling SNF in air in the FHF and DTF, not an upper bound.

A design evaluation should be considered to address clad unzipping based on the lower burnup of the rod ends, regardless of whether the rods are PWR or BWR. This evaluation should be a bounding case that includes low burnup, large holes, and a maximum number of failed rods in a damaged fuel can or damaged fuel assembly (Sections 5.3 and 5.4).

5.1.3 Defect Size Effects On Oxidation

EPRI (1986 [DIRS 127313], p. iii) determined that both the size and shape of the cladding defect appear to influence the time to cladding splitting. For high temperatures (above 283°C), the time to cladding splitting was longer for the sharp small defect than for the large circular defect. This effect diminished as the fuel temperature decreased. The cladding breaches were induced by pressurizing the sample rods at elevated temperatures. Breach sizes ranged from 8 to 52 μm (EPRI 1986 [DIRS 127313], p. 2-2). These breaches are usually axial cracks with pinhole protuberances through the outer cladding surface. The large circular defect was a hole of 760 μm drilled in the fuel rod cladding. For an 8 μm defect at 325°C, it was found that the incubation time (defined as the time when a through-the-wall cladding crack starts to propagate) was 455 hours as opposed to 79 hours for a 760 μm defect at 325°C. The incubation time for a 27 μm defect at 360°C was found to be between 52 and 60 hours, while the incubation time for a 760 μm defect at the same temperature was 20 hours (EPRI 1986 [DIRS 127313], Table 3-3). Assuming that the breaches of 8 to 52 μm represent pinhole leaks or hairline cracks, the minimum incubation time is 52 hours at 360°C.

Using the same data as EPRI (1986 [DIRS 127313]), Einziger and Strain (1984 [DIRS 172756], p. 605) discuss the effects of cladding damage size on incubation time. In this report, it is stated that the incubation time varied inversely with the size of the original defect implying that the defect size was inhibiting access of oxygen through the cladding to the fuel. This effect is shown in Einziger and Strain (1984 [DIRS 172756], Figure 8).

As presented in the above discussion, there is evidence that the cladding defect size affects the incubation time. However, given the uncertainty in the definition of pinhole failures or hairline cracks and the relatively low number of defects with pinhole or hairline cracks, the effect of initial cladding defect size will be conservatively neglected. It is recommended that the conservative approach be taken that will not take credit for the oxidation retardation effects of pinholes and hairline cracks, such that pinholes and hairline cracks will be treated as if they were large defects with no retardation of the oxidation. This results in the most conservative result for incubation time.

5.2 KEY PARAMETERS FOR OXIDATION

A parametric evaluation was performed to show the effects of time and temperature at different burnups on the oxidation rate using the nominal case equation from *Clad Degradation – Dry Unzipping* (CRWMS M&O 2000 [DIRS 149230]). The nominal case equation is Equation 5-2 that used the nominal case input values for $K_{2.4}$ and $K_{7.5}$. Tables 5-1 and 5-2 provide the results of the parametric study. The equations used to create these tables are based on empirical data for burnups between 15 to approximately 40 GWd/MTHM and temperatures between approximately 230°C to 380°C. Other parameters do affect the incubation times, but once oxidation starts to occur, their affect is minimal. These tables were developed for 100 hrs, 500 hrs, and 1,000 hrs, and for burnups between 10 and 50 GWd/MTHM at temperatures from 200°C to 570°C.

Future work should be done to verify that the key parameters used are conservative and no other parameters are significant(Section 5.4).

OFFICIAL USE ONLY

Table 5-1. Damaged Fuel Parameters at 200°C, 300°C, and 350°C

Temperature=200C, Time=100hrs

Burnup (GWd/MTHM)	Plateau (h)	Total Incubation Time (h)	Length of rod split (cm)	Fraction of rod split		
10	1,498.6	6,972.6	0.0	0.00%	PWR	
20	19,042.1	24,516.1	0.0	0.00%	PWR	
30	241,953.5	247,427.4	0.0	0.00%	PWR	
40	3,074,313.3	3,079,787.2	0.0	0.00%	PWR	
48	*23,494,222.6*	*23,499,696.5*	*0.0*	*0.00%*	*PWR*	*Avg. Burnup*
50	39,062,885.3	39,068,359.3	0.0	0.00%	PWR	
10	1,498.6	6,972.6	0.0	0.00%	BWR	
20	19,042.1	24,516.1	0.0	0.00%	BWR	
30	241,953.5	247,427.4	0.0	0.00%	BWR	
40	*3,074,313.3*	*3,079,787.2*	*0.0*	*0.00%*	*BWR*	*Avg. Burnup*
50	39,062,885.3	39,068,359.3	0.0	0.00%	BWR	

Temperature=300C, Time=100hrs

10	1.2	53.2	2.2	0.61%	PWR	
20	10.1	62.1	1.8	0.49%	PWR	
30	82.5	134.5	0.0	0.00%	PWR	
40	672.9	724.8	0.0	0.00%	PWR	
48	*3,606.1*	*3,658.0*	*0.0*	*0.00%*	*PWR*	*Avg. Burnup*
50	5,486.7	5,538.7	0.0	0.00%	PWR	
10	1.2	53.2	2.2	0.73%	BWR	
20	10.1	62.1	1.8	0.59%	BWR	
30	82.5	134.5	0.0	0.00%	BWR	
40	*672.9*	*724.8*	*0.0*	*0.00%*	*BWR*	*Avg. Burnup*
50	5,486.7	5,538.7	0.0	0.00%	BWR	

Temperature=350C, Time=100hrs

10	0.1	9.0	14.0	3.89%	PWR	
20	0.6	9.4	13.9	3.87%	PWR	
30	4.0	12.9	13.4	3.73%	PWR	
40	27.4	36.3	9.8	2.72%	PWR	
48	*128.6*	*137.4*	*0.0*	*0.00%*	*PWR*	*Avg. Burnup*
50	189.1	198.0	0.0	0.00%	PWR	
10	0.1	9.0	14.0	4.67%	BWR	
20	0.6	9.4	13.9	4.65%	BWR	
30	4.0	12.9	13.4	4.47%	BWR	
40	*27.4*	*36.3*	*9.8*	*3.27%*	*BWR*	*Avg. Burnup*
50	189.1	198.0	0.0	0.00%	BWR	

Temperature=200C, Time=500hrs

Burnup (GWd/MTHM)	Plateau (h)	Total Incubation Time (h)	Length of rod split (cm)	Fraction of rod split		
10	1,498.6	6,972.6	0.0	0.00%	PWR	
20	19,042.1	24,516.1	0.0	0.00%	PWR	
30	241,953.5	247,427.4	0.0	0.00%	PWR	
40	3,074,313.3	3,079,787.2	0.0	0.00%	PWR	
48	*23,494,222.6*	*23,499,696.5*	*0.0*	*0.00%*	*PWR*	*Avg. Burnup*
50	39,062,885.3	39,068,359.3	0.0	0.00%	PWR	
10	1,498.6	6,972.6	0.0	0.00%	BWR	
20	19,042.1	24,516.1	0.0	0.00%	BWR	
30	241,953.5	247,427.4	0.0	0.00%	BWR	
40	*3,074,313.3*	*3,079,787.2*	*0.0*	*0.00%*	*BWR*	*Avg. Burnup*
50	39,062,885.3	39,068,359.3	0.0	0.00%	BWR	

Temperature=300C, Time=500hrs

10	1.2	53.2	20.9	5.80%	PWR	
20	10.1	62.1	20.5	5.68%	PWR	
30	82.5	134.5	17.1	4.74%	PWR	
40	672.9	724.8	0.0	0.00%	PWR	
48	*3,606.1*	*3,658.0*	*0.0*	*0.00%*	*PWR*	*Avg. Burnup*
50	5,486.7	5,538.7	0.0	0.00%	PWR	
10	1.2	53.2	20.9	6.96%	BWR	
20	10.1	62.1	20.5	6.82%	BWR	
30	82.5	134.5	17.1	5.69%	BWR	
40	*672.9*	*724.8*	*0.0*	*0.00%*	*BWR*	*Avg. Burnup*
50	5,486.7	5,538.7	0.0	0.00%	BWR	

Temperature=350C, Time=500hrs

10	0.1	9.0	75.6	21.00%	PWR	
20	0.6	9.4	75.5	20.98%	PWR	
30	4.0	12.9	75.0	20.83%	PWR	
40	27.4	36.3	71.4	19.83%	PWR	
48	*128.6*	*137.4*	*55.8*	*13.50%*	*PWR*	*Avg. Burnup*
50	189.1	198.0	46.5	12.91%	PWR	
10	0.1	9.0	75.6	25.20%	BWR	
20	0.6	9.4	75.5	25.17%	BWR	
30	4.0	12.9	75.0	25.00%	BWR	
40	*27.4*	*36.3*	*71.4*	*23.79%*	*BWR*	*Avg. Burnup*
50	189.1	198.0	46.5	15.50%	BWR	

Temperature=200C, Time=1000hrs

Burnup (GWd/MTHM)	Plateau (h)	Total Incubation Time (h)	Length of rod split (cm)	Fraction of rod split		
10	1,498.6	6,972.6	0.0	0.00%	PWR	
20	19,042.1	24,516.1	0.0	0.00%	PWR	
30	241,953.5	247,427.4	0.0	0.00%	PWR	
40	3,074,313.3	3,079,787.2	0.0	0.00%	PWR	
48	*23,494,222.6*	*23,499,696.5*	*0.0*	*0.00%*	*PWR*	*Avg. Burnup*
50	39,062,885.3	39,068,359.3	0.0	0.00%	PWR	
10	1,498.6	6,972.6	0.0	0.00%	BWR	
20	19,042.1	24,516.1	0.0	0.00%	BWR	
30	241,953.5	247,427.4	0.0	0.00%	BWR	
40	*3,074,313.3*	*3,079,787.2*	*0.0*	*0.00%*	*BWR*	*Avg. Burnup*
50	39,062,885.3	39,068,359.3	0.0	0.00%	BWR	

Temperature=300C, Time=1000hrs

10	1.2	53.2	44.2	12.29%	PWR	
20	10.1	62.1	43.8	12.17%	PWR	
30	82.5	134.5	40.4	11.23%	PWR	
40	672.9	724.8	12.9	3.57%	PWR	
48	*3,606.1*	*3,658.0*	*0.0*	*0.00%*	*PWR*	*Avg. Burnup*
50	5,486.7	5,538.7	0.0	0.00%	PWR	
10	1.2	53.2	44.2	14.74%	BWR	
20	10.1	62.1	43.8	14.61%	BWR	
30	82.5	134.5	40.4	13.48%	BWR	
40	*672.9*	*724.8*	*12.9*	*4.29%*	*BWR*	*Avg. Burnup*
50	5,486.7	5,538.7	0.0	0.00%	BWR	

Temperature=350C, Time=1000hrs

10	0.1	9.0	152.6	42.38%	PWR	
20	0.6	9.4	152.5	42.36%	PWR	
30	4.0	12.9	152.0	42.21%	PWR	
40	27.4	36.3	148.4	41.21%	PWR	
48	*128.6*	*137.4*	*132.8*	*36.88%*	*PWR*	*Avg. Burnup*
50	189.1	198.0	123.5	34.29%	PWR	
10	0.1	9.0	152.6	50.85%	BWR	
20	0.6	9.4	152.5	50.83%	BWR	
30	4.0	12.9	152.0	50.65%	BWR	
40	*27.4*	*36.3*	*148.4*	*49.45%*	*BWR*	*Avg. Burnup*
50	189.1	198.0	123.5	41.15%	BWR	

OFFICIAL USE ONLY