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BECHTEL
SAIC COMPANY LLC

Commercial Spent Nuclear Fuel Handling in Air Study

Prepared for:
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Office of Repository Development
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Commercial Spent Nuclear Fuel Handling in Air Study

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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 Génerale 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.

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FIGURES

TABLES

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.

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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]).

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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

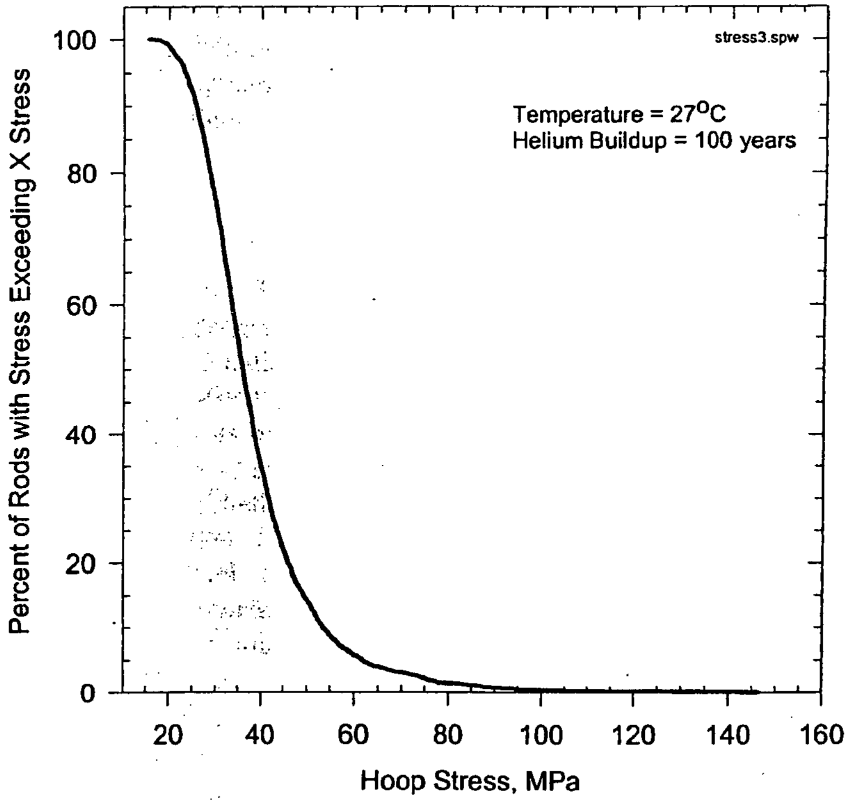
A statistical model was used to evaluate the creep of the cladding at both 400°C and 570°C. The 400°C limit represents the current design for normal operation and the 570°C limit represents the current design for off-normal/accident operations. As discussed in Section 4.2, an examination of surface oxidation 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.

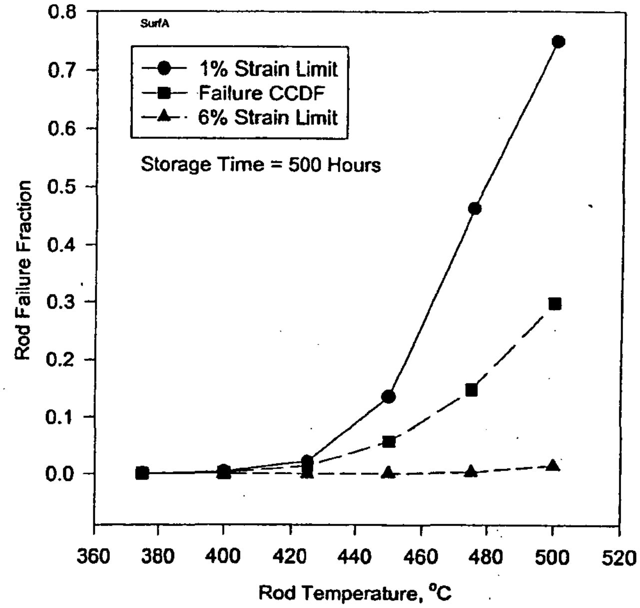


Figure 4-2. Rod Failure Rate versus Storage Temperature for 500 Hour Storage

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.

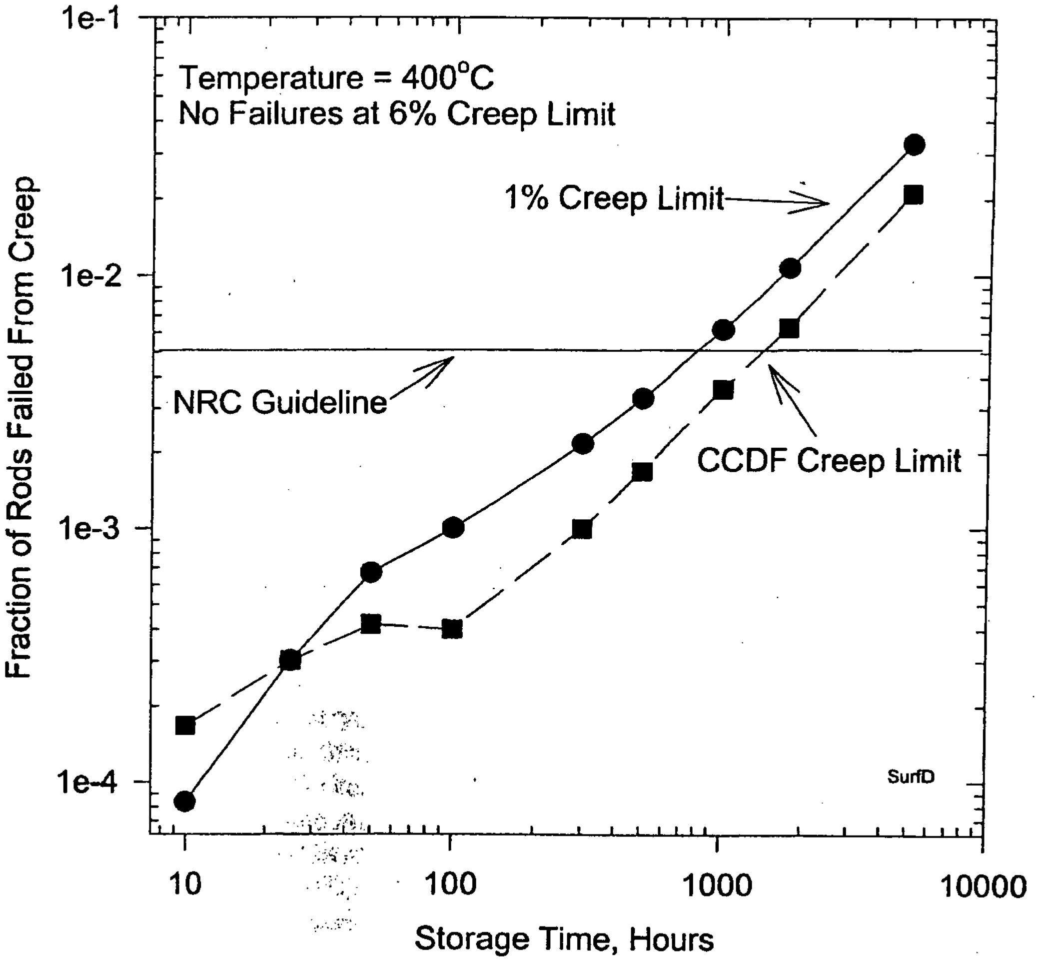


Figure 4-3. Rod Failure Rate versus Storage Times for Storage at 400°C

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.

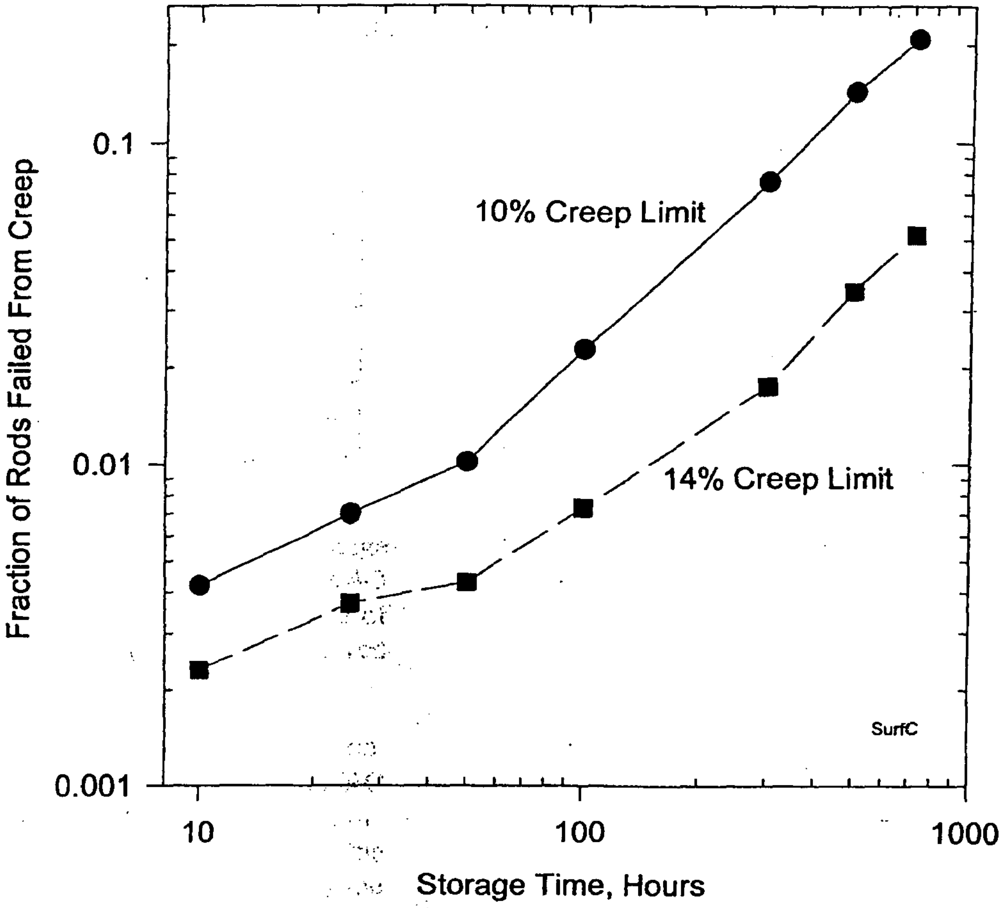
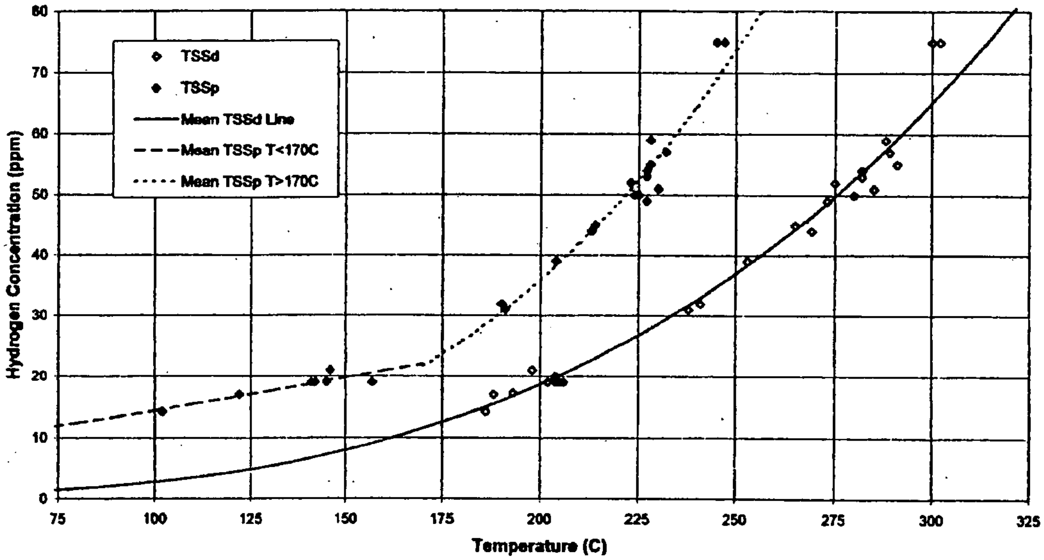


Figure 4-4. Creep Failures at 570°C as a Function of Storage Time

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.

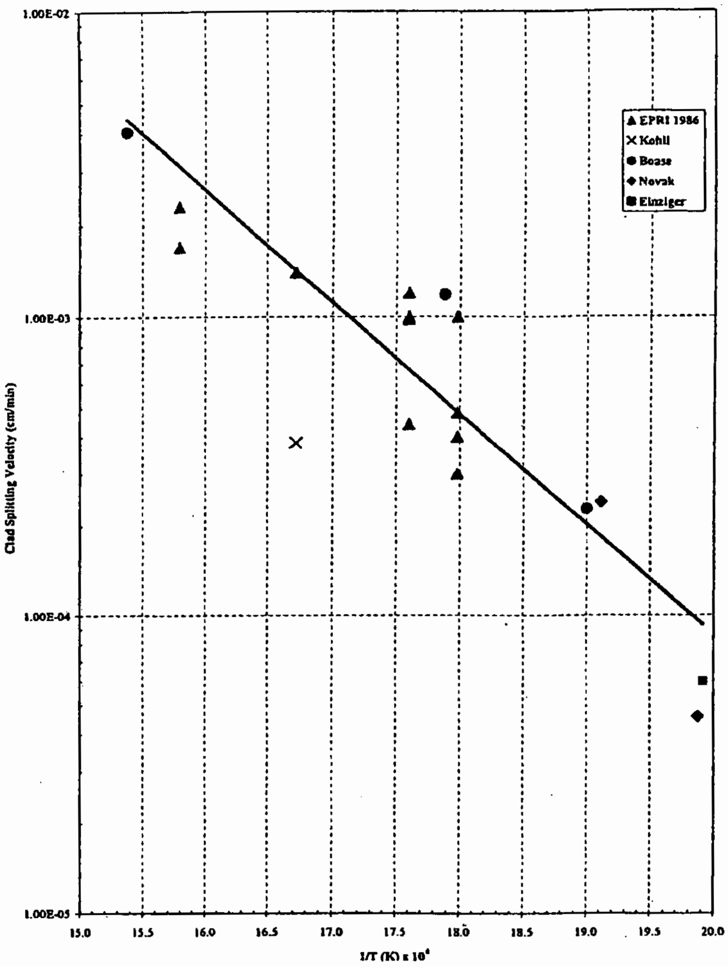


Figure 5-1. Clad Unzipping Velocity

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).

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=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=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=100hrs

Burnup (GWd/MTHM)	Plateau (h)	Total Incubation Time (h)	Length of rod split (cm)	Fraction of rod split		
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=300C, Time=500hrs

Burnup (GWd/MTHM)	Plateau (h)	Total Incubation Time (h)	Length of rod split (cm)	Fraction of rod split		
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=300C, Time=1000hrs

Burnup (GWd/MTHM)	Plateau (h)	Total Incubation Time (h)	Length of rod split (cm)	Fraction of rod split		
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=100hrs

Burnup (GWd/MTHM)	Plateau (h)	Total Incubation Time (h)	Length of rod split (cm)	Fraction of rod split		
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=350C, Time=500hrs

Burnup (GWd/MTHM)	Plateau (h)	Total Incubation Time (h)	Length of rod split (cm)	Fraction of rod split		
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*	*15.50%*	*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=350C, Time=1000hrs

Burnup (GWd/MTHM)	Plateau (h)	Total Incubation Time (h)	Length of rod split (cm)	Fraction of rod split		
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	

Table 5-2. Damaged Fuel Parameters at 400°C, 500°C, and 570°C

Temperature=400C, Time=100hrs

Burnup (GWd/MTHM)	Plateau (h)	Total Incubation Time (h)	Length of rod split (cm)	Fraction of rod split		
10	0.0	2.0	41.7	11.57%	PWR	
20	0.1	2.0	41.6	11.56%	PWR	
30	0.3	2.3	41.5	11.54%	PWR	
40	1.8	3.8	40.9	11.36%	PWR	
48	*7.5*	*9.5*	*38.5*	*10.68%*	*PWR*	*Avg. Burnup*
50	10.8	12.7	37.1	10.30%	PWR	
10	0.0	2.0	41.7	13.88%	BWR	
20	0.1	2.0	41.6	13.88%	BWR	
30	0.3	2.3	41.5	13.84%	BWR	
40	*1.8*	*3.8*	*40.9*	*13.63%*	*BWR*	*Avg. Burnup*
50	10.8	12.7	37.1	12.36%	BWR	

Temperature=400C, Time=500hrs

Burnup (GWd/MTHM)	Plateau (h)	Total Incubation Time (h)	Length of rod split (cm)	Fraction of rod split		
10	0.0	2.0	211.6	58.78%	PWR	
20	0.1	2.0	211.6	58.78%	PWR	
30	0.3	2.3	211.5	58.75%	PWR	
40	1.8	3.8	210.9	58.57%	PWR	
48	*7.5*	*9.5*	*208.4*	*57.90%*	*PWR*	*Avg. Burnup*
50	10.8	12.7	207.1	57.51%	PWR	
10	0.0	2.0	211.6	70.54%	BWR	
20	0.1	2.0	211.6	70.53%	BWR	
30	0.3	2.3	211.5	70.50%	BWR	
40	*1.8*	*3.8*	*210.9*	*70.29%*	*BWR*	*Avg. Burnup*
50	10.8	12.7	207.1	69.02%	BWR	

Temperature=400C, Time=1000hrs

Burnup (GWd/MTHM)	Plateau (h)	Total Incubation Time (h)	Length of rod split (cm)	Fraction of rod split		
10	0.0	2.0	424.1	100.00%	PWR	
20	0.1	2.0	424.1	100.00%	PWR	
30	0.3	2.3	424.0	100.00%	PWR	
40	1.8	3.8	423.3	100.00%	PWR	
48	*7.5*	*9.5*	*420.9*	*100.00%*	*PWR*	*Avg. Burnup*
50	10.8	12.7	419.5	100.00%	PWR	
10	0.0	2.0	424.1	100.00%	BWR	
20	0.1	2.0	424.1	100.00%	BWR	
30	0.3	2.3	424.0	100.00%	BWR	
40	*1.8*	*3.8*	*423.3*	*100.00%*	*BWR*	*Avg. Burnup*
50	10.8	12.7	419.5	100.00%	BWR	

Temperature=500C, Time=100hrs

10	0.0	0.2	217.9	60.54%	PWR	
20	0.0	0.2	217.9	60.54%	PWR	
30	0.0	0.2	217.9	60.54%	PWR	
40	0.0	0.2	217.9	60.53%	PWR	
48	*0.1*	*0.3*	*217.8*	*60.49%*	*PWR*	*Avg. Burnup*
50	0.1	0.3	217.7	60.48%	PWR	
10	0.0	0.2	217.9	72.65%	BWR	
20	0.0	0.2	217.9	72.65%	BWR	
30	0.0	0.2	217.9	72.64%	BWR	
40	*0.0*	*0.2*	*217.9*	*72.63%*	*BWR*	*Avg. Burnup*
50	0.1	0.3	217.7	72.57%	BWR	

Temperature=500C, Time=500hrs

10	0.0	0.2	1,091.2	100.00%	PWR	
20	0.0	0.2	1,091.2	100.00%	PWR	
30	0.0	0.2	1,091.2	100.00%	PWR	
40	0.0	0.2	1,091.2	100.00%	PWR	
48	*0.1*	*0.3*	*1,091.1*	*100.00%*	*PWR*	*Avg. Burnup*
50	0.1	0.3	1,091.0	100.00%	PWR	
10	0.0	0.2	1,091.2	100.00%	BWR	
20	0.0	0.2	1,091.2	100.00%	BWR	
30	0.0	0.2	1,091.2	100.00%	BWR	
40	*0.0*	*0.2*	*1,091.2*	*100.00%*	*BWR*	*Avg. Burnup*
50	0.1	0.3	1,091.0	100.00%	BWR	

Temperature=500C, Time=1000hrs

10	0.0	0.2	2,182.9	100.00%	PWR	
20	0.0	0.2	2,182.9	100.00%	PWR	
30	0.0	0.2	2,182.8	100.00%	PWR	
40	0.0	0.2	2,182.8	100.00%	PWR	
48	*0.1*	*0.3*	*2,182.7*	*100.00%*	*PWR*	*Avg. Burnup*
50	0.1	0.3	2,182.6	100.00%	PWR	
10	0.0	0.2	2,182.9	100.00%	BWR	
20	0.0	0.2	2,182.9	100.00%	BWR	
30	0.0	0.2	2,182.8	100.00%	BWR	
40	*0.0*	*0.2*	*2,182.8*	*100.00%*	*BWR*	*Avg. Burnup*
50	0.1	0.3	2,182.6	100.00%	BWR	

Temperature=570C, Time=100hrs

10	0.0	0.0	544.7	100.00%	PWR	
20	0.0	0.0	544.7	100.00%	PWR	
30	0.0	0.0	544.7	100.00%	PWR	
40	0.0	0.0	544.7	100.00%	PWR	
48	*0.0*	*0.1*	*544.7*	*100.00%*	*PWR*	*Avg. Burnup*
50	0.0	0.1	544.6	100.00%	PWR	
10	0.0	0.0	544.7	100.00%	BWR	
20	0.0	0.0	544.7	100.00%	BWR	
30	0.0	0.0	544.7	100.00%	BWR	
40	*0.0*	*0.0*	*544.7*	*100.00%*	*BWR*	*Avg. Burnup*
50	0.0	0.1	544.6	100.00%	BWR	

Temperature=570C, Time=500hrs

10	0.0	0.0	2,724.4	100.00%	PWR	
20	0.0	0.0	2,724.4	100.00%	PWR	
30	0.0	0.0	2,724.4	100.00%	PWR	
40	0.0	0.0	2,724.4	100.00%	PWR	
48	*0.0*	*0.1*	*2,724.4*	*100.00%*	*PWR*	*Avg. Burnup*
50	0.0	0.1	2,724.4	100.00%	PWR	
10	0.0	0.0	2,724.4	100.00%	BWR	
20	0.0	0.0	2,724.4	100.00%	BWR	
30	0.0	0.0	2,724.4	100.00%	BWR	
40	*0.0*	*0.0*	*2,724.4*	*100.00%*	*BWR*	*Avg. Burnup*
50	0.0	0.1	2,724.4	100.00%	BWR	

Temperature=570C, Time=1000hrs

10	0.0	0.0	5,449.0	100.00%	PWR	
20	0.0	0.0	5,449.0	100.00%	PWR	
30	0.0	0.0	5,449.0	100.00%	PWR	
40	0.0	0.0	5,449.0	100.00%	PWR	
48	*0.0*	*0.1*	*5,449.0*	*100.00%*	*PWR*	*Avg. Burnup*
50	0.0	0.1	5,449.0	100.00%	PWR	
10	0.0	0.0	5,449.0	100.00%	BWR	
20	0.0	0.0	5,449.0	100.00%	BWR	
30	0.0	0.0	5,449.0	100.00%	BWR	
40	*0.0*	*0.0*	*5,449.0*	*100.00%*	*BWR*	*Avg. Burnup*
50	0.0	0.1	5,449.0	100.00%	BWR	

As understanding of fuel oxidation increases, more parameters may be identified as important. Some examples of parameters of this type that may be important include the presence of moisture, the presence and concentration of chemical pellet additives and impurities, initial enrichment, initial fuel pellet grain size, initial fuel pellet density and pore configuration, and the presence of ozone (O_3). The investigation of fuel oxidation parameters is beneficial to refining equations that may be needed to calculate oxidation rates.

5.3 FAILED FUEL PERFORMANCE ISSUES

The following issues related to failed fuel performance were evaluated:

- Exposure to Air Duration—Evaluations have been performed using fuel in air times of 100 hours, 500 hours, and 1,000 hours (Tables 5-1 and 5-2). Based on the present facility designs, it will be difficult to limit the amount of time that an assembly is exposed to air when equipment malfunctions occur. However, by using the parametric results for multiple different times, it can be estimated what the oxidation rate will be such that consequences can be determined.

- Flaw Size and Location—The flaw size for failed fuel rods and location information will not be provided by the utilities and cannot be determined at the repository. Flaw size impacts the incubation time and only has an impact at higher temperatures due to the shorter incubation times. However, it is conservative to assume that incubation time is always calculated for large flaw sizes because this will result in the shortest incubation time. This result would be bounding since the flaw size will likely not be known. This is how the equations in Clad Degradation – Dry Unzipping (CRWMS M&O 2000 [DIRS 149230]) were developed. This document is the basis for Equations 5-1 through 5-4 (Section 5.1.1).

- Assembly Burnup—The assembly burnup information that will be provided by the utilities will be an average burnup. Since the axial burnup is not uniform (and differs between fuel designs), the burnup at the location of the cladding defect will not be known. As shown in Section 6.6.5, lower burnups are more conservative and should be used to estimate oxidation rates.

- Heat Generation During Oxidation—An analysis has been performed to determine if heat generation during oxidation and due to oxidation has an appreciable effect on the overall temperature of a fuel rod.

  A typical PWR assembly has an active length of 360 cm (Sanders et al. 1992 [DIRS 102072], Table I-3), 208 fuel rods per assembly, and an average loading of 420 kg of uranium. This results in 2,019 grams of uranium per rod, or 8.48 moles per rod, or 0.0236 mole/cm. Using an unzipping rate of 2.3×10^{-3} cm/min results in an oxidation rate of 5.45×10^{-5} moles/min or 3.25×10^{-3} moles/hour. Since the uranium in the fuel rod is initially UO_2, the energy released from oxidation of UO_2 to U_3O_8 must be determined. The free energy of formation of UO_2 is -1,031.8 kJ/mol and U_3O_8 is -3,369.5 kJ/mol (DOE 2000 [DIRS 152658], Table 1-3). Therefore, the heat of formation from UO_2 to U_3O_8 is –274.1 kJ/mol. Therefore, the heat generation rate is

1.23 kJ/hour or 0.34 watts. Considering an average of 2.2 failed fuel rods per assembly leads to a heat generation rate of 0.75 watts per assembly. If it is assumed that the average decay heat of an assembly is 1,100 watts, the contribution of 2.2 failed fuel rods per assembly is only 0.07 percent of the total decay heat of the assembly. Therefore, heat generation during CSNF oxidation is not significant when compared to the total decay heat of a CSNF assembly unless a significant number of rods in an assembly oxidize completely.

- Oxidation in High Humidity—The work of McEachern and Taylor (1998 [DIRS 113270]) has demonstrated that the oxidation of fuel in air is not affected by humidity until the relative humidity exceeds about 50 percent. It is likely to then accelerate oxidation until the relative humidity is high enough to limit oxygen to half of atmospheric. The clad tests at ANL show that in 1.5 years at 175°C, clad oxidation (fuel side) can be extensive under high humidity conditions. It is unknown what higher temperatures will do to the oxidation rate in a steam environment (McEachern and Taylor 1998 [DIRS 113270]; Cunnane et al. 2003 [DIRS 162406]).

- Criticality for Damaged Fuel Cans—It is estimated that less than twelve rods within a single assembly or canister can be fully oxidized before criticality becomes an issue (Section 6.6.7). Twelve rods would be approximately 26 kg of uranium fuel material. If it were assumed that 25 percent of the rods placed in a damaged fuel assembly can fully oxidize, (52 fuel rods) the quantity of fissile material released would exceed the criticality limit. Based on this information, an alternative handling method may need to be developed for these cans and so that they can be segregated from the rest of the fuel inventory. Future work needs to be performed on the handling of damaged fuel cans.

- Gamma Radiolysis Impacts on Oxidation Rates—It has been shown that gamma radiolysis increases the oxidation rate of fuel by air and can cause fuel oxidation at temperatures much lower than 360°C (Sunder and Miller 1996 [DIRS 126463]; McEachern et al. 1998 [DIRS 113270]). Also, gamma fields will be much higher in a fuel assembly than on the surface of small (and isolated) fuel fragments used in many studies whose data have been used to derive the empirical oxidation rates in this report. Gamma radiolysis impacts should be negligible because of the high flow rates of air in the surface facility and will not allow buildup or radiolysis products as happened in the reference study. Although if moisture is present, formation of radicals and hydrogen could be an issue for the cladding. It is recommended that further work be performed to confirm that radiolysis is not an issue and that not enough moisture is present to cause problems with cladding in the surface facility design (Section 5.4).

5.4 FINDINGS

- Fuel oxidation incubation time, fuel oxidation, and the extent of cladding failure were estimated for fuel temperatures of 200°C, 300°C, 350°C, 400°C, 460°C, 500°C, and 570°C, and times of 100 hrs, 500 hrs, and 1,000 hrs (Tables 5-1 and 5-2). At temperatures of 350°C to 400°C and times greater than 100 hrs, oxidation starts to become an issue.

- Time and temperature are the primary factors affecting the oxidation rate of fuel in air (Sections 5.4 and 6.6.5).
- Defect size, type, burnup, and fuel type also influence oxidation time, but to a much smaller degree. All cladding defect size effects shall be conservatively neglected. This results in a conservative result for incubation and oxidation time (Section 5.1.3).

The following issues were identified for further investigation:

- Additional work should be performed to verify that the time and temperature parameters used to evaluate fuel oxidation are conservative and no other parameters are significant (Section 5.2). This includes the effects of gamma radiolysis on fuel oxidation rates (Section 5.3).
- Further evaluations need to be performed on handling requirements for damaged fuel cans, including the potential for criticality when failed fuel contained in a damaged fuel can oxidizes (Section 5.3)
- The document, *Clad Degradation – Dry Unzipping* (CRWMS M&O 2000 [DIRS 149230]), should be revised or replaced with a new and revised calculation. The revision would give a single approved source document for dry oxidation calculations. Revising the dry oxidation analysis/model report to a current calculation will provide a referenced basis for dry oxidation evaluations. Currently there are equations published by T. Ahn, R.E. Einziger, McEachern and Taylor, B. Hanson and others. All these equations differ slightly. Some address various burnups and cladding breach size while others are for bare fuel. A single source document would be useful in that it will provide a consistent set of equations to use for calculating oxidation.

  The equation used for the parametric evaluation (results contained in Tables 5-1 and 5-2) that used the nominal case equation from *Clad Degradation – Dry Unzipping* (CRWMS M&O 2000 [DIRS 149230]) was developed using data for oxidized fuel for burnups between 15 and approximately 40 GWd/MTHM and data for fuel at temperatures between approximately 230°C and 380°C. It is recommended as a first step that the equations in *Clad Degradation – Dry Unzipping* be recreated into a calculation clearly stating the bounding limits of the equations (burnup and temperature) and incorporate any data not available when the original equations were developed. The next step (longer term goal) is to critically examine the assumptions made to create the equations and have experiments performed for burnup times and temperatures that are expected at the repository. Then revise the calculation containing the equations for lower and higher burnups and for the higher handling temperatures expected at the repository as the information becomes available from experiments or other sources.

- The times of 100 hrs, 500 hrs, and 1,000 hrs, temperatures of 400°C for normal operations and 460°C for off-normal/accident recovery operations, and burnups (average burnups of 40 PWR and 48 BWR GWd/MTHM) are the time and recommended normal and off-normal temperature input values used in the calculations for this report to assist in decision making for future work. Future work should be done to verify these parameters for each specific handling operation and this information should be documented.

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6. CONTAMINATION AND DOSE CONSEQUENCES

This section discusses the consequences from handling failed fuel in air. As discussed in Section 5, oxidation of the fuel (UO_2) to higher oxides, such as U_3O_8, may occur, which may lead to unzipping of the fuel rod cladding. Releases of fuel fines and other radionuclides from the breached fuel rod cladding would then result in contamination of the fuel transfer cells or other process rooms and dose consequences to workers and the public.

6.1 CURRENT DESIGN AND CONSEQUENCES OF OXIDATION

6.1.1 Preclosure Safety Analysis

The relevant modeling assumptions and parameters are from *Preclosure Consequence Analyses for License Application* (BSC 2005 [DIRS 171607]) and are summarized in Appendix C.

In summary, for fuel rods with pinhole or hairline crack failures, a fuel fine ARF of 3×10^{-5} is used to calculate the normal operation doses. For the purposes of calculating doses, it is assumed that 1 percent of the total fuel rods received at the repository have pinhole or hairline crack failures. For fuel rods with cladding failures greater than pinhole or hairline cracks, a fuel fine ARF of 1.2×10^{-4} is used to calculate the normal operation doses. This ARF is applicable to 154 fuel rods per year that have cladding failures greater than pinhole or hairline cracks (BSC 2005 [DIRS 171607]).

The ARF for gases is 0.3; for volatile radionuclides, the ARF is 2×10^{-4} for fuel rods with any cladding failure (BSC 2005 [DIRS 171607], Table 5). The resultant doses are summarized in Table 6-1.

Table 6-1. Normal Operation Dose Consequences—Current Design

Receptor	Other Sources[a] (mrem)	Oxidation Contribution[b] (mrem)	Total Dose[c] (mrem)	Dose Limit (mrem)
Worker	1,842[d]	—	1,842	5,000 mrem
Onsite Public	4.33	0.0966	4.43	100 mrem
Offsite Public	0.366	0.00789	0.374	15 mrem

NOTES: [a] Other Sources = Total Dose—Oxidation Contribution.

[b] Oxidation contribution from MACCS2 Run 2 (BSC 2005 [DIRS 171607]) to the public, not calculated for workers.

[c] Total doses for the public from BSC 2005 [DIRS 171607], Table 22.

[d] Total doses for worker consists of the maximum dose to the FHF or DTF cask/waste receipt operator from BSC 2005 [DIRS 171607], Table 27 and the normal operation and Category 1 doses from BSC 2005 [DIRS 171607], Table 29.

No release from oxidation is considered during Category 1 or Category 2 event sequences.

6.1.2 Postclosure Safety Analysis

Oxidation of fuel is discussed in *Waste-Form Features, Events, and Processes* (BSC 2004 [DIRS 170020]). The preclosure oxidation dose is not modeled in the TSPA (BSC 2004 [DIRS 170020], p. 6-18). Therefore, for CSNF, preclosure oxidation will not significantly affect the TSPA dose. The TSPA states that for CSNF, the potential oxidation of fuel during preclosure handling operations has negligible consequences to the reasonable maximally exposed individual.

6.2 RELEASE FRACTIONS FROM OXIDATION

The splitting of the cladding due to oxidation may result in releases of U_3O_8 dust (or fuel fines), gaseous, and volatile radionuclides.

6.2.1 Fuel Fines/Particulates

Davis et al. (1998 [DIRS 103711], p. 9) determined the ARF of particles from powder due to the oxidation of UO_2 pellets in air. The ARF was based on experiments performed by Iwasaki et al. (1968 [DIRS 172518]). In these experiments, unirradiated fuel pellets were oxidized in air at temperatures between 500 and 700°C. The particle size distributions at different temperatures were then determined. Using this particle size distribution, Davis et al. (1998 [DIRS 103711]) determined that 12 percent of the powder/particles were small enough to become airborne. Davis et al. further stated that 1 percent is expected to be respirable. Davis et al. cites a DOE source for the respirable fraction (RF); however, no specific section is called out. It is possible that Davis et al. used the data for chemically nonreactive compounds (DOE 1994 [DIRS 103756], Section 4.4.1.1) as the source of the RF.

A review was undertaken to determine the reasonableness of the ARF × RF value used for oxidized particles. For free fall spill experiments (DOE 1994 [DIRS 103756], Section 4.4.3.1.2), the bounding ARF and RF were determined to be 2×10^{-3} and 0.3 for the maximum spill height ($ARF \times RF = 6 \times 10^{-4}$). For releases of powder due to vibration or shock (DOE 1994 [DIRS 103756], Section 4.4.3.3.1), the bounding ARF and RF were determined to be 1×10^{-3} and 1 for powderlike contamination. However, the bounding RF for agglomerated particles was considered to be 0.1. For large falling object impact or induced air turbulence (DOE 1994 [DIRS 103756], Section 4.4.3.3.2), the bounding ARF and RF were conservatively set to be 1×10^{-2} and 0.2. It was also noted that these bounding values were a factor of 5 higher than the largest measured value. For complete oxidation of uranium metal at temperatures greater than 500°C (DOE 1994 [DIRS 103756], pp. 4-2 and 4-3), the bounding ARF and RF were determined to be 1×10^{-3} and 1. Complete oxidation of uranium metal in a flowing air atmosphere resulted in an ARF of 1.0×10^{-3} as determined by DOE (1994 [DIRS 103756], p. 4-3). Using the same data provided by Iwasaki et al. (1968 [DIRS 172518]), DOE (1994 [DIRS 103756], Section 4.3.1.3.1) indicates that an ARF and RF of 6×10^{-3} and 0.01 are appropriate for commercial spent fuel.

Very little data exist regarding the release of fuel particles from fuel rods with cladding damage. Two studies did quantify the amount of fuel fines that were found following the tests. Lorenz et al. (1980 [DIRS 100990] Test HBU-5) indicate that 5.85 mg of fuel remained on the test holder in a test performed at 500°C. In this test, the releases ended after 4 hours due to plugging the defect. Einziger and Cook (1980 [DIRS 100990], p. 65) indicate that 2.9 g of fissile material fell out of the crack after oxidation at 229°C for 2,235 hours. Using Equation 5-1, it takes 1,172 hours for the fuel to oxidize to $UO_{2.4}$. During that time, it is not expected that any fuel would fall out from the rod. Therefore, the amount of time available for the fuel particles to fall out is 1,063 hours. Assuming a linear relationship results in a release rate of 1.4×10^{-3} mg/h for the Lorenz test and 2.7 mg/h for the Einziger and Cook test. Using the higher rate and assuming a handling time of 100 hours results in 270 mg of fissile material released. As discussed in Section 5.3, a PWR rod has an average loading of 2,019 g of uranium; therefore, the ARF is 1.4×10^{-4}.

The above discussion indicates that using an ARF of 1.2×10^{-3} with an RF of 1 for oxidation of fuel bounds the available experimental data. From Section 5.2.2, oxidation during normal operations at 360°C may result in unzipping between 2.2 percent and 5.4 percent of a rod with a cladding defect. Rounding up to 10 percent and using the ARF of 1.2×10^{-3} for bare fuel pellets, an effective ARF of 1.2×10^{-4} is recommended for fuel rods that are expected to be vulnerable to oxidation and cladding unzipping.

6.2.2 Fission Gases

Release of fission product gases from oxidation, such as krypton, have been determined by Einziger (1991 [DIRS 166177], p. 95) and Lorenz et al. (1980 [DIRS 100990], Tests HBU-5 and HBU-6). Einziger (1991 [DIRS 166177] p. 95) indicates that approximately 7 percent to 30 percent of the ^{85}Kr inventory is released from the fuel upon oxidation to U_3O_8 for advanced gas-cooled reactor fuel. Einziger (1991 [DIRS 166177], p. 95) states that although no similar tests have been done for light water reactor fuel, it is expected that the trends will be similar.

Einziger (1991 [DIRS 166177], p. 95) indicates that 8.2×10^{-4} of the gas is released for each centimeter of fuel oxidized. As discussed in Section 5.2.2, for handling operations of 100 hours, about 15.5 cm of the cladding may be split for a breached BWR rod. This would result in 1.27×10^{-2} of the gas in the rod released. Lorenz et al. (1980 [DIRS 100990]) in test HBU-5 reported that 0.53 percent of the ^{85}Kr was released due to oxidation. The test rod segment was used previously in a test where 0.63 percent was released. Lorenz et al. (1980 [DIRS 100990], Figure 11) indicates that ^{85}Kr releases ended after about 4 hours. Lorenz et al. suggest that this is due to plugging of the defect hole. As can be seen in Lorenz et al. (1980 [DIRS 100990], Figure 11), even if plugging did not occur, the release of ^{85}Kr was reaching an upper limit. Therefore, no more than 0.6 percent of the total ^{85}Kr would have been released due to oxidation. In test HBU-6, which was performed at 700°C instead of 500°C a fresh rod segment was used. It is assumed that the release rate of ^{85}Kr is roughly linear once the temperature has reached a steady 700°C (Lorenz et al. 1980 [DIRS 100990], Figure 14). From these data, a release rate of 0.14 percent/hour is estimated. Thus, for a 100-hour duration, approximately 14 percent of the total ^{85}Kr would be released. Both the Einziger (1991 [DIRS 166177]) and Lorenz et al. (1980 [DIRS 100990]) data show that the releases of ^{85}Kr are below the 30 percent fraction recommended by BSC (2004 [DIRS 172371], Table 11).

The release rate of iodine in the dry air test at 700°C (Lorenz et al. 1980 [DIRS 100990], Test HBU-6) was found to be 9.6×10^{-5}/hour; thus, for 100 hours, the total iodine RF is 9.6×10^{-3}, which is well below the 30 percent fraction recommended by BSC (2004 [DIRS 172371], Table 11).

Voloxidation experiments indicate that in the process of oxidation of UO_2 to U_3O_8 essentially removed all of the tritium, 17 to 22 percent of the ^{14}C, 7 to 17 percent of the ^{85}Kr, and less than 8 percent of the ^{129}I. Except for tritium, these release fractions are below the 30 percent fraction recommended by BSC (2004 [DIRS 172371], Table 11).

6.2.3 Volatile Radionuclides

Reexamination of Spent Fuel Shipment Risk Estimates (Sprung et al. 2000 [DIRS 152476], NUREG/CR-6672, p. 7-45) discusses the releases of iodine, cesium, and ruthenium from fuel following oxidation. The data used by Sprung et al. (2000 [DIRS 152476]) come from experiments performed by Lorenz et al. (1980 [DIRS 100990]) on highly irradiated fuel at 700°C. Sprung et al. (2000 [DIRS 152476], p. 7-45) indicate that the releases of Cs, I, and Ru increased respectively by factors of 54.6, 22.4, and 20,200, respectively, when the experimental atmosphere was dry air rather than steam. The experiments by Lorenz et al. (1980 [DIRS 100990]) were performed for various temperatures in both steam and dry air atmospheres. Three steam tests (HBU-1, HBU-2, and HBU-4) and two dry air tests (HBU-5 and HBU-6) results are reviewed. The release rate of Cs in the steam atmosphere was found to be highly dependent on temperature, suggesting a diffusional release mechanism from the fuel matrix. This release rate was significantly enhanced in the dry air atmosphere at 700°C (Lorenz et al. 1980 [DIRS 100990], Test HBU-6); however, the release rate was not appreciably different at 500°C (Lorenz et al. 1980 [DIRS 100990], Test HBU-5). The cladding temperatures expected at the repository will be well below the temperatures used in these tests; therefore, it is expected that the release of cesium will be much lower. Conservatively assuming the release rate at 500°C represents the expected release rate at the repository, the total RF of cesium after 100 hours of oxidation would be approximately 1.3×10^{-7}. The release rates of ruthenium were all well below the release rates for cesium except for test HBU-6, where the release of ruthenium was approximately a factor of 2 lower than cesium. These RFs are at least one order of magnitude smaller than the 2.0×10^{-4} recommended for use in consequence analyses for Cs and Ru by BSC (2004 [DIRS 172371], Table 11).

6.2.4 Receipt of High-Heat Load Fuel

The above discussion assumed that the duration of normal handling operations in air will be approximately 100 hours and at expected heat load and temperature conditions. There will be instances where the fuel is exposed to air for longer periods of time or fuel can be brought in with higher heat loads; thus, at higher temperatures. For example, fuel may be unloaded from the transportation cask and placed in the staging cell of the DTF. Fuel may be in the staging cell for up to 30 days.

The average heat load for a PWR assembly is 601 watts and the average heat load for a BWR assembly is 191 watts (BSC 2004 [DIRS 167441], Table 19).

Fuel assemblies with heat loads greater than the average heat loads discussed above will be received at the repository. However, these assemblies will be shipped in transportation casks whose maximum heat load is limited; therefore, they will be mixed with assemblies with lower heat loads. The heat load limitations of the waste packages will also result in blending these high heat load assemblies with low heat load assemblies. Therefore, the maximum temperatures calculated for the transportation casks and waste packages will apply to these high heat load assemblies. Per BSC (2004 [DIRS 172741], Table 30), the maximum temperature for a transportation cask in air is 371°C and 304°C for a waste package. At 371°C, less than 7 percent of the fuel rod cladding would unzip after 100 hours, and at 304°C, less than 1 percent of the fuel rod would unzip after 100 hours. Therefore, the conditions for the average fuel and the above calculated RFs are valid for the high heat load fuel during the normal transfer operations.

From BSC (2004 [DIRS 171778], Table 16), the maximum temperature for a 600 watt assembly in the staging area with a 2 m/s ventilation flow rate is less than 160°C. In addition, the staging cell ventilation can be designed to ensure that cladding temperature does not exceed 230°C for this average assembly. Little or no oxidation is expected to occur at these temperatures and for the expected duration of storage in the handling cell (up to 30 days).

If high heat load fuel is staged in the staging area for up to 30 days, the temperature in the staging area will be higher than the 230°C. Considering the 95 percent high heat load fuel assembly of 1,600 watts (BSC 2003 [DIRS 165990], Tables 2, 7, and 8), the maximum temperature in the staging area with a 2 m/s ventilation flow rate is 260°C per BSC (2004 [DIRS 171778], Table 16). At this temperature, the incubation time, as calculated using the methodology described in Section 5.2.1, is greater than 1,000 hours for burnups of 30 GWd/MTHM. For burnups of less than 30 GWd/MTHM, the incubation time may be less than 30 days. Using the rate of crack propagation determined by Equation 5-5 at 260°C, or 2.6×10^{-4} cm/min, the maximum cladding crack length after 30 days is estimated to be approximately 7 cm. This crack is less than the crack estimated for the average fuel; therefore, the ARF of 1.2×10^{-4} is valid for the high heat load fuel. Using BSC (2003 [DIRS 165990] Tables 2, 7, and 8), it is estimated that 7,647 total assemblies will have heat loads greater than 1,600 watts. Assuming that 3.7 percent (sum of damaged, known, and unknown leakers) of the assemblies have cladding failures (see Table 6-2 for BWR assemblies), 283 high heat load assemblies would be subject to oxidation (or about 11 assemblies per year).

The FHF does not have a staging area; fuel assemblies may be staged in aging casks in the FHF. The loading of the aging cask will depend on the heat load of the fuel assembles such that the maximum cladding temperature does not exceed 400°C. Thermal analyses are being prepared to determine the expected temperature during staging in the aging casks. The expected amount of oxidation and resultant release fractions will be determined when the thermal analyses become available.

6.3 HIGH TEMPERATURE FUEL OXIDATION

6.3.1 Normal Operation High Temperature Fuel

Assuming that the temperature of the worst case fuel rod in the assembly is at the design temperature of 400°C, the amount of fuel subject to oxidation will be a worst case. The calculation of the amount of fuel subject to oxidation is performed using the methodology discussed in Section 5.1. Instead of the nominal values in Equations 5-1 and 5-2, the bounding values are used. The fuel pellet gap ratio, x, and the strain, s, used in Equation 5-4 are set to zero (i.e., no gap and no strain required to split the cladding). Using this methodology, the total incubation time is approximately 0.5 hour.

The crack propagation velocity at 400°C using the correlation shown in Equation 5-5 is then 7.1×10^{-3} cm/min. At this rate and for approximately 100 hours, 42.3 cm of the fuel rod may unzip, or 13 percent of a PWR rod, and 12 percent of a BWR rod. Rounding up to 20 percent and using the ARF of 1.2×10^{-3} discussed in Section 6.2.1, an ARF of 2.4×10^{-4} is recommended for the high temperature fuel.

6.3.2 Off-Normal Operation with High Temperature Fuel

A loss of ventilation for an extended period of time with fuel in the transfer cell would result in increases in temperature. It is assumed that ventilation is not available for up to 30 days. In these conditions, if the fuel was in a cask or waste package, the temperature may be above 450°C (BSC 2005 [DIRS 172741], Table 33). At this temperature any rod with damaged cladding is expected to fully oxidize and unzip. Therefore, it is recommended that an ARF of 1.2×10^{-3} for fuel fines and volatiles, which corresponds to the bare fuel ARF described in Section 6.2.1, be used for this condition.

6.4 EXPECTED FAILED FUEL ARRIVAL RATES AT THE REPOSITORY

As discussed in Section 3.2.3, the number of failed assemblies expected to be received at the repository is estimated to be 2.8 percent for PWR fuel and 3.4 percent for BWR fuel. The failure characterization discussion in Section 3.2.3 indicates that industry experience shows that 10 percent of these damaged assemblies are known leakers that are not placed in damaged fuel cans (i.e., 0.28 percent of PWR and 0.34 percent of BWR fuel arriving at the repository would be identified leakers [pinhole or hairline crack or less]). Section 3.2.3 indicates that another 10 percent would be unidentified leakers.

Table 6-2 lists the number of assemblies with clad damage that are expected to be received at the repository in any one year. The values in this table are based on a receipt of approximately 222,000 (127,700 BWR and 94,400 PWR) assemblies over the life of the repository (BSC 2003 [DIRS 165990], Table 2) and an operational life of 25 years.

Table 6-2. Yearly Receipt of Damaged Assemblies

	BWR		PWR		
Assembly Type	Percent of all Assemblies	Number of Assemblies Per Year	Percent of all Assemblies	Number of Assemblies Per Year	Total Assemblies Per Year
Damaged—canned	3.0	154	2.5	95	249
Known leakers	0.34	17	0.28	11	28
Unknown leakers	0.34	17	0.28	11	28
Total assemblies		188		117	305

EPRI (1997 [DIRS 100444], p. 4-1) indicates that, on the average, 2.2 rods per failed assembly are failed. Therefore, about 548 failed rods in 249 assemblies per year are expected to be damaged fuel in cans with wire mesh end closures, and 123 failed rods in 56 assemblies are expected to be leakers with pinhole or hairline cracks or less (both known and unknown).

6.5 INPUTS TO CONSEQUENCE ANALYSIS

In summary, the following inputs are used to determine doses during normal handling operations and off-normal conditions.

Normal Operations

- As discussed in Section 6.4, 548 failed rods per year are expected to be damaged fuel in cans and 123 failed rods per year are expected to be leakers (both known and unknown). Therefore, 671 fuel rods received at the repository per year have cladding failures subject to oxidation. This number does not include the fuel rods from reconstitution or fuel pieces and debris that may be included in failed fuel canisters.
- For fuel rods with cladding failures of any kind, the ARF for tritium is 1.0.
- For fuel rods with cladding failures handled at expected normal temperature conditions, a fuel fine ARF of 1.2×10^{-4} should be used to calculate the normal operation doses.
- For fuel rods with cladding failures of any type, the ARF of gaseous radionuclides, such as noble gases and iodines, is 0.3.
- For fuel rods with cladding failures of any type, the ARF of volatile radionuclides, such as cesium and ruthenium, is 2×10^{-4}.
- Fuel assemblies placed in the staging area of the DTF will be at lower temperatures than in the transportation cask or in the waste package; therefore, the ARFs for normal operations can be used to calculate the doses.

Off-Normal Conditions:

- For fuel rods with cladding failures, a fuel fine and volatile ARF of 2.4×10^{-4} and a gaseous ARF of 0.3 should be used to calculate the doses.
- Fuel assemblies affected by loss of ventilation for extended periods of handling can be assumed to be fully oxidized. For these assemblies, a fuel fine and volatile ARF of 1.2×10^{-3} and a gaseous ARF of 0.3 should be used to calculate the doses.
- Fuel assemblies placed in the staging area will be at lower temperatures than in the transportation cask or in the waste package; therefore, the ARFs for normal operations can be used to calculate the doses.

6.6 CONSEQUENCE ANALYSIS

Section 6.1.1 states that the current consequence analysis considers burst rupture and subsequent localized oxidation of 1 percent of fuel rods for a nominal throughput of 3,000 MTHM per year. This analysis assumed oxidation of 154 failed fuel rods with 10 percent of the entire cladding length unzipped during normal operations. The total number of fuel rods subject to oxidation, however, is expected to be 671 instead of 154 (Section 6.5). A preliminary normal operation dose analysis was performed to determine the effect of increasing the number of fuel rods subject to oxidation. In addition, radiation analyses were performed to determine the effects of the releases of oxidized fuel on the contamination levels in the transfer cell and on the HEPA filters. Doses from a loss of ventilation event were also determined.

In addition to the calculations described above, bounding analyses were also performed for the fuel rods, assuming they were fully oxidized. Finally, a sensitivity analysis was performed to determine the effects of varying the oxidation parameters on the resultant doses.

6.6.1 Normal Operation Doses

The normal operation doses are calculated using the methodology described in BSC (2005 [DIRS 171607]). The major input parameters and assumptions are listed below.

6.6.1.1 Input Parameters and Assumptions

- Source term basis (BSC 2004 [DIRS 167441], Section 6.1.1.1):
  - Average PWR fuel (4 percent enrichment, 48 GWd/MTU burnup, and 25 yr cooled).
- Number of damaged fuel rods subject to oxidation: 671 rods/yr (Section 6.5).
- ARFs for oxidized fuel (Section 6.5)
  - Gas ARF = 0.3
  - Cs ARF = 2.0×10^{-4}
  - Fuel fines/particulates = 1.2×10^{-4}.
- Crud effective ARF = 0.0015 (BSC 2004 provides justification for lower crud ARFs (BSC 2004 [DIRS 172371], p. 32). For the purposes of this calculation, the crud accident ARF is reduced by a factor of 10 since no drop is assumed.)
- HEPA filter efficiency = 99 percent each for two banks in a series (total reduction factor of 10,000), BSC (2005 [DIRS 171607], Assumption 4.8).
- Mass of uranium in the fuel assembly (based on BSC 2004 [DIRS 167441], Table 16)
  - PWR assembly = 429.7 kg
  - BWR assembly = 180.2 kg.

- Total effective dose equivalent (TEDE) dose limits (BSC 2005 [DIRS 171607], Table 1)
  - Offsite public = 15 mrem/yr
  - Onsite public = 100 mrem/yr
  - Worker = 5,000 mrem/yr.

6.6.1.2 Results

The resultant doses are shown on Table 6-3. As shown, the oxidation contribution is approximately 30 percent of the total dose for the public. The amount of airborne material lost due to oxidation is 166 grams for PWR fuel and 184 grams for BWR fuel. The total amount of fuel material escaping the cladding breached by fuel oxidation has not been evaluated due to the uncertainties in modeling this physical phenomenon.

Table 6-3. Normal Annual Operation Dose Consequences

Receptor	Other Sources (mrem)[a]	Oxidation Contribution (mrem)	Total Dose (mrem)	Dose Limit (mrem)
Worker	1,842	1.38	1,843	5,000
Onsite Public	4.33	1.99	6.32	100
Offsite Public	0.366	0.18	0.55	15

NOTE: [a] Other sources from Table 6-1.

6.6.2 Dose Rate from Contamination

Releases of fuel fines from oxidized fuel may accumulate over time in the fuel transfer cells. This accumulated activity would pose a direct dose hazard to workers entering the fuel transfer cell for maintenance operations. An analysis was performed to determine the exposure level due to the accumulated activity. For this analysis, the FHF was chosen since the fuel transfer cell is smaller than the DTF transfer cell; thus, concentrating the activity that results in higher dose rates.

6.6.2.1 Input Parameters and Assumptions

- All of the rods subject to oxidation received in a year are processed through the FHF, 671 rods/yr (Section 6.5).

- FHF throughput of 80 waste packages/yr
  - PWR—equivalent to 1,680 assemblies per yr
  - BWR—equivalent to 3,520 assemblies per yr.

- Contamination uniformly distributed over the floor of the FHF transfer cell.

- Source term basis—same as normal operations (Section 6.6.1.1) with the following exception:
  - Gas ARF = 0.0.
- 100 percent of the airborne activity released is deposited on the surfaces of the FHF transfer cell.
- Fuel material (U_3O_8 powder, pellets) that may escape from the breached cladding during fuel handling operations and relocated on the cell surfaces is not considered.
- Accumulation of activity for one year (assumes yearly cleanup of the fuel transfer cells).
- No decay of activity during accumulation.

6.6.2.2 Results

The resultant dose rates from the activity accumulated over one year are shown in Table 6-4. Although no operational limits have been set to date, a rate of 100 mrem/hour has been suggested. As can be seen in Table 6-4, this suggested limit is exceeded, which indicates that more frequent decontamination will be required, also see Section 6.7.

Table 6-4. Dose Rates Three Feet Above Floor

Source	Dose Rate (mrem/hour)	
	PWR	BWR
Crud	91	940
Oxidation	780	680
Total	870	1,630

6.6.3 Dose Rate Calculation from Activity Buildup on HEPA filters

Releases of fuel fines from oxidized fuel will be picked up by the air flow and will accumulate over time on the HEPA filters. This accumulated activity may challenge the filter shielding design basis discussed in BSC (2004 [DIRS 171405]). An analysis was performed to determine the exposure level due to the accumulated activity.

6.6.3.1 Input Parameters and Assumptions

- Facility: DTF 1 or DTF 2
  - Throughput for DTF 1: 1,500 MTHM/yr
  - Throughput for DTF 2: 1,500 MTHM/yr
- Source term basis—same as normal operations (Section 6.6.1.1) with the following exception

- Gas ARF = 0.0
- HEPA filter parameters
- No. of HEPA filters per facility: 72 (4 trains, each with 2 × 9 arrays)
- Filter size: 2' × 2' × 1' each
- Source distribution: uniform over all filters
- Accumulation of activity for one year (assumes yearly cleanup of the fuel transfer cells).
- No decay of activity during accumulation.

6.6.3.2 Results

The resultant dose rates from the activity accumulated over one year are shown in Table 6-5. The total dose rate is below the DTF shielding design dose rate of 3,200 mrem/hour (BSC 2004 [DIRS 171405], p. 64).

Table 6-5. Surface Dose Rate at Each Filter

Source	Dose Rate (mrem/hour)
Crud	470
Oxidation	1,090
Total	1,560

6.6.4 Dose Calculation from Resuspension of Radioactive Materials—Loss of HVAC

Activity accumulated over time on the surfaces of the transfer cell could be a source of dose from the facility. A loss of HVAC for the fuel transfer cell would result in a loss of confinement with subsequent leakage of resuspended radioactive materials to the worker occupied areas and to the environment. For this calculation, a conservative assumption was made that upon loss of HVAC, confinement is also lost and the activity is resuspended and mixed in the secondary confinement zone. Loss of HVAC will impact confinement, however, cell walls, airlocks, and penetration seals will continue to provide a barrier to movement of contamination out of the primary confinement zone. Currently, this event is not considered in the PCSA consequence calculations (BSC 2005 [DIRS 171607]). For this analysis, the FHF was chosen since the secondary confinement areas are much smaller than the DTF secondary confinement areas, which result in higher airborne concentrations for the same resuspension rate.

6.6.4.1 Input Parameters and Assumptions

- All of the rods subject to oxidation received in a year are processed through the FHF, 671 rods/yr (Section 6.5).

- FHF throughput of 80 waste packages/yr (equivalent to 1,680 PWR assemblies/yr)
  - FHF bounds DTF for worker dose, which is limiting.
- Source term basis—same as normal operations (Section 6.6.1.1) with the following exception:
  - Gas ARF = 0.0.
- 100 percent of the airborne activity released is deposited on the surfaces of the FHF transfer cell.
- Fuel material (U_3O_8 powder, pellets) that may escape from the breached cladding during fuel handling operations and relocated on the cell surfaces is not considered.
- Accumulation of activity for one year (assumes yearly cleanup of the fuel transfer cells).
- No decay of activity during accumulation.
- Resuspension of activity (DOE 1994 [DIRS 103756])
  - During normal HVAC operations = 4×10^{-5}/hr
  - During loss of HVAC = 4×10^{-6}/hr.
- Loss of HVAC and subsequent loss of confinement.
- Activity resuspended and mixed in the secondary confinement zones of the FHF.
- Worker and onsite public doses are calculated for 2 hours (assumes evacuation).
- Offsite public doses are calculated for 30 days (assuming HVAC is restored after 30 days).

6.6.4.2 Results

The resultant doses are presented in Table 6-6. The results are compared to the Category 1 dose limits.

Table 6-6. Loss of HVAC TEDE Inhalation Doses

Receptor	Dose (mrem)	Category 1 Dose Limit (mrem)
Worker	413	5,000
Onsite Public	1.11	100
Offsite Public	2.18	15

6.6.5 Sensitivity Analysis

A sensitivity analysis is performed to determine the importance of the parameters used to calculate the oxidation and unzipping rates. The parameters used in Equations 5-1 through 5-5 of Section 5.2 are varied as shown below. The results of the sensitivity analysis are compared to the results for normal operations presented in Sections 6.6.1 through 6.6.4.

6.6.5.1 Parameters Varied

- Cladding gap—triangular distribution
  - Minimum = 0 mm
  - Maximum = 0.25 mm
  - Most likely = 0.03 mm
- Strain—triangular distribution
  - Minimum = 0 percent
  - Maximum = 6.5 percent
  - Most likely = 1 percent
- Volume ratio of $UO_{2.4}$ to UO_2 (z_1)—uniformly distributed between 0.98 to 0.9929
- Temperature—triangular distribution
  - Minimum = 300°C
  - Maximum = 400°C
  - Most likely = 370°C
- Time—triangular distribution
  - Minimum = 90 hrs
  - Maximum = 200 hrs
  - Most likely = 120 hrs
- Burnup—uniformly distributed between 10 and 45 MWd/Kg M.

6.6.5.2 Sensitivity Results

The sensitivity analysis determined the significance of the six parameters that were varied using a Monte Carlo analysis method. A rank correlation coefficient was calculated for each of the parameters and output doses presented in Section 6.6.5.3. The higher the correlation between the input and output, the more significant the input is in determining the output values. The correlation coefficients are provided in Table 6-7.

Table 6-7. Sensitivity Analysis Results

Parameter	Correlation Coefficient
Temperature	0.895
Time	0.286
Burnup	-0.238
Strain	-0.0499
Cladding Gap	-0.0301
Volume Ratio	0.0171

As shown above, the resultant doses are strongly dependent on temperature and time. Burnup has a negative correlation, which indicates that assuming low burnup for the fuel results in conservative doses. The other three parameters have a very small effect on the resultant doses; therefore, conservative values for these parameters can be used without unduly affecting the doses.

6.6.5.3 Dose Calculation Sensitivity Analysis Results

In the following paragraphs, the doses calculated using the sensitivity analysis parameters were compared to doses calculated using the parameters discussed in Sections 6.6.1 through 6.6.4. As can be seen, the doses using the recommended ARFs are always higher than the mean calculated using the sensitivity analysis.

The normal operation doses are calculated using the input parameters and assumptions described in Section 6.6.1.1 with the exception of the ARFs. The ARFs for fuel fines and volatile radionuclides were varied dependent on the variables listed in Section 6.6.5.1. The results are presented in Table 6-8 for the material deposited and Table 6-9 for the resultant doses.

Table 6-8. Amount of Airborne Material Deposited (kg/yr) Sensitivity Analysis Results

	Mean	5%–95%	Section 6.6.1.2
PWR assemblies	0.112	0.0–0.235	0.166
BWR assemblies	0.124	0.0–0.259	0.184

Table 6-9. Normal Operation Dose Sensitivity Analysis Results—TEDE

Receptor	Mean (mrem)	5%–95% (mrem)	Table 6-3 (mrem)
Worker	0.98	0.17–1.87	1.38
Onsite Public	1.42	0.25–2.70	1.99
Offsite Public	0.13	0.022–0.25	0.18

The worker dose rates from the accumulated activity are calculated using the input parameters and assumptions described in Section 6.6.2.1 with the exception of the ARFs. The ARFs for fuel fines and volatile radionuclides were varied dependent on the variables listed in Section 6.6.5.1. The results are presented in Table 6-10 for the resultant dose rates.

Table 6-10. Worker Dose Rate Sensitivity Analysis Results—3 feet above floor

Source	Dose Rate—3 ft above floor (mrem/hr)		Table 6-4 (mrem/hr)	
	PWR	BWR	PWR	BWR
Crud	91	940	91	940
Oxidation—mean	769	677	780	680
Oxidation 5%–95%	747–790	665–688	N/A	N/A

The dose rates from the accumulated activity on the HEPA filters are calculated using the input parameters and assumptions described in Section 6.6.3.1 with the exception of the ARFs. The ARFs for fuel fines and volatile radionuclides were varied dependent on the variables listed in Section 6.6.5.1. The results are presented in Table 6-11 for the resultant dose rates.

Table 6-11. HEPA Filter Dose Rate Sensitivity Analysis Results

Radioactivity Source	Dose Rate on Filter Surface (mrem)		
	Mean	5%–95%	Table 6-5
Oxidation	1,070	1,040–1,100	1,090
Crud	470	—	470
Total	1,540	—	1,560

The loss of HVAC doses were calculated using the input parameters and assumptions described in Sections 6.6.4.1 and 6.6.3.1 with the exception of the ARFs. The ARFs for fuel fines and volatile radionuclides were varied dependent on the variables listed in Section 6.6.5.1. The results are presented in Table 6-12 for the resultant inhalation TEDE doses for this event.

Table 6-12. Loss of HVAC Sensitivity Analysis Dose Results

Receptor	Mean (mrem)	5%–95% (mrem)	Dose Limit (mrem)	Table 6-6 (mrem)
Worker	279	0.418–583	5,000	413
Onsite Public	0.750	0.00113–1.57	100	1.11
Offsite Public	1.47	0.00221–3.08	15	2.18

6.6.6 Bounding Dose Analysis

A bounding analysis was performed to determine the consequences of assuming that the damaged fuel received at the repository will completely oxidize. This bounding analysis uses the parameters discussed in Sections 6.6.1 through 6.6.4 with the exception of the ARFs. To account for a fully oxidized fuel rod, the fuel fine ARF was increased to 1.2×10^{-3} as recommended in Section 6.5. This ARF was also used for the volatile radionuclides since it is assumed that any volatiles bound with the airborne fuel fines will also be airborne. The results of this analysis are presented in Tables 6-13 through Table 6-17.

Table 6-13. Bounding Amount of Airborne Material Deposited (kg/yr)

	Bounding	Section 6.6.1.2
PWR assemblies	1.7	0.166
BWR assemblies	1.8	0.184

Table 6-14. Bounding Normal Operation TEDE Doses—TEDE

Receptor	Bounding (mrem)	Table 6-3 (mrem)
Worker (mrem)	12	1.38
Onsite Public (mrem)	18	1.99
Offsite Public (mrem)	1.6	0.18

Table 6-15. Bounding Maintenance Worker Dose Rates

Source	Dose Rate—3 ft above floor (mrem/hr)		Table 6-4 (mrem/hr)	
	PWR	BWR	PWR	BWR
Crud	91	940	91	940
Oxidation	4,800	4,200	780	680
Total	4,900	5,100	870	1,630

Table 6-16. Bounding HEPA Filter Dose Rates

Radioactivity Source	Dose Rate on Filter Surface (mrem/hr)	
	Bounding	Table 6-5
Oxidation	6,700	1,090
Crud	470	470
Total	7,200	1,560

Table 6-17. Bounding Loss of HVAC Inhalation TEDE Doses

Receptor	Bounding (mrem)	Table 6-6 (mrem)
Worker	4,100	413
Onsite Public	11.1	1.11
Offsite Public	21.8	2.18

6.6.7 Criticality

A preliminary study looked at spherical (optimum) geometries of oxidized uranium fuel from approximately six and twelve fuel rods at a range of moderator and uranium densities to cover the possible range of the corrosion products and moderation levels. The conditions with the highest effective neutron multiplication factor, k_{eff}, were identified and compared to the upper subcritical limit (USL) for a similar class of configurations. Cases where the k_{eff} exceeds the USL are considered critical. All the cases for six rods worth of oxidized fuel have k_{eff} values below the USL and so they are subcritical. Some of the cases for twelve rods worth of oxidized fuel have k_{eff} values above the USL so they are critical under conditions of optimum moderation and geometry.

The amount of material in twelve rods worth of oxidized fuel is approximately 25 kg. As shown in the previous sections, the maximum amount of airborne material released is 1.8 kg for the average failed fuel (2.2 failed rods per assembly) subject to full oxidation. This value is well below the amount required for criticality concerns.

6.6.8 Discussion of Results

The previous sections show that for normal operations and the off-normal events considered in this study, doses to workers and the public from handling fuel in air are well within the established limits. This remains to be evaluated for other off-normal events involving fuel oxidation.

It has also been shown that the amount of material released from the fuel due to oxidation is well below criticality levels if decontamination is performed during the life of the facility.

6.7 CONTAMINATION IN THE FUEL TRANSFER CELL

The level of contamination due to oxidation of the fuel presents a challenge to hands-on maintenance in the fuel transfer cell, which requires frequent decontamination. The total amount of fuel powders or pellets that may be released during the handling of fuel with breached cladding has not been determined. There exists little or no information in the literature regarding this issue. Anecdotal evidence presented in many studies suggests that some fuel fines are released and drop off (Einzinger 1991 [DIRS 166177], p. 92; Lorenz 1980 [DIRS 100990], p. 34); however, the amount lost and deposited on surfaces is not available.

Since the amount of material that may be released from the breached cladding is not available, an effort was made to determine the level of contamination present in the fuel transfer cell above which work conditions would be undesirable (either because of radiation exposure or because of the resource expenditure required for recovery).

Criteria were established such that it would be undesirable to allow the level of contamination on the transfer cell surfaces to result in dose rates in excess of 100 mrem/hr at one foot (30 cm). These criteria were chosen to prevent the area from being posted as a high radiation area due solely to the presence of the contamination.

Of the radionuclides released during fuel oxidation, only three were considered to be significant gamma dose rate contributors: Co-60, Ba-137m, and Eu-154. The amount of these three radionuclides, when deposited over the surface area of the transfer cell floor, which would deliver 100 mrem/hr at one foot, was then calculated. Applying a factor relating the relative abundance, the total amount of each radionuclide present (which would deliver 100 mrem/hr at one foot) was then determined. This resulted in approximately 1.5×10^9 dpm/100cm^2. This value includes alpha and gamma emitters for which typical GM-based surface contamination instruments have very low measurement efficiencies. As a matter of practicality, the activity of beta emitters (i.e., those easily detected by a GM type instrument) was determined. This resulted in approximately 1.0×10^9 dpm/100cm^2 of measured activity to deliver 100 mrem/hr at one foot.

In addition to dose rate considerations, airborne radioactivity due to resuspension of the surface contamination should be considered. This is necessary to evaluate whether airborne radioactivity concentrations due to resuspension would be acceptable at the surface area concentrations based solely on dose rate. Once the dose rate based criteria were determined, an evaluation was conducted to determine whether airborne activity due to resuspension of surface contamination was an additional concern. Assuming all activity released from the oxidized fuel is deposited (i.e., no ventilation removal) and subject to resuspension results in a conservative calculation. Utilizing a resuspension factor of 1×10^{-6} m^{-1}, an airborne concentration of 850 DAC was obtained (assumes all resuspended activity is respirable). To determine whether this number was acceptable, a full facepiece respirator with a protection factor of 100 was considered to be utilized by any worker entering the area. This would result in an inhaled concentration of approximately 8.5 DAC. It was assumed that a decontamination or maintenance activity would not exceed four hours of duration in this airborne radioactivity concentration. This would result in an exposure to airborne radioactivity of 8.5 DAC × 4 hr, or 34 DAC-hrs.

As an additional data point, a calculation to determine the amount of time it takes to reach the activity level noted above was performed. Using the information from Section 6.6.2, the time (assuming a constant release rate) to deposit the activity was determined to be approximately 40 days. If the bounding analysis assumptions of Section 6.6.6 are used, the time to deposit the activity was determined to be approximately 4.3 days.

While the contamination levels calculated above (1×10^9 dpm/100cm^2) and resultant dose to workers could theoretically be managed, the levels are much higher than desirable. When considered with the frequency at which this level of contamination occurs (based on the calculation), the practicality of operating in this radiological environment is considered unacceptable. At such high levels of contamination, the protective actions required to be taken for the worker become so onerous that work productivity and efficiency would be significantly impacted. Additionally, the potential for the spread of contamination to areas of lesser contamination or clean areas is significantly increased. A spread of contamination from the transfer cells to the positioning cells would increase the chance to contaminate the transfer room and cross-contaminate the transporter. The impacts are a loss of productivity and an increase in the amount of areas and equipment requiring discontamination. Based on operational health physics experience, the maximum contamination levels that still allow a practical operational efficiency for routine operations is in the range of 100,000 to 500,000 dpm/100cm^2. With such high contamination levels, the presence of alpha emitting radionuclides from spent fuel oxidation will create additional hazards not considered in this study. Alpha contamination risks should be further evaluated (Section 6.9).

Based on the assumptions provided in Section 6.6.2, the contamination level from the releases of one fuel rod from oxidation results in a contamination level of approximately 2×10^7 dpm/100cm^2. This indicates that decontamination activities may have to be performed after handling any fuel with breached cladding.

6.8 REVIEW OF DOSE CONSEQUENCE AND CONTAMINATION ISSUES

Issues related to dose consequences and contamination are discussed below.

6.8.1 Evaluation of Issues

1. Perform fuel sipping and segregation of leakers prior to loading transportation casks.
2. Ensure fuel is consistently characterized prior to shipment.
   These two issues are outside the scope of this study and will be considered as potential future actions for consideration.
3. Review as low as is reasonably achievable (ALARA) considerations for worker doses. Issue 18 was considered to be part of this issue. It was noted that the normal dose calculations show a dose of 2,200 mrem per year to the workers while the ALARA goal is to minimize the number of individuals that have the potential of receiving more than 500 mrem per year. A detailed ALARA analysis will be performed as the design progresses.
4. The consequences due to releases from unbolted transportation casks prior to docking in the transfer cell and unsealed waste packages after loading in the transfer cell need to be considered. There is limited potential for release of contamination from the fuel transfer cell to the cask preparation area under normal operations because the unloading port of the transfer cell should be closed from the cask preparation area, except during cask unloading operations. During cask unloading operations, the loaded cask is first positioned under the hot cell unloading port and mated to the transfer cell port collar. The unloading port is opened only after the seal between the port collar and the cask surface is confirmed. In the cask preparation area, with the cask closure and the cask containment boundary O-ring seals firmly placed on the cask, there is a potential for release of material from the cask since the inner cask lid is unbolted in this area. There is also a similar concern with loaded, unsealed waste packages after removal from the transfer cell loading port. These operations will be evaluated in more detail for potential operator dose and ALARA considerations as the design progresses. Current Nuclear Engineering calculations consider any potential leakage from these operations to be part of the normal dose consequences.
5. Fraction of pinhole/hairline cracks in failed fuel. This issue is related to the apparent discrepancy between the EPRI (1997 [DIRS 100444]) estimate and Section 3.2.3 estimate of the amount of pinhole/hairline crack spent fuel. This issue is discussed in Section 3.2.3. As discussed in Section 5.1, given the uncertainty in the definition of

pinhole or hairline cracks, the effect of initial cladding defect size has been conservatively neglected in the oxidation rate determination.

6. Dose model for loss of HVAC condition. The concern is that the dose is higher when the release is to a small volume as opposed to a larger volume. This is not an issue for doses due to release through normal ventilation systems; however, when the HVAC systems are not operational, there may be leakage due to loss of confinement into occupied areas. The dose analysis shown in Section 6.6.4 assumed instantaneous release of the resuspended activity into the occupied areas. This is a conservative assumption that results in bounding dose estimates.

7. HEPA filter credit of 99 percent for each bank in series. HEPA filters are tested to provide a 99.97 percent removal efficiency; therefore, taking credit for only 99 percent per bank in series is conservative and appropriate.

8. Definition of damaged/failed fuel is needed. This issue is discussed in Section 3.2.3.

9. Material control and accountability. This issue is a project wide issue and requires further evaluation. This is an open item in PCSA calculations.

10. Criticality. As discussed in Section 6.6.7, criticality could be an issue if the material in twelve fuel rods is released into an optimum geometry with optimum moderator. The amount of material in twelve rods worth of oxidized fuel is approximately 25 kg. As shown in Sections 6.6.1 through 6.6.6, the maximum amount of airborne material released is 1.8 kg for the bounding case well below the amount required for criticality concerns. However, as discussed in Section 6.7, the total amount of fuel material that may be released during handling operations from fuel rods with breached cladding has not been determined.

11. RFs for the pinhole/hairline crack fuel. RFs for fuel fines, volatile radionuclides and gaseous radionuclides have been developed as reported in Section 6.2. Uncertainties in these values suggests that further work is required to define the RFs.

12. Particle size for filtered and unfiltered releases—important when using ICRP-68 (ICRP 1995 [DIRS 172721]) and ICRP-72 (ICRP 1996 [DIRS 152446]). ARF values for fuel fines formed during the oxidation of CSNF to U_3O_8 used in this study were derived from the oxidation of fuel pellets and uranium metal at temperatures greater than 500°C. The size of the fuel particles decreases as the temperature of oxidation decreases. Liu et al. (1992 [DIRS 172864]) determined the particle size distribution of oxidized fuel. The particle size distribution was found to be much coarser for higher oxidation temperatures than for lower oxidation temperatures. In the range of 300°C to 500°C the median particle sizes were 7.5 and 16 μm, respectively. Iwasaki et al. (1968 [DIRS 172518], Table 1) show a similar behavior, although the experiments were performed at much higher temperatures. Therefore, it is expected that releases of fuel fines will be in the respirable range. The ARF values for the oxidation of fuel at temperatures lower than 500°C should be reevaluated.

13. Filter loading—dose (part of ALARA) - filter replacement schedule. Filter loading dose calculations were performed and are reported in Section 6.6.3 for normal expected conditions. The results shown in Table 6-5 indicate that the dose rates after one year of accumulation are below the shielding design basis. The level of contamination on the filter may exceed the 10 CFR Part 61 [DIRS 103735] Class C waste limits by a factor of 10.

14. Dealing with the spread of contamination and potential contamination of a waste package, a facility, and equipment (i.e., remote maintenance, decontamination processes, and dealing with resulting radwaste). The concern is that the contamination from the oxidized fuel will spread from the transfer cell onto the surfaces of the transportation cask, waste package, and equipment. The level of airborne contamination was determined in Section 6.6.1.2 to be 166 grams for PWR fuel and 184 grams for BWR fuel for normal operations. The bounding estimate shown in Section 6.6.6 for a fully oxidized fuel rod was 1.7 kg for PWR fuel and 1.8 kg for BWR fuel. The potential for the spread of contamination and the potential impacts on design and operations need to be evaluated further. This should include the risks of alpha contamination.

15. The additional diffusional releases (due to fuel oxidation) following a Category 1 (fuel assembly drop) need to be modeled and accounted for in the dose consequence analysis. Current PCSA Category 1 event consequence analyses do not consider the source from oxidation following a drop or collision for longer than 20 hours. However, it does account for an additional diffusional release for a period of less than 20 hours (Lorenz et al. 1980 [DIRS 100990], High Burnup Fuel Tests 5 and 6) following a drop or collision. It is implicitly assumed that the dropped fuel assembly will be picked up and sent to wet remediation in less than 20 hours. It was noted that the time following the event until the fuel assemblies are recovered has not been determined. It is suggested that bounding assumptions such as full oxidation be considered in the analysis.

16. Thermal effects (due to the heat generation of fuel) on important to safety (ITS) equipment operating in the transfer cell should be evaluated. Heat generation from the fuel is an issue that will be addressed during the design process, along with radiation effects on equipment. The project equipment qualification process will address this issue.

17. Normal operations, off-normal operations, and accident conditions must be defined. Identification of equipment relied upon for each of the conditions of operation is needed. Section 7 includes a discussion on the fuel handling process. This identifies the normal operations and off-normal operations. This has been reviewed by PCSA to determine if the hazards analysis and categorizations cover the conditions described. This process description is used in this study for consequences. Accident conditions that are ITS are described in the *Preclosure Consequence Analysis for License Application* (BSC 2005 [DIRS 171607]). Adjustments will be made in PCSA as the design is refined.

18. Radioactive contamination may affect workers and equipment. Estimate the potential impact of a radioactive contamination environment on operation and maintenance of highly reliable equipment; alpha radioactive contamination and dose during normal operations must be considered from an ALARA perspective. The dose rates calculated for the normal operation accumulation of activity shown in Table 6-4 and the bounding case accumulation shown in Table 6-15 are above the recommended level of 100 mrem/hour. This result indicates that frequent decontamination operations need to be performed to ensure that the dose rates are below the recommend value. The design and operating alternatives are discussed in Section 7.

6.9 FINDINGS

Issues requiring further evaluation are discussed below:

1. RFs of gaseous fission products, volatile fission products, and fuel fines due to oxidation of damaged fuel. RFs from normal operations and off-normal conditions are needed. Recommended values are presented in Section 6.2 based on limited experimental data. This should also include an evaluation of the quantity of fuel material (oxidized or not) that may fall from the rod breached by fuel oxidation. The RFs due to oxidation need to be incorporated in *Commercial SNF Accident Release Fractions* (BSC 2004 [DIRS 172371]). The formation of RuO_4 in an ozone environment due to radiolysis should also be considered.

2. To support the ARFs, the amount of fuel that is oxidized during normal operations and off-normal or accident conditions needs to be determined. The oxidation is strongly dependent on the temperature that the fuel is subject to and the time that the fuel is exposed to air. This issue needs to be resolved since preliminary bounding calculations assuming fully oxidized damaged fuel result in unacceptable consequences to workers, public, and HEPA filter loading. To resolve this issue, further thermal analyses need to be performed along with detailed process duration estimates. The thermal analyses should include calculations of fuel in transportation casks, waste packages, and aging casks and during handling operations for normal and off-normal conditions.

3. Credible off-normal or accident conditions need to be defined. Section 6.3 defines several off-normal conditions that need to be reviewed by PCSA to determine if they should be considered and further defined (Category 1 or 2 or beyond category 2). Current PCSA consequence analyses for Category 1 and Category 2 events do not consider oxidation following a drop or collision. PCSA needs to review the event sequences and determine the consequences of oxidation. Such determination may require revision to several PCSA documents such as *Categorization of Event Sequences for License Application* (BSC 2004 [DIRS 171429]) and *Consequence Analysis* (BSC 2005 [DIRS 171607]).

4. Currently, a loss of ventilation with subsequent loss of confinement is not considered in the PCSA consequence analysis to be a credible event. A loss of confinement could result in leakage out of the transfer cell into normally occupied areas. Bounding

calculations using very conservative assumptions indicate that the worker dose approaches the limits and offsite public dose consequences exceed the limits for Category 1 events. This event needs further work to determine what category it falls into and what design features need to be implemented to mitigate or eliminate this event. Items such as allowable leakage rates from the primary confinement may need to be defined. This event may require revision to several PCSA documents such as the *Categorization of Event Sequences for License Application* (BSC 2004 [DIRS 171429]) and *Consequence Analysis* (BSC 2005 [DIRS 171607]).

5. The allowable filter loading on the HEPA needs to be defined. A bounding analysis indicates that the filter shine dose rates may exceed the shielding design criteria. In addition the loading on the HEPA filters need to be determined to support a review of disposal options for the used HEPA filters (low-level radioactive waste [LLW], greater than Class C, and HLW).

6. The current PCSA consequence analysis (BSC 2005 [DIRS 171607]) consider 154 fuel rods being subject to oxidation (Section 6.1.1). As discussed in Section 6.5, the number of fuel rods subject to oxidation is 671. The PCSA consequence analysis (BSC 2005 [DIRS 171607]) needs to be revised to reflect this recommendation.

7. The preliminary criticality analysis discussed in Section 6.6.7 needs to be finalized. Parameters such as specific geometries (e.g., the base of the staging racks, drains, sumps, or anywhere else that the oxidized fuel could accumulate); what range of chemical forms the oxidized fuel will take (e.g., schoepite: $UO_3{:}_2H_2O$); and the amount of moderation to consider should be analyzed. The specific number of rods worth of oxidized fuel necessary for criticality over the range of conditions should be determined. The criticality study should also consider the total amount of potentially loose fissile material; the potential for the oxidation of fuel assemblies with a greater than average number of failed rods as well as off-normal events where the quantity of unclad fuel available for oxidation increased.

8. Based on the discussion in Section 6.7, oxidation of bare fuel in air and the resultant release of radioactivity produces a radiological environment with a high risk of alpha contamination that will not practically support routine work activities in the fuel transfer cell. While work could be performed in this environment if mandatory, it is not considered ALARA and manned entry would require maximum respiratory and contamination personnel protective equipment for workers. The recommendation is to apply design features that would either limit the contamination initially to be released or provide for totally remote maintenance. Utilizing a design target range of 100,000 dpm/100cm^2 to 500,000 dpm/100cm^2 may provide an operationally manageable level of contamination from both a worker protection and operational efficiency perspective.

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7. PREVENTION, MITIGATION, AND RECOVERY ACTIONS

The scope of this section is threefold. Section 7.2 presents an analysis and evaluation of the fuel handling process steps in the surface facilities and the identification of potential hazards due to handling CSNF in air. Section 7.3 describes recommended design features and operating strategies to prevent, mitigate, and recover from the consequences of handling CSNF in air. Section 7.4 discusses the issues and recommendations for further steps in the design development.

7.1 DESCRIPTION OF PROCESS FOR HANDLING FUEL IN AIR

The following sections summarize the major process steps in the FHF and DTF.

7.1.1 Cask, CSNF, and Waste Package Handling Process Summary—FHF

Until the DTF becomes operational, transfers of uncanistered CSNF from transportation casks and loaded aging casks to waste packages for emplacement underground are carried out in the FHF. The facility is also used to transfer canistered and uncanistered CSNF to aging casks for aging. CSNF that is too hot to be directly emplaced underground is transferred from transportation casks to aging casks and moved to an aging pad until adequately cooled. The FHF does not have a separate staging area. However, within the FHF, a single aging cask may be used as a means to temporarily stage SNF during waste transfer operations.

7.1.1.1 Cask/Waste Package Preparation

A transportation cask, aging cask, or waste package arrives at the FHF on a railcar, truck, or aging cask transporter. The transportation cask, aging cask, or waste package is moved into the entrance vestibule and inspected. Impact limiters and personnel barriers are removed as necessary using the 30-ton auxiliary hook of the 200-ton entrance vestibule crane. A trolley is prepared for the transportation cask, aging cask, or waste package being received. The 200-ton entrance vestibule crane is used to move the transportation cask, aging cask, or waste package from the railcar or truck trailer to the trolley. Once the transportation cask, aging cask, or waste package is secured to the trolley, the trolley is moved from the entrance vestibule to the preparation room. In the preparation room, the loaded transportation cask or loaded aging cask interior gas is sampled and vented. The cask outer lid(s) is removed. The loaded cask is brought into the main transfer room with a lid lifting fixture attached to the inner lid. The loaded cask is moved from the incoming trolley and placed on one of three trolleys that support the fuel transfer bays. Inner lid bolts are removed and a docking ring is installed on top of the transportation cask, aging cask, or waste package. The transfer trolley is then moved into one of the three fuel transfer bays for subsequent docking to the fuel transfer room.

7.1.1.2 CSNF Transfer

For uncanistered CSNF transfer involving a loaded cask, waste transfer is performed in the fuel transfer room with the cask and waste package positioned in the fuel transfer bays. The CSNF transfer system is used for the actual transfer of BWR or PWR fuel assemblies using the spent fuel transfer machine and grapple.

7.1.1.3 Waste Package Closure

Waste package closure operations include welding waste package lids to the waste package, inerting the waste package, mitigating the stress associated with the welding, and performing nondestructive examination of the waste package closure welds. These operations are performed in the waste package positioning cell using remotely controlled equipment in a waste package closure cell that is located directly above the waste package positioning cell.

7.1.1.4 Cask/Aging Cask Restoration

The restoration of a loaded aging cask and empty transportation cask is performed in the main transfer room, the preparation room, and the entrance vestibule. The process begins after the cask lid is inserted and the cask is undocked from the fuel transfer bay. The cask is moved into the main transfer room. The cask lid is secured, and the cask is transferred to the import/export trolley. The trolley and cask then move into the preparation room. The remaining lid bolts are inserted and torqued to prescribed limits. The interior of the aging cask is inerted.

7.1.1.5 Remediation

During normal handling operations, the FHF has limited capacity to perform in-process remediation on casks, canisters, CSNF assemblies, and waste packages. Remediation is generally limited to those operations required to allow the waste form to be safely handled and placed into a waste package. In the event an off-normal item cannot be remediated in the FHF, the item is sent to the DTF for remediation. Remediation of CSNF assemblies is carried out in the fuel transfer room.

7.1.1.6 Disposal of Damaged Fuel Cans

The 12-PWR waste package has been sized to accept single-assembly-sized damaged fuel cans that fit within the contractual envelope (viz., 9" by 9" by 14' 10" (10 CFR 961.11 [DIRS 118049], Appendix B, Table 1) that has been modified to accommodate the length of such a canister (DOE 2002 [DIRS 158398], Section 11.3). The internal cavity of that waste package configuration is 201.6 in. (BSC 2003 [DIRS 164976] and BSC 2004 [DIRS 169766], Table 24, p. 47). It should be noted that the most recent compilation of information on existing cans found a maximum length of only 189 in. (BSC 2003 [DIRS 165990], Appendix G, p. G-3). Note that such cans from BWRs would probably also be inserted in this waste package.

Since these cans are designed to be placed in spent fuel pool storage racks at the reactor sites, the handling features of the cans should be comparable to those on intact fuel assemblies so that traditional handling tools may be used; however, damaged fuel cans can have different handling features than intact fuel assemblies. In addition, many damaged fuel cans are not able to be handled as a unit with the contained assembly. However, the Integrated Interface Control Document (DOE 2002 [DIRS 158398], p. 21) seems to admit the possibility of removable handling features. It is noted in the 2002 Waste Stream Projections Report (BSC 2003 [DIRS 165990], Appendix G, p. G-2) that specific can-handling data are not currently available in the CRWMS M&O database (BSC 2003 [DIRS 165990], Appendix G, p. G-4).

There are some cans that do not conform to the standard dimensions (BSC 2003 [DIRS 165990], Table G-1, p. G-4) and these must be addressed on a case-by-case basis. For those that fall within the envelope, they probably can be directly inserted into the 12-PWR waste package. For those that are larger, the cans may require the opening of the damaged fuel can and removal of the fuel assembly or rods. Alternatively, it may be possible to adjust the basket design to accommodate the larger cans, but the impact must be assessed for postclosure performance as well as from operational considerations.

7.1.2 Cask, CSNF, and Waste Package Handling Process Summary—DTF

The CSNF processing functions conducted in the DTF are similar to those performed in the FHF, but are more extensive in scope. The major CSNF processing functions in the DTF are summarized in the following sections.

7.1.2.1 Transportation Cask and Aging Cask Preparation

A transportation cask containing CSNF arrives from the transportation cask buffer area on a site rail transfer cart (SRTC). The cask is moved through the cask entrance vestibule and into the cask receipt area. Impact limiters and personnel barriers are removed as necessary. Using the 200-ton cask handling crane, the cask is upended from the horizontal to vertical position, lifted off the SRTC, and moved through a door to a trolley located in the trolley transfer room. The crane is used to place the cask on the trolley where it is secured in a vertical position. Shield doors in the facility are opened and closed as the trolley moves through the facility. The trolley and cask are moved to the cask turntable room. The turntable aligns the trolley to the rails leading to the cask preparation room where the cask is manually prepared for unloading. Cask preparation consists of removing the cask lid bolts, sampling the cask internal atmosphere, and installing the docking rings. When the cask is ready for docking and unloading, the trolley is moved back to the turntable room and the turntable aligns to the rails leading to the cask docking room. The trolley is moved to a turntable in the cask docking room that aligns to the rails leading to the desired cask docking port. The trolley positions the cask under the docking port and confinement is established between the cask and the waste transfer cell. Aging casks to be loaded or unloaded follow the same processing steps as described above for transportation casks.

If a transportation cask or aging cask contains an aging canister or a dual-purpose canister (DPC) with CSNF that is to be loaded into a waste package, the transportation cask or aging cask is processed in a different location based on the need to open the DPC. After being placed on a trolley, the cask containing the DPC is moved through the cask preparation room to the DPC preparation room. The DPC transportation cask or aging cask is prepared for docking, which includes removing the cask lid bolts, sampling and venting the cask atmosphere, installing a canister lifting fixture, and installing docking rings. The trolley and cask then move to the DPC docking room. Confinement is established between the DPC transportation cask or aging cask and the DPC cutting/waste package dry remediation cell. The DPC is transferred by a crane to a DPC cutting station in preparation for unloading the DPC.

7.1.2.2 Waste Package Preparation

When a waste package is ready for loading, it is moved into the DTF on an SRTC. There are two waste package processing lines. The waste package and SRTC are moved through the waste package entrance vestibule and into the waste package receipt area. Using the 50-ton waste package handling crane, the waste package is lifted off of the SRTC and moved through a door to a trolley located in the waste package trolley transfer room. The crane places the waste package on the preconfigured trolley where it is secured in place. The trolley and waste package are moved to the waste package preparation room where the waste package docking ring is installed. When the waste package is ready for docking and loading, the trolley and waste package are moved to the docking cell. The trolley positions the waste package under the docking port, and confinement is established between the waste package and the waste transfer cell. After the waste package has been loaded, the inner waste package lid is installed and the waste package is moved to a positioning cell for closure operations.

7.1.2.3 CSNF Transfer

CSNF transfer operations are conducted in heavily shielded, reinforced concrete transfer cells. CSNF is transferred between transportation casks, aging casks, and waste packages in the waste transfer cell. CSNF arriving in DPCs that have been cut open are also unloaded and transferred into waste packages in the waste transfer cell. After a waste package has been loaded and the inner lid installed, the waste package is undocked from the waste transfer cell. The trolley and waste package are moved from the cask docking cell to the docking ring removal cell. The docking rings are removed and the trolley and waste package are then moved to the waste package handling and staging cell. The waste package is transferred by crane to one of the trolleys leading to a waste package positioning cell.

7.1.2.4 Waste Package Closure

When a waste package is ready to be sealed, it is moved by trolley into a waste package positioning cell. The waste package trolley positions the waste package under the waste package closure cell where the closure system welds the inner lid and installs and welds the remaining two waste package lids. The waste package is also inerted and the external lid weld is stress relieved. Operators control the waste package closure process remotely using welding, inerting, nondestructive examination, and stress mitigation equipment designed for remote maintenance, repair, and replacement.

7.1.2.5 Aging Cask Restoration

After an aging cask is loaded, the cask lid is reinstalled, the waste transfer cell port plug is installed, and the aging cask is undocked from the waste transfer cell docking port.

The trolley and cask are moved to the cask docking room turntable, which rotates to send the trolley to the turntable in the cask turntable room, which rotates to send the trolley into the cask restoration room.

Cask restoration is performed manually via access from above the cask on a platform in the cask restoration room. Specific activities include removing docking rings, which are sent to the docking rings decontamination area before being returned to their dedicated storage area; installing and bolting lids; testing for leaks; and inerting the cask and conducting radiological surveys to ensure the cask can be released.

7.1.2.6 Remediation

Three types of remediation operations are performed in the DTF: dry remediation, wet remediation, and waste package remediation. Remediation actions performed on transportation casks can also be performed on aging casks.

- Dry Remediation—Dry remediation can be used to resolve a CSNF problem that may have occurred in a transportation cask. When an off-normal CSNF condition that requires remediation is detected, the transportation cask is closed and moved to the cask docking/dry remediation room for resolution. Dry remediation areas are located away from normal process areas to prevent interference with required facility throughput. Major equipment used in dry remediation includes a 15-ton service crane, turntable, trolley, and master slave manipulator.

- Wet Remediation—A transportation cask with failed fuel may be placed in the remediation pool located in the wet remediation area and the fuel transferred from the transportation cask into staging racks pending further action. When corrective action is identified and approved, the remediation of the failed fuel may be performed in the remediation pool.

- Waste Package Remediation—The main function of the waste package remediation subsystem is to open a defective waste package. Once a defective waste package is opened, it is moved under the unloading port of the waste transfer cell for CSNF transfer. Waste package remediation typically is performed inside the DPC cutting/waste package dry remediation cell.

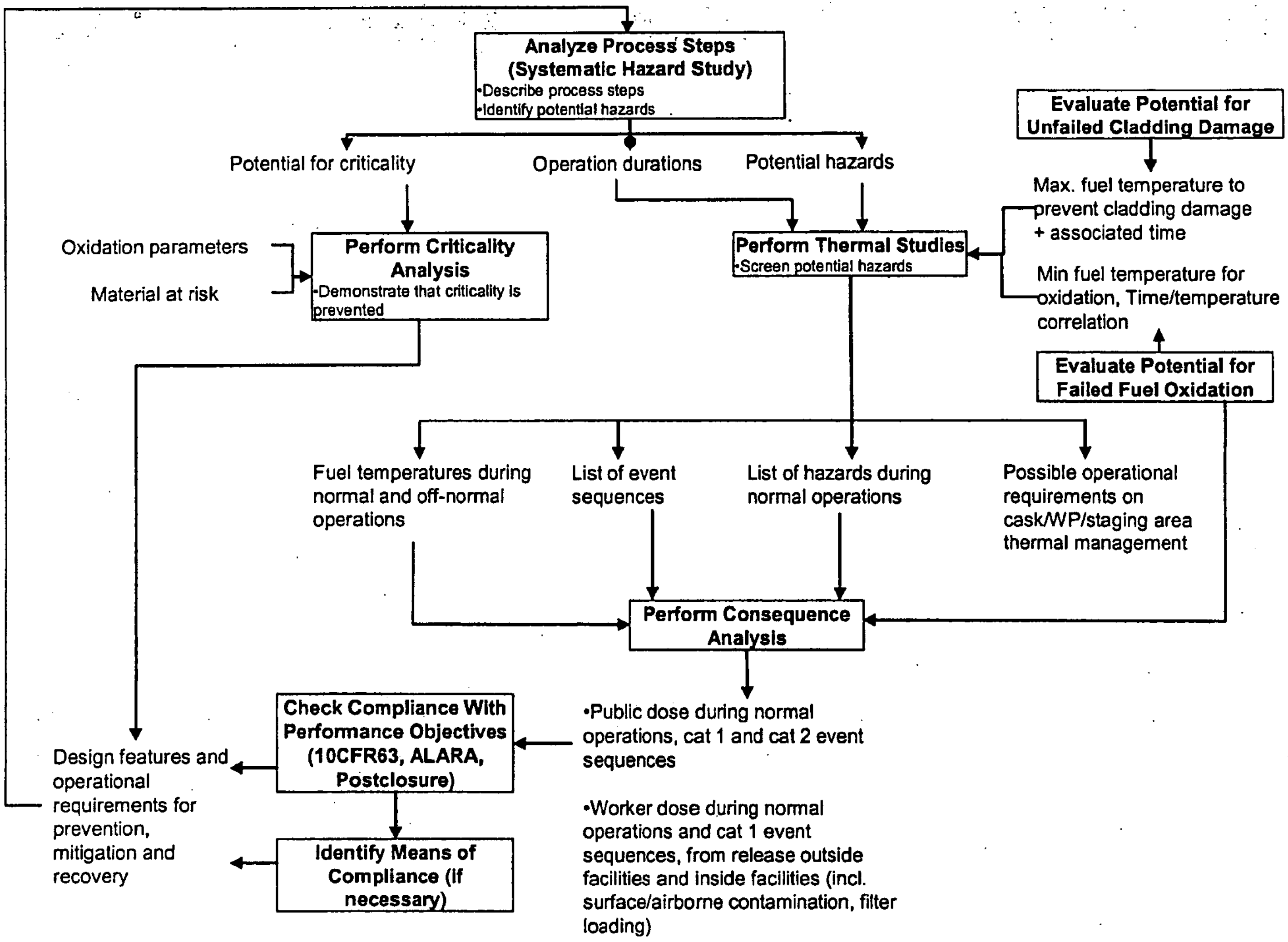
7.2 PROCESS ANALYSIS

7.2.1 Purpose

A systematic analysis of process steps where CSNF is handled in air has been performed to define the appropriate design features that prevent, mitigate, or recover from related hazards. This section covers part of an overall process that is summarized in Figure 7-1 and Appendix D.

The purpose of this section is to summarize the potential hazards associated with handling fuel in air identified in Appendix D. Contamination requirements for the current design are summarized in Appendix B. As part of an iterative process, the study should be updated when additional results from thermal evaluations and consequence calculations become available. Any process modifications or implementation of design features may modify, add, or remove hazards.

The study was performed only for the FHF. Subsequently, the study's conclusions and recommendations have been discussed and developed for operations performed in the DTF.



NOTE: WP = waste package; Cat 1 = Category 1.

Figure 7-1. Fuel in Air Study Process

7.2.2 Assumptions and Basis

CSNF assemblies (some of which may contain failed fuel rods) are assumed to be handled in the DTF and FHF surface unloading areas as part of normal dry transfer operations.

As discussed in Section 3.2.3, these surface transfer facilities are expected to receive and process failed fuel at an average of 4.4 percent of the total CSNF fuel assembly receipts with an average of 2.2 failed rods in each assembly. Failed fuel is defined as fuel suffering defects equal to or greater than a pinhole or hairline crack.

As discussed in Section 5, when in an air environment, UO_2 fuel with failed cladding oxidizes to U_3O_8. During the oxidation process, the oxidized fuel swells and may cause failure of the cladding. The consequences are releases of nuclear material during CSNF handling operations. Consequences of this release of material are discussed in Section 6.

The process for handling failed fuel in damaged fuel cans is not yet detailed in current design documents, and the related hazards have not yet been evaluated. At this stage of the analysis, processing damaged fuel cans will be the same as processing intact fuel. However, handling CSNF assemblies in cans has a direct impact on fuel temperature and should be specifically evaluated in thermal calculations, which may have an impact on consequences and any resulting limiting conditions of operation (time versus temperature) for CSNF handling.

7.2.3 Potential Hazards in FHF

Potential hazards identified in the FHF study are presented in Appendix D by briefly describing each potential hazard, its initiating event (if applicable), and its potential consequences, followed by criteria or specific studies that are necessary to determine whether the hazard will occur. Some of the hazards have been identified in current PCSA documents and should be reviewed according to the issues on handling fuel in air discussed in Sections 5 and 6. Additional hazards are also presented for further analysis.

7.2.4 Summary of Potential Hazards in the DTF

In the DTF for normal operations, CSNF is handled in air during operational processes similar to those in the FHF. They include cask preparation, SNF transfer, and waste package closure operations. A systematic hazard study for the DTF has yet to be performed. However, the types of potential hazards are expected to be similar between the DTF and FHF.

CSNF is also handled in air during processes specific to the DTF. They include DPC cutting and handling, waste package remediation, cask dry remediation, SNF staging, and parts of cask wet remediation operations. These processes create specific types of hazards that will be identified and reviewed as the design progresses.

7.3 POTENTIAL DESIGN FEATURES

The determination of potential design features to address handling CSNF in air is based on the current information about thermal issues and event consequences. Updated thermal and consequence calculations may impact this list. The feasibility of implementing the potential design features in the current FHF or DTF design has not been evaluated.

7.3.1 FHF

7.3.1.1 Prevention/Mitigation Design Features

The potential design features for the prevention or mitigation of hazards from handling CSNF in air can be grouped in the following seven types.

Filtration—The purpose is to minimize the release of contamination outside the facilities:

- Cask sampling system connected to HEPA filters (building ventilation or process ventilation system). This is the current design.
- Cask venting system connected to HEPA filters (building ventilation or process ventilation system). This is the current design.
- Waste package purging system connected to HEPA filters (building ventilation or process ventilation system). This is the current design.

Cooling Systems—The purpose is to minimize the potential for unfailed fuel cladding degradation and fuel oxidation and to control cask and waste package temperature:

- Cask cooling system on the cask exterior at various process stations (preparation area, docking, and other areas where required)
- Waste package cooling system on the waste package exterior at various process stations (docking, closure, and other areas where required)
- Fuel transfer room HVAC to maintain fuel temperature below limits
- Cooling of the internal cavity of cask or waste package with a cold inert gas.

Fuel Inerting—The purpose is to prevent the oxidation of failed fuel in a cask, aging cask, and waste package and by limiting the fuel exposure to air to only the transfer between cask, aging cask, or waste package. The use of a heavy gas, such as Argon, should be considered to minimize dispersion and the mixing of the inert gas with the non-inerted atmosphere surrounding the container:

- Vent and refill cask with an inert gas during preparation process.
- Maintain cask inerting throughout the unloading process.
- Inert docked waste package and maintain inerting until the waste package is inerted with helium during the closure process.

- Inert docked aging cask and maintain inerting until the cask is sealed.
- Transfer SNF in an inerted tube or inert the entire fuel transfer room to prevent oxidation during SNF transfer between cask, aging cask, or waste package.

Confinement—The purpose is to prevent the spread of contamination inside the facilities:

- Confine cask sampling/venting operations (minimize leakage from tools and components).
- Confine waste package venting operations (minimize leakage from tools and components).
- Maintain a partial vacuum in the transportation cask cavity during preparation operations and maintain a differential pressure between cask cavity and occupied areas.
- Maintain a differential pressure between waste package cavity and occupied areas (initial waste package cavity is at the fuel transfer room pressure).
- Reduce air change rates in areas affected by fuel oxidation.
- Maintain the confinement of the fuel transfer room in case of a loss of ventilation by minimizing its leakage rate and/or relying on natural convection to preserve a negative pressure.
- Maintain static confinement of unsealed waste package (e.g., filter release from an unsealed closed waste package to the rooms where it is handled) and unbolted aging cask.
- Contain release from oxidation of a CSNF assembly during its transfer in the fuel transfer room.

Ventilation System—The purpose is to maintain the air conditioning function of the HVAC in order to keep room and cask/fuel temperatures within acceptable limits:

- Backup HVAC for cask preparation area.
- Backup HVAC for main transfer room.
- Backup HVAC for transfer bay.
- Backup HVAC for fuel transfer room. The fuel transfer room ventilation is currently credited in the consequence evaluation of CSNF assembly collision/drop events. Additional reliability requirements may be developed to address thermal issues.

- Backup HVAC for waste package positioning cell and closure cell.
- Backup system for specific cask or waste package cooling systems, if implemented.
- Backup power supply for ventilation equipment.

Mechanical Equipment—The purpose is to minimize the probability of failure causing damage to CSNF or increasing the duration of handling CSNF in air:

- Main transfer room crane minimum reliability for drop events. This is the current design.
- Crane and trolley redundant design features to limit immobilization time. This is the current design.
- Fuel transfer machine minimum reliability for drop events and other failures. This is the current design for drop events.
- Ensure that mechanical equipment can be fed by alternative power supply. This is the current design for all mechanical equipment used for critical lifts.
- Lock waste package lid to secure its content prior to lifting an unsealed waste package. This is the current design.

SNF/HLW Transfer System—The purpose is to minimize the duration of handling CSNF in air:

- Start operations only if downstream process is available (to limit potential staging/waiting time of an unbolted cask or unsealed waste package with fuel in air). This is the current design approach.
- Examine fuel prior to de-inerting casks in order to segregate failed fuel and apply a specific unloading process to minimize or prevent its handling in air.
- Reduce exposure time in air of known failed fuel by transferring it directly to a waste package (no aging cask).
- Stage known failed fuel in an inerted location in case of process delays or unavailability of handling equipment.

7.3.1.2 Recovery Design Features

The possible recovery design features for handling CSNF in air during off-normal events can be grouped to address the following issues.

SNF/HLW Transfer and Cask Preparation Systems—The purpose is to be able to complete an operation when normal operations design features have failed:

- Recovery devices on crane/trolley to move cask back to the preparation area where it can be cooled down and/or inerted
- Recovery devices on transfer trolley to move cask back to the main transfer room where it can be cooled down and/or inerted
- Recovery devices on crane/trolley to move waste package back to the main transfer room or fuel transfer bays where remote operations can put the waste package in a safe configuration
- Recovery devices on crane to put cask or waste package lid/port cover back in place
- Recovery devices on fuel transfer machine to complete movement with SNF assembly.

SNF/HLW Transfer and Cask Preparation Systems—The purpose is to be able to inert a cask or waste package to recover from an off-normal SNF in air handling events:

- Specific mobile system with appropriate tools to inert a cask and an aging cask in the main transfer room
- Specific system to inert a cask and an aging cask while still docked to the fuel transfer room
- Specific remote system with appropriate tools to inert a waste package in the main transfer room
- Specific remote system to inert a waste package while still docked to the fuel transfer room.

SNF/HLW Transfer and Cask Preparation Systems—The purpose is to recover from contamination release:

- System to adjust an unbolted cask pressure while cask or aging cask is in the main transfer room or docked
- System to adjust the waste package pressure while waste package is still docked or when waste package is in the main transfer room (remotely)
- Remotely maintained handling equipment within the fuel transfer room
- Fuel transfer room or other room decontamination capability: remote or hands-on decontamination systems, adequate cell surface (e.g., stainless steel)
- Decontamination capability of cask interior (including retrieval and disposal of fuel material)

- Decontamination capability of cask or waste package exterior
- Shielding of HEPA filters and remote maintenance capability for HEPA filters.

7.3.2 DTF

The potential design features presented for the FHF are also applicable to address similar processes and related hazards in the DTF. Those processes include cask preparation, SNF transfer, and waste package closure operations. For processes specific to the DTF, such as DPC cutting and handling, waste package remediation, cask dry remediation, SNF staging, and parts of cask wet remediation operations, specific design features may need to be developed.

The DTF provides an enhanced operational flexibility due to a larger number of CSNF transfer lines and waste package closure cells, a physical separation between operations on casks and empty waste packages, and loaded waste packages. The potential release of contamination within the DTF would also be more controlled through its confinement zoning (airlocks).

7.4 REVIEW OF THE IDENTIFIED ISSUES

7.4.1 Purpose

The purpose of this section is to review issues identified and provide recommendations.

7.4.2 DTF Systematic Hazard Analysis

CSNF is handled in air in the DTF during processes specific to this facility that have not been evaluated in the FHF systematic hazard study. These processes such as remediation and DPC cutting create specific types of hazards that will be identified through a hazard analysis similar to the one presented in Section 7.3.1.

A systematic hazard analysis for the DTF for fuel in air issues will be performed as part of the established hazards analysis design process.

7.4.3 Damaged Fuel Cans

Damaged fuel cans will be used to package known failed fuel and must be evaluated to considered:

- The thermal analysis results and maximum temperatures of the fuel rods
- The consequences of fuel oxidation (peak dose when a damaged fuel can with a high number of failed rods [e.g., 50 rods] is received)
- Criticality because of the number of potentially failed fuel rods in a damaged fuel can.

It is recommended to update the systematic hazard analysis and the thermal and consequences calculations to take into account the damaged fuel can.

7.4.4 Thermal Calculations and Analyses

Further analyses are required to complete the evaluation of the potential hazards identified above in Section 7.3. Additional inputs should be provided by further thermal calculations to obtain temperature/time parameters for handling fuel in air.

The following evaluations are recommended:

- A comprehensive thermal analysis of the process from cask receipt operations to waste package closure to provide an evaluation of fuel temperatures, cask temperatures (neutron shield, seals), and room/cell temperatures at each process step, during normal and off-normal operations (Section 7.5).
- This comprehensive analysis should include the case of fuel assemblies or fuel rods received in damaged fuel cans.
- An analysis to provide an evaluation of relative pressures and air leakages between rooms in case of a loss of the HVAC system (Section 7.5).
- An analysis of the cask and waste package for the potential release of contamination when lids are in place, but not sealed and during process steps (e.g., cask preparation, fuel transfer, and waste package closure) for normal operations and loss of HVAC (Section 7.5).
- It is recommended to perform further thermal calculations and analyses to take into account the different configurations of the cask, aging cask, waste package, and CSNF during all the process steps in the DTF and FHF (Section 7.5).

7.4.5 Consequence Analyses

The following consequences should be evaluated to identify prevention/mitigation/recovery design features:

- Release to the environment: public dose, onsite worker dose
- Release to a room or cell: facility worker dose, room/cell surface and airborne contamination, room or cell ventilation filter loading (e.g., secondary confinement system HEPA filter loading for a release in the main transfer room of the FHF)
- Validation of HEPA filter effectiveness to mitigate releases (assess potential presence of RuO_4 gas and RuO_2 solid)
- Evaluation of the quantity, form, and physical properties of fuel material (U_3O_8 powder, pellets) that could potentially escape from damage fuel assemblies and accumulate in casks, waste packages, staging racks, and transfer cell.

7.4.6 Contamination Source Term

If no prevention plans and specific features to avoid the oxidation of failed CSNF in air are implemented for the transfer to surface facilities, based on the Section 6 evaluations, there will be a high potential for the spread of nuclear materials in the remote process rooms, such as the fuel transfer room (Room 2001), in the FHF, and in rooms where hands-on operations are required, such as the main transfer room (Room 1003) in the FHF.

The current design philosophy and concepts used for the FHF and DTF are based on proven designs for intact fuel assemblies where the only source of contamination is from corrosion products (crud). If fuel were allowed to oxidize as part of normal operations, the contamination source term would include alpha emitters, volatile products, and gases. Additional design concepts should be considered to control the spread of contamination, including static containment (tight enclosures, specific seals).

It is recommended to perform evaluations of contamination control, including the HVAC, waste management, and airborne monitoring systems (Section 7.5).

7.4.7 HVAC HEPA Filters

The releases from CSNF oxidation events and the buildup of contamination in the fuel transfer cells will have an impact on the design of the HVAC systems to account for alpha emitters and gases (e.g., washing columns, scrubbers, and metallic prefilters may be required to protect HEPA filters and to bring releases to ALARA levels, even if releases are compliant with 10 CFR Part 63 [DIRS 156605] acceptance criteria). HEPA filters' loading and change out frequency would be increased. This will result in additional radioactive waste generation, additional personnel exposure, and reduction in throughput of the facilities.

As it is documented in *Fuel Handling Facility Description Document* (BSC 2004 [DIRS 169630]) and *Fuel Handling Facility General Arrangement Ground Floor Plan* (BSC 2004 [DIRS 171716]), the remote HEPA filter room (Room 1007) is adjacent to the fuel transfer bays. Trolley tracks connect Room 1005 with Room 1007 through a shield door between the two rooms. This opening is used to transfer the used contaminated HEPA filters into the shielded HEPA filter boxes that are placed on a trolley. Room 1005 is the transfer bay that is used for the cask. Given the requirements established on the cask level of contamination when leaving the repository (non-exclusive use thresholds), and in order to avoid additional extensive decontamination operations and worker exposure, this area should stay as clean as possible. This study did not address the potential risk of contamination of the trolley, transfer bay, and adjacent rooms due to contamination leakages from the HEPA filter and HEPA filters' room that are contained within the primary confinement zone.

As the oxidation of the failed CSNF may have an effect on the HEPA filters change out, it is recommended to analyze the risk of contamination due to the HEPA filters change out operation and to perform an evaluation of its consequences, including personnel exposure, waste generation, and impact on facility throughput (Section 7.5).

7.4.8 LLW/HLW Management Systems

The current design, based on *Project Functional and Operational Requirements* (Curry 2004 [DIRS 170557], 1.4.4.2.6.2-1) and upper level requirements in the *Project Requirements Document* (Canori and Leitner 2003 [DIRS 166275], PRD 014/P-001, 014/T-016), does not include provisions for managing HLW generated during operations at the repository. Only LLW and Resource Conservation and Recovery Act of 1976 [DIRS 103936] waste are addressed.

Decontamination operations of the transfer cell and casks may generate HLW. Fuel fines and particulates will be collected during decontamination operations. The present study does not address how those wastes will be managed.

It is recommended that evaluations be performed on the management of the HLW that may be generated as the result of fuel oxidation (Section 7.5).

7.4.9 Material Control and Accountability

As a result of the oxidation of failed CSNF during the transfer operations from the cask to the waste package, nuclear material is likely to escape from the fuel rod and will not follow the CSNF from the cask or aging cask to the waste package. Those nuclear materials will be captured by both the HVAC filters and decontamination system that will be used to decontaminate and remove the nuclear material from the contaminated casks, docking rings, and process areas. This type of loose nuclear material needs to be accounted for in the material control and accounting program.

It is recommended that the material control and accounting plan consider the possibility of fuel oxidation and evaluate the design to determine if additional equipment and processes will be necessary to characterize and account for the material that may escape from the assemblies due to oxidation (Section 7.5).

7.4.10 Fire Protection System

The transfer cell in both the DTF and FHF are equipped with a fire suppression system. The review of this system is not a part of this study.

It is recommended that the fire hazard analysis be evaluated to consider the presence of oxidized fuel and perform a review of the fire suppression system to account for fuel oxidation consequences such as contamination buildup in the transfer cell (Section 7.5).

7.4.11 Throughput and Operating Cost

Throughputs, ease of operations, and life cycle costs impacted by the issues discussed above, as well as by operating procedures to be identified as part of followup activities, are not addressed in this study.

A single decontamination operation may require a shutdown of the FHF for a week or more. It will have an impact on both the throughput and operating cost of the facility.

Casks and waste packages have very low contamination requirements. Appendix B lists contamination requirements for the current design. For the casks, those requirements are defined in 49 CFR 173.443 [DIRS 165690] and *Project Functional and Operational Requirements* (Curry 2004 [DIRS 170557]). They are 0.4 Bq/cm^2 of beta and gamma emitters and 0.04 Bq/cm^2 of alpha emitters. For the waste package, the derived contamination levels are documented in *Recommended Surface Contamination Levels for Waste Packages Prior to Placement in the Repository* (Edwards and Yuan 2003 [DIRS 164177], Section 6.1). They are 0.042 Bq/cm^2 for alpha emitting nuclides and 12 Bq/cm^2 for beta and gamma emitting nuclides. In order to meet these requirements and avoid excessive decontamination operations of the transportation convenience, cask, and waste package, the process cells should stay as clean as possible.

It is recommended that design features be implemented to minimize contamination of the process cells and cross contamination of casks and waste packages. The impact of decontamination operations on the throughput and operating cost should also be evaluated (Section 7.5).

7.4.12 Monitoring CSNF Time and Temperature

Issues related to monitoring fuel handling operations are as follows:

- It will be difficult to monitor and track the temperature of each assembly to ensure that the temperature remains below the allowable limit.
- It will be difficult to accurately keep track of how long each of the 220,000 assemblies is exposed to air.

It is recommended that specific studies and thermal calculations be performed, based on bounding fuel characteristics, to define realistic operations requirements and avoid having to track and measure the temperature for each assembly (Section 7.5).

7.4.13 HVAC Design and Contamination Control

The HVAC system design includes areas with relatively high air velocities. With a high potential for contamination, controlling the spread of contamination will be difficult. It is recommended that the HVAC system be evaluated for minimizing the spread of contamination (Section 7.5).

7.5 FINDINGS

Further design analyses are necessary to validate and better define recovery means to prevent or mitigate fuel oxidation in the surface facilities. These activities are detailed in Section 7.4 and include:

- Thermal analyses and HVAC design (cooling systems) to reduce fuel temperature levels as low as possible to prevent or slow the oxidation process.

- Evaluate the concepts for inerting the fuel assemblies when still in the cask, staging area, and waste package before closure. The objective is to limit the duration when the fuel is in air.
- Specific recovery systems will need to be developed to refill cask, staging, and waste package cavities with inert gas as needed or to cope with fuel handling mechanical equipment failure when fuel is in air.
- Review confinement issues for each step of the process considering the presence of fuel fines.
- Specific operating procedures will need to be developed to limit the duration where fuel is in air.

The following is the summary of the key issues identified in Section 7:

- Thermal management of fuel temperatures and monitoring of temperatures.
- Criticality assessment should consider failed fuel rods received in DFCs and high-level liquid waste resulting from cask rinsing and cell decontamination.
- The overall contamination control design philosophy and HVAC design should be reviewed to determine the consequences of handling oxidized fuel.
- The Material Control and Accounting plan should be reviewed to consider how oxidized fuel and other fuel not contained in fuel cladding will be accounted for.
- The fire analysis should be revised to account for accumulation of fuel material in the transfer cells.
- Current design does not yet include the management of all radioactive waste that will be generated during maintenance, filter change out and decontamination activities.

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8. CONCLUSIONS AND RECOMMENDATIONS

The purpose of this study is to evaluate handling CSNF in air and identify issues, consequences, and operational considerations during processing and packaging activities in the Yucca Mountain surface facilities. This section summarizes the significant conclusions and recommendations presented in Sections 3 through 7. These conclusions and recommendations provide a basis for decisions, ongoing work, future evaluations, and a path forward to further support design development and the license application.

8.1 CONCLUSIONS

Key conclusions are listed below with references to the sections of the study where the issue is discussed in more detail.

Design Bases for Handling CSNF in Air—Section 3 of the study addresses design bases for handling CSNF in air. This includes handling unfailed CSNF, handling failed CSNF, and the expected quantities of failed CSNF and types of damage.

- The surface facilities should use a 400°C normal maximum operating temperature for CSNF cladding. This limit is also discussed in Section 4 of the study. The likely cladding failure or degradation modes do not appear to cause failure of unfailed fuel at this temperature. This conclusion is similar to the COGEMA experience at La Hague and is consistent with guidance provided in NRC ISG 11 (NRC 2003 [DIRS 170332]).

- The surface facilities should use a 460°C proposed off-normal maximum temperature for CSNF cladding. This limit is also discussed in Section 4 of the study. The off-normal operations temperature of 570°C used in NRC ISG 11 (NRC 2003 [DIRS 170332]) may result in cladding oxidation and cladding creep issues when CSNF is handled in air.

- For failed CSNF, the surface facilities should use the same temperatures for normal and off-normal operations as for unfailed CSNF. This subject is also discussed in Sections 5 and 6. Further evaluations may determine that additional temperature limits for handling failed fuel are necessary.

- Based on fuel assembly characterization at receipt, failed fuel quantities and types of damage are described in Table 3-4. Approximately 3.3 to 4.4 percent of fuel assemblies received are expected to have an average of 2.2 failed fuel rods per assembly. It is estimated that 90 percent of the failed fuel will be known and identified prior to shipment to the repository. The remaining 10 percent failed fuel is expected to be pinhole leaks and hairline cracks and will not be identified prior to shipment.

CSNF Oxidation—CSNF is projected to arrive at the repository at a rate of about 9,000 assemblies per year. 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 may allow air to come in contact with the uranium dioxide fuel, UO_2). The current surface waste handling facilities design was developed to handle CSNF in air. During surface facility handling operations, 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 during normal waste handling operations. At temperatures above 250° C, the UO_2 fuel exposed to air will begin to oxidize. The rate of oxidation depends on time and temperature. When in an air environment, UO_2 fuel with damaged cladding has been observed to oxidize and generate U_3O_8 powder. During the oxidation process, the oxidized fuel swells and may cause further failure of the CSNF cladding and release of the powder.

As discussed in Section 5, there are numerous technical studies on the mechanisms and behavior of fuel oxidation. These studies provide information that can be used to evaluate the consequences of fuel oxidation during handling operations. However there are several key areas, discussed below, where limited information is available.

- 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 have been estimated to have a secondary effect on fuel oxidation. These estimates need to be confirmed.
- Oxidation rate—The methodology for calculating the rate of oxidation needs to be further evaluated and documented. There are several methods for calculating oxidation rates, Section 5.6.2. An updated version of *Clad Degradation – Dry Unzipping* (CRWMS M&O 2000 [DIRS 149230]) would provide a referenced basis for dry oxidation evaluations.
- Release fractions during oxidation—Release fractions of gaseous fission products, volatile fission products, and oxidized fuel fines need to be determined. The study uses release fractions published in technical reports contained in the references to this study. However these release fractions may not be conservative and may underestimate the amount of radioactive materials released during fuel handling operations (Section 6).

Because of the uncertainties in evaluating CSNF oxidation, the results in this study should be considered preliminary and should not be represented as conservative or bounding.

Contamination and Dose Rates—The contamination levels and dose rates calculated in this study are dependent on the mass of fuel oxidized and the percent released from the cladding. Based on the evaluation in Section 6, a release of approximately 0.2 to 2 kg of fuel fines and volatile radionuclides (Table 6-8 and Table 6-13) may occur during annual operation in the FHF transfer cell. As discussed in the CSNF Oxidation conclusion above, there is uncertainty in the rate of oxidation and release fractions, so this estimate may not be conservative or bounding. Since the amount of material that may be released from breached cladding is uncertain, the level of contamination that would result in a dose rate of 100 mrem/hr at one foot above the floor was calculated. This amount of contamination is equivalent to a contamination level of 1.5×10^9 dpm/100 cm^2, and it is estimated that it will take approximately 4 to 40 days of operation to deposit this level of activity dependent on the assumed release fraction (Section 6.7). While these contamination levels and dose rates could be managed, the levels are much higher than desirable. When considered with the calculated rates of contamination buildup, operation in this environment is considered unacceptable.

Material Control and Accounting—As a result of fuel oxidation during handling operations, oxidized material is likely to be released from fuel rods and will not be contained during the fuel handling process. This issue is discussed in Section 7.4.9. The oxidized material will be collected by the HVAC HEPA filters and during decontamination of the transfer cells. This oxidized material will be difficult to control and account for as required under the Yucca Mountain material control and accounting plan.

Criticality—The preliminary criticality analysis discussed in Section 6.6.7 concludes that the expected amount of oxidized material released is less than the amount determined to be a criticality concern. The uncertainty with oxidation rates and release fractions needs to be evaluated further to determine if this preliminary conclusion is valid for normal and accident events.

Other Key Conclusions—Other key conclusions from the study are that the PCSA consequence analysis needs to consider fuel oxidation after a Category 1 fuel drop event (Section 6), and the radwaste streams resulting from operations and decontamination activities need to be evaluated and included in the design basis (Section 7).

8.2 RECOMMENDATIONS

Recommendations are presented for future work and evaluations to support facility design development and the license application. Section 8.2.1 provides recommendations for additional development work to support the design bases for handling CSNF. 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. Recommendations related to this approach are presented in Section 8.2.2.
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. Recommendations associated with this approach are presented in Section 8.2.3.
3. Option 3, limit handling of CSNF in air to small time durations. The short time that the fuel is exposed to air during transfer is not expected to result in fuel oxidation. Recommendations associated with this approach are presented in Section 8.2.4.

This study provides a technical basis for evaluating fuel oxidation and the consequences to the design and operations. Prior to determining which of the three options should be selected, additional reviews should be performed to further identify and evaluate concepts that could be implemented at Yucca Mountain.

8.2.1 Design Bases

Independent of the decision to select an option, there are several recommendations for establishing a basis for handling CSNF with a potential for oxidation.

Design Basis Temperatures and Times—Additional work is required to establish the off-normal/accident scenario temperature(s) and duration(s) (e.g. loss of HVAC event). The design bases need to have a well defined accident temperature limit and time duration for accident conditions with supporting justifications. The times of 100 hrs, 500 hrs, and 1,000 hrs, temperatures of 400°C for normal operations and 570°C for off-normal/accident operations, and burnups (average burnups of 40 PWR and 48 BWR GWd/MTHM) are bounding values in the calculation for this study to assist in decision making for future work. Future work should be done to verify these parameters for each specific handling operation. In addition, calculations are needed 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, then mitigating features may need to be implemented.

Cladding Performance—A cladding performance model is needed to confirm the cladding creep performance at various times and temperatures for normal, off-normal and accident conditions. Further evaluations are needed to determine cladding performance under these conditions considering expected fuel burnup and cladding materials such as stainless steel and advanced alloy materials such as M5 and ZIRLO. The cladding failure modes listed below were briefly reviewed in this study and should be evaluated in more detail to confirm the conclusions reached: stress corrosion cracking; delayed hydride cracking; hydrogen redistribution; hydride reorientation and pumping; irradiation embrittlement; and strain rate embrittlement.

Failed Fuel Quantities—Further evaluate failed fuel quantities and types of damage to establish a basis for design of the surface facilities, operations, and the safety analyses. For example, margins could be added to the failed fuel estimates in Table 3-4 to ensure a conservative basis for the design.

Material Control and Accounting—The Material Control and Accounting plan should be reviewed to consider how fuel materials not contained in fuel cladding will be accounted for.

PCSA Evaluations—The current PCSA consequence analysis (BSC 2005 [DIRS 171607]) considers 154 fuel rods being subject to oxidation (Section 6.1.1). As discussed in Section 6.4, the number of fuel rods subject to oxidation is approximately 671. Off-normal or accident conditions need to be reviewed. Section 6.3 defines several off-normal conditions that need to be reviewed by PCSA to determine if they should be considered and further defined (Category 1 or 2 or beyond category 2). Current PCSA consequence analyses for Category 1 and Category 2 events do not consider oxidation following a drop or collision. PCSA needs to review the event sequences and determine the consequences of oxidation. Such determination may require revision to PCSA documents such as *Categorization of Event Sequences for License Application* (BSC 2004 [DIRS 171429]) and *Consequence Analysis* (BSC 2005 [DIRS 171607]).

8.2.2 Prevent Routine Fuel Oxidation, Option 1

Recommendations for preventing and minimizing the potential for fuel oxidation are discussed in Section 7. Many of the issues in the Section 8.1 key conclusions would be eliminated or significantly reduced if normal fuel handling operations prevented fuel oxidation. Potential design features and operating strategies for Option 1 are listed below.

- Cooling systems—Minimize the potential for cladding degradation and fuel oxidation by controlling cask and waste package temperature using: cask cooling system on the cask exterior at process stations; waste package cooling systems on the waste package exterior at process stations; fuel transfer room cooling; and cooling the interior of the cask or waste package with a cold inert gas (Section 7).

- Fuel inerting—Prevent oxidation of failed fuel by inerting the cask during the preparation process; the cask throughout the unloading process; waste package throughout the loading process up to waste package inerting during the closure process; the aging cask throughout the loading process up to inerting during the cask restoration process; and the CSNF transfer process by inerting a transfer tube on the spent fuel transfer machine or the entire transfer room (Section 7).

- Minimize off-normal fuel oxidation events—If an inert gas design is used, include the ability to recover from a loss of the inerting system by using a backup/mobile system with appropriate tools to inert a cask or waste package (Section 7).

- Segregate failed fuel—Segregate failed fuel from unfailed fuel at the utility sites before shipment to the repository. Failed fuel could be placed in sealed, inerted canisters to avoid fuel oxidation or an unsealed canister that confines any oxidation materials and prevents the spread of contamination.

8.2.3 Handle CSNF in Air, Option 2

Section 7 of the study reviews the fuel handling process steps in the surface facilities and identifies potential hazards associated with handling CSNF in air. The areas of concern discussed in Section 8.1 would need to be addressed in the design development of Option 2. Potential design features and operating strategies are described below to accommodate and recover from the consequences of handling CSNF in air.

- Fuel oxidation process—As discussed in Section 8.1, further work is needed to provide a referencable basis for the phenomenon and parameters controlling fuel oxidation, the fuel cladding failure rate, the oxidation rate, and release fractions during oxidation.

- Equipment reliability—To minimize the potential for equipment failure trigger and lenghten an oxidation event, supporting systems such as HVAC and SNF/HLW transfer systems should be reviewed to ensure appropriate reliability and redundancy are provided. This includes the HVAC functions to control temperature and confine radioactivity to areas where waste forms are processed. Mechanical handling functions and equipment would be relied upon to prevent damage to CSNF and minimize the time CSNF is handled in air (Section 7).

- Recovery devices—To avoid scenarios where CSNF may be exposed to air for indeterminate time intervals, the design would have the capability to recover from failures of cranes and trolleys by moving casks, waste packages, or fuel assemblies to an area where it can be cooled and/or inerted (Section 7).

- Contamination recovery—Recover from contamination of equipment or an area by: adjusting cask or waste package pressure while in the main transfer room or docked in the fuel transfer room; remotely maintaining handling equipment in the fuel transfer room; capability to decontaminate the fuel transfer room either remote or hands-on, with appropriate cell surface material (e.g. stainless steel); capability to decontaminate the cask interior; capability to decontaminate the exterior of a cask and waste package; and appropriate shielding of remote HEPA filters and remote maintenance capability for HEPA filters (Section 7).

- Confinement—Review how confinement would be maintained for each step of the process, considering the presence of oxidized fuel materials. Minimize the spread of contamination by: confining cask sampling/venting operations; confining waste package venting operations; maintaining a negative pressure in the cask cavity and waste package cavity; reducing HVAC air velocity in areas affected by fuel oxidation; confining the fuel transfer room to preclude out leakage during a loss of HVAC; maintaining static confinement of unsealed waste package and unbolted aging cask; and confining releases of oxidized fuel during transfer in the fuel transfer room. The overall contamination control design philosophy and HVAC design should be reviewed to determine the consequences of handling oxidized fuel. Designing systems and processes to preclude spread of contamination, even if the contamination and the dose to the public and workers consequences are acceptable, is a good defense-in-depth design and safety philosophy (Section 7).

- Contamination levels—Based on the discussion in Section 6.7, oxidation of bare fuel in air and the resultant release of radioactivity produces a radiological environment with a high risk of alpha contamination that will not allow routine work activities in the fuel transfer cell. While work could be performed in this environment if mandatory, it may not be considered ALARA and manned entry would require maximum respiratory and contamination personnel protective equipment for workers. The recommendation is to apply design features that would either limit the contamination initially or provide for remote maintenance. Using a design target range for maximum contamination levels of 100,000 dpm/100 cm^2 to 500,000 dpm/100 cm^2 may provide an operationally manageable level of contamination from both a worker protection and operational efficiency perspective.

- Criticality—Criticality evaluations should be expanded to consider the following.

  - Damaged fuel cans will contain failed fuel. Criticality control assessments should be performed to consider oxidation of failed fuel rods received in DFCs.

  - The fire hazards analysis should be reviewed to consider the accumulation of oxidized fuel material in the transfer cells.

  - The preliminary criticality analysis discussed in Section 6.6.7 needs to be finalized. Parameters such as specific geometries (e.g., the base of the staging tubes in DTF, drains, sumps, and other locations where oxidized fuel could accumulate); what range of chemical forms the oxidized fuel will take (e.g., schoepite: $UO_3{\cdot}2H_2O$); and the amount of moderation to consider should be analyzed.

  - The number of rods of oxidized fuel necessary for criticality over the range of conditions should be determined.

  - The criticality evaluation should also consider the total amount of potentially loose fissile material, the potential for the oxidation of fuel assemblies with a greater than average number of failed rods as well as off-normal events where the quantity of unclad fuel available for oxidation is increased.

- Radwaste—The current design does not yet include the management of radioactive waste that will be generated during maintenance, HEPA filter change out and decontamination activities. The allowable HEPA filter loadings need to be defined. A bounding analysis indicates that the HEPA filter shine dose rates may exceed the shielding design criteria. The loading on the HEPA filters are required to determine the disposal options for the used HEPA filters (low-level radioactive waste [LLW], greater than Class C, or HLW) (Section 7).

- Loss of HVAC—Currently, a loss of ventilation and subsequent loss of confinement is not considered in the PCSA consequence analysis to be a credible event. Loss of HVAC confinement could result in leakage out of the transfer cell into normally occupied areas. Bounding calculations using very conservative assumptions indicate that the worker dose approaches the limits and offsite public dose consequences exceed the limits for Category 1 events. This event needs further work to determine what category it falls into and what design features need to be implemented to mitigate or preclude this event. Items such as allowable leakage rates from the primary confinement may need to be defined. This event may require revision to several PCSA documents such as the *Categorization of Event Sequences for License Application* (BSC 2004 [DIRS 171429]) and *Consequence Analysis* (BSC 2005 [DIRS 171607]) (Section 6).

8.2.4 Minimize Time CSNF is Handled in Air, Option 3

This approach is a combination of Options 1 and 2 and would keep CSNF in an inert environment except for short periods when CSNF 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 CSNF is exposed to air during the 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.

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APPENDIX A
GLOSSARY

aging cask—A large shielded container used for transferring, staging, or aging CSNF within the geologic repository operations area that meets all applicable regulatory requirements.

aging cask transporter—A large vehicle designed to straddle, safely lift, and transport an aging cask or transportation cask from the surface waste handling facilities to the aging pads.

aging pads—Structures and systems built in the surface geologic repository operations area (GROA) near the North Portal for aging CSNF and staging SNF and HLW prior to emplacement in the subsurface repository.

Alloy 22 (UNS N06022)—A high-nickel alloy used for the outer barrier of the waste package.

as low as is reasonably achievable—A regulatory requirement defined in 10 CFR 20.1003 [DIRS 104787] that means making every reasonable effort to maintain exposures to radiation as far below the dose limits in this part as is practical consistent with the purpose for which the licensed activity is undertaken, and taking into account the state of technology, economics of improvements in relation to the state of technology, economics of improvements in relation to benefits to the public health and safety, and other societal and socioeconomic considerations and in relation to the utilization of nuclear energy and licensed materials in the public interest.

boiling water reactor—A nuclear power reactor in which water passing as coolant through the core is turned to steam by direct use of fission heat from the uranium oxide fuel. Steam for driving the turbine generator is formed within the reactor vessel itself rather than in an external heat exchanger and, after being condensed, returns as feedwater to the reactor vessel.

bridge crane—A large overhead crane used for material handling that spans across rails on either side of a structure.

burnup—A measure of nuclear-reactor fuel consumption expressed either as the percentage of fuel atoms that have undergone fission or as the amount of energy produced per initial unit weight of fuel.

canister—The metal structure containing some forms of SNF and HLW (e.g., HLW immobilized in vitrified-glass or SNF assemblies) that facilitates handling, storage, transportation, or disposal or a combination of these.

cask—A large, shielded container for shipping, transferring, or storing SNF or HLW that meets all applicable regulatory requirements.

cladding—The metallic outer sheath of a fuel rod element generally made of a zirconium alloy or stainless steel. It is intended to isolate the fuel element from the external environment.

codisposal—A packaging method for the disposal of radioactive waste in which more than one type of waste, such as DOE SNF and HLW, is combined in waste packages. Codisposal takes advantage of otherwise unused space in waste packages and is a cost-effective method of limiting the reactivity of individual waste packages.

commercial high-level waste—HLW created by the processing of SNF discharged from reactors owned by public or private utilities, companies, and corporations.

commercial spent nuclear fuel—Radioactive waste created in nuclear reactors owned by public or private utilities, companies, and corporations. There are over 100 sites in the United States that currently generate or have generated such waste.

confinement—The control of radioactive contamination within a designated boundary. Often used to represent a design feature that controls or restricts the spread of contaminants.

containment—(1) A barrier or structure that serves to prevent the escape of radioactivity from a designated boundary or (2) a design feature used to prevent or minimize radioactive releases to the environment.

criticality—The condition in which nuclear fuel sustains a chain reaction. It occurs when the effective neutron multiplication factor of a system equals one. The condition is avoided in all events at the repository.

defense high-level radioactive waste—HLW created by processing SNF discharged from reactors constructed and operated by the U.S. government, its agencies, and contractors.

disposable canister—A sealed metal canister specifically designed and licensed for long-term disposal and waste isolation in the repository.

docking ring—A remotely operated sealing mechanism used to install a contamination barrier between a cask or waste package and the SNF/HLW transfer cell prior to waste transfer operations.

drip shield—A corrosion-resistant engineered barrier that is placed above the waste package to prevent seepage water from directly contacting the waste package for thousands of years. The drip shield also offers protection to the waste package from rockfall.

Dry Transfer Facility—One of several waste handling buildings constructed without water basins near the repository North Portal used to receive, handle, process, transfer, package, seal, and transport SNF and HLW for emplacement. Abbreviated DTF 1 or DTF 2.

dual-purpose canister—A sealed metal container used to transfer, store, and transport SNF and HLW from a reactor site to a storage and repository site. The NRC, pursuant to 10 CFR Part 71 [DIRS 104091] and Part 72 [DIRS 127267] for transportation and storage, issues Certificates of Compliance for packages that may include shipping casks and canisters as a component. Thus, the term dual-purpose.

emplacement—The placement and positioning of waste packages in the repository subsurface emplacement drifts.

emplacement drift—A straight underground tunnel running from one subsurface access main to an exhaust main where remote operations occur to place waste packages for long-term geological disposal.

emplacement pallet—A welded structure, of varying size, used to support each waste package during transport, emplacement, and long-term waste isolation in the repository.

engineered barrier—Any component of the engineered barrier system, such as the drip shield, waste package, or invert, that is designed to meet performance requirements of 10 CFR Part 63 [DIRS 156605].

event sequence—A series of actions, occurrences, or both within the natural and engineered components of a GROA that could lead to exposure of individuals to radiation. Event sequences that are expected to occur one or more times before permanent closure of the GROA are referred to as Category 1 event sequences. Other event sequences that have at least one chance in 10,000 of occurring before permanent closure are referred to as Category 2 event sequences. Event sequences with a less than one in 10,000 chance of occurring before permanent closure are categorized as beyond Category 2 events.

fission product—Any nuclide, either radioactive or stable, that arises from fission, including both the primary fission fragments and their radioactive decay products. Also called daughter product or decay product.

fuel assembly—A number of fuel rods held together by plates and separated by spacers, used in a reactor. This assembly is sometimes called a fuel bundle.

fuel blending—The process of loading low heat output waste with high heat output waste in a waste package to balance its total heat output. This process applies only to CSNF.

gas tungsten arc welding—A fabrication process where a high-voltage electric arc between a tungsten electrode, the metal being welded, and filler metal is used to weld components. The process is performed in a monatomic inert gas mixture environment to shield the molten metal from air. A recognized method of welding by the American Welding Society.

geologic repository—A system for the disposal of radioactive waste in excavated geologic media. A geologic repository includes the engineered barrier system and portion of the geologic setting that provides isolation of the radioactive waste.

geologic repository operations area—An HLW facility that is part of a geologic repository, including both surface and subsurface areas, where waste handling activities are conducted.

grapple—A mechanical handling device or end-effector used to safely and remotely couple a SNF assembly, HLW canister, and waste handling equipment for lifting, transferring, and packaging operations.

heavy-haul truck—A large, tractor-towed, multi-axle, multi-wheel, transport vehicle used to haul heavy oversize loads across U.S. highways and roads. Such vehicles may be used to transport large rail shipping casks weighing 100 to 150 tons to the repository by highway.

high-heat spent nuclear fuel—CSNF assemblies arriving at the repository that emit heat at a higher rate than other assemblies due to their young age (i.e., 5 to 10 years since reactor discharge) or high burnup levels. Typically associated with assemblies generating greater than 1,000 watts.

high-level waste—The highly radioactive material resulting from reprocessing SNF, including liquid waste produced directly in reprocessing; any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and other highly radioactive materials that the NRC rules must receive permanent isolation.

impact limiters—Removable engineered structures fastened to either end of an NRC-certified transportation cask that reduce the structural loads on a cask body that would result from a severe transportation accident. Typical impact limiters contain a deformable material such as aluminum honeycomb, wood, or engineered foam surrounded by a thin metal or composite skin. Impact limiters are removed from the cask during handling or loading and are fastened in place on the cask prior to transportation.

important to safety—A preclosure safety classification assigned to repository systems, structures, and components (SSCs) that is required to function to protect public or worker health and safety pursuant to regulatory criteria in 10 CFR 63.111 [DIRS 156605].

important to waste isolation—A postclosure safety classification assigned to natural barriers and geologic repository engineered barrier systems that is required to function to protect the maximally exposed individual pursuant to the regulatory criteria in 10 CFR 63.113 [DIRS 156605].

inner lid—A stainless steel engineered barrier component that is used to close and seal the stainless steel inner vessel of the waste package.

inner lid spread rings—Stainless steel engineered barrier components that are remotely installed and welded to structurally secure, close, and seal the inner lid to the inner vessel of the waste package.

inner vessel—The inner stainless steel cylindrical structure of the waste package used to provide structural integrity for this engineered barrier system component.

interpass temperature—The maximum allowable temperature that the weldment may be at immediately prior to the next weld pass.

legal-weight truck—A tractor-towed, multi-axle, multi-wheel, transport vehicle used to haul loads that meet the U.S. Department of Transportation size and weight limits (80,000 lbs maximum), across U.S. highways and roads. Such vehicles will be used to transport small shipping casks weighing 20 to 25 tons to the repository by highway.

license application—A formal submittal to the NRC initially to construct a repository for the disposal of SNF and HLW.

low-level radioactive waste—Radioactive waste producing small quantities of ionizing radiation and that is not classified as HLW, transuranic waste, or byproduct tailings containing uranium or thorium from processed ore. Usually generated by hospitals, research laboratories, and nuclear industry facilities such as the Yucca Mountain repository.

metric tons of heavy metal—A metric ton is a unit of mass equal to 1,000 kg (2,205 lb). Heavy metals are those with atomic masses greater than 230. Examples include thorium, uranium, plutonium, and neptunium. The term usually pertains to heavy metals in SNF and HLW. For SNF, MTHM is approximately equal to MTU.

middle lid—An engineered barrier component that is the first of two lids used to close and seal the Alloy 22 outer vessel of the waste package.

monitored geologic repository—A system, requiring licensing by the NRC, intended or used for the permanent underground disposal of SNF and HLW. A geologic repository includes (a) the geologic repository operations area, and (b) the geologic setting within the controlled area that provides isolation of the radioactive waste.

multi-canister overpack—A disposable canister designed for repository disposal, containing DOE SNF produced by the N-Reactor at the DOE Hanford Site.

multi-purpose canister—The metal structure containing some forms of SNF and HLW (e.g., HLW immobilized in vitrified-glass or spent fuel assemblies) that meets all applicable regulatory requirements for the handling, storage, transportation, and disposal in the repository.

National Transportation Project—An organization created by DOE, and managed by BSC, for the purpose of managing the transportation element of the Civilian Radioactive Waste Management System. This element is responsible for carrying out the mission of design, procurement, licensing, and deployment of transportation systems, casks, canisters, and equipment for the DOE Office of Civilian Radioactive Waste Management. Also see Nevada Transportation Project.

nondestructive examination—Testing methods that determine the properties, structure, serviceability, or quality of a part or component without limiting its usefulness.

North Portal—An opening from surface facilities to the subsurface facilities of the repository that allows access for personnel, emplacement equipment, waste packages, and air to enter into the underground tunnel and ramp to the repository emplacement drifts.

off-normal—A term used to define SNF, HLW, operations, and processes that are not expected during normal repository activities.

outer corrosion barrier—The external Alloy 22 shell of the waste package that provides the engineered barrier its long-term corrosion protection.

outer lid—The outermost Alloy 22 lid is an engineered barrier component that is used to close and seal the Alloy 22 outer vessel of the waste package. See also: middle lid.

overpack—A shielded enclosure used to protect: (1) a DPC for the purpose of staging and (2) an aging canister during transport or aging.

preclosure—The period of time before and during closure of the repository.

pressurized water reactor—A type of nuclear steam supply system that uses clad uranium-oxide fuel elements, cooled and moderated by primary coolant water under high pressure, in a pressure vessel to prevent the water from boiling. An external heat exchanger or steam generator is used to boil secondary coolant water and generate steam. The primary coolant passes through tubes in the steam generator to boil the secondary coolant. The steam is passed through a turbine-generator to produce electricity.

prime mover—A heavily loaded vehicle used to haul transportation casks, transfer casks, and aging casks to and from the repository surface facilities.

protected area—An area within the surface repository operations area enclosed by security fences, control gates, lighting, and access detection systems. This area includes the facilities and transportation systems required to receive and ship rail and truck waste shipments, prepare shipping casks for handling, and load waste forms into waste packages for underground emplacement. It also includes the facilities and systems required to treat and package site-generated, low-level radioactive waste for offsite disposal.

radioactive—The property possessed by some elements (e.g., uranium) of spontaneously emitting alpha, beta, or gamma rays by the disintegration of atomic nuclei.

radioactive decay—The process in which one radionuclide spontaneously transforms into one or more different radionuclides called decay products or daughter products.

radioactive waste—HLW and other radioactive materials, including SNF, that are received for emplacement in the repository.

restricted area—An area in which access is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. Restricted area does not include areas used as residential quarters, but separate rooms in a residential building may be set apart as a restricted area.

scenario—A well defined, connected sequence of features, events, and processes that can be thought of as an outline of a possible future condition of the repository system.

site—The area surrounding the GROA for which DOE exercises authority over its use.

site rail transfer cart—A small railcar designed for the sole purpose of transferring various transportation casks from the Transportation Cask Receipt Building to the waste handling facilities located at the surface GROA. The railcar is also used to queue and stage casks at the transportation cask buffer area.

spent fuel transfer machine—Mechanical handling equipment used to remotely transfer highly radioactive SNF assemblies between casks, canisters, and waste packages for disposal in the repository.

spent nuclear fuel—Used fuel elements and the associated hardware withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated by reprocessing. Used fuel elements that have been irradiated (burned) in a reactor to the extent that it no longer makes an efficient contribution to a nuclear chain reaction. Used fuel that is highly radioactive compared to unused fuel and generates significant decay heat.

spent nuclear fuel/high-level waste transfer cell—A heavily shielded reinforced concrete and steel structure containing SNF, HLW canisters, and mechanical handling equipment used to remotely transfer highly radioactive waste forms between casks and waste packages for disposal in the repository.

staging area—An area in the surface waste-handling facilities in which SNF or HLW is temporarily stored for future handling, packaging, or emplacement.

subsurface facilities—The repository underground SSCs.

surface facilities—The repository surface SSCs.

transfer cask—A heavily shielded container system that meets applicable regulatory requirements for the transfer of SNF or HLW between surface waste transfer and aging facilities. Also called an aging cask.

transportation cask—A heavily shielded container system that has been certified by the NRC to ship SNF or HLW to the geologic repository using public highways, roads, and rail systems.

transportation cask buffer area—A queuing and staging area located near the North Portal surface waste handling facilities for site-specific railcars and transportation casks.

Transportation Cask Receipt/Return Facility—A building located near the North Portal surface waste handling facilities used for transferring of transportation casks from their offsite railcars and truck trailers to a site-specific railcar.

ultrasonic test—A non-destructive examination process that uses high frequency sound waves to detect and locate structural discontinuities in materials and volumetric defects in welds. A recognized method of non-destructive examination by the American Society of Mechanical Engineers.

U.S. Department of Energy spent nuclear fuel—Radioactive waste forms created by irradiation of uranium in defense-related nuclear reactors or DOE-owned or fueled commercial reactors. The major contributor to this waste form is the N-Reactor fuel currently stored at the Hanford, Washington site. This waste form also includes naval SNF.

visual test—A non-destructive examination process that uses manual, video, or photographic inspection procedures to detect component surface defects.

vitrified high-level radioactive waste—A type of processed HLW where the waste is mixed with glass-forming chemicals and put through a melting process. The melted mixture is then put into a canister where it becomes a dry solid log of vitrified waste in a glass matrix.

waste form—A generic term that refers to the different types and configurations of HLW and SNF.

waste handling facility—A generic term describing a surface building used to receive, handle, process, transfer, package, seal, and transport SNF and HLW for emplacement in the repository.

waste package—A metal vessel that is designed to contain waste for emplacement. The waste package includes the waste form and any canisters, spacing structures or baskets, and other absorbent materials placed internally to the vessel or permanently attached to the outer surface of the vessel. Other terms used to describe the waste package are: (1) empty waste package—an open/new waste package that contains no waste forms; (2) loaded waste package—an open waste package containing waste forms that is unsealed or only temporarily closed; (3) closed waste package—a sealed waste package containing waste forms that is undergoing final welding, stress mitigation, examination, handling, or aging; and (4) completed waste package—a sealed waste package ready for emplacement.

waste package remediation—An operation or area for repair of waste packages that has failed the weld inspection processes; that is defective, damaged, or off-normal; or that has been selected for retrieval from the repository for performance confirmation examinations.

waste package transporter—A large, heavy, shielded, self-powered vehicle used to haul completed waste packages in a horizontal orientation from the surface facilities to the underground repository for emplacement.

waste stream—A scenario, schedule, specification, and set of input assumptions used to forecast the quantity, rate, and type of HLW arriving at the geologic repository over time.

zircaloy—A family of alloys of zirconium that may have any of several compositions. These alloys are frequently used as a cladding material for SNF pellets.

APPENDIX C
PRECLOSURE CONSEQUENCE ANALYSIS FOR LICENSE APPLICATION ASSUMPTIONS

The following assumptions are taken from the *Preclosure Consequence Analysis for License Application* (BSC 2005 [DIRS 171607], Section 4). The assumptions, except for Assumption 14, are used for calculating public dose. Assumption 14 is used for calculating worker dose. The basis for the assumptions is included in *Preclosure Consequence Analysis for License Application* (BSC 2005 [DIRS 171607], Section 4).

1. Waste forms involved in normal operations and in Category 1 and Category 2 event sequences include PWR or BWR spent fuel assemblies, HLW, and naval SNF.

2. At the highest nominal receipt rate, 3,000 MTHM (DOE 2004 [DIRS 171945]) of CSNF pass through the Canister Handling Facility, DTF 1, DTF 2, and FHF each year. It is assumed that for the purposes of calculating worker and public doses, fuels received are PWR spent fuel assemblies. Using an average PWR assembly weight of 0.475 MTHM per spent fuel assembly (BSC 2004 [DIRS 169061]), the highest nominal throughput is 6,316 spent fuel assemblies per year.

3. One percent of the fuel rods received at the repository are modeled as having defect sizes equal to pinhole leaks or hairline cracks, and the fission product gases, volatile species, and fuel fines are released. Releases from one percent of the 6,316 spent fuel assemblies are used as the source term for the calculation of normal operations doses.

4. The HEPA filters of the surface facility HVAC systems are assumed to be unavailable to remove radionuclides for Category 2 event sequences. HEPA filters are assumed to be functioning for surface facility normal operations and Category 1 event sequence dose calculations.

5. For Category 1 and Category 2 event sequences, radioactive materials are released in a 1-hr duration.

6. For calculating public doses from normal operations (ORNL 1998 [DIRS 164761]), the release duration is assumed to be 24 hr. The release is assumed to result in an acute individual exposure during plume passage and a chronic individual exposure to ground contamination and contaminated food after plume passage. The period of long-term exposure to ground contamination and intake of contaminated food is one year.

7. It is assumed that radionuclides are released from surface facilities during normal operations via the exhaust stack.

8. For normal operations and Category 1 event sequences, a two-stage HEPA filtration system with a particulate removal efficiency of 99 percent per stage is assumed. This gives a combined efficiency of 99.99 percent for two stages, and a HEPA leak path factor (LPF) of 10^{-4}. It is further assumed that the HVAC system is removing

particulates and cesium in air through two stages of HEPA filters in series that are protected by prefilters, sprinklers, and demisters.

9. Drops or collisions of canistered single CSNF rods in transportation casks and waste packages are assumed to be Category 2 event sequences.

10. For Category 1 and Category 2 event sequences, the DR is assumed to be 1.0 for CSNF. For Category 2 event sequences, the DR is assumed to be 1.0 for HLW in a canister. The DR is the fraction of fuel rods that is assumed to fail by cladding breach during an event sequence or the fraction of HLW that is damaged by crush or impact, or both. Bounding DRs are used for Category 2 event sequences involving naval SNF (Gisch 2004 [DIRS 171579]).

11. For Category 2 event sequences, the cask leak path factor, $(LPF)_{cask}$, is assumed to be 0.1 for SNF in a transportation cask, 0.01 for HLW in a canister in a transportation cask, and 1.0 for naval SNF in a transportation cask or canister. The $(LPF)_{cask}$ is the fraction of the ARF that reaches the ventilation system after local deposition, consisting of plate-out and gravitational settling, within a transportation cask.

12. For normal operations and Category 1 event sequences, the facility leak path factor, $(LPF)_{fac}$, is conservatively assumed to be 1.0 inside a waste transfer cell. The $(LPF)_{fac}$ is the fraction of the ARF that reaches the ventilation system after local deposition, consisting of plate-out and gravitational settling, within a surface facility.

13. It is assumed that the HVAC system is operating and no airborne material released from Category 1 event sequences leaks into space occupied by workers who work in rooms adjacent to a waste transfer cell in a DTF or FHF.

14. For normal operations and Category 1 event sequences, it is assumed that for radionuclides released from a waste transfer cell within a surface facility, the HVAC system is operating and airborne radionuclides are vented through the building exhaust stack, dispersed into the atmosphere, and then reenter the building through the building ventilation system air intakes. It is assumed that for radionuclides released from the subsurface facility, airborne radionuclides are dispersed into the atmosphere and reenter the subsurface facility through the subsurface ventilation system air intakes.

15. The maximally exposed individual is defined as an individual located at a distance that corresponds to the approximate distance between the surface facility or subsurface repository and the nearest point of public access on the repository site boundary, which lies to the west. The proposed land withdrawal area boundary is assumed to be the site boundary. A site boundary distance of 11 km is used to calculate χ/Q values from radiological releases from the surface facility. This distance corresponds to the distance from the DTF ventilation exhaust shaft to the nearest point on the site boundary that is the closest point where any member of the public could be standing or living at the time of a postulated radiological release. A site boundary distance of

8 km is used to calculate χ/Q values from radiological releases from the subsurface repository. This distance corresponds to the approximate distance between the subsurface repository and the nearest point of public access on the site boundary, which lies to the west.

16. It is assumed that the fission product gas, volatile species, and crud ARFs for breaches of intact CSNF assemblies and rods in Table 5 of the *Preclosure Consequence Analysis for License Application* (BSC 2005 [DIRS 171607]) are applicable to releases of fission product gases, volatile species, and crud during the oxidation of damaged CSNF in air (BSC 2004 [DIRS 172371]).

17. The releases from the subsurface exhaust shafts during normal operations and the releases from the surface facilities from Category 1 and Category 2 event sequences are assumed to be at ground level.

18. For normal operations, and Category 1 and Category 2 event sequences, no credit is taken for charcoal absorbers to remove radionuclides.

19. It is assumed that the canister handling system is designed so that a drop of an HLW canister inside a surface facility will not exceed a drop height of 276 in. (23 ft).

20. The maximally exposed individual at the site boundary is assumed to receive doses from the inhalation, resuspension inhalation, air submersion, groundshine, and ingestion pathways for a period of 8,760 hr. The onsite individual member of the public, at 100 m or 3 km away from a DTF, FHF, or subsurface exhaust shaft, is assumed to receive doses from inhalation, resuspension inhalation, air submersion, and groundshine pathways for a period of 2,000 hr.

21. It is conservatively assumed that radionuclides are released from a height of 30 m during surface facility normal operations. This release height is used to calculate χ/Q values for use in dose calculations for surface facility normal operations.

22. For members of the public at the site boundary, the period of long-term exposure to ground contamination and intake of contaminated food is one year.

23. It is assumed that 154 failed fuel rods per year are expected to be vulnerable to oxidation and cladding unzipping.

24. An ARF of 1.2×10^{-4} is assumed for fuel rods expected to be vulnerable to oxidation and cladding unzipping.

25. Little or no oxidation is expected to occur for intact fuel or fuel with pinhole or hairline cracks during fuel handling operations in the repository. The fuel fine ARF of 3×10^{-5} for a burst rupture is conservatively used for fuel with pinhole leaks or hairline cracks where little or no fuel oxidation occurs.

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

D1 POTENTIAL HAZARDS DURING NORMAL OPERATIONS

D1.1 CASK VENTING OR SAMPLING SYSTEM CAUSING A RELEASE OF RADIONUCLIDES TO THE ENVIRONMENT FROM FAILED FUEL OR CRUD

Venting and sampling operations create a direct communication path between the contaminated cask cavity and the environment. The main sources of potential contamination are the presence of crud or the presence of failed fuel. Crud is present on both intact and failed fuel. Since failed fuel is assumed to be processed along with intact fuel, this hazard will occur and is considered in the draft SAR (Section 1.8).

D1.2 WASTE PACKAGE VENTING SYSTEM CAUSING A RELEASE OF RADIONUCLIDES TO THE ENVIRONMENT FROM FAILED FUEL OR CRUD

Venting operations create a direct communication path between the contaminated waste package cavity and the environment. The main sources of potential contamination are the presence of crud or the presence of failed fuel. Crud is present on both intact and failed fuel. Since failed fuel is assumed to be processed along with intact fuel, this hazard will occur.

D1.3 RISE IN FUEL TEMPERATURE CAUSING CLADDING DEGRADATION

The cask handling process modifies the cask thermal equilibrium and may cause a rise in fuel temperature due to the handling of casks in a vertical position, the replacement of the inert gas within the cask cavity by air, and varying ambient conditions. A thermal analysis of the cask-handling process from cask receipt to cask unloading should be performed to evaluate fuel temperatures at each step. A similar thermal analysis is needed for fuel in a waste package, from waste package loading to closure. A maximum allowable fuel cladding temperature of 400°C is currently being considered for normal operations.

Preliminary thermal analyses (BSC 2004 [DIRS 171778]; BSC 2005 [DIRS 172741]) indicate that cladding temperatures will remain below 400°C. If confirmed, these analyses would demonstrate that this hazard cannot occur during normal operations. As discussed in Section 4, cladding creep rupture may be time dependent, and may occur at 400°C for a fraction of the total fuel rod inventory if the fuel handling/transfer operations were to require an extended period of time.

D1.4 INCREASE OF CASK OR WASTE PACKAGE PRESSURE CAUSING A RELEASE OF CONTAMINATION

An increase of the internal pressure in an unbolted cask or unsealed waste package may result in a leakage of contamination. An analysis of the corresponding process steps should provide the changes in container cavity pressure between process stations.

Pressure criteria for an unbolted cask or unsealed waste package should be based on leakage rates or results from leak-tightness testing.

D1.5 RISE IN CASK TEMPERATURE CAUSING A LOSS OF NEUTRON SHIELD EFFICIENCY OR DAMAGE TO CASK COMPONENTS

The cask-handling process modifies the cask thermal equilibrium and may cause an increase of the cask neutron shield temperature due to handling casks in a vertical position and varying ambient conditions. High temperatures could also damage cask confinement seals. A thermal analysis of the cask-handling process should provide an evaluation of cask and cask seal temperatures at each step. Temperature limits for cask neutron shields and thermal criteria for cask seals are defined in the cask SARs as shown in Table D-1.

Table D-1. Temperature Criteria for Cask Components

		Neutron Shield		Seals	
Cask Type	**Waste Type**	**Maximum Operating Temperature (°F)[a]**	**Max Allowable Temperature (°F)**	**Maximum Operating Temperature (°F)[a]**	**Max Allowable Temperature (°F)**
NAC-LWT	CSNF	238	**350**	227	**550/735/800**
GA-4/9	CSNF	TBD	**TBD**	TBD	**TBD**
TN 68	CSNF	244	**300**	234	**536**
NAC-STC	CSNF	285	**300**	190	**500**
	DPC	270/288	**300**	190/228	**500**
HI-STAR	DPC				
MPC 68		264/255	**300**	TBD	**TBD**
MPC 24		271/292	**300**		
MP-187	DPC	TBD	**250**	249/283	**600/700**
MP-197	DPC	249	**300**	217	**400**
TS-125 FuelSolutions	DPC	289/284	**300/350**	246/262	**932/662**
NAC-UMS	DPC	286/293	**300**	206 to 266	**300**
MSC	CSNF	TBD	**TBD**	TBD	**TBD**
TN 32	CSNF	TBD	**TBD**	TBD	**TBD**

NOTE: [a] Normal Transport Conditions: 38°C (100°F) ambient temperature, Full solar insulation, SNF in Helium.

D1.6 CRUD SUSPENSION AND DEPOSITION CAUSING SURFACE AND AIRBORNE CONTAMINATION OF A CELL

The presence of crud results in airborne contamination and surface contamination of the FHF. The consequences of releasing radionuclides to the environment through the cell ventilation system are currently bounded by the consequence analysis for cask sampling/venting operations in the draft SAR (Section 1.8). The dose for workers entering the cell is being calculated. The contamination buildup in the Fuel Transfer Room and the HEPA filters is discussed in Section 6.

D1.7 CONTAMINATED SURFACES OF A CASK OR WASTE PACKAGE DOCKING PORT CAUSING A RELEASE OF CONTAMINATION

During undocking operations, potentially contaminated parts of the docking port are in contact with the Fuel Transfer Bay atmosphere. This could lead to the contamination of the Waste Package or cask exterior. Re-suspension of deposited radioactive particulates may cause an airborne contamination within these rooms. The contamination buildup in the facility and the HEPA filters is discussed in Section 6.

D1.8 OXIDATION OF FAILED FUEL CAUSING A RELEASE OF CONTAMINATION

As discussed in Section 4, above a minimum temperature, the uranium in fuel rods with a failed cladding is susceptible to oxidation in air. A thermal analysis of the fuel-handling process from cask receipt operations to waste package closure operations should provide an evaluation of fuel temperatures at each step. Based on Einziger (1991 [DIRS 166177]) and Lorenz (1980 [DIRS 100990]), the oxidation hazard is not considered for cladding temperatures below 200°C or 230°C. For higher temperatures, a correlation with the oxidation incubation time has not been defined yet, but the hazard is likely to occur during normal operations, given high expected CSNF temperatures.

The event affects room/cell surface and airborne contamination levels, radionuclide releases from the facilities, as well as HEPA filter loading. These issues are discussed in Section 6. Filter changeout and disposal options need to be addressed.

D1.9 OXIDIZED FUEL CAUSING A CRITICALITY

When fuel oxidation cannot be ruled out, the potential for the oxidized fuel to aggregate in an unfavorable geometry and result in a criticality event should be addressed.

The criticality safety criterion is a k_{eff} lower than 0.95.

D2 POTENTIAL HAZARDS DURING OFF-NORMAL OPERATIONS

D2.1 FAILURE OF THE CASK VENTING OR SAMPLING SYSTEM CAUSING A RELEASE OF CONTAMINATION

Cask venting and sampling systems contain components located in occupied rooms. Their failure could pose a contamination hazard to workers who could be exposed to an unfiltered release.

D2.2 FAILURE OF THE CASK OR WASTE PACKAGE DOCKING PORT CAUSING A RELEASE OF CONTAMINATION

The hazard during normal operations is likely to be worsened by a failure of the docking system (docking port malfunction or incorrect undocking procedure). Airborne contamination within the fuel transfer bays and filter loading are to be evaluated.

D2.3 LOSS OF BUILDING VENTILATION CAUSING A RISE IN FUEL TEMPERATURE

A loss of building ventilation may be caused by a failure of the HVAC equipment, a loss of power, or a seismic event. A loss of building ventilation is likely to increase room temperatures and modify heat exchanges with the cask or waste package content. The resulting hazards are similar to items 3 and 4 in Section 7.3.3.1. A thermal analysis of the fuel-handling process from cask receipt operations to waste package closure operations should provide an evaluation of fuel temperatures at each step following a loss of building ventilation.

For intact fuel, 570°C is considered in the current design to be the maximum allowable fuel cladding temperatures during off-normal operations. As discussed in Section 4, fuel cladding failure may occur at such a temperature.

For information, preliminary thermal analyses (BSC 2004 [DIRS 171778]; BSC 2005 [DIRS 172741]) indicate that cladding temperatures would not exceed 460°C.

A study of cask or waste package pressure in case of a loss of building ventilation remains to be performed.

D2.4 LOSS OF BUILDING VENTILATION CAUSING A TEMPERATURE RISE IN CASK TEMPERATURE

A loss of building ventilation may be caused by a failure of the HVAC equipment, a loss of power, or a seismic event. A loss of building ventilation is likely to increase room temperature and modify heat exchanges with the cask content. The resulting hazard is similar to items 4 and 5 in Section 7.5.3.1. A thermal analysis of the process should provide an evaluation of the cask's neutron shield temperature (and that of other components) at each step.

The limiting temperatures may be identical to the values for normal operations as defined in the casks SAR (see Table 1), or a higher limit may be justified.

D2.5 LOSS OF BUILDING VENTILATION CAUSING A TEMPERATURE RISE IN ROOM OR CELL TEMPERATURE

A loss of building ventilation may be caused by a failure of the HVAC equipment, a loss of power, or a seismic event. A loss of building ventilation is likely to increase the temperature in rooms where loaded casks, waste packages or CSNF assemblies are present. The resulting hazard is damage to the room or cell static confinement and equipment. A thermal analysis of the process should provide an evaluation of room/cell ambient and local temperatures in case of a loss of building ventilation.

Limiting temperatures exist for concrete (65°C and 177°C, according to the code ACI-349-2001 [DIRS 158833] and steel. Appropriate criteria should be defined for cell ambient temperature to prevent a loss of static confinement (e.g., 70°C is considered for the T0 transfer cell at the La Hague facility, based on lead glass shield window and manipulator wall tube properties).

D2.6 LOSS OF BUILDING VENTILATION CAUSING A LOSS OF FUEL TRANSFER ROOM DYNAMIC CONFINEMENT

A loss of building ventilation may be caused by a failure of the HVAC equipment, a loss of power, or a seismic event. A loss of building ventilation is likely to cause a release of contamination outside the cell if the cell's differential pressure with surrounding rooms is no longer maintained. Thermal analyses of this class of events together with appropriate design control features are being evaluated.

D2.7 MECHANICAL EQUIPMENT FAILURE PREVENTING THE COMPLETION OF A PROCESS STEP

An equipment failure may cause fuel to be exposed to air longer than the normal process duration for which sensitive parameters have been calculated. Those parameters include fuel cladding temperature, cask temperature, cask and waste package pressure, and oxidation rate. The resulting hazards are similar to items 3, 4, 5, and 7 Section 7.3.3.1. A preliminary thermal analysis has been performed that provides temperatures reached after longer than expected process durations. Similar analyses and results are required for the evolution of a container's internal pressure. The impact on the oxidation rate is already considered to be time-dependent in the consequence analysis.

D2.8 CRANE DROPPING A CASK OR WASTE PACKAGE AND CAUSING A RELEASE OF CONTAMINATION

A release of contamination following the drop of a cask or waste package from a crane is currently categorized as a Category 2 event in the SAR (Section 1.7), and the consequences of breached fuel are evaluated (Section 1.8). However, the dose is not only due to the immediate release from breached fuel, but also to subsequent fuel oxidation. This issue is discussed in Section 6.7.2. Some fuel temperatures (> 400°C) are high enough to cause almost immediate oxidation. No credit shall be taken for recovery actions, as their implementation is bound to be a very time-consuming process. This event also affects the HEPA filter loading and subsequent filter change and disposal operations.

D2.9 CRANE DROPPING A CASK OR WASTE PACKAGE AND CAUSING A CRITICALITY EVENT

When fuel oxidation cannot be ruled out, the potential for the oxidized fuel to aggregate in an unfavorable geometry and result in a criticality event should be addressed. During a drop event, the potential for criticality is increased, as the quantity of breached fuel rods subject to oxidation is greater.

The criticality safety criterion is a k_{eff} lower than 0.95.

D2.10 HEAVY EQUIPMENT DROPPING ONTO A CASK OR WASTE PACKAGE CAUSING A RELEASE OF CONTAMINATION

A release of contamination following the drop of heavy equipment onto casks or waste packages is currently categorized in the SAR (Section 1.7) as a beyond Category 2 event when low lifting heights prevent damage to the fuel inside the container, or as a Category 2 event when drop heights exceed the allowable limits. An additional consequence to be considered is the release of contamination from the oxidation of the breached fuel.

D2.11 HEAVY EQUIPMENT DROPPING ONTO A CASK OR WASTE PACKAGE AND CAUSING A CRITICALITY EVENT

When the drop of heavy equipment causing damage to the fuel is not a beyond Category 2 event and when fuel oxidation cannot be ruled out, the potential for the oxidized fuel to aggregate in an unfavorable geometry and result in a criticality event should be addressed. During a drop event, the potential for criticality is increased, as the quantity of breached fuel rods subject to oxidation is greater.

The criticality safety criterion is a k_{eff} lower than 0.95.

D2.12 DROP/COLLISION OF A CSNF ASSEMBLY BY THE FUEL TRANSFER MACHINE CAUSING A RELEASE OF CONTAMINATION

A release of contamination following the drop or collision of an assembly being handled by the fuel transfer machine is currently categorized as a Category 1 event in the SAR (Section 1.7), and the consequences of breached fuel are evaluated (Section 1.8). However, the dose is not only due to the immediate release from breached fuel, but also to subsequent fuel oxidation. Some fuel temperatures (> 400°C) are high enough to cause almost immediate oxidation. No credit shall be taken for recovery actions, as their implementation is bound to be a very time-consuming process. This event also affects the HEPA filter loading and subsequent filter change and disposal operations.

D2.13 DROP/COLLISION OF A CSNF ASSEMBLY BY THE FUEL TRANSFER MACHINE CAUSING A CRITICALITY EVENT

When fuel oxidation cannot be ruled out, the potential for the oxidized fuel to aggregate in an unfavorable geometry and result in a criticality event should be addressed. During a drop or collision event, the potential for criticality is increased, as the quantity of breached fuel rods subject to oxidation is greater.

The criticality safety criterion is a k_{eff} lower than 0.95.

D2.14 HANDLING EQUIPMENT DROPPING ONTO A CSNF ASSEMBLY CAUSING A RELEASE OF CONTAMINATION

A release of contamination following the drop of handling equipment onto a CSNF assembly is currently categorized as a Category 2 event in the SAR (Section 1.7), and the consequences of breached fuel are evaluated (Section 1.8). However, the dose is not only due to the immediate release from breached fuel, but also to subsequent fuel oxidation. Some fuel temperatures (greater than 400°C) are high enough to cause almost immediate oxidation. No credit shall be taken for recovery actions as their implementation is bound to be a very time-consuming process. This event also affects the HEPA filter loading and subsequent filter change and disposal operations.

D2.15 CASK OR WASTE PACKAGE TROLLEY DERAILMENT CAUSING CASK OR WASTE PACKAGE TIPOVER AND FUEL DAMAGE (CONTAMINATION, CRITICALITY)

This hazard is prevented by design and categorized as a beyond Category 2 event.

NOTE: Fire is not considered as an initiating event; according to the fire analysis methodology, it is already prevented by design from resulting in damage to the waste form (BSC 2004 [DIRS 171488]).

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APPENDIX E
FUEL CHARACTERIZATION BASED ON FUEL INSPECTION RECORDS

E1 PURPOSE

The purpose of this study was to characterize the quantities and types of fuel failures based on fuel inspections at commercial plants. This information will be used to better understand the failure characteristics of the fuel to be received at the repository and validate numbers used in surface facility studies (Eric R. Siegmann at Framatome-ANP).

E2 SCOPE

Fuel inspection reports and records for past poolside inspections will be reviewed. Information to be identified is:

1. Inspection techniques
2. Number of failed assemblies and rods present at the refueling
3. Failed rods per failed assembly
4. Assembly reconstitution or recaging
5. Failure characteristics.

E3 RESULTS

A series of fuel inspection reports and records were reviewed to collect statistics on failed fuel characteristics. Data from both PWRs and BWRs were reviewed and most of the inspection work was on the Framatome supplied fuel. The specific plants or utilities will not be identified because the specific fuel inspection records are company confidential and cannot be released without utility approval.

E3.1 INSPECTION TECHNIQUES

Through the monitoring of primary water chemistry and cover gas, the utilities know if there is any failed fuel in the reactor when they start a refueling. If the number of failed rods is small, they may also know how many rods are failed and what batch they are from. Most utilities do not have the staff or equipment to perform fuel inspections and generally the fuel supplier performs the fuel inspection work if any is required. The simplest and minimum inspection is fuel sipping. The fuel is placed in a mast and after a change in pressure; the water and cover gas above the assembly are sampled for fission products. Sipping may identify whether an assembly contains a failed rod or rods but not which rod is failed or the cause of the failure.

The assemblies could then be inspected. Sometimes the failed rods are on the outside of the assembly and can be identified by visual inspection. These signs include hydride discoloration or blisters on the cladding, cracks, and colored plumes from fuel and fission gas wash out. Failures of interior rods might not be visible. The failed rods can sometimes be identified by viewing the top of the assembly because the failed rods have an accelerated axial growth from the additional hydrogen in the cladding.

Ultra-sound inspection (UT) is a common method to detect which rod is failed. UT probes are slid between the rods of the assembly and the signal dissipates if water is present in the bottom of the fuel rod. The presence of moisture allows the high frequency signal to transfer from the cladding to the fuel pellet, which dissipates the signal that is returned to the inspection probe. If the rod fails right before the refueling or has tight contact between the pellet and cladding, a false reading could occur. In the first case, little water might enter the rod. In the second case, the good contact might give a false failure reading.

Eddy current inspections are a more reliable way of detecting rod failure. To use this method, however, the rod must be removed from the assembly. Eddy current inspections can identify the size of the cladding defects or degree of partial cladding penetration.

Most utilities do sipping to identify failed assemblies. Shortened outage durations have put pressure on the abilities of utility fuel groups to perform failure cause analysis. Some only look at peripheral rods and do not bother with a detailed fuel characteristic study. Some will do UT to identify which rods are failed and do visual inspections of the outer rods. Fewer utilities had actually pulled the damaged rods and conducted eddy current tests on the rods and full visual inspections. Recent Institute of Nuclear Power Operations activities, including fines for operation with fuel failures and pressure to ascertain fuel failure causes, may result in more detailed failed fuel inspections.

E3.2 NUMBER OF FAILED ASSEMBLIES PRESENT AT THE REFUELING

Table E-1 summarizes the UT inspections performed by Framatome-ANP from 1986 through 1998. In that period, over 6,600 assemblies were inspected and a failure rate of 3.5 percent was identified. As noted earlier, this failure rate might be too high (conservative) because fuel without failure indication would not be inspected. Also, some known failures were not inspected (a nonconservative effect). The rod failure data in Table E-1 show a rod failure rate of 0.026 percent. These failure rates are consistent with the rates reported (Section 3) that were generated with general EPRI and DOE data. In that study the overall assembly failure rate was 3.3 to 4.4 percent. 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 rate defined by 0.01 and 1 percent (producing a median failure rate of 0.1 percent) (BSC 2004 [DIRS 172895]). The rod failure rate developed in this report is in the range used in the TSPA.

Table E-1. Summary of Framatome-ANP UT Inspections through 1998

Plant ID	Number of Assemblies Inspected	Number of Failed Fuel Assemblies	Number of Rods Inspected	Number of Failed Rods
12	117	13	24,336	16
1	94	22	19,552	51
13	104	8	21,632	8
11	177	2	36,816	2
12	396	26	82,368	38
17	154	10	32,032	13
13	181	4	37,648	4
17	178	3	37,024	4

Table E-1. Summary of Framatome-ANP UT Inspections through 1998 (Continued)

Plant ID	Number of Assemblies Inspected	Number of Failed Fuel Assemblies	Number of Rods Inspected	Number of Failed Rods
5	220	21	45,760	28
1	177	5	36,816	6
1	25	5	5,200	19
11	377	2	78,416	2
6	124	5	25,792	5
12	180	3	37,440	4
1	178	11	37,024	14
5	187	23	38,896	64
2	206	0	54,384	0
13	177	4	36,816	5
11	177	2	36,816	2
9	76	1	20,064	1
19	72	1	19,008	1
17	173	10	35,984	18
13	177	1	36,816	1
5	164	2	34,112	2
11	177	2	36,816	2
3	76	1	20,064	1
9	136	1	35,904	1
6	182	2	37,856	2
12	178	2	37,024	2
2	184	1	48,576	1
1	181	1	37,648	2
13	177	16	36,816	22
17	193	3	40,144	8
9	193	2	50,952	3
5	157	1	32,656	1
6	176	2	36,608	2
1	2	2	416	2
13	177	10	36,816	11
9	193	2	40,144	2
13	177	0	36,816	0
Total	6,650	232	1,436,008	370
Percent failed		3.5		0.026

In an inspection campaign, a total of 331 assemblies at one reactor, both in the spent fuel pool and in a cycle offload, were inspected. These assemblies contained 68,848 rods. A total of 12 rods in 9 assemblies were found to be failed. For this plant, the assembly failure rate was 2.7 percent and rod failure rate was 0.02 percent. These failure rates are also consistent with the rates generated with DOE and EPRI data (Section 3).

E3.3 FAILED RODS PER FAILED ASSEMBLY

The above data and discussion are for fuel assemblies. Assemblies have different numbers of rods in them; therefore, the rod reliability is different than the assembly reliability. PWR assemblies vary in design with the earlier plants having 14 × 14 rods (about 164 rods per assembly since not all locations have fuel rods in them). The newer designs are 17 × 17 (about 264 rods per assembly). The number of rods in a PWR assembly cannot be readily 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 adjacent 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 of an assembly to fail. In one case of grid fretting, 32 failed rods were identified in one PWR assembly. In that same batch of fuel was an assembly with 25 failed rods. Manufacturing failures tend to be single rod failures, though some manufacturing events have lead to the failure of many rods in many assemblies, as in the case of a bad batch of cladding. For the early period (<1986), the number of rods failed per failed assembly averaged 2.2 (EPRI 1997 [DIRS 100444]) but this has decreased to 1.4 rods per PWR assembly (Yang et al. 1991 [DIRS 125302]). Table E-1 shows that for the fuel surveyed by Framatome through 1998, the number of failed rods per failed assembly averaged 1.6 rods per assembly. This is consistent with Yang et al. 1991 [DIRS 125302]). A conservative number of failed rods per failed assembly is 2.2 and can be used for future calculations. As noted earlier, this failure rate might be too high (conservative) because fuel without failure indication would not be inspected. Also, some known failures were not inspected (a nonconservative effect).

A survey of the fuel inspection reports was performed to characterize the number of failed rods in each failed assembly. A survey of 222 failed assemblies with information on failed rod failures found that the average assembly contained 1.4 failed rods. This is consistent with the EPRI (1997 [DIRS 100444]) and Yang et al. (1991 [DIRS 125302]) estimates of 2.2 to 1.4 rods per assembly and shows that the use of the earlier value of 2.2 failed rods per failed assembly is conservative. Table E-2 gives the breakdown. Over 90 percent of the failed assemblies had one or two failed rods.

Table E-2. Number of Rods Failed per Assembly from the Survey of 222 Failed Assemblies

Failed Rods/Failed Assemblies	% of Assemblies
1	76
2	15.8
3	4.1
4	1.8
5	0.5
6	0.9
8	0.9

E3.4 ASSEMBLY RECONSTITUTION OR RECAGING

Some fuel assemblies can be reconstituted or recaged. When reconstituting a PWR assembly, the damaged rod is replaced with either an inert rod or uranium rod. For BWRs, the damaged rod is replaced with a low enriched rod or a rod salvaged from a discharged assembly. The fuel cage (or skeleton) is the frame of grids (spacers), instrument tube, guide tubes (for control rods), and upper and lower endfittings. The fuel rods are inserted into this cage after removing the upper endfitting. Recaging occurs when the undamaged rods are wanted for reuse and either the damaged rod cannot be removed (rod is broken) or the cage is damaged. For fuel that is scheduled to go back into the reactor, reconstitution is most common. For assemblies being discharged, reconstitution occurs at times if the damaged rod is being removed for root cause studies. Table E-3 lists the assemblies that were reconstituted during a nine-year period. While most of the assemblies were first cycle failures, some latter cycle assemblies were reconstituted. Table E-4 gives the history of recaging for a nine-year period and shows that recaging is less frequent. Using the data in Tables E-1, E-3, and E-4, it is estimated that 65 percent of the failed assemblies are either reconstituted or recaged based on the historical frequency of these events (see Table E-5). This reconstitution rate could be high because the data reviewed had two specific events that led to a large number of reconstitutions. Reconstitution varies among utilities in that one utility reconstitutes all failed assemblies, while another rarely does.

Some early assembly designs were not able to be easily reconstituted. Westinghouse and Babcock and Wilcox were not able to reconstitute before the mid 1980s when removable top nozzles were introduced. Combustion Engineering designs were always able to be reconstituted.

The survey of recent fuel inspection reports identified only 31 assemblies that were reconstituted or recaged. While this number seems low, the reconstitution normally takes place after the fuel inspection and is not always documented in the fuel inspection report. Most of the time, the reports concentrate on the characteristics of the failed rods and not on its disposition.

Table E-3. Reconstituted Assemblies from 10/91 through 6/01

Fuel Assembly ID	Root Cause	After Cycle	Use
542	not determined	1	reuse
4TH	no defect	2	reuse
4T0	no defect	2	reuse
59A	no defect	1	reuse
591	no defect	1	reuse
59D	no defect	1	reuse
4FA	2 debris failures	3	root cause
4EE	1 fretting failure	3	root cause
4CW	no defect	2	reuse
4C6	1 debris failure	3	root cause
4K1	1 debris failure	3	root cause
5JK	unknown	1	reuse
5JW	fretting	2	not used
5KZ (K33)	1 debris failure	1	reuse

Table E-3. Reconstituted Assemblies from 10/91 through 6/01 (Continued)

Fuel Assembly ID	Root Cause	After Cycle	Use
6HN (T39)	1 debris failure	1	reuse
6DD	no defect found	1	reuse
5RZ	no defect found	2	reuse
579	no defect found	3	root cause
4MZ	1 debris failure	4	root cause
4LV	1 debris failure	4	root cause
6UX (AA36)	1 debris failure	1	reuse
6N2	1 unknown	1	reuse
6N5	1 unknown	1	reuse
6KZ	1 unknown	1	reuse
550	fretting	3	root cause
5UR	no defect found	2	root cause
71L	1 unknown	1	reuse
71Z	2 unknown	1	reuse
72C	1 unknown	1	reuse
72E	3 unknown	1	reuse
72G	1 unknown	1	reuse
72V	1 unknown	1	reuse
72Y	4 unknown	1	reuse
72Z	1 unknown	1	reuse
730	1 unknown	1	reuse
731	1 unknown	1	reuse
736	1 unknown	1	reuse
6AG	1 unknown	2	reuse
6CG	no defect	2	reuse
71G	crud, no failures	1	reuse
71F	crud, no failures	1	reuse
71E	crud, no failures	1	reuse
71D	crud, no failures	1	reuse
71A	crud, no failures	1	reuse
70P	crud, 1 failed rod	1	reuse
70N	crud, no failures	1	reuse
70K	crud, no failures	1	reuse
70J	crud, no failures	1	reuse
70H	crud, no failures	1	reuse
70G	crud, 6 failed rods	1	reuse
70F	crud, no failures	1	reuse
70E	crud, no failures	1	reuse
70D	crud, no failures	1	reuse
6ZV	crud, no failures	1	reuse
6ZU	crud, no failures	1	reuse
6ZP	crud, no failures	1	reuse

Table E-3. Reconstituted Assemblies from 10/91 through 6/01 (Continued)

Fuel Assembly ID	Root Cause	After Cycle	Use
6ZG	crud, no failures	1	reuse
6ZC	crud, 1 failed rod	1	reuse
5ZZ	crud, no failures	1	reuse
5ZY	crud, no failures	1	reuse
5ZV	crud, 1 failed rod	1	reuse
5ZP	crud, no failures	1	reuse
5ZJ	crud, no failures	1	reuse
51D	no defect found	3	root cause
719	crud, no failures	1	reuse
602	crud, no failures	1	reuse
601	crud, no failures	1	reuse
600	crud, no failures	1	reuse
5KL (K22)	1 creep collapse	3	root cause
6K6	no defect found	2	reuse
7KR	1 unknown	1	reuse
7PJ	repaired grid	1	reuse
7P6	repaired grid	1	reuse
7P1	repaired grid	1	reuse
750	repaired grid	2	reuse
74U	repaired grid	2	reuse
73U	repaired grid	2	reuse
65C	slipped grid	3	test equipment
7R8	1 unknown	1	reuse
72K	1 unknown	2	reuse
71P	1 unknown	2	reuse
6C7	pilgering defect	3	root cause
5YU	2 high bumup rods	2	hot cell exam
K45	8 high bumup rods	3	hot cell exam
AB26	1 unknown	3	root cause
AB36	no defect found	3	root cause
AB26	1 unknown	3	root cause
AB36	no defect found	3	root cause
1003	repaired grid	1	reuse
9WW	repaired grid	1	reuse
9WE	repaired grid	1	reuse
K17	no defect found	3	root cause
AD06	no defect found	2	reuse
AD55	no defect found	2	reuse
AD61	no defect found	2	root cause

Table E-4. Recaged Assemblies from 1/98 through 11/96

Old Fuel Assembly ID	New Fuel Assembly ID	Root Cause	After Cycle
4CX	4CX	1 debris failure	1
4HD	538	2 fretting failures	1
41B	539	1 debris failure	2
1F6	53A	5 debris failure	1
488	53F	1 debris failure	1
48N	5PA	1 debris failure	1
48X	5PE	unknown	1
48Z	5PC	1 debris - 1 unknown	1
4A2	5PD	1 debris failure	1
4RW	53D	end fitting damaged	2
5WJ (V27)	5PG(RC01)	unknown	1
44J(G69)	5PM(S49)	nfi requested cage	3
53L	RAA	unknown	1
71N	5PK	unknown	1
73C	5PJ	unknown	1
776	5PU	damaged fuel assembly	1
7L0	5PV	unknown	1
7R1	5PX	weld contamination	1
7PF	536	damaged grid	1
7RT	5PW	damaged grid	1
7T0	8HH	damaged grid	1
7T1	8HJ	damaged grid	1
7TM	8HK	damaged grid	1

Table E-5. Estimation of Fraction of Failed Assemblies that are Reconstituted or Recaged

Operation	Record Start	Record End	Number of Assemblies	Rate, Assembly/yr
Reconstituted	09/1991	06/2001	95	8.84
Recage	01/1989	11/1996	23	2.58
Failed Assemblies	06/1986	09/1998	232	17.51
Percent reconstituted or recaged				65%

E3.5 FAILURE CHARACTERISTICS

The size of the defect (pinhole or hairline crack or larger) depends on the cause of the failure, the location of the failure, and the timing of the failure.

The cause of the failure affects both the size of the initial cladding penetration and timing of the failure. Two failure mechanisms, manufacturing defects and pellet cladding interaction (PCI), tend to produce hairline cracks or pinholes, but these tend to enlarge from secondary failure, including unzipping. The other failure mechanisms tend to cause larger defects. The following is a brief discussion of the failure mechanisms.

1. Manufacturing Defects—These failures are normally small cracks that are caused by weld failure or faults in the cladding. They tend to occur in the first cycle. The frequency of these failures has decreased as manufacturing techniques have improved. These failures could be characterized as pinhole or hairline crack if no secondary degradation occurs. Figure E-1 shows an example of a manufacturing defect, a faulty end cap weld.

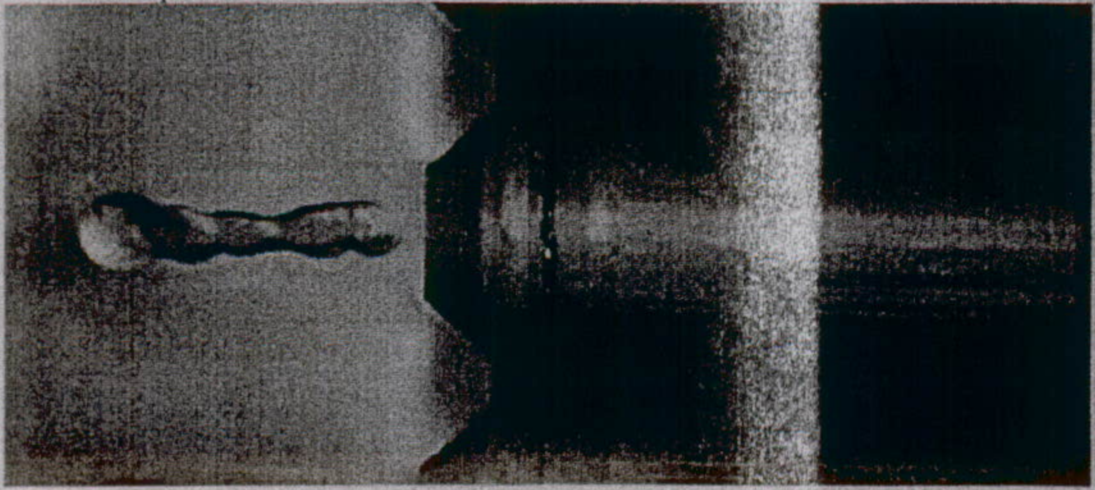


Figure E-1. Example of End Cap Weld Failure

2. Grid Fretting—These failures in PWRs tend to be larger than a pinhole or hairline crack because the fretting continues after rod failure. They tend to occur in the outer row if they occur from flow-induced vibration, such as cross flow from loss-of-coolant-accident holes in the baffle. These tend to occur in later fuel cycles primarily due to fuel cycle designs that place high burnup fuel in locations most susceptible to the fretting phenomenon. This problem is being addressed with better grid designs but has not yet been completely eliminated. Figure E-2 shows an example of grid fretting.

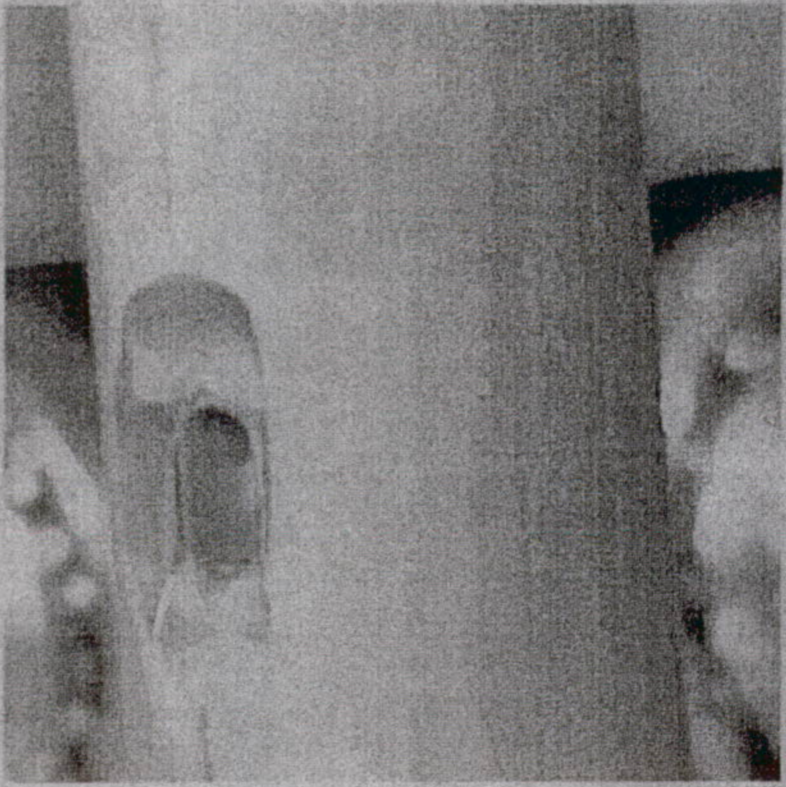


Figure E-2. Example of Grid Fretting

3. Debris Fretting—These failures tend to be larger than a pinhole or hairline crack because the fretting continues after rod failure. They can occur in pairs because the debris can vibrate between two adjacent rods. This problem has been addressed with better primary system maintenance and the introduction of debris screens at the bottom of the assemblies. Its frequency has decreased over the years but not eliminated. Figure E-3 gives an example of debris fretting.

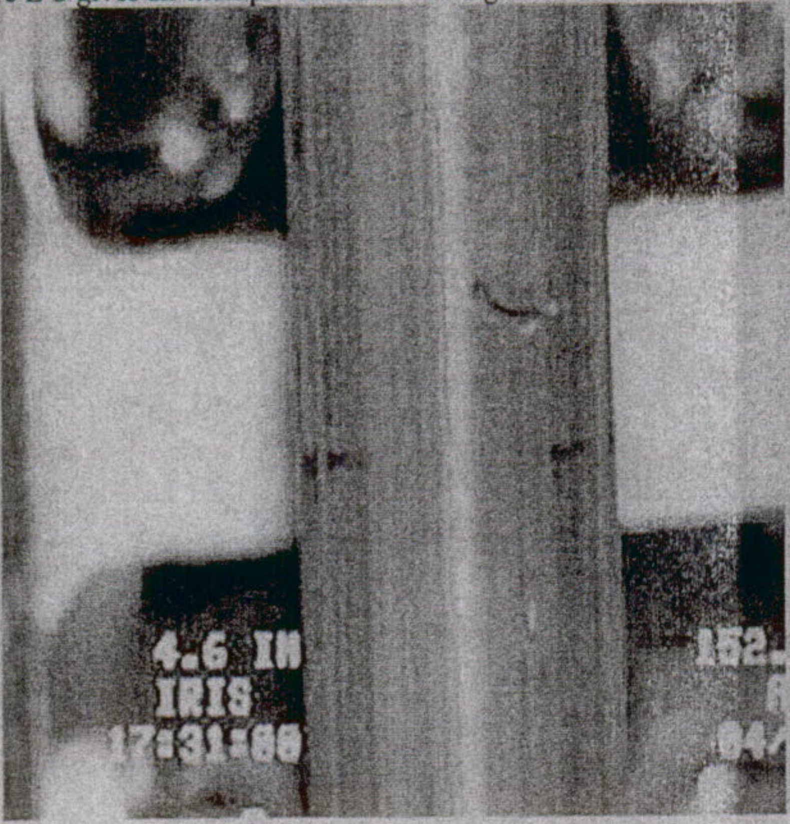


Figure E-3. Example of Debris Fretting

4. Pellet Cladding Interaction—PCI normally is associated with BWRs but recently is believed to have occurred in some PWRs. PCI occurs during rapid power changes when pellet expansion stresses the cladding. It produces small hairline cracks in the cladding that, if no secondary failures occur, could be characterized as a leaker and not placed in damaged fuel cans. As discussed later, secondary degradation usually occurs and produces additional cracks and blisters. Figure E-4 gives two examples of PCI cracks.

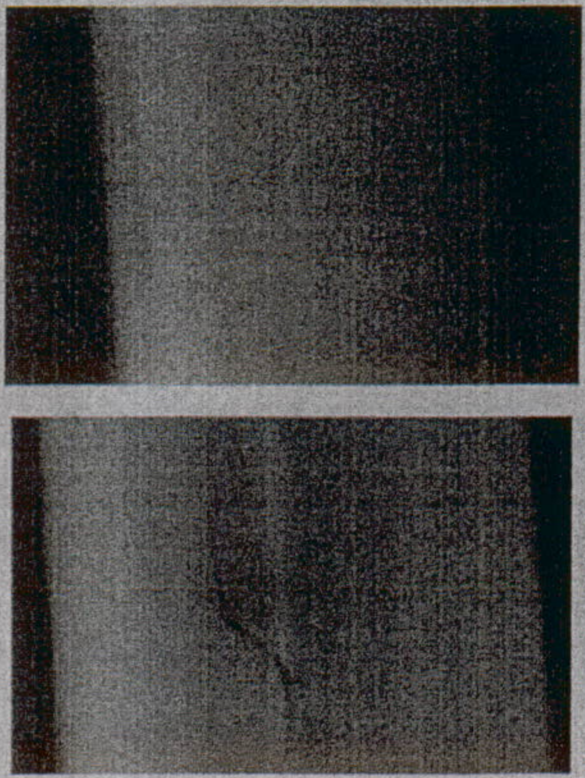
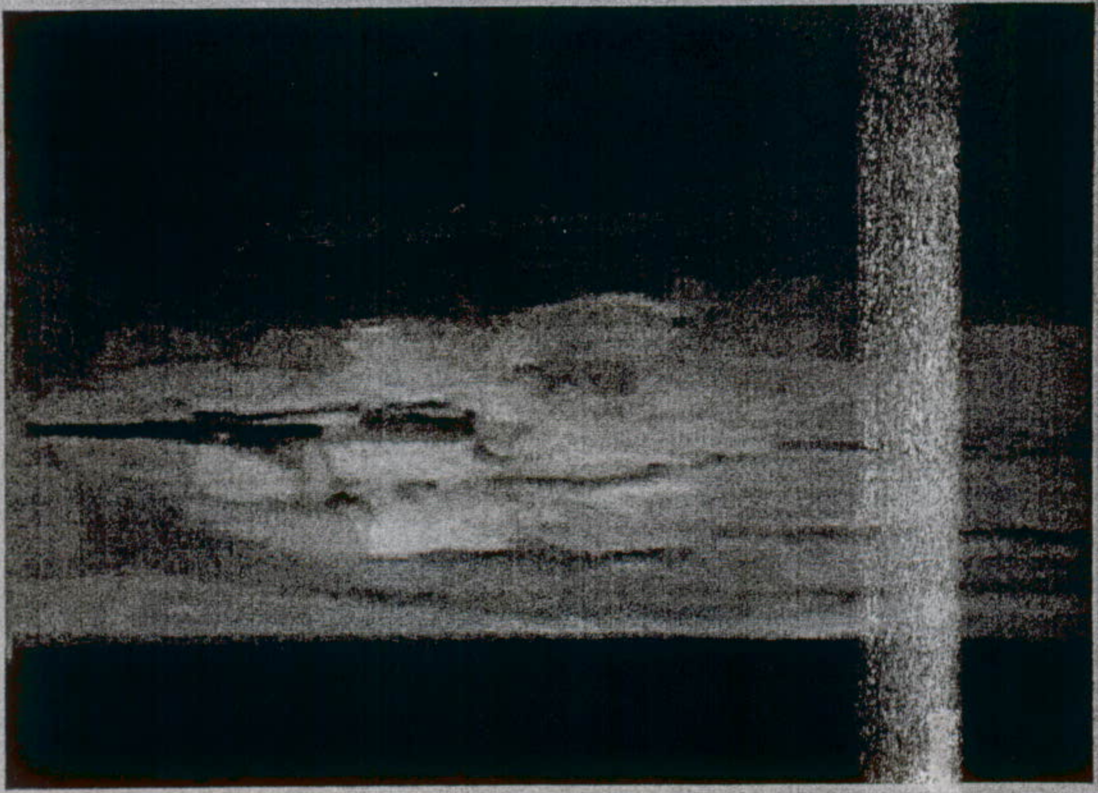


Figure E-4. Examples of PCI Cracks

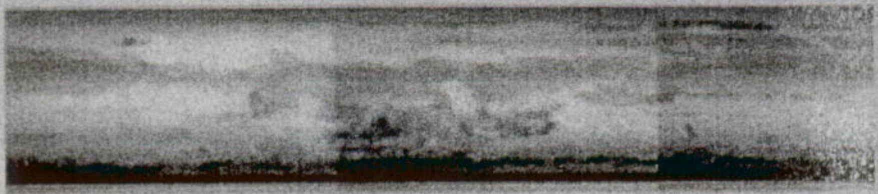
5. Crud and Corrosion—This failure tends to occur in BWRs but has been seen at PWRs as well. It usually occurs because the water chemistry is not maintained correctly and results in accelerated cladding corrosion in larger areas than pinholes because it occurs under larger crud deposits. It was common in early BWRs that had copper in their condensers, but this was corrected with condenser retubing. It has recently occurred in a few BWRs and is being studied. Figure E-5 gives two examples of corrosion failures.



Through-Wall Defect

300° at 117"-119.5" from BEP

Mild Oxide Damage



100° at 117.5"-120" from BEP

Figure E-5. Examples of Corrosion Failures

The location of the failure could affect how a utility would classify the failure. If the failure is on the outer row of the assembly, with visual inspection, a utility might characterize failure as being small. If the failure is in the assembly interior and cannot be characterized, the utility will have to declare the assembly as damaged without investing additional effort and cost in characterizing it.

The timing of the failure can also affect the fuel classification. If the failure does not occur right before the refueling, then there is time for secondary degradation to occur. When water enters the cladding, it can react with the fuel and cladding interior and in time, leads to the hydriding of the cladding. This will produce hydride blisters (Figure E-6), cracking (Figure E-6), axial splits (Figure E-7) and even rod breakage. Figure E-8 show the condition of seven rods that first failed from PCI. At failure, there were small hairline cracks located near the middle of the rods. As the figure shows, after discharge, there are multiple cracks, hydride blisters (some through wall), and splits.

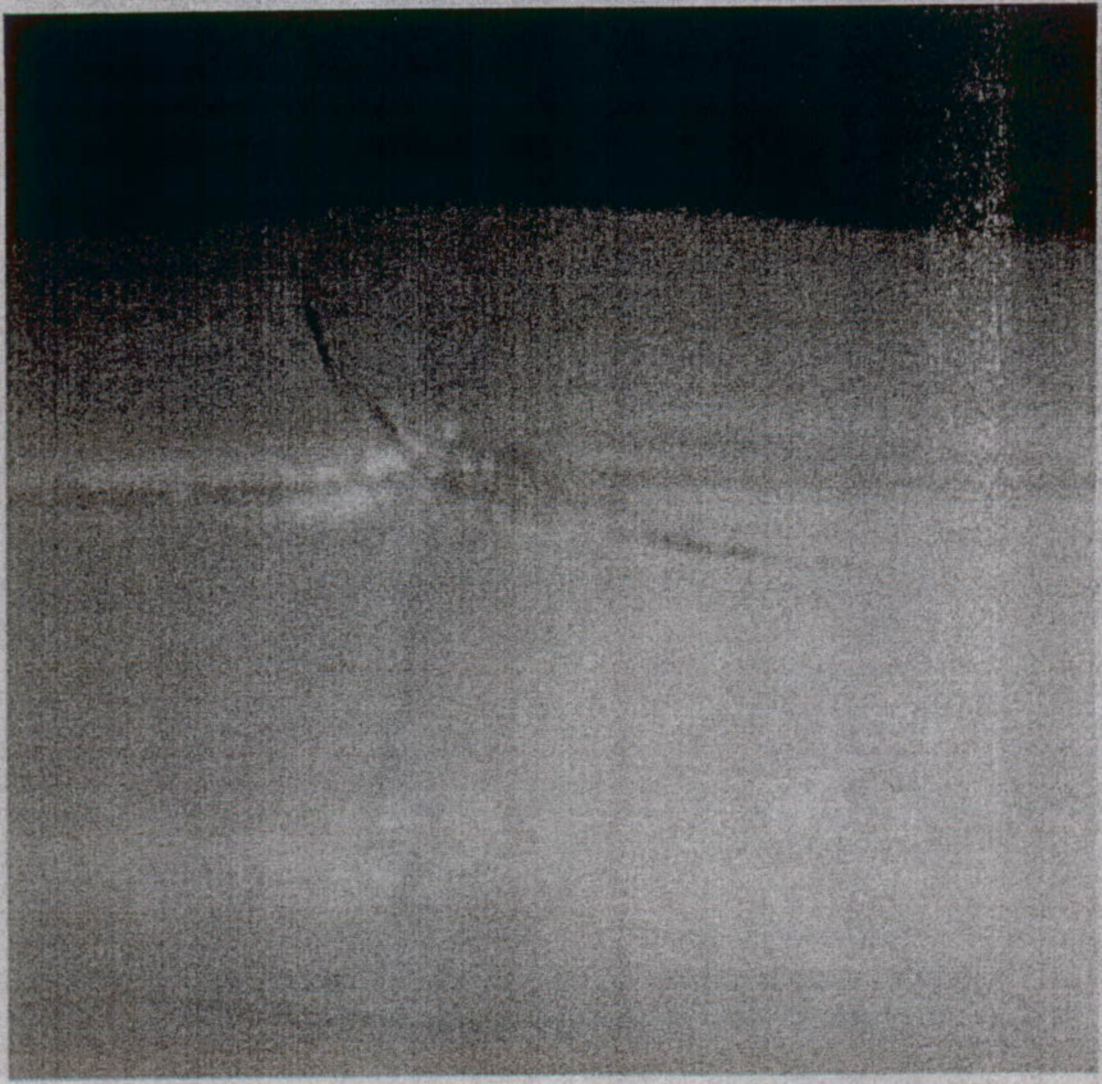


Figure E-6. Example of Hydride Blister with Cracks

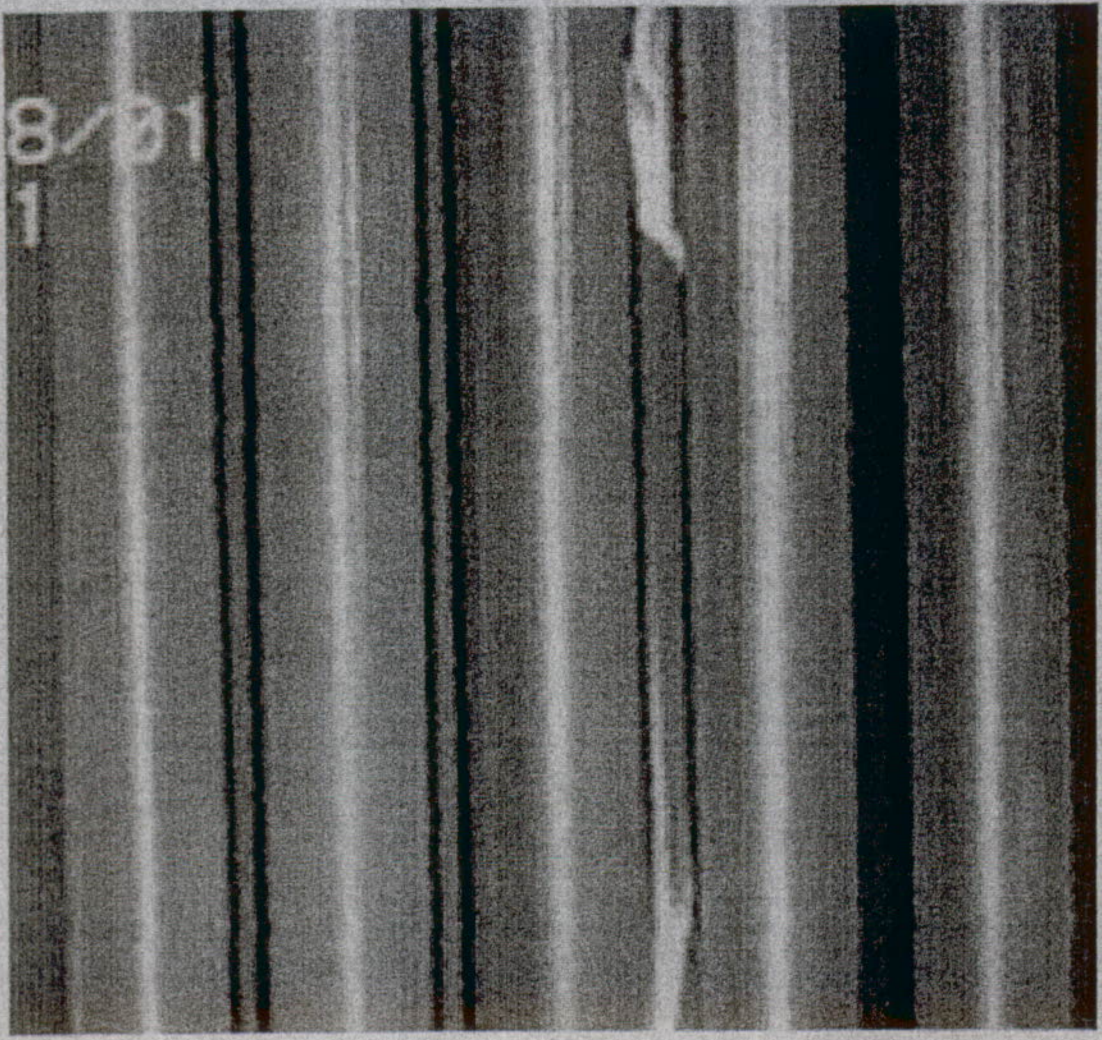


Figure E-7. Example of Axial Crack

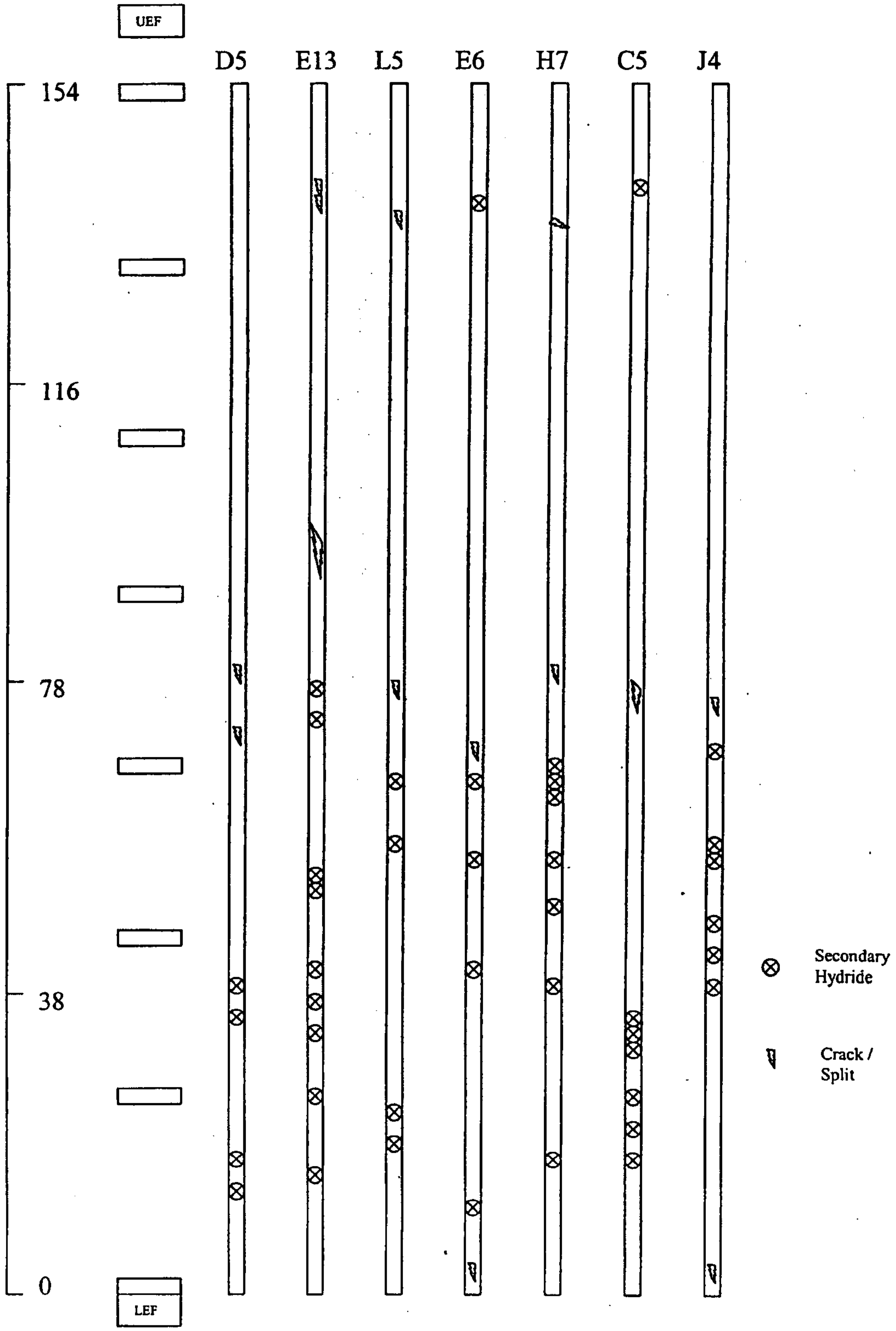


Figure E-8. Diagram Showing Secondary Failures after Initial PCI Cracking

The above discussion of the failure mechanisms, including secondary failures, suggests that most fuel failures would be characterized as damaged in terms of ISG-1 (NRC 2002 [DIRS 164018]) and would be placed in damaged fuel cans before shipping. The review of inspection records also shows that most failures would be described as larger than hairline crack or pinhole and would be classified as damaged. The earlier estimate (Section 3) is that 90 percent of the failed fuel would be characterized as damaged and seems valid.

E4 CONCLUSIONS

The review of the Framatome-ANP fuel inspection records showed an assembly failure rate of about 3.5 percent and rod failure rate of about 0.026 percent. This is generally consistent with the earlier study of DOE and EPRI data (Section 3) 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 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 eight 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 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 supports the earlier estimate that only 10 percent of the failed fuel would be characterized as intact and not placed into damaged fuel cans.