Contract No:

This document was prepared in conjunction with work accomplished under Contract No. 89303321CEM000080 with the U.S. Department of Energy (DOE) Office of Environmental Management (EM).

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NASNF Processing and Packaging Technical Study

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September 2020

SRNL-TR-2020-00313, Revision 0

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Printed in the United States of America

Prepared for
U.S. Department of Energy

SRNL-TR-2020-00313
Revision 0

Keywords: *ABD, L-Area, H-Area, Electrolytic Dissolver, Fuel, Packaging*

Retention: *Permanent*

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Prepared for the U.S. Department of Energy under contract number DE-AC09-08SR22470.

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ACKNOWLEDGEMENTS

The contributions of the following team members are greatly appreciated:
C. W. James
K. Karanth
T. S. Rudisill
W. G. Dyer
Other significant contributors include N. M. Askew and J. E. Laurinat.

EXECUTIVE SUMMARY

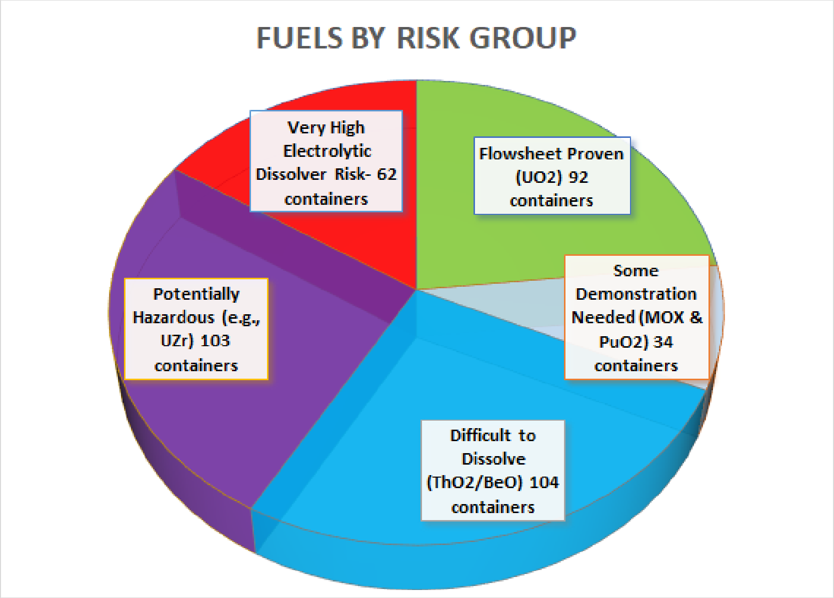
The Accelerated Basin Deinventory (ABD) Program is designed to accelerate the deinventory of L-Basin and accelerate the SNF disposition mission. Spent fuel will be dissolved in H-Canyon with no recovery of Uranium-235 (U-235). The dissolver solutions will be temporarily stored, then transferred to the tank farms and immobilized in the Defense Waste Processing Facility. ABD accelerates basin closure, significantly reduces programmatic risk and greatly reduces the lifecycle budget requirements for the site by eliminating the need for a SNF drying and packaging project. The ABD approach represents a significant change to the clean-up approach for the Savannah River Site. The ABD Program requires changes in how H-Canyon is configured and operated.

The ABD program will require development of a strategy for the dispositioning the Non-Aluminum Spent Nuclear Fuel (NASNF), some of which will be very challenging to handle and dissolve as described in this report. Not all the NASNF is suitable for processing through the electrolytic dissolver. The strategy may involve dissolution of the majority of the NASNF inventory in the electrolytic dissolver combined with some other disposition alternative(s) for the fraction that is not suitable for electrolytic dissolution. This report highlights the risks and challenges associated with the baseline approach of using the electrolytic dissolver to process all of the NASNF. In addition, the characteristics of the NASNF and how those characteristics relate to the risks associated with electrolytically dissolving the specific fuel types.

The NASNF inventory contains a variety of shapes, sizes, materials of construction, cladding and fuel composition. Currently, the ABD program assumes dissolving this material in the electrolytic dissolver, starting in FY28 and finishing in FY33. The purpose of this assessment is to bin the NASNF inventory into categories of fuel and evaluate the feasibility, risks and impacts associated with electrolytic dissolution at SRS. L-Basin contains approximately 20 MTHM (Metric Tons Heavy Metal) of NASNF with approximately 700 kg fissile content. The 395 containers are configured as 371 bundles, 23 oversized cans, and 1 bucket.

Through this assessment, the NASNF inventory is put into 5 categories based on increasing risk:

1. Flowsheet proven (UO_2) – 92 containers
2. Some demonstration needed (MOX & refractory core) – 34 containers
3. Difficult to fully dissolve (ThO_2, BeO) – 104 containers
4. Potentially hazardous (e.g. UZr) – 103 containers
5. Very high dissolving risk or high U losses in electrolytic dissolver – 62 containers



These five categories are addressed, as appropriate, based on the 10 risks identified as:

1. Physical size of NASNF bundles requires unproven “stray currents” for electrolytic dissolution
2. Impact of bundle bottom dissolution on fuel dropping and buoyancy, and bundle protrusions
3. Sludge behavior and removal following Zr-clad dissolutions
4. Other potential concerns related to ZrO_2 sludge properties, reactivity, and disposition
5. Dissolution and corrosion impacts for refractory oxide pellet fuels (NASNF categories 2 & 3)
6. UZr fuel hazardous acid reactions (NASNF category 4)
7. Limited understanding of very high risk NASNF and alternative approaches (NASNF category 5)
8. H_2 and autocatalytic reactions due to metal fines
9. Oversized items lack material characterization and may need repackaging
10. Extensive processing time and potentially unacceptable corrosion rates due to material configuration and bundling

Risks 1 through 4 are common to all NASNF processing and must be addressed for dissolution of even proven technologies (UO_2 core fuels). The issue is not whether the material could successfully be processed, but what is the potential impact associated with the risk. Understanding the risks through further research and demonstration can reduce or eliminate them. Risks 5 through 7 are unique to one or two fuel categories and have specific impacts associated with each risk and fuel type. Risks 8 and 9 involve characterization and repackaging risks common to many fuels. Risk 10 addresses the incremental impact of the current bundling and explores rebundling strategies to reduce the processing time and the expected corrosion associated with dissolving these fuels. Based on the Risk 10 assessment, it is likely that rebundling would be required to reduce the processing time and corrosion to acceptable levels, but even rebundling might not allow processing all these fuels within a four-year window, which is the current target of the project.

Based on this assessment, the fourth and fifth fuel categories pose the greatest program risk, with extensive research and demonstration needed to address the risk associated with electrolytically dissolving these materials, should that approach be adopted. However, varying levels of research, development, and demonstration of dissolution of each fuel type is required to reduce risk and develop flowsheets. Further, as noted above, given the significant potential impacts of these risks, a technical evaluation of options is warranted.

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LIST OF ABBREVIATIONS

ABD	Accelerated Basin Deinventory
ANN	Aluminum Nitrate Nonahydrate
Bq	Becquerel
CCO	Criticality Control Overpack
CV	Confinement Vessel
DOE	Department of Energy
EBC	Equivalent Boron Content
EBR	Experimental Breeder Reactor
EBWR	Experimental Boiling Water Reactor
ED	Electrolytic Dissolver
ER	Electrorefiner
ERR	Elk River Reactor
FCA	Fast Critical Assembly
FEC	Failed Equipment Can
FFTF	Fast Flux Test Facility
FG	Fuel Grade
FGE	Fissile Gram Equivalents
HEU	High Enriched Uranium
HTGR	High Temperature Gas-cooled Reactor
HWCTR	Heavy Water Component Test Reactor
HTRE	Heat Transfer Reactor Experiment
IAEA	International Atomic Energy Agency
INL	Idaho National laboratory
KAMS	K-Area Materials Storage
LANL	Los Alamos National Laboratory
LEU	Low Enriched Uranium
LPR	Linear Polarization Resistance
M	Molar
MC&A	Materials Controls and Accountability
MCO	Multi-Canister Overpack
MOX	Mixed-Oxide Fuel
mR	Millirem
mph	Mils per hour
mpy	Mils per year

MT	Metric Ton
MTHM	Metric Ton Heavy Metal
NASNF	Non-Aluminum Spent Nuclear Fuel
NCS	Nuclear Criticality Safety
Np	Neptunium
OSC	Over Sized Can
ORNL	Oak Ridge National Laboratory
PG	Power Grace
PNNL	Pacific Northwest National Laboratory
PTFE	Teflon
Pu	Plutonium
RH	Remote Handled
SCFM	Standard Cubic Feet per Minute
SISC	Short Inner Storage Can
SS	Stainless Steel
S&S	Safeguards and Security
SRE	Sodium Reactor Experiment
SRNL	Savannah River National Laboratory
SRS	Savannah River Site
TD	Theoretical Density
TFEN	Tubular Fuel Element- Natural
TRU	Transuranic
U	Uranium
UN	Uranyl Nitrate
UPuZr	An alloy containing 70% U, 20% Pu and 10% Zr, by weight
WAC	Waste Acceptance Criteria
WG	Weapons Grade
WIPP	Waste Isolation Pilot Plant
µg	Microgram
Zr	Zirconium

1.0 Introduction

The Accelerated Basin Deinventory (ABD) Program is designed to accelerate the deinventory of L-Basin and accelerate the SNF disposition mission. Spent fuel will be dissolved in H-Canyon with no recovery of Uranium-235 (U-235). The dissolver solutions will be temporarily stored, then transferred to the tank farms and immobilized in the Defense Waste Processing Facility. ABD accelerates basin closure, significantly reduces programmatic risk and greatly reduces the lifecycle budget requirements for the site. The ABD approach represents a significant change to the clean-up approach for the Savannah River Site. The ABD Program requires changes in how H-Canyon is configured and operated.

The ABD program will require development of a strategy for the dispositioning the Non-Aluminum Spent Nuclear Fuel (NASNF), some of which will be very challenging to handle and dissolve as described in this report. Not all of the NASNF is suitable for processing through the electrolytic dissolver. The strategy may involve dissolution of the majority of the NASNF inventory in the electrolytic dissolver combined with some other disposition alternative(s) for the fraction that is not suitable for electrolytic dissolution. This report highlights the risks and challenges associated with the baseline approach of using the electrolytic dissolver to process all of the NASNF. This report provides information about the characteristics of the NASNF and the processing challenges and risks associated with handling and processing via the electrolytic dissolver.

The NASNF inventory contains a variety of shapes, sizes, materials of construction, cladding and fuel composition. To support the processing of Fast Critical Assembly (FCA) fuel from Japan, H Canyon is resurrecting the 6.3D Electrolytic Dissolver (ED). After completion of the FCA campaign, the 6.3D ED is available to process other NASNF assemblies in L-Basin. The purpose of this assessment is to bin the inventory into fuel types and evaluate the feasibility, risks and impacts associated with electrolytic dissolution of NASNF at SRS. L-Basin contains approximately 20 MTHM (Metric Tons Heavy Metal) of NASNF with approximately 700 kg fissile content. The 395 containers are configured as 371 bundles, 23 oversized cans, and 1 bucket.

1.1 Background

The electrolytic dissolver (Figure 1-1) development started with initial studies and small-scale testing taking place in 1958-1959 followed by full-scale pilot testing between 1960-1968. The working dissolver was installed in H-Canyon in 1968. After installation, the electrolytic dissolver in H-Area successfully operated from 1969 to 1980. During that period, 33 MTU (Metric Tons Uranium) and approximately 16 MT of metal cladding (stainless steel (SS), zirconium, nickel-chromium, Incaloy) and 1213 metal bundles (SS and aluminum) was processed in 182 batches. Pickett (1988) provides a summary of the fuels processed and the processing rates attained.[1] Figure 1-2 shows the wide variety of UO_2 core fuel types processed through 1980.

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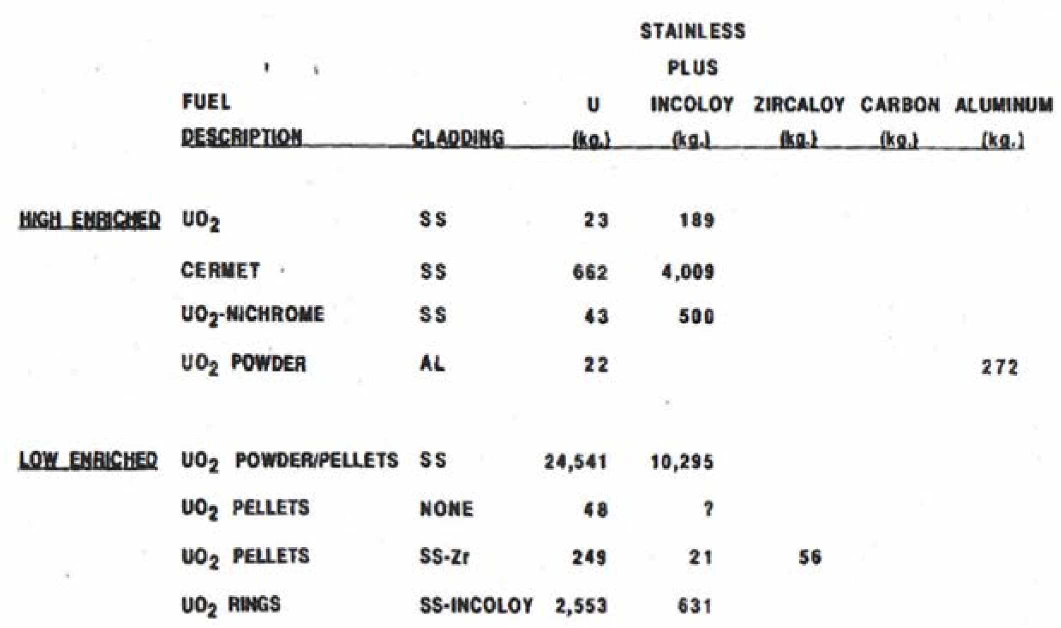
[1] A recent unpublished review found the Pickett summary failed to count 10 MT of SS, a few MT of uranium, and 75 kg of Zr clad fuel. Thus, the values cited above were from the recent CAB report, which corrected for these omissions.

Figure 1-1. H-Canyon Electrolytic Dissolver Top View.



	FUEL DESCRIPTION	CLADDING	U (kg.)	STAINLESS PLUS INCOLOY (kg.)	ZIRCALOY (kg.)	CARBON (kg.)	ALUMINUM (kg.)
HIGH ENRICHED	UO_2	SS	23	189			
	CERMET	SS	662	4,009			
	UO_2-NICHROME	SS	43	500			
	UO_2 POWDER	AL	22				272
LOW ENRICHED	UO_2 POWDER/PELLETS	SS	24,541	10,295			
	UO_2 PELLETS	NONE	48	?			
	UO_2 PELLETS	SS-Zr	249	21	56		
	UO_2 RINGS	SS-INCOLOY	2,553	631			

Figure 1-2. Types of Fuels Processed in the H-Canyon Electrolytic Dissolver through 1980.

Although large quantities of stainless steel clad NASNF were processed, only 212 kilograms of zirconium-clad fuel were processed.[2] Although zirconium-clad fuel dissolution has been demonstrated, the zirconium dissolution rate is half that of stainless steel and sludge buildup is much greater than for stainless steel clad fuel. The generally recognized capacity for stainless steel clad tubular fuel is 50 kg/day while the dissolution rates for aluminum and zirconium are 30 and 25 kg per day, respectively.[3] These rates were based on fuels that fit within the niobium basket, so rates for current packaged fuels would be even lower because they would not fit into the active zone of the Nb basket and thus rely on stray currents to dissolve. The previous fuels processed in the 70s were mostly stainless steel bundled. In contrast, almost all the fuel today is aluminum bundled. Processing rates are projected to be low, as discussed in more detail in Section 6.0. Bull

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[2] A recent unpublished review of these data found a total of 212 kg of Zr dissolved rather than the 56 kg reported in Figure 1-2.

[3] See DP-896 (1964), Pickett (1988), Bull and Koonce (1970), and DPSTM-200-HM: the predicted rates for SS, Al, and Zr metal dissolution in the Electrolytic Dissolver are, respectively, 0.5, 0.3, and 0.25 grams per amp-hour. However, for tubular fuel, the rates fall to 0.36, 0.22, and ~0.18 grams per amp-hour. See also Dyer (2019), Table 1, rows 3&4 avg. for SS bundled tubular is 50 kg/day. Lower Al bundled rates discussed later.

and Koonce (1970) estimated the overall time at-current, after adjusting for charging, probing, and transferring, was 86% of an operating hour. Hence, at a minimum, for every hour spent at-current there would be 16% of an hour for operations not at-current, assuming no other downtime. Assuming 40 operating weeks per year and 70% attainment, there would be 4,000 hours at-current for the ED. While only a few percent of the stainless-steel cladding formed solids, 85% of the Zircaloy is converted to an oxide resulting in about 0.33 gallons of sludge settling to the tank bottom per kilogram of zirconium dissolved.

1.2 Key Principles of Operation

The dissolver is a large, eight-foot diameter vessel with the electrochemical cathode/anode assembly in the center and cooling coils to remove heat and air sparging to provide solution mixing and dilution of off-gas to meet safety requirements. The design allows dissolution of up to three bundles at a time through three charging chutes into a three-partitioned niobium basket where the dissolution mostly takes place. Spray nozzles and a slightly tapered bottom facilitate removal of solids that leave the basket and collect in the vessel. The initial nitric acid concentration is 10 M. Between dissolutions of a fuel batch, nitric acid may be added when the dissolver is shut down to maintain the dissolver concentration above 7 M, which is required to minimize corrosion of the niobium components and minimize formation of hydrogen.

Electrolytic dissolution occurs within a niobium basket insert located between a platinum-faced cathode and anode, which do not dissolve. A simplified electrochemical cell is shown in Figure 1-3. The metal cladding and bundles dissolve electrochemically. Metallic cores dissolve electrochemically and chemically after they are exposed while dissolution of oxide fuel cores is solely chemical. As previously mentioned, dissolution of UO_2 core fuels has been demonstrated, but less than 25% of the NASNF requiring processing contains UO_2 cores. Hence, demonstration is needed, and additional processing time is expected for the other fuel core types.

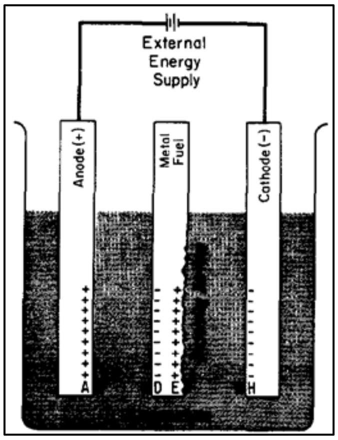


Figure 1-3. Simplified Electrochemical Cell.

1.3 Planned Use of the Electrolytic Dissolver for NASNF

The NASNF dissolution will be concurrent with dissolution of ASNF in the chemical dissolver except during the last year. The plan assumes dissolution starts in FY28 with NASNF intact, UO_2 core fuels that do not require repackaging. The processing of fuels requiring repackaging and flowsheet demonstration would begin in FY29. The baseline approach assumes the majority of the bundles (nominal 5” diameter) do not require repackaging and that oversized fuels, which typically contain failed or damaged fuel, need minimal repackaging.

1.4 Overview of Fuels and Packaging

Through this assessment, the NASNF inventory is put into 5 categories based on risk (Figure 1-4). Further discussion on each category is provided throughout this report.

1. Flowsheet proven (e.g. UO_2) – 92 containers
2. Some demonstration needed (e.g. Mixed Oxide (MOX) & refractory core$_2$) – 34 containers
3. Difficult to fully dissolve (e.g. ThO_2, BeO) – 104 containers
4. Potentially hazardous (e.g. UZr) – 103 containers
5. Very high dissolving risk or high U losses in electrolytic dissolver – 62 containers

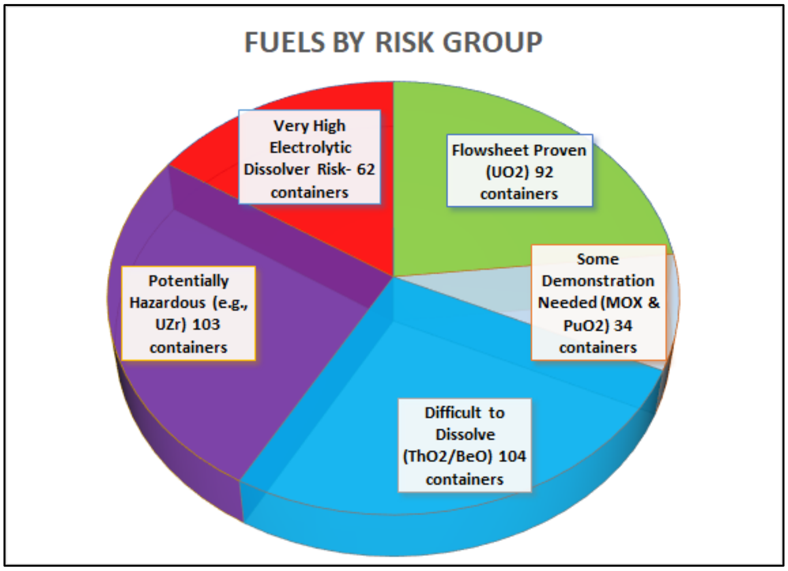


Figure 1-4. Fuel Groups Categorized by Risk.

Table 1-1 provides a breakdown of the estimated number of containers after rebundling. Also included is the mass (in MT) of heavy metal, aluminum, zirconium, stainless steel, and Pu-239 fissile gram equivalents (FGE), with the estimated electrolytic dissolution time in years. The 411 bundles reflect the estimated number after repackaging oversized items. The number of bundles (411) reflects additional bundles from repackaging changes as 16 additional bundles are created as oversized bundles are split. Because the various risks are interrelated, the discussion will begin with those activities common to all the fuels. Next, each of the risk areas are examined in order of increasing risk. Then, the cross-cutting issues such as those related to characterization and repackaging of some fuels will be addressed.

Table 1-1. Breakdown and Estimated Electrolytic Dissolving Time for Each Group.

Fuel Group	ED Bundles #	Heavy Metal (MT)	Aluminum (MT)	Zirconium (MT)	Stainless Steel (MT)	FGE (MT)	Time for ED (Year)
UO_2 core	92	3.6	1.4	1.7	0.7	0.1	0.9
MOX and Refractory Core	147	9.0	2.4	0.4	4.1	0.5	1.4
UZr metal Core	110	7.6	2.0	2.3	0.0	0.1	1.2
Highest risk	62	0.2	0.8	1.3	0.1	0.1	0.5
Total	411	20.3	6.6	5.7	4.9	0.8	4.0
	ED= Electrolytic Dissolver; MT= Metric Ton; Zr includes Nb; SS includes Hastelloy & other alloys						

2.0 Risks Associated with Processing All NASNF Fuel Categories (Risks 1 – 4)

There are four risks common to processing all NASNF, and which must be addressed to dissolve even the lowest risk group (e.g. UO_2 core fuels):

- Risk 1. Physical size of NASNF bundles relies on uncharacterized "stray currents" for electrolytic dissolution and may result in requiring extended dissolution times.
- Risk 2. Dissolution of bundle bottoms using stray currents may result in issues related to fuel dropping, buoyancy, and protrusions requiring intervention by the crane or extended dissolution times.
- Risk 3. Dissolution of Zr cladding is known to produces large volumes of sludge, which has not been fully characterized and could be difficult to remove.
- Risk 4: Properties related to sludge, material reactivity, and disposition of dissolver solution require further characterization, which could result in additional scope and schedule impacts.

The issue is not whether these risks would prohibit processing. Rather, the focus is on the potential impact associated with the risk, understanding the risks can be reduced (or found worse than expected), but not eliminated through further research and demonstration. A high spot (-30%+50%) estimate of the potential impact of these risks on the processing schedule, even if successfully demonstrated, is shown in Table 2-1. More work is needed to mature and validate these estimates. In summary, the four main risks could have a combined impact (without repackaging) of more than an additional year of processing. The following sub-sections will also discuss in further detail the impacts of each risk and the demonstrations needed to address the risks.

Table 2-1. High-Spot Estimate of Impacts Even if Successful with Demonstration.

Fuel Group	ED Bundles (#)	Sludge (Gal)	Stray Current Impact (Years)	Physical Impact (Years)	Sludge Removal (Years)	Cumulative Impact (Years)
UO_2 core	92	569	0.03	0.02	0.01	0.06
MOX and Refractory Core	147	164	0.09	0.06	0.09	0.24
UZr metal Core	110	770	0.06	0.04	0.08	0.17
Highest risk	62	446	0.03	0.02	0.03	0.07
Total	411	1949	0.21	0.14	0.20	0.55

2.1 Risk 1: Physical size of NASNF bundles relies on uncharacterized "stray currents" for electrolytic dissolution

Figure 2-1 shows the existing basket, which has a tapered design with a limiting width of 5 1/8" at the top which decreases to 1" at the bottom. Without repackaging, the current bundle diameter is approximately 5", however there is a known, substantial variation in the dimensions of the bundle (Table 2-2). See Table 5-1 for breakdown and number of bundles by fuel category. These variations mean that the bundles may barely fit into the tapered basket, resulting in the bottom being principally outside the full current zone. Hence, the dissolution of the bundle bottom requires stray currents, which certainly exist at substantial levels, but the rate of dissolution by stray currents has not been demonstrated. In contrast, the Fast Critical Assembly (FCA) can has a smaller diameter and fits deeper in the basket, with the bottom 4" to 6" within the full current zone.

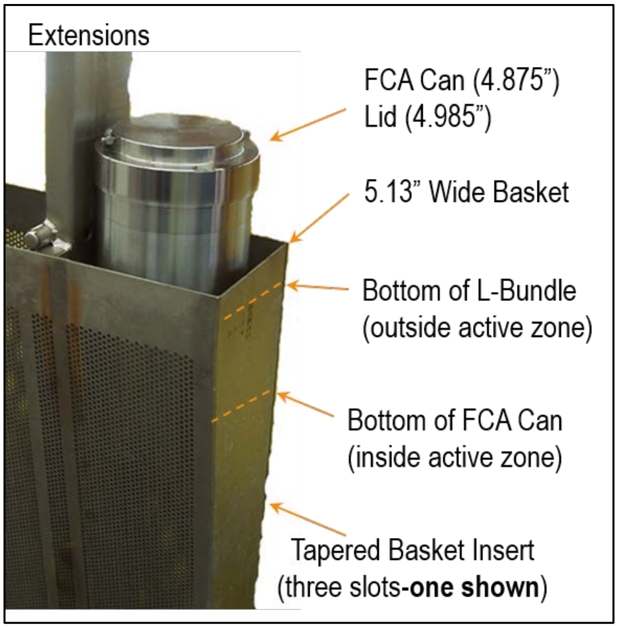
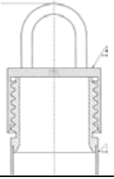


Figure 2-1. View of 1/3 of Dissolver Basket.

Table 2-2. Dimensions of Fuel Bundles (in Inches).

Container	Picture	Tube	Tol.+/-	Straightness	Lid
L/EBS Bundle		5.00	0.025	0.14	5.105
GP Tube Bundle		5.00	0.015	0.14	5.057
5" Square Tube		5.00	0.062	0.0625	5.26
Tubular Fuel Element-Natural (TFEN) Can		5.00	0.01	0.05	5.38
B-Can		4.00	0.01	0.05	5
Failed Equipment Can (FEC)		5.00	unknown	unknown	flange

2.1.1 Impact of Risk 1

Even if successfully demonstrated, the use of stray current dissolution would likely extend the overall program length by 0.21 years (-30%+50%). Although cautioning that demonstration was required, Caracciolo (1964a) performed stray current calculations to evaluate the feasibility of processing Na-bonded fuel, where the reaction could be managed at a controlled rate. The estimate postulated that in the first 5 cm above the full current zone the corrosion rate would be 40 mils per hour (mph), equivalent to 0.8 g/cm^2-hour for stainless steel. Given aluminum electrolytically corrodes at 60% of the rate for stainless steel, the aluminum dissolution rate is therefore estimated to be 0.5 g/cm^2-hour within the first 5 cm above the full current zone. At these corrosion rates, it is reasonable to believe that the bottom of a bundle (0.25" thick L-Bundle, 0.313" thick square tube bundle, and ½" for Failed Equipment Cans (FECs)) should dissolve through in less than 4 hours; however, occurring concurrently, corrosion is reducing the width, so that in less than 2-3 hours the bottom should be open or in the full current field. On the other hand, there is evidence that the stray current field might result in a slower dissolution rate than indicated by Caracciolo's stray current calculation. Without the cathode shield, DP-811 reported a corrosion rate of about 100 mpy (mils per year), which are equivalent to approximately 0.01 mph in an area between the cathode and the tank bottom. Given that the region between the cathode and the tank bottom is a greater distance away from the active current zone and presents a different geometrical aspect relative to the current, the dissolution rates in this region are expected to be much lower. As such, data available in this region do not disprove the Caracciolo estimate but point to the need for characterization of the stray current field and associated dissolution rates.

2.1.2 Development Needs for Risk 1

Task 1a: Demonstrate bench-scale stray current dissolution ($0.1M).

Task 1b: Demonstrate scale stray current dissolution at pilot, prototypical scale ($3M).

The bench-scale dissolver will contain a niobium basket between two platinum electrodes. Bench-scale testing includes suspending simulated Al and SS cans containing diverse surrogate fuels (e.g. SS, Zr-2, Zr-4, Hastelloy X) outside the active current zone. Additionally, samples with simulated flanges or middle canister supports exceeding the basket dimensions will be dissolved. Pilot scale testing is required since the bench-scale dissolver cannot match the geometric, kinetic, thermal, mass transfer, and current/potential conditions of the H-Canyon dissolver. The pilot will be more prototypic (full size with one well) and able to integrate multiple test parameters together, providing a more comprehensive understanding of electrolytic dissolution of the various assemblies. SRNL recommends performing electrolytic dissolution tests in a pilot-scale dissolver with experiments to evaluate reaction mechanisms and support development of a computational fluid dynamics model (dissolution fundamentals, impact of stray currents, etc.). If developed in conjunction with a pilot scale dissolver, the model can assist with assessing flow sheet changes (current density, temperature, electrolyte, etc,) during operation of the 6.3 ED without having to perform testing.

Initial estimates of cost and schedule are shown in Appendix A and will be refined as necessary.

2.2 Risk 2: Impact of bundle bottom dissolution on fuel dropping, buoyancy, and protrusions

Even if successful in demonstrating stray current viability, there are several potential risks regarding physical constraints. First, the lids and protrusions of several bundle types (e.g. square tube bundles, FECs, and TFEN cans will not fit into the basket so the bundle remnant may have to be removed using the crane. Second, with few exceptions, the sum of the tolerances plus the straightness (Table 2-2) leave little cushion relative to the basket well size; hence, it is possible that some bundles could get stuck, requiring crane intervention. Third, floating cans could slow dissolution. Fourth, for the bundle to pass down, its rate of descent is limited by stray current corrosion at a rate of several inches of length per hour. Hence, even though the fuel cladding would fall into the full current field, the bundle itself would take a day or so to corrode completely via stray current. A fifth consideration is damage to the dissolver due to falling fuel.

2.2.1 Impact of Risk 2

It is reasonable to assume that every carrier bundle might require removal by the crane or some mechanical or chemical other intervention. The model projects that the net impact from interventions would be approximately 0.14 years (-30%+50%) extension of processing time.[4] These estimates assume that crane intervention is preferable to the delay that would result from waiting on slow bundle descent. The potential concerns are so diverse and the impacts so significant that additional modeling and demonstration are needed to better understand the concern and address potential mitigation strategies. These strategies could include eliminating protrusions, puncturing cans, or repackaging problematic bundle configurations before shipment, and modeling to show that the basket would not be damaged by fuel falling into the basket.

2.2.2 Development Needs for Risk 2

Task 2a: Demonstrate physical considerations (e.g., damage to dissolver from fuel dropping, floating cans, and protrusions) during pilot demonstration for stray current dissolution. (Cost included in Section 2.1.2)

Task 2b: Create a CAD (computer-aided design) model of the dissolver to better understand physical fuel fit issues and a physical model to evaluate impacts of different fuel configurations, gravity/dropping, etc. ($0.2M).

Initial estimates of cost and schedule are shown in Appendix A and will be refined as necessary.

2.3 Risk 3: Sludge behavior and removal following Zr-clad dissolutions

SRS Zr-clad fuel processing experience is limited to Zr-clad UO_2 (with 212 kg Zr),[5] and removal of the sludge proved difficult. Therefore, additional demonstration and development of sludge-removal methods are required. Caracciolo (1964b) found that 2 wt. % of SS-clad UO_2 fuels, 8 wt. % of SS clad cermet, fuels and 85 wt. % of zirconium clad fuels ended up as sludge. After washing to remove residual uranium, the report found that 0.25% and 1.6% of the original uranium was retained in the stainless-steel particulate sludge and ZrO_2 sludge, respectively.[6]

During pilot testing, the sludge removal system consisting of 16 spray nozzles was found to remove most of the sludge, but multiple removal washes were required.[7] However, DP-824 (1963) noted that "ZrO_2 sludge was difficult to remove when it was allowed to accumulate on the dissolver bottom." While stainless steel dissolution generates 0.01 gallons of sludge per kg processed, zirconium generates 0.33 gallons per

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[4] The estimate was derived assuming (1) a crane intervention every charge with 2 hours per impact; (2) the net floater impact is 1 hour per charge (considering the floater would not dissolve until the bundle dissolved) or 1 day per charge were the cycle limited by the bundle dissolution. (3) The impact of gravity drop is unknown, but if damage occurred every 50 charges, the average impact to troubleshoot and replace the basket would be ½-1 hour per charge. These add up to 1/6 (1/12 -1/3) day per charge. Modeling and demonstration might prove these to be over- or under-estimates.

[5] A recent unpublished review of these data found a total of 212 kg of Zr dissolved rather than the 56 kg reported in Figure 1-2.

[6] The percent of the particles smaller than 44 micron (325 mesh) was for the stainless steel and zirconia sludge, respectively, 40% and 3%, with the particle shape of the zirconia sludge being flakes and that of the stainless steel sludge being flakes where particles are greater than 177 micron in size and being spheres if smaller than 177 microns in size. The loose bulk density of the stainless-steel particulate sludge and ZrO_2 sludge, were, respectively, 4 and 0.9 g/cc. He found the velocity to suspend the particles in a vertical column to be approximately 0.7 ft/sec for the stainless steel and 0.25 ft/sec to suspend the zirconia

[7] The sludge removal setup required 16 Spraco (FL-14) wide-angle flat spray (160°), operating at a pressure ~60 psig with each nozzle delivering 10 gpm to each spray, for a total of 160 gpm. The spray operating technique employed six intermittent cycles of 10 seconds on and 5 seconds off, with continuous operation of steam jet to remove solids. The spray nozzles are also suitable for use as an air sparge to provide agitation during dissolution. A minimum total air flow of 25 scfm to the 16 sprays was required to uniformly agitate the solution. The aerated mounds over the 16 outlets rose only 3 inches above the general liquid level, so will not diminish the "freeboard" appreciably. The design intent was to wash sludge and then transfer out the sludge to a collection tank. The sludge is rinsed and agitated with the sprays to remove any residual uranium, then allowed to settle, and the rinse solutions are jetted out to a collection tank. Tests in a 10-inch pipe mockup of the plant waste header system showed that both the ZrO_2 sludge and the stainless steel sludge can be moved through the plant waste header. Settling of the ZrO_2 sludge from a 2.5 vol % water slurry was prevented by reducing the flow into the header provided flow is limited and controlled.

kg processed (per DP-1500). The projected amount of sludge over the ABD program would be 2,000 gallons, which is ten times more sludge than was generated over the ten-years of operation in the 1970s.

2.3.1 Impact of Risk 3

Once a successful sludge removal method is demonstrated in pilot scale testing, it is recommended that the zirconium sludge be flushed from the dissolver after every batch. The model[8] projects that the net impact would be approximately 0.2 years (-30%+50%) extension of processing time for doing routine flushes.

TA 2-834 (1975) concluded that the "dissolver sludge removal system has not performed satisfactorily, removing only a small fraction of the sludge expected to be present." At that time, the sludge accumulation was about 3", and subsequent dissolving programs brought the level up to 6"-7" in places around the pot (per Woods (2020)), where it is today[9]. TA 2-834 (1975) argues that a 7" deep sludge level does not pose an electrical safety issue to the cathode, but could result in increased arcing and pitting of the electrode and the localized accumulation of vapor forming on the electrodes, which could require operational intervention to redistribute the sludge. However, the review did not consider significant accumulation well above the cathode, which could restrict heat removal.

2.3.2 Development Needs for Risk 3

Task 3: Conduct pilot testing with a full-scale tank bottom to ensure existing spray sweeps remove solids. The test should address the impact of corrosive degradation of the spray nozzles to ensure that sludge can be removed even with corroded nozzles ($0.5M).

Initial estimates of cost and schedule are shown in Appendix A and will be refined as necessary.

2.4 Risk 4: Other potential concerns related to ZrO_2 sludge properties, reactivity, and disposition

There are several risks associated with dissolving Zr for which there is insufficient evidence to posit a real impact, but more than enough evidence to require additional consideration, development, and demonstration. These include Zr processing risks related to sludge, such as potential higher uranium content, potential buildup of sludge in the basket, sludge disposal concerns, and the need to understand and address different properties for sludge from UZr processing. First, Caracciolo (1964b) reports that the uranium content of zirconium (Zr-2) sludge to be 1.6% (on a dry basis) after washing and drying, which is much higher than the assumed 0.25% for stainless steel cermet sludge. Similarly, DP-824 (1963) reports that "7% of the uranium in the fuel assemblies sorbed in the sludge (this was not expected on the basis of previous tests), but it was easily desorbed until less than 0.5% remained." Although all reports, i.e., DP-1500 (1979), all the TAs, and pilot testing (DP-546 (1961) and DP-647 (1961)), universally agree that, after leaching, less than 0.5% of the uranium remained, some data indicate the potential for higher uranium content prior to leaching. At a minimum, these variable results and potentially higher concentrations prior to leaching should be addressed as part of the criticality control strategy. See Forstner to Zeyfang (1974) for an initial assessment for one fuel type.

Second, Perkins (1974)[10] states that UZr_3 may be formed in zirconium-clad UO_2 fuels during fabrication and reactor operation at temperatures exceeding 350 °C. Although these conditions are unlikely, before processing all UO_2 core fuels or other fuel cores with zirconium cladding, they must be carefully researched to ensure that there is no potential for UZr_3 being present. One such fuel that has this concern is the EBWR (Experimental Boiling Water Reactor) UO_2-ZrO_2 fuel per Perkins (1974).

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[8] Four-hour flush per batch assumed.

[9] It should be noted that sludge was allowed to accumulate for the first 3.5 years of operation (until October 1972, 8.7MT metal processed). It was estimated that there was 55 gallons of sludge at a thickness of 1.75 inches in the dissolver. A visual check was never performed.

[10] See also Gens (1958) and Allison (1960).

Third, there is evidence that zirconium oxide sludge could accumulate in the basket and impede dissolution or require operator intervention, based on lessons learned in the first zirconium cladding dissolution (TA 2-776 (1972)). The accumulation concerns were enough that, according to TA 2-827 (1974), the second Zr cladding dissolution was performed using a different basket to avoid probable accumulation of zirconium oxide sludge in the basket. Therefore, basket design can be important for sludge accumulation and should be considered for the diverse fuel types that will be dissolved.

Fourth, although Caracciolo (1964b) demonstrated removal of ZrO_2 and SS sludge, the properties as well as the transfer and flush strategy for SS and ZrO_2 sludges differed. Reasons for additional study include:

- The volume being transferred is orders of magnitude greater than in the previous program.
- The demonstration used viscosity altering additives (sodium carboxymethylcellulose is not currently allowed by H-Tank Farm) without additional study to determine downstream impacts.
- The planned transfers all involve mixed SS-ZrO_2, whereas the testing found radically different transfer strategies and concentrations were needed for unmixed SS and ZrO_2 sludges because of their different properties. Testing also found that plugging was possible if the sludge transfers were not performed properly as shown in DP-740 (1962).
- The current expectation for the sludge disposition is that it will be transferred out of the Canyon to H Tank Farm as liquid waste. However, by the 1980s, the assumed disposition path (DP-1500) was not for the sludge to be transferred to liquid waste but transferred to a canister in a tank where it would be collected, dried, and then removed for disposal as solid waste. Additional work is needed to understand downstream impacts related to sludge disposition.
- Sludge from UZr processing (as discussed in Section 3.2) is expected to have a higher entrained uranium content and could contain potentially reactive uranium metal, hydrides, and denser and intractable double oxides of U and Zr. Wilson (1972) cautions that UZr_3 solids from either UZr or intermetallic dissolutions are difficult to dissolve and can be pressure-sensitive and potentially-explosive on contact with many different chemicals. The sludge from dissolving UZr could likely be quite different from the limited Zr cladding sludge handled to date. See Section 3.2 on UZr processing for more detail.
- A related concern with processing all fuel-types is addressing the NCS (nuclear criticality safety) implications of failed fuel, where the cladding barrier cannot be counted as a defense for keeping particulates in the basket. Forstner (1968) requires cladding as an assumed nuclear safety control; therefore, a nuclear criticality assessment is needed to ensure failed fuel is suitable for processing in the electrolytic dissolver.

2.4.1 Impact of Risk 4

The impacts of the four additional concerns are too tentative to estimate, but the potential impact can be described qualitatively as:

- Higher uranium in the sludge could impact the NCS safety strategy and well loading limits so additional NCS study is warranted and/or additional hot nitric acid leaches and cleanouts could be required with a delay of a day or so per batch.
- Since the potential for UZr_3 intermetallic exists, the fuels could be treated using the flowsheet for UZr fuels (Section 3.2.3).
- The impact of sludge accumulation in the basket could include, per SRNS-TR-2019-0107 (2019), a "reduced …dissolving rate" due to current bypassing fuel and due to the sludge occupying a large portion of the area between electrodes. The other potential impact would be slower dissolution of the core fuel material since the zirconium sludge would inhibit chemical exchange, lowering the dissolution rate. Another potential impact is downtime to replace the basket. Modification of the basket design might eliminate the problem.

- Since there is only a limited sludge storage capacity outside the dissolver, it is imperative that a prompt and reliable sludge disposition path be developed and demonstrated.

2.4.2 Development Needs for Risk 4

Task 4a: Develop a flowsheet for sludge transfer of various sludges (mixed, higher density U-ZrO_2 particles, UZr double oxides) to the H-Tank Farm and demonstrate with cold pilot ($0.2M).

Task 4b: Develop conceptual approach and assess feasibility, including Materials Controls and Accountability (MC&A), Safeguards and Security (S&S), and transportation concerns, for solid waste disposal of sludge ($0.2M).

Task 4c: Evaluate all Zr clad oxide fuel types for potential UZr_3 at oxide-Zr cladding interface ($0.1M).

Task 4d: Assess potential for higher amount of uranium entrainment during zirconium dissolution ($0.2M).

Task 4e: Assess potential for zirconium oxide sludge build-up in the dissolver basket as part of the pilot demonstration ($0.2M).

Initial estimates of cost and schedule are shown in Appendix A and will be refined as necessary.

3.0 Risks Associated with Individual Fuel Categories (Risks 5 – 10)

3.1 Risk 5: Dissolution and corrosion impacts for refractory oxide pellet fuels (NASNF Categories 2 & 3)

The UO_2-core fuels comprise a low-risk group contained in 92 bundles. Of the 92 bundles, only two bundles require repackaging. The major subgroups are as follows:

- One bundle of UO_2 core, SS-clad fuel
- Five bundles of UO_2-SS core fuel with SS cladding
- The remainder are UO_2 core with Zr or mixed Zr/SS cladding

Table 3-1 provides a summary of the UO_2-core fuel. The dissolution rate of this fuel-type is well proven with electrolytic dissolution.[11] However, the dissolution rate can be impeded by local considerations such as sludge covering the pellets, which can limit chemical exchange or create stagnant zones with lower acid concentrations. Even for pure UO_2, Uriarte and Rainey (1965) indicate that 1-2.3 % could remain undissolved. Hence, it is likely that some form of cleanout would be required between batches to ensure dissolution of all uranium. Additionally, work is required to verify nuclear safety charge limits are not exceeded, especially for the five bundles containing HEU (high enriched uranium).

Electrolytic dissolution of UO_2 fuel with SS cladding is well proven. The potential impacts associated with bundle size and zirconium cladding were addressed in Sections 2.0 and 6.0. Note that there is no incremental impact assumed for core dissolution because dissolution is concurrent with electrolytic dissolving and only incremental corrosion from chemical processing is addressed in this report.

Table 3-1. UO_2 Core Fuel Assumptions.

Fuel Core Risk Group	ED Bundles	Heavy Metal kg/can	FGE kg/can	Al kg/can	SS kg/can	Zr kg/can	wells per charge	charges per batch	Incr. Impact yrs	Est. 304L Corr. (mils)
UO_2 core*	92	39.4	1.3	15.5	7.5	18.5	2.3	7.1	0.00	0

* Low Risk

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[11] DP-491 (1960) established the chemical dissolution rate for sintered UO_2 pellets in 10 M nitric acid (@ 106 °C) to be around 370 mph.

MOX and Refractory fuels require special flowsheet considerations since oxides are unaffected by the electrolytic dissolver and require additional chemical dissolution once the cladding has been electrolytically dissolved. Table 3-2 provides a summary for each core fuel type in the Refractory/MOX risk group. The average masses of HM, FGE, Al, SS, and Zr are provided with the incremental impact in years. Note that all estimates are high spot (-30% - 50%). The MOX core fuel is considered medium risk while BeO and ThO_2 fuels are considered high risk. The projected development cost is $0.9M.

Table 3-2. Incremental Impact for MOX/Refractory Core Dissolving.

Fuel Core Risk Group	ED Bundles	Heavy Metal kg/can	FGE kg/can	Al kg/can	SS kg/can	Zr kg/can	Avg. wells per charge	Avg. charges per batch	Incr. Impact (yrs)	Est. 304L Corr. (mils)
MOX core*	34	38.2	1.9	23.3	11.2	11.0	1.6	2.9	0.13	5
BeO†	36	3.3	3.0	20.5	9.6	0.0	1.0	7.7	0.07	0
ThO_2†	77	103.6	3.9	11.9	45.7	0.0	1.1	1.0	0.67	31
Total	147	--	--	--	--	--	--	--	0.87	36

* Medium Risk
† High Risk

3.1.1 MOX (PuO_2/UO_2)

The fuels that comprise MOX are:

- MOX cores with zirconium or mixed zirconium and stainless-steel cladding. All but three bundles have less than 25% plutonium on an overall bundle average basis. (30 bundles)
- The three higher Pu content bundles contain greater than 25% Pu, with one can containing pure PuO_2, albeit only 22 grams of plutonium. (3 bundles)
- This group also contains one Sodium Reactor Experiment (SRE) uranium carbide core, stainless-steel clad bundle, which is treated in this group to simplify the number of categories. (1 bundle)

Risk Overview: Although the cladding dissolves electrochemically, the fuel meat dissolves chemically; hence, equipment and process constraints (e.g., corrosion rates, poison precipitation, and dissolution rate) could limit time cycles and performance. Several additional unique risks follow.

First, the existing nuclear safety controls are based on enriched uranium dissolution, and they need to be revised to address plutonium-bearing fuels. One significant concern with these fuels is the potential of plutonium residues that could leave the basket and accumulate in the sludge, complicating NCS control. Wilson (1972) argues that processing such fuels in the electrolytic dissolver requires reliance upon the neutron poison not as a secondary but as a primary control. Unlike UO_2 core fuel, which likely dissolves before it leaves the cladding, the basket cannot be credited for retaining MOX fuels because of their slower, potentially incomplete, and unquantifiable variability.[12]

Second, Uriarte and Rainey (1965) report the dissolution rate for MOX fuels is four to five times slower than for UO_2 fuels. The low surface area pellet form further reduces the rate. Although nitric acid dissolves most of the plutonium in low-Pu content MOX fuels, cleanouts with heated nitric acid containing potassium

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[12] Wilson argues, because of the slower, potentially incomplete, and unquantifiable variability of dissolution of MOX fuels, "the neutron poison provides the fundamental protection against a nuclear excursion, which might result from release of fissile particles in the unconfined dissolver space."

fluoride are likely needed prior to solution transfer. See also Tallent and Mailen (1977) for similar PuO_2 rate data.

- To avoid transfer of undissolved PuO_2 to other vessels during solution removal, see Wilson (1972) for basis; and
- to dissolve residual heels, Burch (1979) observes the acid dissolution of two types of MOX fuel with Pu contents of 1.5 and 2%. Even after two rinses, the fuel left from 1 to 3.5 weight percent of the original plutonium undissolved, and it required hot HNO_3-KF assisted dissolution for recovery. Further, dissolution rates vary among fuels and are sensitive to local concentrations and pellet properties. For example, Uriarte and Rainey (1965) indicate the dissolution rate for 100% theoretic density PuO_2 is 3,000 times slower than for 72% theoretical density fuel. Dissolution can also be impeded by local conditions such as sludge covering the pellets or stagnant zones.

Third, the ability to dissolve at or near boiling is required. The 6.3D dissolver cooling loop (currently used to remove heat from the dissolver) was designed for water cooling and optional steam heating, with a blank on the steam line (See TA-2 834 (1975)) to avoid the potential criticality safety risk of concentration through evaporation. NCS review is required as design changes are needed to ensure easy transition from cooling to heating.

There is one SRE container with uranium carbide fuel meat. Based on Ferris and Bradley (1965), the dissolution of uranium carbide in 2 to 16 M nitric acid at temperatures ranging from 25 to 120 °C, does not produce any hydrogen, carbon monoxide, or gaseous hydrocarbons, and instead results in uranyl nitrate, soluble organic acids, NO_2, NO, and CO_2.[13] Hence, although the dissolution should not pose flammable gas safety concerns, further study is warranted to ensure that the rate of reaction is controlled.

The model estimates the net impact for additional chemical processing and cleanouts for the MOX fuels to be 0.17 years (-30+50%). Underlying assumptions include 1.6 wells charged per dissolution and 2.3 charges per batch, with the principal limitation being the pellet volume, which often fills the basket, thus limiting charging to one well in such cases. A cleanout requires several hours up to a day of operating time to heat/cool, dissolve, transfer, flush, and account. The three high Pu assay containers are estimated to require a week to dissolve. Uriarte and Rainey (1965) found that with 14 M HNO_3 and 0.05 M KF, dissolution of a PuO_2 pellet was only 85% complete in 86 hours. Complete dissolution would take longer.

Hence, cleanouts with heated nitric acid and fluorides would likely be required to guard against plutonium buildup to stay within yet-to-be determined NCS controls, and to ensure MC&A and S&S requirements are addressed. At times, a complete dissolution will require higher concentrations of nitric acid containing low (approximately 0.1 M) fluoride levels with elevated temperatures. Given the complex and competing factors, work is required to develop the ideal flowsheet that minimizes corrosion of the dissolver components and avoids precipitation of gadolinium. See Section 3.2 for related fluoride concerns.

3.1.2 Refractory Fuels- ThO_2

The 68 existing bundles (77 bundles after repackaging oversized cans) of ThO_2 core fuels are all stainless steel clad, comprised of Dresden (ThO_2-2%UO_2) and Elk River Reactor (ERR) (ThO_2-4%UO_2) fuel. Most are intact, but some bundles and assemblies contain failed rods. Each bundle, on average, contains approximately 100 kg ThO_2 per bundle. Electrolytic dissolution would take about a day to dissolve the stainless steel, with the thoria rods (approximately 1/2" diameter by 60" long) remaining in the basket. Thoria dissolution requires a high temperature dissolving cycle.

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[13] It was found that 50-80% of the carbide carbon was converted to CO_2. Resultant soluble organic species include oxalic acid (0-11% of the carbide carbon), mellitic acid (2-9% of the carbide carbon), and "unidentified compounds which were probably aromatics highly substituted with -COOH and -OH groups."

This fuel type has the same risks as MOX dissolving, with the principal difference being that longer times under high temperature and corrosive dissolver conditions are required due to the more refractory nature of the highly sintered pellets with low surface area.

First, the dissolution rate of thoria-type fuels is slow, i.e., approximately 0.22 g/cm^2-hour for 95% TD (theoretical density) without additives. A review of two dozen sources adjusted to a common basis found good agreement with two data clusters.[14] One cluster, where the dissolving conditions are close to those planned and include irradiated fuel data, suggests the dissolution should take 1.2 days (250 mil pellet at a rate of 8.9 mph) using 10 M nitric acid, 0.05 M fluoride, and 0.025 M ANN (aluminum nitrate nonahydrate). This rate could be reduced with higher acid molarity and full boiling, but with higher corrosion (See Section 3.2.2 for discussion on fluoride corrosion and its effects). The other cluster, based on conditions often different than those planned, suggests it might take twice as long.

Second, all reports universally agree that fluoride levels of approximately 0.05 M or less are adequate. The small amount of fluoride increases the rate four orders of magnitude compared to nitric acid alone. Adding complexants like aluminum only had a limited impact on the rate, whereas dissolving PuO_2 is highly sensitive to minor complexant levels.

Third, the dissolution rate is highly sensitive to pellet density, percent uranium in the fuel, and additives like CaO, MgO, and CaF_2 which enhance sintering and subsequent dissolving. Were additives present, the dissolution could be reduced by 12 hours. The Dresden fuel has a very high density (97% theoretical) and likely no additives, which means it could take 1.5 -2 days to dissolve. Likewise, the Dresden fuel with 2% thorium would dissolve slower than the ERR with 4% uranium. Overall, the Dresden and ERR rates are likely offsetting, so the generic rate works for the overall average.

Fourth, dissolution is limited to one bundle (pellets fill the basket) at a time due to current bundling. the volume of pellets from one of the current bundles is sufficient to completely fill all three compartments of the basket.[15] The current basket design has partitions to isolate each compartment, so with the current basket design, the fuel would have to be repacked into three times as many containers; thus increasing the container count by 136. An NCS safety scope study is needed to verify the feasibility of designing a basket for the ThO_2 fuel without partitions.

Fifth, NCS limits need to be established/evaluated, allowing for potentially higher limits for a single bundle or repackaging. The fuel contains substantial U-233 which must be addressed in NCS loading and sludge limits. Further, like PuO_2 residues, thorium residues can contain substantial undissolved SNM (special nuclear material), and the criticality implications outside the basket and in the sludge must be addressed. This information is needed early since it could impact repackaging assumptions.

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[14] The most solid cluster includes Blanco (1962), Bond (1958), Phillips & Huber (1968), Hyder (1966),). Moore (1957). These sources support a rate of 30 g/m^2-min (or 0.22 g/cm^2-hour) which implies a penetration rate of 8.9 mils/hour at density=9.5 g/cm^3. Dissolving a pellet with a 250 mil pellet radius would take 28 hours. The second cluster includes Das & Bharadwaj (2013), Nayak & Sehgal (2019), Akobori & Shiratori (1994), Burch (1979), Roberts (1965), and Culler (1964). The assumed 99.9% completion corresponds to the probing basket depth were the basket full is minimized fissile accumulation. See Hart (1979) for details on ERR and Dresden like theorical density. Most of the rate sensitives were from Phillip and Huber who performed a complex multivariate analysis with irradiated sol-gel fuel particles. To illustrate method, Bond indicates a dissolution rate of 0.5 g/cm^2-hour for ~93% TD Consolidated Edison pellets in boiling 10 M HNO_3 with low Th molarity. But when adjusting for TD from 93% to 95%, using sub-boiling temperature, having ANN present to reduce corrosion, adjusted for no additive (1.5-2X), the rate is close to the 0.22 g/cm^2-hour used as the basis in this report. Hyder (1966) found no difference with irradiation. Hutton, in reviewing literature found some evidence of increased rate. Another consideration is that most of rate data was in well agitated tanks. The ED has less mixing.

[15] The limiting process variable in charging is that the fuel rods from one bundle would fill the basket, driving the cycle to a two -step operation. Assuming ~125 kg/bundle, adjusting for oxide and density would imply 15 liters (125/.88/9.5) of fuel. The basket volume is less than 28 liters (2.6”X18”x36”). Assuming a 50% packing fraction, the pellets from one bundle would fill the basket. There are currently 38 bundles of ERR and 30 Dresden bundles.

Sixth, there is a substantial risk of rod blockage because the rods (as currently packaged) occupy a large fraction of the cross-sectional area. Since they would not dissolve during electrolytic dissolving a cluster of rods constrained by the ever-narrowing basket could block the bundle from dropping down in the basket and keep the top portions of the cladding (outside the full current zone) from dissolving. Consequently, the top portions of the cladding (outside the full current zone) would be prevented from dissolving.

Scoping and optimization studies are required to better estimate the potential time cycle and overall impact. It is estimated that the net impact for additional chemical processing and cleanouts for the refractory ThO_2 fuels would be 0.67 (-30+50%) years. The dissolver has three wells, but the number of wells used in a dissolving cycle depends on NCS limits, SS and Zr concentrations, and pellet volume. For this fuel, the underlying assumption is that only one well is charged per dissolving cycle with only one charge per batch, with the principal limitation being the pellet volume. A cleanout requires several hours up to a day of operating time to heat/cool, dissolve, transfer, flush, and perform material accounting. The Model (Appendix B) estimated corrosion using the assumed reference flowsheet to be as high as 31 mils. As discussed in Section 6.0, rebundling the fuel could cut the processing time and corrosion in half and possibly allow use of the current basket. Developmental opportunities are discussed in Section 3.1.5.

3.1.3 Refractory Fuels containing BeO

There are 36 bundles containing Hastelloy-clad enriched uranium oxide rods with 30 percent BeO by mass, but the BeO occupies two-thirds the volume of fuel. The rod assemblies are stored in aluminum short inner storage cans (SISCs) within the bundle.

First, NCS limits need review, and repackaging could be required: The historical NCS limits (Forstner, 1978) for General Gas-Cooled Reactor material (UO_2-40% BeO) allow only 2.6 kg U-235 charged to only two wells provided the U concentration at time of charging is 0 g/l. Many of the bundles contain more HEU than allowed in a well per this limit. Either some bundles must be repackaged or the NCS limits must be revised. Second, after the Hastelloy dissolves and the baskets are left with BeO pellet fragments, a cleanout run is required to dissolve the BeO, which dissolves slowly. Bull (1972) reports that leaching the undissolved BeO pellet remnants with boiling 6 M HNO_3 for 42 hours is needed to ensure dissolution.

The model estimates the net impact for additional chemical processing and cleanouts for the Refractory BeO fuels would be about 1 month. Underlying assumptions include 1 to 2 wells charged per dissolution and up to six charges per batch. The BeO cleanout requires a half day to heat and cool and 2 days to dissolve, transfer, flush, and perform material accounting. No fluoride is required, so corrosion is limited.

3.1.4 Development Needs for Risk 5

Development work for Risk 5 includes modeling, developing, and demonstrating flowsheets for MOX, ThO_2, and BeO/UO_2 fuels, and performing corrosion studies for all fuel types.

Task 5a: Review literature, assess, and refine processing rates and corrosion estimates for MOX and refractory fuels to support mid-FY21 Non-intact decision ($0.1M).

Task 5b: Complete lab scale flowsheet studies using simulated pellets to define fluoride levels, processing time, sludge cleanout, and evaluate other concerns ($0.5M).

Task 5c: Perform lab/bench-scale corrosion studies (fluoride impact on Nb, Pt, and SS parts) over entire range of bounding conditions for all fuel types. ($0.15M) [Also applies to Risks 6 & 7]

Task 5d: Model and assess corrosion for all fuel types ($0.2M).

Task 5e: Assess viability of NCS limits for failed fuel and for refractory fuels containing ThO_2 and PuO_2, which are likely to not remain in the basket and could be transferred out of the dissolver to other tanks. ($ not included since NCS is a H-Canyon function)

Initial estimates of cost and schedule are shown in Appendix A and will be refined as necessary.

3.2 Risk 6: UZr fuel hazardous acid reactions (NASNF Category 4)

The flowsheet for safely dissolving UZr fuels requires significant development to avoid potential explosions as described in Section 3.2.1. It also requires high (approximately 0.5 M) fluoride levels and must address concerns with K-Gd-F precipitation, potentially reactive metallic fines, and corrosion potential. Applicable fuel groups include:

- One group of UZr fuel is principally comprised of intact EBWR plates (U-5%Zr-1.5%Nb) with a few intact HWCTR (Heavy Water Component Test Reactor) tubes comprised mostly of U-2%Zr.[16] Although a few of the HWCTR tubes have a variety of cores ranging from 0-2% Mo or Si, the overwhelming majority are U-2%Zr. Most are packaged in Al L-bundles or TFEN cans, with one can having a co-packed SS clad oxide assembly. (71 bundles)
- The other group is comprised of mostly HWCTR damaged rods with a few damaged EBWR assemblies. Table 3-3 provides a summary for the key risk groups and UZr core fuel. Note that all estimates are high spot (-30% - 50%). The average mass of HM, FGE, Al, SS, and Zr are provided with the incremental impact in years. These fuels are mostly inside Over Sized Cans (OSCs). See risk and impact discussion below for details (32 existing bundles but repacked into 39 bundles).

Table 3-3. Incremental Impact for UZr Core Dissolving.

Fuel Core Risk Group	ED Bundles	Heavy Metal kg/can	FGE kg/can	Al kg/can	SS kg/can	Zr kg/can	Avg. wells per charge	Avg. charges per batch	Incr. Impact yrs	Est. 304L Corr. (mils)
UZr Intact*	71	89.6	1.3	14.6	0.2	29.8	1.1	1.0	0.47	93
UZr Defective*	39	33.3	0.3	28.1	0.6	6.0	2.5	1.0	0.13	29
Total	110								0.60	122

* Potentially Reactive

3.2.1 UZr Risk Overview:

Extensive reports[17], confirmed by multiple historic incidents, show the potential for explosions when dissolving U-Zr alloys (or intermetallic U-Zr) in nitric acid (Figure 3-1). All agree that alloys containing less than 1% zirconium are safe to dissolve. All also agree that alloys containing between 1-30% zirconium pose a risk, but Martin and Field (1958) insist the potential exists even for alloys that are mostly zirconium. Larsen (1954) argues that above the molten solubility of zirconium in uranium (less than 1%) UZr_3 can form as "a finely divided epsilon [called delta in Figure 3-1] phase in a matrix of alpha-uranium." The presence and quality of this phase depends on the composition, homogeneity, and previous thermal history. Dissolving the fuel matrix in nitric acid can leave behind UZr_3 particles, which can "undergo a rapid oxidation with explosive violence," with copious quantities of water vapor from the heat of reaction and NO_x from nitrate reduction. If no explosion occurs, the phase is slowly oxidized to form double dioxides of uranium and zirconium in a mole ratio of one to three, which are stable and difficult to dissolve. Martin and Field (1958) posit that there are possible explanatory mechanisms other than UZr_3, which include the

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[16] The 3.75"x58" EBWR assemblies, each have 6 plates (with gaps between plates), are in 5" Al EBWR cans. The EBWR Zr cladding is 0.02" thick with 0.0225" side plates.

[17] Larsen (1954), Martin and Field (1958), Gens (1958), Swanson (1985).

reaction between nitrates and finely divided metal fragments produced from dissolution. Gens (1958) found niobium just as hazardous as zirconium.

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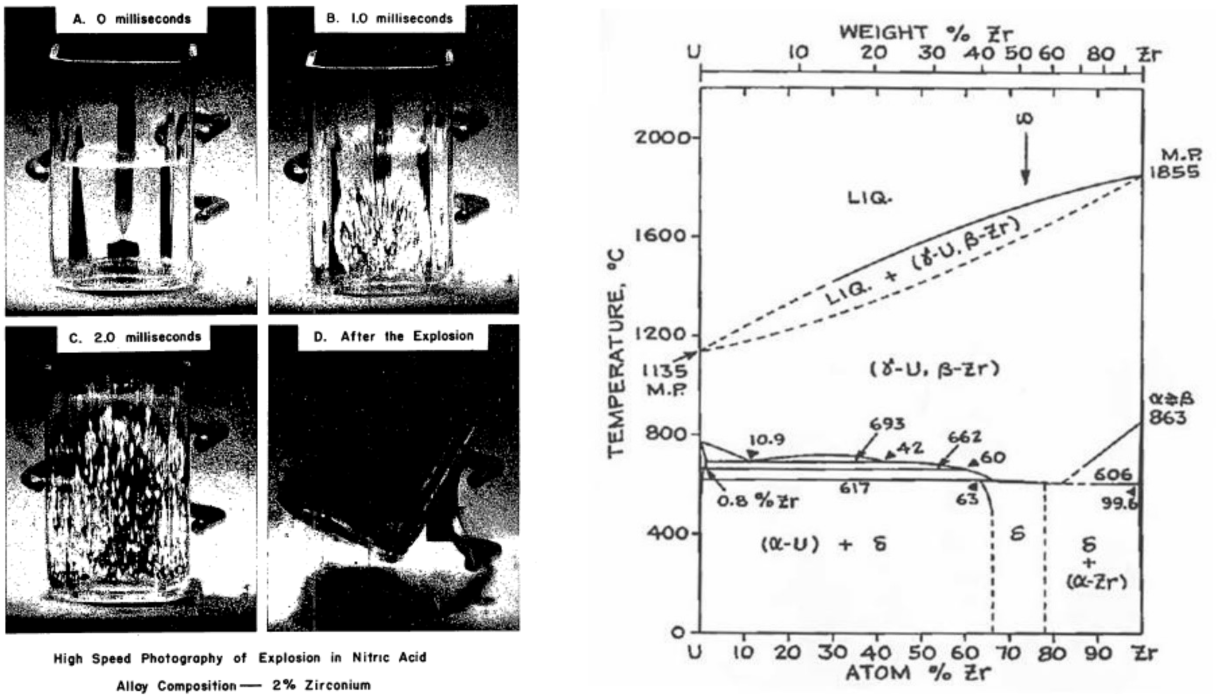

High Speed Photography of Explosion in Nitric Acid
Alloy Composition — 2% Zirconium

Figure 3-1. Explosion from dissolving U-2%Zr alloy (left); Zirconium-uranium phase diagram (right).

Another potential risk is that high fluoride levels are required. Gens (1958) summarized the proven control strategy: “In all cases, there apparently is no explosion hazard if a F/Zr ratio of 4 in solution can be maintained with certainty throughout the dissolution and if no fluoride-complexing agent other than zirconium is present.” The latter qualification implies that the fluoride to zirconium ratio must sufficiently be greater than 4:1 in order to account for other chemical species such as aluminum. To address this concern, Laurinat (2020b) estimates the required fluoride concentrations could be as high as 0.46 M, with a free fluoride level of 0.24 M, assuming one charge per dissolving with <0.06 M Zr.

Safe electrolytic dissolution at approximately 0.5 M fluoride levels must be demonstrated. The key to eliminating the explosion potential of U-Zr alloys is to manipulate the dissolving chemistry so that the U does not preferentially dissolve leaving high surface area UZr_3. Since electrolytic dissolution of uranium metal is much faster than zirconium dissolution, it is not inherently safe and has no effect on dissolving oxides or UZr_3.[18] Further, electrolytic dissolving disintegrates rather than dissolves zirconium, producing large qualities of ZrO_2 containing lesser quantities of UZr_3 as double (U-Zr) oxides which do not dissolve readily. Martin and Field argue that once a protective black powder layer forms, “it would take hours for the fluoride to deactivate it”. Therefore, the solution would be the presence of a “rapid-acting solvent for zirconium”, such as fluoride, to be freely present in a minimum 4:1 ratio. The proposed flowsheet (approximately 0.5 M fluoride in nitric acid) is adequate for safety but the dissolution rate is slower than the more proven Zirflex process.[19]

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[18] Per DP-1500 uranium metal dissolves electrochemical ~10 times faster than zirconium. In contrast, per Sreenivasulu (2016), uranium dissolves 4-6 times faster than zirconium from U-6% zirconium alloy at a rate of 4-8 times faster at room temperature but less than twice as fast when boiling (see figures 1 and 4).

[19] See Martin and Field (1958); Gens (1958), Clark and Kibbey (1958), and Occhipinti and Owen (1964) for rate data.

When electrolytically dissolving UZr fuels, uranium losses to sludge are likely high and essentially insoluble. The 2% and 5% Zr alloys have not been tested in the electrolytic dissolver. Larson (1954) reports, "It was observed at the outset that the 2 without alloy did not dissolve cleanly in nitric acid. The residues contained both zirconium and uranium and were impervious to nitric acid leaching."[20] Based on the reaction mechanisms without fluoride, perhaps up to 1-2% of the U-2%Zr alloy and 5-6% of the U-5%Zr-1.5%Nb alloy could end up as an intractable residue in the sludge, similar to but with lower losses than the Zr-low U fuel (section 3.3.1). It is believed that the only way to avoid such solids is to dissolve in a superabundance of fluoride.[21]

Per Gens (1959), the dissolution of uranium metal cores does not generate hydrogen. Vander Wall (1960) estimates zirconium dissolving produces 2.9% hydrogen in 8 M HNO_3, with less hydrogen at higher acid concentrations and more hydrogen at lower acid content. The air sparging should handle these levels.

The EBWR dissolution leaves Nb_2O_5 (niobium oxide) as an oxide with 0.03% uranium. Thus, 90 kg of niobium would end up as an oxide. Therefore, the implications on sludge transfer need assessment.

3.2.2 Fluoride risks include precipitation and high corrosion

Higher fluoride concentrations pose several risks that require further investigation. First, a study is needed to ensure that the higher fluoride concentration will not result in precipitation of UF_4 or neutron poisons (via K-Gd-F precipitation) or similar imbalances. Second, corrosion rates for SS and alumina are troubling at higher fluoride concentrations. Initial FCA laboratory testing by Truong (2019) using LPR (Linear Polarization Resistance) found corrosion rates were not adversely impacted with concentration of fluoride less than 0.1 M and gadolinium less than 2 g/liter. Truong reports that niobium corrosion rate estimates ranged from 10 mpy with electrolytic dissolution without KF at 60 °C to 37mpy (0.0042 mph) with 0.2 M KF at up to 60°C. Assuming 0.35M F^- (geometric mean during run with initial 0.5 M F^- and final 0.24 M, i.e., minimum needed for 4:1 F/Zr ratio) the implied Nb corrosion rate would be approximately 0.075 mph. However, Gens (1962) indicates that high acid and fluoride levels are extremely corrosive to niobium, so the rate could be higher (Figure 3-2).[22] In regard to SS corrosion, assuming fluoride complexing from 0.1 M Al and 0.03 M Zr (average during run), the model estimates 135 mpy, which, when adjusted for time running, would be approximately 122 mils.[23] Mikeska and Bennison (1999) report the alumina dissolution rate in aqueous HF is approximately 1,000 mils per year, but the rate with 10 M HNO_3 with 0.5 M HF is likely much lower (e.g., Warren (1962) recovered little uranium from an alumina matrix with 0.2 M KF). Preliminary results indicate platinum corrosion is low (Section 6.0). Rebundling could cut the dissolution time and corrosion by as much as half. Third, another potential concern is the need to explore options other than KF to provide the needed fluoride level, since KF is not sufficiently soluble at the needed fluoride

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[20] Larson notes, "Therefore, it would appear probable that zirconium present in excess of the amount of solid solution in uranium is insoluble in nitric acid. The oxide residue is apparently formed by the oxidation of a phase in which excess zirconium is combined with uranium in an atom ratio of approximately three to one," Larsen also notes, "Low zirconium alloys may be contacted with nitric acid without incident provided no disturbance is permitted to occur. The coating which forms on the surface is black and explosive in· the early stages of the dissolution, becomes gray on further treatment, and, eventually, on completion of the dissolution, becomes a completely passive residue. The residue contains uranium, but prolonged leaching with nitric acid fails to remove it." Regarding the >1% Zr alloys, Larson also "observed that dangerous explosions (see inset) did occur during the course of nitric acid attack on this metal.".

[21] Gerke (1953) found that no solids formed when dissolving unalloyed uranium and zirconium metal plate (with a mass ratio of 17) where the fluoride/zirconium ratio was greater than 4 and the acid concentration was greater than 3 M, but there remains lack of proof that this principle applies to different U/Zr mass ration.

[22] However, corrosion would be lower (perhaps 30%) at 60 °C, Were Nb corrosion to follow the Zr diffusion limited rate model (see Vander Wall (1960), the expected corrosion rate at 8 M would follow the curve of 5*HF concentration at boiling, which would indicate at 60 °C (assuming 30% of boiling) a rate of 0.05 g/cm^2-hour.

[23] The model estimates 135 mpy, adjusting for the time actually dissolving. To cross check the model, DP-486 (1960) gives the corrosion rate of 304-L SS at 60 °C for 0.1 and 0.5 M HF at 10 M HNO_3 to be, respectively 52 and 197 mpy and at 6 M HNO_3 to be, respectively 44 and 300 mpy. The geomean and average of these rates are respectively 108 and 148 mpy, giving a range of potential corrosion, in agreement with the model.

levels for UZr dissolution. Therefore, ensuring dissolution can occur at the correct fluoride concentrations without adverse side effects is critical.

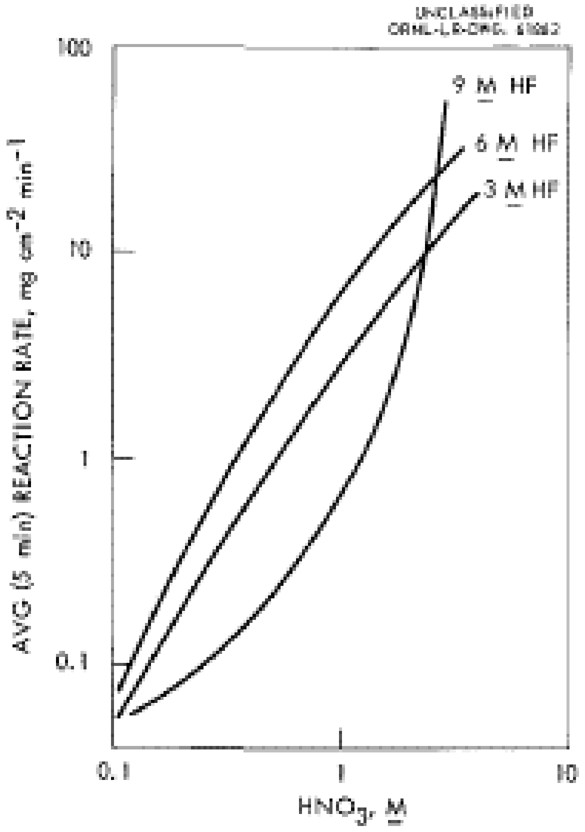

Figure 3-2. Niobium Reaction Rate in Varying Concentrations of Boiling HF.

3.2.3 Flowsheet Selection for UZr Fuels

There are four possible options for selecting and demonstrating a safe flowsheet for electrolytically processing UZr fuel:

Option 1: Demonstrate that electrolytic dissolution without fluoride poses no UZr explosive hazard. This would be difficult to prove. Even were there no incidents during testing and the sludge found passive, it does not prove there is no hazard.[24]

Option 2: Run the electrolytic dissolver with sufficient fluoride (e.g., 0.5 M) for safety. Although sufficient fluoride can eliminate both high losses to sludge and the safety risk, the combination of fluoride and electrolytic dissolution requires demonstration and proper balancing. The reason for concern is that TNX testing proved that even with fluoride present (at 0.03 M) during electrolytic dissolution the high uranium losses and high ZrO_2 sludge levels were not eliminated. However, it might be possible to prove that at 0.5 M fluoride with electrolytic dissolution that these losses could be avoided. If not, the heavier U-Zr oxide particles would be difficult to remove, pose substantial criticality concerns, and could complicate disposal options due to the potential for containing potentially reactive particulates.

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[24] The early work (e.g., Larsen) clearly shows the risk is during dissolving and that the sludge becomes passive as it oxidizes, and an initiating event (such as a bump or disturbance) is needed to result in an explosion, so it would be difficult to prove safety under a robust safety envelope. Martin and Field (1958) add, "energy requirements to initiate the explosion are modest." Moreover, fluoride is needed to facilitate the dissolution of the UZr_3 by Zr complexation, thereby eliminating the explosion potential. The electrolytic dissolving time is slightly over two days.

Option 3: Use the electrolytic function as little as possible (perhaps at the beginning and end) and rely on 60 ºC, 7-10 M HNO_3, 0.5 M fluoride, and 0.25 M ANN chemical dissolution, which should dissolve cleanly, eliminating the ZrO_2 sludge and high losses to the sludge, while providing safety. This option has the greatest probability of success. Although demonstration is required, there is firm basis for its viability. The chemistry and rates for zirconium, zirconium alloy, and aluminum dissolution for this flowsheet approach are well established.

- Zr dissolution: Vander Wall (1960) found the dissolution rate constant for HF in 8-10 M nitric acid at boiling to be approximately 6 g/cm^2-hour times the HF concentration. Dissolution rate in 0.5 M fluoride and correcting for 60 °C[25] is estimated at approximately 0.84 g/cm^2-hour. Adding aluminum to reduce corrosion would lower this rate, and the rate falls as zirconium is dissolved. See Clark and Kibbey (1958) and DP-872 (1964) for supporting empirical confirmation, positive (3X) improvement with good mixing, variable sensitivity, and evidence that the proposed flowsheet dissolves cleanly.[26] However, it takes approximately 1 hour to break through the oxide layer before effective dissolution can continue.
- Alloy dissolution: Gens and Baird (1959) report an average rate for EBWR alloy (U-5%Zr-1.5%Nb) at boiling between 7-10 M HNO_3 of approximately 3 and approximately 0.9 mg/cm^2-hour, respectively, for 0.5 and 0.1 M F^-, but the rate falls with Al and Zr complexation of F^-. They also report complete dissolution of an EBWR heavy assembly (60 kg alloy with 3,117 cm^2 surface area without side plates) in ten hours at boiling in 7 M HNO_3, 0.7 M fluoride, 0.7 M ANN, and with terminal 0.09 M Zr concentration. The average rate was 1.9 g/cm^2-min. At 60 ºC, the rate would be 26% of the rate at boiling; therefore, dissolution would take approximately 40 hours per assembly.[27] Multiple reports (e.g., DP-318 (1958), DP-439 (1959), and DP-839(1963)) address dissolution of various low zirconium (2-10% Zr or Mo) alloys, all indicating that the alloy dissolution rate is faster than dissolving zirconium or EBWR alloy.[28] With two assemblies per bundle, the chemical dissolution duration would be sequential and take over three days, but if repackaged, the dissolution and the corrosion rate per assembly would be cut in half. See Section 6.0 for details.
- Aluminum dissolution. Straumanis and Wang (2016) indicate a rate for dissolving (via complexation) aluminum metal with fluoride equal to 28+275*(fluoride normality) in mm^3/cm^2-min. At this rate (5 to 6.8 g/cm^2-hour), zirconium not aluminum dissolution is the limiting factor. Vander Wall (1960) states that pure aluminum dissolves at the same rate as zirconium in mixed nitric acid-fluoride flowsheets.

Option 4: See discussion of a low acid-high fluoride flowsheet in Section 3.2.2 which would only be used as a backup approach if the 0.5 M fluoride approach proves unsuccessful.

With current bundling, chemical dissolution would take 1½ days per assembly compared to slightly over two days for the electrolytic dissolution reference case, which would take longer as explained in Section

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[25] 60 ºC rate is approximated as ~ 26% rate at boiling based on interpolating from chart in Vander Wall report.

[26] Clark and Kibbey (1958) show a rate for zirconium at 8 M HNO_3, 0.1 M Zr present, with 0.5 M fluoride, and at 90 ºC to be 0.43 g/cm^2-hour, but the rate decrease to 0.0007 (near zero) g/cm^2-hour at 0.1 M fluoride. They report that good mixing could improve these rates by a factor of three. Scaled to 60 ºC using curves from DP-872 (1964), a rate of ~0.25 g/cm^2-hour would be likely.

[27] Calculation assumed EBWR heavy bundle has a cladding surface area of 4,182 cm^2 (without side plates) and a core surface area of 3,117 cm^2, and an average dissolution rate of 0.6 g/cm^2-hour at 60 ºC. There are 14 kg of Zr cladding and 60 kg of alloy. No surface area credit was given for side plates, since the assembly actually has less area that its active plates. Pending flowsheet development and demonstration, an initial 0.5 M fluoride with 0.25 (TBD) moles Al should give an average rate ~0.6 g/cm^2-hour, which should dissolve an EBWR heavy bundle in 40 hours at 60 ºC plus 3 hours for initial electrolytic dissolving.

[28] Gens and Baird (1969) found rates 3.6 times higher at 11.5 M acid with 1.1 M fluoride: 1.1 M ANN at boiling. DP-239 (1959) found rate for U-2%Zr alloy at boiling and between 7-9 M HNO_3 to be 9.1 and 1.4 g/cm^2-min., respectively, for 0.4 and 0.1 M fluoride. Likewise, Sreenivasulu (2016) dissolved U-5%Zr alloys in boiling 15.7 M HNO3 with 0.05 M HF within 5 hours, with completeness indicated by the disappearance of black sold residue; however, by implication, the formation of black residues was not eliminated and the associated dissolution rate of the black solids was less than that of the alloy. DP-318 (1958) observed a dissolution rate of U-10%Mo alloy at 3.9 g/cm^2-hour in 5 M HNO_3 and 0.1 M F^- at boiling. DP-319 found the rate up to 3 times faster with good stirring.

6.0. However, were the assemblies repackaged to one assembly per bundle and two bundles charged at a time, then chemical dissolution would take less time. Similarly, the 304L and Nb corrosion, which is troublingly high, would also be reduced as the cycle time is reduced. Together, these two factors (rate and corrosion) would drive rebundling, as discussed in Section 6.0, to maximize throughput and reduce corrosion.

3.2.4 Development Needs for Risk 6

Development work for Risk 6 includes modelling and experimental studies to refine and validate a UZr flowsheet and address K-Gd-F precipitation, process safety, uranium losses to sludge, and other potential flowsheet parameters.

Task 6a: Review literature, assess, and refine processing rate and corrosion estimates for UZr fuels to support mid-FY21 Non-intact decision ($0.05M).

Task 6b: Perform a modeling and repackaging assessment to develop the optimal UZr approach. Options to address include (1) to decide if repackaging UZr material to one assembly per bundle is warranted and (2) to identify an optimal fluoride strategy that could allow two current bundles per charge. Identify any process feedback information needed for safety ($0.2M).

Task 6c: Lab/bench scale studies to refine and validate UZr flowsheet and address K-Gd-F precipitation, process safety, uranium losses to sludge, etc. ($0.35M).

Task 6d: Demonstrate with irradiated UZr fuel the dissolution rate, minimal losses, sludge impact, and stable fluoride interactions ($1.5M).

Initial estimates of cost and schedule are shown in Appendix A and will be refined as necessary.

3.3 Risk 7: Understanding very high risk NASNF and alternative approaches (NASNF Category 5)

Three fuel groups totaling 62 bundles pose the highest risk for electrolytic dissolution because they will not completely dissolve or will leave excessive fissile material in the sludge. Previous electrolytic dissolution studies with Zr-low U fuels or Th-U fuels were unsuccessful in eliminating high losses to sludge. Other unsuitable fuels include seven bundles containing mostly BeO and seventeen bundles with mostly ZrO_2. Both would be limited by the fluoride concern. The original team screening identified these as very high risk based on an initial review that these fuels could require multi-molar H_2SO_4, HF-NH_4F, and HCl solutions. However, subsequent review found potential approaches that could work but would require demonstration as discussed in this section. Table 3-4 shows the average mass of uranium, FGE, Al, SS, and Zr per bundle, the charging statistics, and the estimated program impact in years.

Table 3-4. Incremental Impact for High Risk Fuel Dissolution

Fuel Core Risk Group	ED Bundles	Heavy Metal kg/can	FGE kg/can	Al kg/can	SS kg/can	Zr kg/can	wells per charge	charges per batch	Incr. Impact yrs	Est. 304L Corr. (mils)
BeO*	7	0.6	0.5	13.8	1.7	0.0	3.0	10.0	0.01	0
ZrO2*	17	3.3	1.7	15.4	4.7	38.3	2.0	6.2	0.09	3
ZrU*	38	3.8	1.3	10.9	0.1	18.4	1.7	1.0	0.47	18
Total	62								0.57	21

* Very high risk

3.3.1 Zr-Low U risks:

Electrolytic dissolution of Zr-9.3%U core fuels (35 containers), based on past studies, could result in high (20-60%) losses of uranium to the sludge. Multiple SRS reports (DP-574 (1961), DP-546 (1961), DP-647(1961), DP-479 (1960), and DP-547(1961)) demonstrate that all efforts to eliminate high uranium losses to the sludge through electrolytic dissolution were unsuccessful. DP-740 found electrolytic dissolution of thorium-uranium alloys resulted in high (70.5% +) losses of the uranium and 79% of the thorium trapped in the sludge. Months of testing concluded that none of the chemical methods tried as additives to the nitric acid electrolyte were successful in avoiding the high uranium loss to the sludge. Likewise, no chemical methods were able to recover the uranium from the sludge. An X-ray pattern of the sludge found a "double oxide of zirconium and uranium, which formed from disintegration of the Zr-low U alloy but the ZrO_2 that is formed contains 10 to 50% of the total uranium. Dissolving using the same flowsheet as for UZr previously discussed in Section 3.2.3, should eliminate the high loss concern. If proven, then the risk of this material group is no different than for UZr alloys.

The backup strategy would be to use a high fluoride (2-5 M) and low acid (1 M) flowsheet that would be suitable for use in 304L stainless steel equipment at 60 ºC, but not likely suitable in the electrolytic dissolver unless used at the end of the campaign. DP-839 (1963) suggests a possible flowsheet for dissolving the Zr-low U fuel at 65 ºC called the "fluoride-chromate process." This flowsheet uses 1 M ANN, 1 M HNO_3, 0.1 M $K_2Cr_2O_7$, and 2 M HF initial with incremental additions of HF as the zirconium dissolves and complexes. The report claims suitability for use in 304L, but the process assumed use of Hastelloy lines to add the HF since the concentrated HF added would corrode 304L stainless steel at a rate of 500-1000 mils/year (approximately 0.1 mils/hour). However, this process was never fully demonstrated and its suitability for use in the electrolytic dissolver would need extensive testing and demonstration. Some potential concerns include (1) using HF or NH_4F, (2) using 1 M HNO_3 acid (DP-1500 requires ≥6M), which could generate significant hydrogen and result in hydrogen embrittlement of the platinum electrode, and (3) balancing corrosion with explosion safety requires study and demonstration. The Zirflex[29] process was intended only as a penetrating or decladding step, leaving undissolved hulls for disposal and is not intended to dissolve uranium. The uranium solubility would be less than 0.1 g/liter under these flowsheet conditions due to the precipitation of UF_4. Assuming NCS issues could be addressed, an additional step to dissolve the uranium would be needed. This approach eliminates the intractably high uranium losses, but so should the higher acid-lower fluoride flowsheet. Demonstration of this flowsheet is needed.

3.3.2 High BeO content risks:

Warren (1962) found 5%UO_2-BeO to be nearly intractable, with the dissolution rate in boiling 10 M nitric acid being approximately 0.015 mg/cm^2-min, with little improvement with 0.2 M fluoride added. Further, Warren states the dissolution rate would be extremely low even in multi-molar H_2SO_4, HF-NH_4F, and HCl solutions. WSRC-TR-2005-00042 (2005) addresses beryllium metal dissolution. This report explains beryllium is hard to dissolve because it complexes so strongly with fluoride that cations like NH_4^+ rather than H^+ are required for dissolution. Fortunately, only 20 mils of penetration are required to dissolve the annular pellets, which should take about one week per batch plus electrolytic dissolution time to dissolve the outer can. Given the pellet volume, it would take two batches to dissolve the HTRE fuel.[30] Although this fuel is classified as high risk, when consideration is given for thickness, this group should be reclassified

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[29] Smith (1960), Swanson (1985); See also, Harmon (1976) proposed the "penetrate-leach" process, an adaptation of the Alniflex flowsheet (initially 2 M HF – 1 M HNO_3 – 1 M $A1(NO_3)_3$ - 0.1 M $K_2Cr_2O_7$) to partially dissolve the zirconium cladding, exposing the fuel core, and also to dissolve any UZr_3 intermetallic compounds, which are insoluble in nitric acid but soluble in a fluoride flowsheet. The process strategy involved adding fluoride incrementally as the zirconium dissolved in 0.5 M increments (bringing the final fluoride concentration to 2-3 M HF) to keep ZrF_4 and AlF_3 from precipitating. Dissolution aimed at a final temperature of 70-75ºC to minimize corrosion.

[30] The annular fuel pellet is 0.28" OD x 0.2" ID x 3.75" long. Hence, the dissolution must penetrate 40 mils of 2.83 density pellet. Warren gives a dissolution rate of 0.015 mg/cm^2-min in 10 M boiling nitric acid were the fuel BeO-5%UO2. This would imply 0.015*60 (min/hr)/1000 (mg/g)/2.83 (g/cc)/2.54 (cm/in)*1000 mils/in = 0.134 mils/hr. To penetrate 20 mils would require ~150 hours (~1 weeks). All the pellets (~14 liters at 50% packing density) would fill into the basket, so it would take at least 2 incremental (not counting electrolytic dissolving) weeks for the overall HTRE dissolution.

and included with the refractory BeO group. See TA-2-683D for previous HTRE processing. Another concern with this fuel is the lack of cladding: Forstner (1968) requires cladding as an assumed nuclear safety control; therefore, NCS assessment is needed.

3.3.3 High ZrO_2 fuel risks:

This fuel group poses the risk of high losses to sludge as well as potential UZr_3 reactivity. Perkins (1973) noted that EBWR UO_2-ZrO_2 fuels should be considered to contain UZr_3 intermetallic, since this fuel was exposed to temperatures in excess to 350 ºC.[31] Seventeen bundles of EBWR fuel contain approximately 9% UO_2, 82% ZrO_2, and 8% CaO core fuel. Prajapati (2014) estimates the dissolution rate for ZrO_2 in nitric acid "appears to be five orders-of-magnitude" slower than for PuO_2.[32] Lowalekar and Raghavan (2004) indicate that approximately 0.5 M HF solution would dissolve zirconia at a rate of 0.07 mils/hour, which might take a week and should provide sufficient fluoride to address the chemical safety concerns.

The backup approach would be to use the same low acid, high fluoride flowsheet discussed as a backup option for the Zr-low fuel in Section 3.3.1 Stevenson and Paige (1968) demonstrated dissolution of ZrO_2-UO_2 fuel in 20 M HF. El-Mahdy (1996) demonstrated that adding fluoride to nitric acid qualitatively increased the dissolution rate. DP-574 (1961) states, "ZrO_2 can be dissolved in boiling 1 M HNO_3 – 5 M HF, but this solution cannot be handled in the type 309SCb vessels in the SRP facility," but more work is required to develop a possible flowsheet, for use in 304L stainless steel. The concern is that the nearly impenetrable zirconia would preclude leaching the uranium. It is likely that the "fluoride-chromate process" advocated in DP-839 (1963) (See Option 4 in Section 3.2.3) would dissolve ZrO_2 at 65 ºC and be compatible with 304L. However, this low nitric acid flowsheet could embrittle the electrodes and is not recommended for use in the electrolytic dissolver.

The model assumed a one-week incremental impact per dissolution for the HTRE BeO-6%UO_2 annular pellets. If the low fluoride flowsheet proves unsuccessful, a high fluoride-low acid flowsheet must be used.

3.3.4 Development Needs for Risk 7

Task 7a: Develop position paper on feasibility of using the same flowsheet for Zr-low U as for UZr. The assumption is likely since fluoride is known to eliminate high uranium losses ($0.1M)

Task 7b: Flowsheet development needed for very high-risk fuels to establish dissolution approach and address issues with high fluoride (corrosion, Gd precipitation, etc.) or alternative reagents (approximately 1 year, $0.6M, + irradiated 1 ½ years, $2M).

Task 7c: Alternative disposition options should be considered ($0.1M).

Initial estimates of cost and schedule are shown in Appendix A and will be refined as necessary.

4.0 Risk 8: H_2 and autocatalytic reactions due to metal fines

Characterization is needed to address potential H_2 generation concerns and autocatalytic reactions with metal fines. Limiting sludge formed in the bundle to less than 3.5 kg per bundle would likely be required. To address vapor space concerns, sampling and characterization of various fuels, non-destructive evaluation (NDE) of all metallic-core fuels, venting of large sealed cans, removal of excess sludge, disposition of

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[31] Hence, these fuels would have to be treated the same as the UZr metal fuels and with a ratio of > 4 moles of fluoride per mole of zirconium, which would limit one charge per batch and require 0.5 M fluoride or greater.

[32] Prajapati (2014) gives a rate in 10 M at 90 ºC corresponding to a penetration rate of 1.9×10^{-7} mils per hour (which is 4-5 orders of magnitude slower than PuO_2). Given the pellets are 0.321" in diameter, ~160 mils penetration would be required for dissolution. Obviously, even allowing for a high rate at boiling, it would take many weeks to complete. Prajapati gives a rate of 2.3×10^{-8} g/m^2-sec at 90 ºC in 12 M HNO_3, which is 0.14×10^{-8} mg/cm^2-min. Assuming boiling acid would be as much as 3 times faster, this gives an implied rate of 0.42×10^{-8} mg/cm^2-min, which gives a penetration rate of 1.7×10^{-8} mils/hour, assuming density of 5.68 g/cm^3. The pellet diameter is 0.321", so the required penetration is 160.5 mils.

sludges, wet shipment, and flowsheet development are needed. There are three aspects of characterization risks: (1) understanding the mechanisms and limits important to safety, (2) characterizing the fuels to stay within these limits, and (3) developing needed technologies such as NDE to support characterization.

Characterization and disposition of sludge (with potentially reactive metallic particulates) is essential. Evidencing the complexity of this task is the fact that Hanford, two-decades after packaging the fuel, is still spending $30 million per year on sludge disposition.[33] Although general characterization is needed for all fuels processed, this section principally focuses on the metallic-core, U-Zr failed fuels, where the characterization needs are an order-of-magnitude more complex.

4.1 Risk 8a: Understanding mechanism and limits important to safety must be defined

Significant corrosion and sludge are expected in the UZr-Defective fuel group. It is likely that approximately half of the HWCTR metal-core fuel (approximately 1 MT total), mostly in oversized cans, has corroded with a substantial fraction as sludge or debris. This estimate is based on cesium measurements when the fuels were moved from RBOF to L Area and indicate that nearly a third of the HWCTR fuel had extensive corrosion. Sautman (2013) reported the cleanup column resin collected 319 of the 1000 curies estimated by Fisher (2010) in the HWCTR fuel. The failed fuel is stored for containment in oversized cans, with a J-tube at the top to allow gas to vent. During repackaging, two Z cans (Figure 4-1-left) were cracked as a result of the volume expansion as metal corrodes to sludge. Although only 22 kg of sludge were collected (6 kg on a filter and 16 kg in the A5 can), the sludge came principally from repackaging of two failed Z-cans. Since the overwhelming majority of the failed fuel was not repackaged, it would not be surprising to find 130-200 kg of sludge in the HWCTR metallic-core cans.[34] Since sludge can contain large quantities of finely divided metal, UZr particulate, and uranium hydride, characterization is essential to safe processing. Contributing causes include fuel damage, the water quality in the cans and the possibility of accelerated corrosion under anoxic (where the oxygen in the water has been essentially depleted) corrosion.

The UZr-Defective fuels have adverse safety properties due to anoxic corrosion. Anoxic corrosion rates for uranium metal are several orders-of-magnitude faster than oxic (clean basin water containing dissolved oxygen) corrosion. Figure 4-1 from Stitt (2012) illustrates the difference. Since some oxygen is always present, it is best to discuss rates at near-oxic and near-anoxic conditions. While normal basin or oxic corrosion creates little hydrogen and sludge with low metallic fine and hydride content, anoxic corrosion generates significant hydrogen and sludge with substantial uranium metal and hydride content. Anoxic corrosion requires a condition where cans are sealed with small leaks to limit oxygen levels. Evidence of such corrosion has been found at both INL (Idaho National Laboratory) and Hanford. At INL, Pahl (2000) documents the conditions found in three leaking bottles of EBR-II metallic core fuel.[35] Figure 4-1 shows neutron radiographs of failed EBR bottles of metallic core fuel before (left) and after (right) water removal. The three leaking containers were approximately ⅓a third filled with water and head space samples found from 28 to 45 psig, with hydrogen contents ranging from 48.9 to 99.94 percent and oxygen levels below the detection limit of 1000 ppm. Although positive pressure was observed, Pahl argues that severe over pressurization is unlikely given the presence of a leak. To explain the observed "well-progressed" but

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[33] For example, even TRU (Transuranic) disposal is limited. Sinkov (2011) cautions that "hydrogen from the U–H_2O reaction poses significant challenges to the safety of … planned sludge transport, storage, processing, and disposal activities," e. g., even TRU disposal is limited to 25 grams per drum due to hydrogen.

[34] Given 22 kg of sludge came principally from cans with 50-60 kg U, and assuming 1/3 to 1/2 of the 1 MT U in HWCTR fuel has failed, would imply by extrapolation a range of (1000*0.33/55)*22 to (1000*0.5/55)*22) based on cesium data and time since repackaging. In addition, Sautman (2013) reports bubbles evidencing failure of many of these cans, with a fall in radiation readings (300-400 mrem/hour) when flushing these cans in RBOF, which further evidence severe corrosion.

[35] The EBR-II metallic core (U-5% Zr, Mo, and other noble metals) is stored underwater at INL in 3600 thin-walled bottles (each bottle 2"D x 30.5" T, containing ~0.7 kg uranium in twelve elements), sealed with a Swagelok fitting (leak tight provided it is tightened correctly). Pahl (1998, 2000), reports that ten leaking containers were examined after sixteen years underwater and found to have various levels of corrosion. In the one case, 20% (based on measured cesium levels in the container water) to 25% (based on visual observation of the settled sludge volume and damaged portions of the fuel) of the original metal had corroded after sixteen years.

incomplete uranium reaction coupled with high hydrogen levels, Pahl postulates that “the original oxygen in the container had long been depleted by chemical reactions or flushed out of the leak,” leading to “an essentially pure hydrogen gas atmosphere” with near-anoxic corrosion.[36] Pahl (1996, 1998) also reports an earlier failed fuel canister with nearly complete corrosion over a shorter period of time due to higher irradiation levels. In both cases, the observed corrosion can only be explained by anoxic corrosion, with the much higher corrosion in the earlier case principally attributed to a much higher burn-up level. In every case, the leaking containers were identified through ultrasonic, NDE measurements. See Pahl (1996, 1998, 2000).

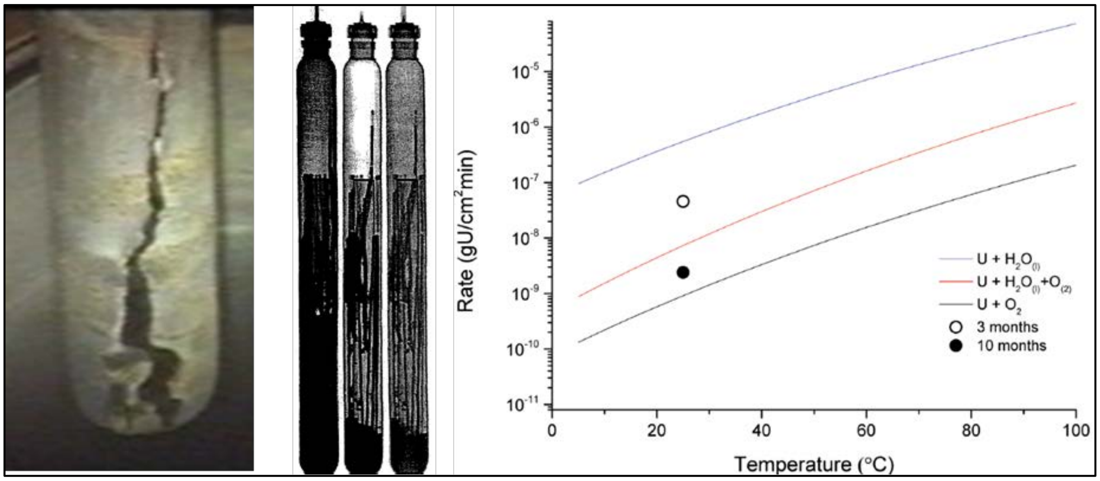

Figure 4-1. Failed Z-can (left); Failed EBR fuel (middle); Anoxic versus Oxic Corrosion (right)[37]

In addition, metallic core fuel poses a risk when transitioning from a wet state to a dry state. Metallic-core fuel is safe wet and safe dry, but serious accidents can occur during the transition from wet to dry. Hanford identified three potential mechanisms that could lead to release of contamination during the transition of uranium core metal from a wet to a dry state:

- A thermal runaway reaction could occur from water vapor reacting with exposed metal if the autoignition temperature is exceeded.
- A hydrogen deflagration (with a maximum rate of pressure rise of 891.3 atm/s) could occur if the lower flammability limit is exceeded.
- An over pressurization with containment failure could result from accumulation of reactant gas.

Hanford’s PSE-003 (1996) postulated the resulting potential onsite dose for the incidents to be 157, 580, and 1,200 rems; thus, Hanford requires safety class and safety significant controls during transition from wet to dry. In all cases, these accidents were principally caused by the quantity and physical and chemical characteristics of the knock-out pot sludge/debris material loaded into the MCO (multi-canister overpack).

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[36] Pahl estimated were cesium in the solution from one bottle released by accident, it would add 5.6 curies to the basin. The measured cesium level correlates to complete corrosion of 1.5 of the 12 elements, or about 20%. Assuming 16 years of corrosion under hypothetical anoxic conditions, Pahl estimates a potential corrosion of 9 grams (initially) U per year times 16 years or 144 grams (not adjusting for average versus initial rate), which would be approximately 20% of the uranium originally present (732 grams given 12 rods times 16 grams per rod). The visual estimate based on sludge accumulation assumes the bottom half of three of the twelve elements (i.e., 25%) had corroded. In contrast, the sodium concentration corresponds to 3.5% of the original sodium. Given the relatively small quantity of sodium and the fact that the reaction of sodium produces only ½ mole of hydrogen per mole reaction, the reaction products are mostly explained by the uranium reaction not the sodium reaction. Although the system contained reactive sodium metal, he notes “uranium corrosion produces most of the hydrogen,” with only a small portion of the sodium reacted, being mostly above the water level.

[37] Left: Z-can; Middle: Neutron radiograph from Pahl, 2000, contrasting water filled (left) with water drained views, showing settled sludge; Right: Stitt’s (2015) figure contrasts the relative reaction rates with and without the presence of oxygen. Stitt (2015) in a more recent study documented the existence in localized hydride layers even in cemented uranium sludge.

Hanford found it necessary to implement controls such as limiting exposed surface area of fuel (1) by opening each container and carefully inspecting and cleaning each element to remove most adhering and settled particulates and (2) by developing a bounding and defensible strategy for estimating exposure surface of damaged fuel. Given the potential reactive surface area for sludge and uranium hydride are, respectively, 1 million and 60 billion times greater per kg than cleaned fuel, establishing a safety basis without repackaging would have been extremely difficult. Nonetheless, the team believes through characterization and adjusting for the differences between the processing and dry storage environments that complete repackaging can be avoided.

As long as the fuel is stored wet, any heat generated by oxidation or corrosion of the fuel can be dissipated into the water. However, a runaway reaction can occur in moist air when there is sufficient reactive surface area, inadequately dissipated heat from oxidation and corrosive reactions with uranium metal, and the temperature exceeds autoignition. Pajunem (1997) shows (Figure 4-2) that control is possible if reactive surface area is limited and the drying temperature is limited to less than 50 ºC. Piepho and Crowe (1998) model (Figure 4-3) the potential runaway reaction that could result if the drying temperature or gas phase conditions is not maintained, though it takes some time to materialize. While Hanford focused on water and oxygen reactions, Delegard and Schmidt (2008) observe that the reaction of uranium metal with HNO_3 is about 10 times more energetic than the reaction of UO_2 with HNO_3; hence, the SRS challenge is exacerbated. See Cooper (1984) for similar autoignition curves for zirconium.

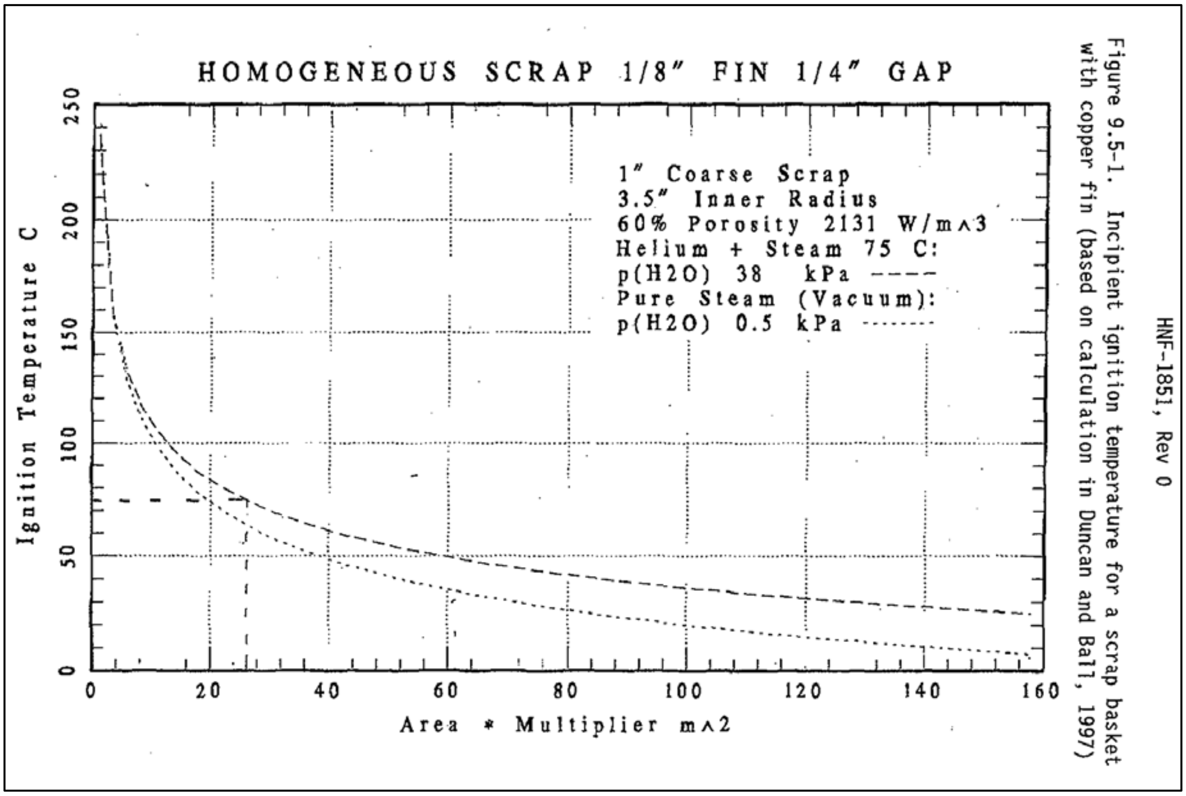

Figure 4-2. Surface area versus autognition temperature-Pajunen (1997).

Figure ES-1. Thermal Runaway Reaction with Nominal and Bounding Values Caused by Loss of Coolant Condition and Closed Multi-Canister Overpack.

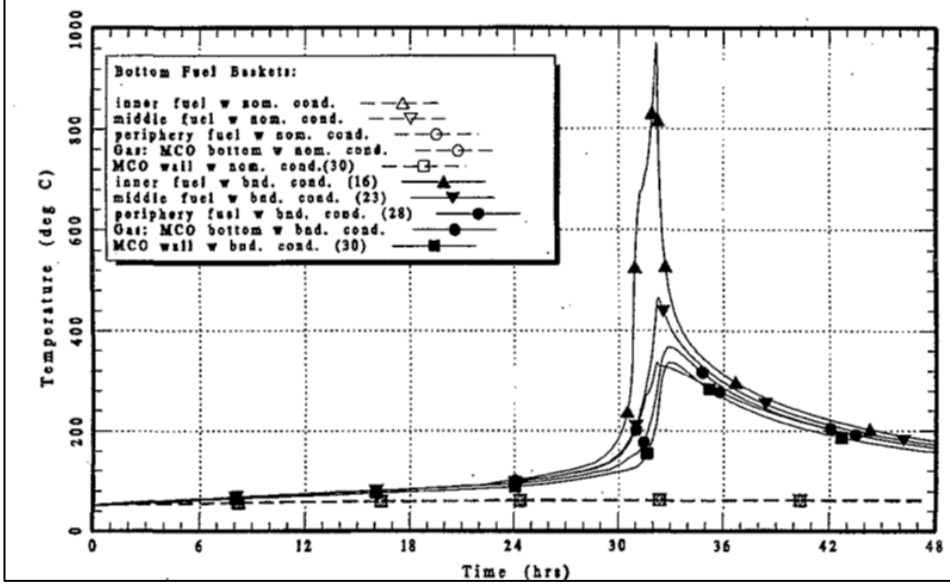

Figure 4-3. Runaway reaction- per Piepho and Crowe (1998)

Although uranium metal can react with water under oxic conditions producing little to no hydrogen , Piepho and Crowe (1998) explained the Hanford safety basis, assumed an anoxic corrosion, 2 moles of hydrogen produced per mole of uranium metal, and 3.5 moles of hydrogen produced per mole of uranium hydride. The maximum corrosion rate for uranium metal of 0.22 g/cm^2-year in ambient water is slow enough that the potential volume of hydrogen per minute would be a fraction of a liter per kg of U in sludge; however, the hydrogen released from the hydride reaction is nearly instantaneous. Although a rapid rise in temperature and pressure (in a closed MCO case) can occur, the system is stable for a day or so before the adverse reaction takes place.

Hydride reactions can create high localized temperatures with an intense release of heat and hydrogen gas and the potential to ignite the uranium metal underneath. Even after removing most of the sludge, Randall (2018), in the final version of the MCO technical handbook, estimates remaining potential uranium hydride in a cleaned scrap basket to be 1.76 g per kg. The cited basis of Pajunen (1998)[38] explains that these estimates are not speculation but empirical results from eight drying tests, and the hydride is proportional to the remaining particulates.[39] Given the large quantity of removed sludge, if the fuel was not repackaged

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[38] Randall assumes 1.56 kg UH_3 per scrap basket (containing 885 kg U assuming 39 assemblies with 22.7 kg U per assembly). Pajunen (1998) bounded the potential hydride present in the particulate remaining on the scrap after cleaning (assuming as much as 8.25 kg of uranium particulate per scrap basket with as much as 9% being hydride. The estimate is further adjusted for particulate formed during drying. Pajunmen argues the expected or design value is 1/20 the safety value. Loscoe (2000) argue that although the hydrides cannot be ruled out, those on fuel exposed to water are likely destroyed by reaction with water unless present in containers where sufficient hydrogen can accumulate to create conditions favorable to further UH_3 formation over prolonged periods of time.

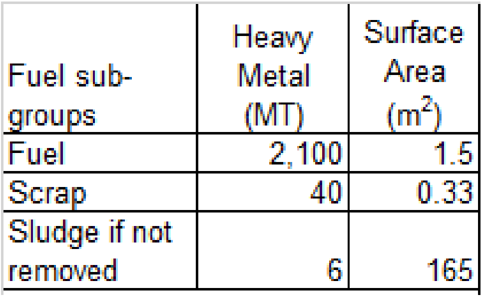
[39] Pajunen (1998) estimates the potential hydride content of particulate to be 4.5 (design case) to 9% (safety case), but in the case for 40 years of storage under a hydrogen atmosphere to be bounded by 54%. These estimates are consistent with "laboratory studies of uranium metal corrosion in water [that] found that 2% to 9% of the metal corrosion product formed is uranium hydride." (Baker, *et al*. 1966). However, these estimates are for low oxygen content water environment and give no credit (for bounding purposes) for the potential destruction of hydride through reaction with water. Pajunen (1998) estimates that for fuel removed from the basin a nominal content of UH_3 in a scrap and fuel basket are 0.5 and 0.113 kg, and the bounding content are 1.56 and 0.324 kg. However, where there is the potential for direct reaction of hydrogen with the metal, the bounding

and cleaned, the potential hydride quantities could easily be an order-of-magnitude or more if the sludge concentrated in a few containers. Baker (1966) estimates that, considering both the production and consumption reactions, the potential for hydrides is about 2 to 9% of the uranium particulates, while Ebner (1996), assuming normal aerated basin water and typically thin corrosion layers, estimates the potential concentration to be 1% or less. Ebner (1996) estimates the reactive surface area of uranium hydride to be 30,000 m^2/kg. The hazard of hydride when spread out over massive metal pieces (which can dissipate the heat) is much less than when combined with high surface area reactive sludge. Ebner estimates that as little as 1% hydride can lower the autoignition temperature (normally 550-595 °C in pure oxygen) of bulk uranium metal to less than 100 °C. Johnson (1996) notes, “There is not a practical basis to demonstrate that all hydrides and metal particles have fully reacted. Also, additional hydrides may form during the period of long-term storage.”

4.2 Risk 8b: Characterizing the fuels to stay within the safety limit

Safety requires a highly defensible quantification of reactive surface area. Unpackaging and cleaning the fuel at Hanford resulted in three material groups, each with a defensible basis for estimating the reactive surface area (See inset Table). The three groups (Table 4-1), ordered in increasing reactivity, were fuel, scrap, and sludge. For each group, Hanford gathered enough samples to have greater than 95% confidence in the estimates. For supporting basis and additional details, see Duncan and Ball (1998) and Pitner (1998) for fuel and scrap classification, Randall (2018) for basket safety and design basis, and McCormick (2014) for basket overview.

Table 4-1. Fuel Sub-Groups with Potential for Increased Reactivity



Fuel sub-groups	Heavy Metal (MT)	Surface Area (m^2)
Fuel	2,100	1.5
Scrap	40	0.33
Sludge if not removed	6	165

1875 out of the Hanford MCO 1940 baskets contained 2,100 MTU and were classified as fuel. The average percentages of intact, breached (minor cladding failure), defected (full break equivalent), and bad fuel were 70.7%, 17.9%, 6.6%, and 4.8%, respectively. For the design case, the reactive surface area for intact fuel was assumed to be zero, defected fuel was assumed to be twice the cross-sectional area, and breached fuel was assumed to be half the cross-sectional area. In contrast, the exposed area of bad fuel was assumed to be 10 times the cross-sectional area. The safety case assumes a reactive surface area of a basket of cleaned fuel to be 0.0008 m^2/kg, with the design basis being 5% of that value. Hanford also assumed that even cleaned and scrap fuel has residual uranium particulate, which approaches 1% of the uranium mass for the bounding scrap case.

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values would be 3.49 and 0.694 kg. The estimates in Randall (2018) are even higher, with a bounding value of 3.12 kg. These estimates do not account for the potential for hydride in particulate sludge, which Pajunen estimates normally being 4.5%, with a bounding content 9% for wet sludge, but in the case for 40 years of storage under a hydrogen atmosphere to be bounded by 54%. These estimates are consistent with “laboratory studies of uranium metal corrosion in water [that] found than 2% to 9% of the metal corrosion product formed is uranium hydride.” (Baker, *et al.* 1966). However, these estimates are for low oxygen content water environment. See also Ebner (1996) for similar estimates.

Scrap fuel is comprised of fuel with bad or gross failures and includes swelled fuel and split cladding. Hanford generated 66 baskets of scrap containing around 40 kilograms of uranium. The safety case assumes a reactive surface area of a basket of cleaned fuel to be 0.005 m^2/kg, with the design basis being 20% of this value.

Hanford sludge is principally a 50:50 mix of $UO_{2.25}$ and $UO_3 \cdot 2H_2O$. The former is the hydrated reaction product with water in the absence of oxygen; the latter is the reaction product with oxygenated water.[40] Unreacted metallic sludge is a significant processing concern due to its high reactive surface area of 0.5 m^2/kg.[41] This estimate assumes 500-micron particles and a hydrogen generation potential of 0.33 liters per hour-kg. Hanford, per Meichle (2020)[42], collected approximately 6 MT of uranium as sludge, settled sludge and almost a metric ton of KOP (Knockout Pot Sludge). The safety basis estimate (design basis is about a third of these values) for the percent uranium found as metal and the distribution are as follows:

- The floor and pit sludge (approximately 2 MT) averaged 12.9% metal but varied widely with KE (K-Area East) sludge having only 0.7% while KW (K-Area West) sludge had 18% metal content. The reason for the difference is that corrosion in KE basins was mainly under near oxic conditions with higher water quality, while the corrosion in KW occurred under more anoxic conditions with poorer water quality.
- The settled sludge (approximately 3 MT) was comprised of fine (<600 micron) but heavy particulate that settled out in collection tanks from KE and KW basins and was found to be 19.4% uranium metal by mass.
- The KOP sludge (approximately 1 MT) was comprised of particulate and debris larger than 500 microns but less than 0.25 inch. Around 29% of the uranium in the sludge was assumed to be metallic uranium.
- If the sludge and debris is not separated (as in the SRS reference case), SRS can assume the Hanford sample results for unseparated sludge. Abrefah (2000) found that in unseparated particulate the average uranium metal fraction is about a third of the total uranium content. This estimate assumes exclusion of the one high sample result taken directly from the center of the breach. SRS can also assume the Schmidt and Plys (2003) and Mellinger (2004) estimate of 500 microns as the design and safety basis value for reactive sludge particle size for non-size-segregated sludge. However, these assumptions require characterization to demonstrate Hanford assumptions are bounding.

4.3 Proposed Processing approach to addressing Risk 8

The following provide key principles to develop a preliminary bounding safety case for dissolution, pending a detailed process hazards assessment.

- In exploring the history of process fires in shear-leach facilities, Swanson (1985) found that the exothermic reactions from small particles and residual hydrides can result in “high temperatures in a reasonably short time under Nuclear Fuel Services dissolver conditions.” He found that when the incidents did occur, which was very infrequent throughout processing, they occurred in baskets outside the liquid level in the vapor space.
- Although not seeing major concern with UH_3 present in small quantities on metal surfaces, Swanson warns that UH_3 and finely divided uranium corrosion products will react readily with HNO_3, contributing to runaway reactions if present in a sufficient quantity. Per Swanson (1985),

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[40] See Schmidt and Delegard (2002). Were corrosion and hydration 100% complete, after many decades, $UO_4 \cdot 4H_2O$ would be the final form. They postulate that the equimolar mixture can be explained by a combined equation (2 U(metal) + 0.375 O_2 + 6.5 $H_2O \rightarrow UO_{2.25} + UO_3 \cdot 2H_2O + 4.5 H_2$). They note, “The “average” dry particle density of the uranium corrosion product mixture, $UO_{2.63} \cdot H_2O$ is calculated to be approximately 6.60 g/cm^3.” They further argue that given infinite time in the presence of water with oxygen, the final sludge would become studtite ($UO_4 \cdot 4H_2O$). Various studies, Leggett and Kemper (1967), Schmidt (2003), and Plys and Schmit (2006) document the progression of uranium corrosion through crumbling or fragmentation of the original metal with terminal particles sizes for N Reactor fuel corrosion ranging from ~400 to 5,000 microns.

[41] Mellinger (2004), citing Schmidt and Plys (2003).

[42] See also PNNL-14228 (2003); Sutter (2009); PNNL-14845 (2004); Knollmeyer (2006); Carro (2010), and Schmidt and Delegard (2002).

the nitric acid reaction with hydride generates 2 moles NO, 2 moles H_2O, and 1.5 moles H_2. Were this to happen instantaneously, the sludge limit would have to be as low as 3.5 kg U assuming 315 g hydride and would result in 163 liters of gas, which if it mixed with the 1000 liter head space would exceed 4% hydrogen. Hence, a K can would not have to be opened, but all the larger cans and FECs would have to undergo sludge removal, if NDE indicates sludge is present.

- Excluding the hydride reaction, other potential sludge-acid reactions could impact operations (e.g., foaming, localized boiling, etc.), but the heat should be removed by the solution, and the localized acid concentration (See Laurinat (2020a)) should remain high enough not to generate high levels of hydrogen.[43]
- The potential for deflagration, which could happen as oxygen and hydrogen mix, when the can seal fails and if pyrophoric material, is above its autoignition temperature. However, any such event would be limited by the small amount of hydrogen in the cans.
- Characterization is limited to establishing that the safety significant properties like mass, particle size, and the reactivity of sludge are bounded by those assumed in extensive Hanford characterization analyses of N-Reactor sludge and debris or to validate new flowsheets where required.

The team believes the following characterization actions are needed:

- Establish a safety basis limit for the maximum sludge and its assumed properties allowed based on potential dissolver solution and off-gas accident scenarios. Pending further study, a sludge limit of 3.5 kg per can to avoid exceeding the LFL in the event of a hydride-acid reaction. More is needed to define whether a bundle sludge mass limit is also needed.
- Sample each diverse type of metallic core fuel to confirm bounding Hanford assumptions.
- Examine by NDE every metallic-core bundle to verify reactive surface area and sludge limits are met.
- Assuming K and small P cans do not require repackaging, open and remove sludge (if found by NDE measurement in excess of limits) from larger OSC, FECs, A cans, B cans, etc. The bundle sludge limit might also require redistribution of K and P cans.
- Ship metallic core fuels from L- to H-Area wet. Although a potential concern is the time when the fuel is carried through open air to be put in the dissolver and then during dissolving until the fuel is submerged below the liquid level.
- Vent large containers that could contain hydrogen. As shown by Pahl (2000), there is a potential for hydrogen in the cans, but given the approximately1,000 liter vapor space of the H-Canyon dissolver, it would take 10 liters of hydrogen to exceed 1% of the tank LFL (lower flammability limit) and four times that amount to reach the LFL. Hence, venting of small cans like K-Cans would likely not be needed, but venting of larger cans would be required, if not already vented. Although there is a potential for positive hydrogen pressure in the cans, as Pahl concluded, the leak should preclude excessive pressure buildup.
- Obtain samples and demonstrate flowsheet where uncertainties remain.

4.4 Development Needs for Risk 8

Fuel should be characterized, with supporting databases and modeling, to mitigate potential H_2 generation concerns and autocatalytic reactions with metal fines.

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[43] Although the localized acid concentration should remain above 3 M, where little hydrogen is generated, the equilibrium (60 ºC) acid concentration, see Maimoni (1980), in the vapor space would be less than 1 M (where the reaction could produce hydrogen). Hence, the conditions would be more favorable for hydrogen generated from corrosion of fines in the vapor space. More study is required.

Task 8a: Review literature, assess, and refine processing rate and corrosion estimates for characterization and NDE feasibility ($0.05 M).

Task 8b: Develop a model (building on the Hanford approach) to identify and assess characterization needs to ensure safe handling and processing. Hanford, ($0.2M).

Task 8c: Develop a characterization and sampling plan with 95% confidence in results and with minimal sampling as done at Hanford per Abrefah (1994) ($0.1M).

Task 8d: Develop and provide QA for supporting databases that map every container to a ready for processing state and provide reliable information for process studies ($0.2M).

Task 8e: Obtain and analyze samples where needed to validate that Hanford data are bounding ($1.1M).

Task 8f: Develop and implement NDE technology to determine whether canned fuels have transitioned into compounds that are not compatible with dissolution process flowsheet assumptions ($0.5M).

Task 8g: Develop basis for wet shipment and explore strategies to optimize cask such as a new rack for HFIR cask ($0.2M). [44]

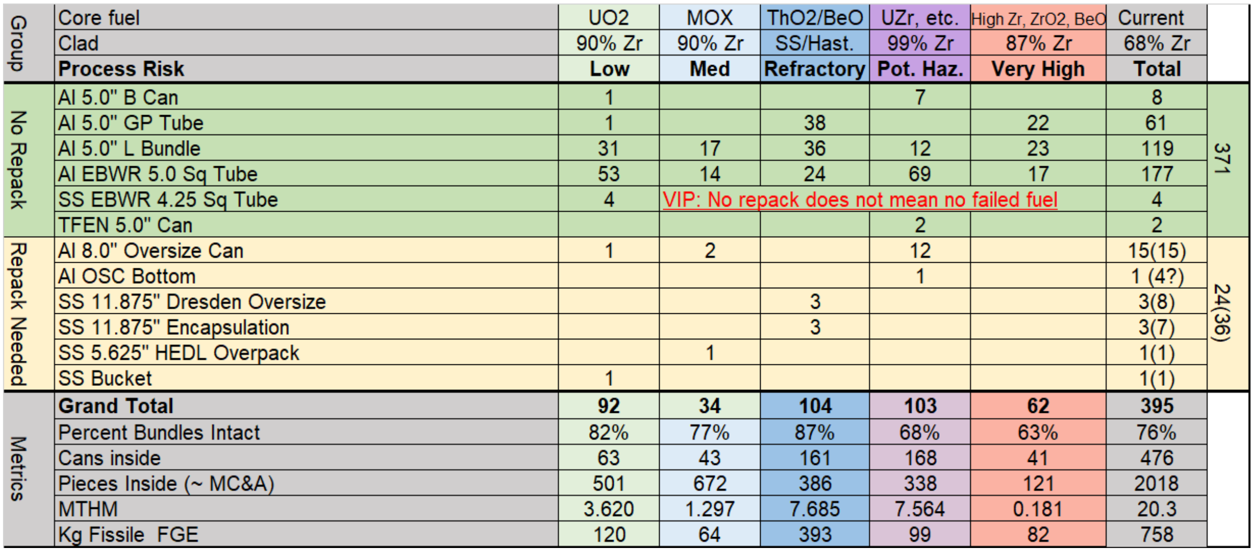
Task 8h: Identify and define pathway for sludge disposition in excess of limit. ($0.3M).

Initial estimates of cost and schedule are shown in Appendix A and will be refined as necessary.

5.0 Risk 9: Oversized items lack of material characterization

Several of the oversized items lack any characterization. Thus, repackaging needs to be defined. In addition, a repackaging approach needs to be developed which includes a feasibility assessment of repackaging without a major basin isolation project. A breakdown of the fuel groups by packaging type is shown in Table 5-1.

Table 5-1. Breakdown by Fuel Risk Group and Packaging (note the total column shows total before and after repackaging).



Group	Core fuel	UO2	MOX	ThO2/BeO	UZr, etc.	High Zr, ZrO2, BeO	Current	
	Clad	90% Zr	90% Zr	SS/Hast.	99% Zr	87% Zr	68% Zr	
	Process Risk	**Low**	**Med**	**Refractory**	**Pot. Haz.**	**Very High**	**Total**	
No Repack	Al 5.0" B Can	1			7		8	371
	Al 5.0" GP Tube	1		38		22	61	
	Al 5.0" L Bundle	31	17	36	12	23	119	
	Al EBWR 5.0 Sq Tube	53	14	24	69	17	177	
	SS EBWR 4.25 Sq Tube	4	VIP: No repack does not mean no failed fuel				4	
	TFEN 5.0" Can				2		2	
Repack Needed	Al 8.0" Oversize Can	1	2		12		15(15)	24(36)
	Al OSC Bottom				1		1 (4?)	
	SS 11.875" Dresden Oversize			3			3(8)	
	SS 11.875" Encapsulation			3			3(7)	
	SS 5.625" HEDL Overpack		1				1(1)	
	SS Bucket	1					1(1)	
Metrics	**Grand Total**	**92**	**34**	**104**	**103**	**62**	**395**	
	Percent Bundles Intact	82%	77%	87%	68%	63%	76%	
	Cans inside	63	43	161	168	41	476	
	Pieces Inside (~ MC&A)	501	672	386	338	121	2018	
	MTHM	3.620	1.297	7.685	7.564	0.181	20.3	
	Kg Fissile FGE	120	64	393	99	82	758	

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[44] Although wet shipment in the 70-ton cask would eliminate run-away reaction concerns, the hydrogen concentration (from existing hydrogen or that generated by radiolysis or chemical reaction) above the liquid space must be kept below the lower flammability limit in air, with the assumed vapor space (approximately1700 liters) of the cask. Askew (email 3/10) estimates at the highest expected average air temperature of approximately 200°C, and using a LFL for hydrogen of 3.5% at this temperature, 1.6 moles of hydrogen are required to reach the flammability limit in the air space of the cask. Askew's preliminary estimate, bounded by wet sludge with a 12.5 watt heat content, is it would take 90 days to reach the limit. In contrast, intact fuel assemblies would be self-shielding and radiolytic hydrogen production would not be a problem.

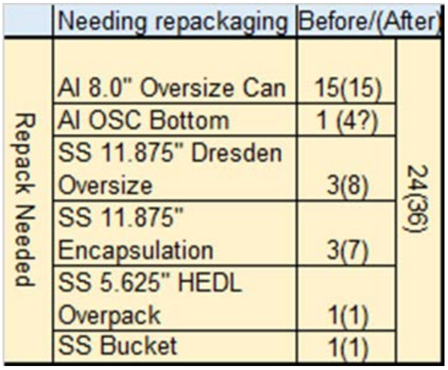
Table 5-1 shows for each risk group the number of each bundle type, grouped by those bundle types that absolutely must be repackaged and those where no repackaging is assumed. Also shown for each risk group are the number of bundles, the percent of the bundles that contain failed or potentially failed fuel, the number of cans inside the bundles, the number of pieces inside the bundles, the mass (kilograms) of heavy metal (U + Pu + Th + Np), and the mass (kilograms) of fissile material. The right hand-column shows the number of containers before and after repackaging, assuming the failed equipment container (stated to be a 5” diameter container but without complete design details available) within the 8” oversized containers would be suitable for electrolytic dissolution.

5.1 Key minimal repackaging risks and assumptions:

Some fuel requires disassembly to fit in a 5” bundle. Where the oversized container is not an 8” OSC (a standard 8” oversized container), oversized fuel assemblies would require disassembly and repackaging. The reason this is not absolutely required for OSCs is that they have a 5” diameter FEC (failed equipment can) inside. Most of the oversized fuels are or have some failed fuel inside. Special tools would be needed and demonstrated to remove rods from oversized assemblies without cutting fuel meat. Note that repackaging is required for some fuels in almost every fuel risk group and that 74% of the bundles contain failed or suspect fuels. Hence, the table presents a highly optimistic scenario from a repackaging perspective, where repackaging is assumed required only where the bundles are greater than the nominal 5” diameter pending further study.

Although over half of the fuel items are failed or are suspect; the failed fuel is in only 26% of the containers. In addition to the 24 oversized containers with failed fuel, an additional 70 or so containers could require repackaging pending further study of the characterization, processing, and repackaging. The group with the most significant risk for complicating repackaging are the metallic core fuels where reactive sludge could be present and require characterization as discussed in Section 4.0 in this report. Table 5-2 shows OSCs that require repackaging. Although the standard OSCs have 5” diameter FEC inside that might fit in the dissolver, the other OSCs must be disassembled and repacked into suitable containers for dissolution.

Table 5-2. OSCs Requiring Repackaging.



	Needing repackaging	Before/(After)	
Repack Needed	Al 8.0" Oversize Can	15(15)	24(36)
	Al OSC Bottom	1 (4?)	
	SS 11.875" Dresden Oversize	3(8)	
	SS 11.875" Encapsulation	3(7)	
	SS 5.625" HEDL Overpack	1(1)	
	SS Bucket	1(1)	

Cs/sludge capture systems with filter and resin columns are needed to repackage OSCs. OSCs have been used for decades at SRS to safely contain damaged or failed fuel. A J-tube at the top of the container allows gas to vent from the container while maintaining confinement of the fuel. Relocation of the fuel from RBOF to L Area that took place between 1997 and 2003 required special controls. Fisher (2010) states, “A special underwater filter/deionizer apparatus was designed, constructed, and deployed to efficiently capture the cesium from the OS canister water prior to opening the OS canister, and to prevent it from dispersal into the general RBOF basin water and mitigate high radiation levels above the water.” The system worked by connecting to preexisting ports on the RBOF oversized cans, which allowed the water and sludge inside to be pulled through a filter, pump, and resin system that captured the cesium contamination, avoiding contaminating the basin. During repackaging, the basin deionizer trapped 8 curies. The RBOF cleanup system worked so well because the old RBOF OS cans were designed to be flushed, with ports and flanges to facilitate cleanout. However, the current L-Area OSCs do not have these fittings to allow flushing.

The inner cans have mechanical seals with high leak potential with fuel inside known to be cut. The OSCs contain diverse, poorly-sealed cans types including K cans (sealed by a bar wedged on the top plate to form a metal-to-metal seal), slug cans (cans containing uranium slugs with hole drilled in lid allowing water contact), and screw-top cans often located within vented 12' tall FEC within the oversized cans. Some of the cans have J-trap seals or openings to allow gas to escape, such as the seven 5" B cans each containing 4" diameter FECs filled with cut pieces of metal. There are A cans filled with cut metal dumped from K-cans. Also, there are two GP tubes with 3" diameter A cans filled with cut pieces. In short, damaged fuel is in direct contact with water and the water in the cans is the poorer-quality water from the basin of origin (e.g., P).

The vented 5" FECs inside the OSCs are known to contain failed fuel (much of it metallic core). Although originally in nine large oversized containers in RBOF, bubbling indicated continuing corrosion. Five of these nine RBOF OSCs were sampled with the highest concentration reported at 3.5×10^8 dpm/ml ^{137}Cs, and one of the OSCs (A5) released 65 curies during movement. Products of this repackaging are in L Basin today and include the bottom of the A5 can (which contains 14 kg of uranium as sludge), an OSC containing the resin column with approximately 250 curies, 13 OSCs (with J-trap vents to prevent activity release to the basin but without connections to facilitate flushing), and one L Bundle, all containing the fuel in its original inner can packaging.

Some cans could have too much reactive sludge. Several cans such as B Cans, failed Z Cans, and smaller FECs have 35-48 kg of uranium metal, likely in the form of reactive sludge and debris. Although the safe mass of sludge has not been defined, it is likely that mass will be well less than 3.5 kg. Consequently, it is highly likely that these cans will have to be opened and cleaned of free sludge. Further, the 5" FECs will likely have to be opened and cleaned of excess sludge.

The conceptual approach for repackaging cans with potential for high sludge content has not been developed. However, it is likely that the three-well Machine Basin Unloading Station in conjunction with a skid mounted cleanup system (with pump, filter, and resin column) could be adapted to minimize basin contamination. Another option is to install needed equipment in the Emergency Basin, which would allow improved confinement of contaminated water. To do this, significant work, tooling, and demonstration would be needed. It is imperative to safety manage, collect, and disposition the potentially reactive degradation products. A decision is needed on whether to package the collected sludge for dissolution or for treatment and disposal as solid waste.

5.2 Development Needs for Risk 9

Develop a repackaging strategy, approach, and any needed new technology, processes, and instruments for repackaging damaged fuel.

Task 9a: Identify repackaging requirements for each fuel, define new containers needed, and develop and QA the supporting database to manage effort ($0.15M).

Task 9b: Assess options and develop a conceptual approach for repackaging damaged fuel bundles and removing and collecting sludge without excessively contaminating basin ($0.3M).

Task 9c: Develop conceptual approach for removing rods from assemblies too large to fit into the electrolytic dissolver ($0.25M).

Task 9d: Develop approach for integrating characterization (developed in task 8) and addressing MC&A

requirements, including development of needed instruments ($0.3M).

Task 9e: Develop options and demonstrate feasibility for in situ passivation of sludge using hydrothermal treatment and for disposal packaging ($0.25M).

Initial estimates of cost and schedule are shown in Appendix A and will be refined as necessary.

6.0 Risk 10: Extensive processing time and potentially unacceptable corrosion rates due to material configuration and bundling

There is a high probability that, without rebundling fuel, electrolytic dissolution would take longer because (1) data show that fuel in aluminum bundles takes much longer, and (2) a number of process constraints (NCS, pebble volume vs. basket volume, fluoride safety limits) would only allow 1 bundle per charge. However, were the fuels rebundled using 4.75” cans, it is possible to regain this lost time as well as reduce processing time. Table 6-1 shows the potential impacts and savings for repackaging. The green versus tan colors indicate what could be done in 4 years. The processing rates assumed a conservative balance between rate and corrosion. It is certainly possible to reduce the process time with higher acid, fluoride, and temperature. If more corrosion risk is accepted, then less process time would be required and the light tan rows could potentially be done within 4 years as well, leaving the dark tan rows (high ZrO_2 and damaged HWCTR metal) for an alternative disposition, such as dry storage.

6.1 <u>Risk 10a: The electrolytic dissolution rate for aluminum bundled fuel lower than expected</u>

The previously stated electrolytic dissolution rates for SS, Al, and Zr, i.e., 50, 30, and 25 kg/day, respectively, are based on fuel bundled in SS cans (because over 70% of fuel in earlier program was bundled in SS cans). However, SRNS-TR-2019-0107 (2019) found the plant demonstrated rates for the smaller portion of stainless steel and zirconium, were about 50% and 25%, respectively, of that expected when bundled in aluminum cans. Although the earlier pilot program demonstrated an adequate zirconium dissolution rate, there was insufficient zirconium clad fuel dissolved to confirm the rate. In fact, SRNS-RP-2019-00278 (2019) reported the Zr dissolution rate was about a third of the rate expected.

<u>Impact</u>: Assuming only a 33% rather than a 50% reduction, the model projects that the electrolytic dissolution would take approximately a half a year longer than the baseline (based on the nameplate assumptions). However, were a decision made to rebundle the fuel from 5” aluminum to 5” steel bundles, the schedule could be reduced by approximately a half a year. The overall change would be a one-year schedule improvement over what it would have been given the slower than planned dissolution of aluminum bundled fuel. Were a 4.75” can used, added savings in reduced risk from stray current dissolution and other physical considerations would also be realized. However, there are additional implications to increasing the repackaging effort in L Area, which would have to be addressed in a trade-off study. Given the H-Canyon operating cost of over $200 millions of dollars per year, the cost savings should more than pay for rebundling in L Area. Except in case where 4.75” SISCs are used, this option only requires replacing the outer bundle, not the inner cans. Assuming 2 bundles repackaged per day, it would take about a year. In most cases, the effort is simply moving intact fuel from one bundle to another. There are issues with long term storage of SS cans in Al racks, but only limited storage for efficient repackaging is required. Note: In some cases, aluminum is more efficient to dissolve, so some new bundles would be Al rather than steel.

6.2 <u>Risk 10b: The electrolytic dissolution and process time impact is substantially greater with one bundle charged</u>

As discussed in each process fuel category, there are numerous places where current packaging would limit the loading to one bundle per charge. For example, NCS would likely limit the H: BeO group to one bundle per charge; pebble volume would limit the H:ThO_2 group to one bundle per charge; and UZr and ZrU fuel loading is limited to one bundle a charge in order to maintain a 4:1 ratio of F/Zr for safety. The process time impact for each fuel group made no adjustment for the potential reduced electrolytic dissolution efficiency.

Although DP-706 (1962) indicates that a steel rod (with 1% of the electrode area) perpendicular to the electrodes but with the rod-ends 1 cm away carried 90% of the current, the efficiency would be much less when the rod is parallel to and further away from the electrodes. The efficiency would be further reduced when consideration is given to basket partitions and other structures that could obstruct one cell taking all the current. Giving a more comparable estimate of the impact, Bull and Koonce (1970) observed, “if the electrodes are wider than the dissolving metal, a fraction of the current will flow between the electrodes and will not be used in dissolution.” This statement was made during a discussion of why the rates for tubes with rods was two-thirds that of the rate for suitcases of plates with wide flat surfaces. Although work is needed to better define the lost electrolytic efficiency, a loss of one-third is assumed (for purposes of a high spot impact estimate) where there is only one bundle per charge. The model indicates the impact could be 0.56 years. Further, chemical dissolution rates, which are typically diffusion limited, would be doubled were two bundles instead of one being dissolved at a time.

6.3 Risk 10c: Tradeoff between rate and corrosion needed to meet timeline and reduce corrosion

The process time estimates were based on conservative assumptions about processing temperature, nitric acid concentration, fluoride levels, and added ANN to reduce corrosion while maintaining a reasonable rate. Additional study, testing, and pilot demonstration are needed to confirm these assumptions. If corrosion is worse than assumed, the estimated process time might have to increase to compensate. However, if the estimates are too conservative or more corrosion risk is allowed, then it could be possible to achieve up to an a third reduction[45] in dissolution time for ThO_2 and UZr fuels. Regardless, repackaging is essential to even considering allowing additional corrosion, because corrosion is unacceptable with current bundling. No improvement is assumed for BeO since the rate is based on proven processing.

The potential saving total is almost 0.5 years. But, crediting this reduction is highly optimistic given the corrosion concerns and is pending a more comprehensive analysis.

6.4 Development Needs for Risk 10

Perform modeling and optimization studies, integrated with lab, bench, and pilot studies, to establish and then validate the optimal process approach. Optimization is vital for timely completion and to reduce corrosion (which is unacceptable with current bundling approach) and involves highly complex tradeoffs.

Task 10a: Prepare a report to support non-intact fuel processing decision. Assess overall NASNF processing viability and rebundling options, accounting for rates, corrosion, and rebundling difficulties ($0.2M).

Task 10b: Develop electrolytic and chemical model for balancing rates and corrosion ($0.2M).

Task 10c: Perform an integrated pilot demonstration to support and validate key model assumptions for the key processes involved ($1.5M).

Initial estimates of cost and schedule are shown in Appendix A and will be refined as necessary.

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[45] For example, the ThO_2 dissolution could be completed in two-thirds of the time were higher initial acid and boiling used. However, this would result in additional corrosion. But, by rebundling to allow two bundles rather than one being dissolved at a time would allow room for more corrosion. These potential changes are purely hypothetical and would require detailed assessment for optimization.

Table 6-1. Summary of impacts and potential savings (years) through repackaging.

Fuel Core* Risk Group	ED Bundles (#)	Baseline ED Time	Stray Current Impact (Yrs)	Physical Impact (Yrs)	Sludge Removal (Yrs)	Sum Chemical Process Impacts	ED Impact if Al bundled	ED one well/ charge Impact	Net impact w/o changes	Savings if Rebundled†	Savings if two wells	Net w/ changes
L: UO2 core	92	0.9	0.0	0.0	0.0	0.0	0.2	0.0	0.3	-0.3	0.0	0.01
M: MOX core	34	0.35	0.02	0.01	0.02	0.13	0.08	0.06	0.3	-0.11	-0.12	0.09
H: BeO:	36	0.20	0.02	0.02	0.00	0.07	0.05	0.06	0.2	-0.09	-0.09	0.04
H: ThO2:	77	0.80	0.05	0.03	0.07	0.67	0.05	0.16	1.0	-0.14	-0.50	0.41
P: UZr:Intact	71	0.92	0.05	0.03	0.06	0.47	0.22	0.21	1.0	-0.30	-0.44	0.30
P: UZr:Defective	39	0.30	0.01	0.01	0.01	0.13	0.08	0.00	0.2	-0.09	0.00	0.14
V: BeO:	7	0.02	0.00	0.00	0.00	0.01	0.01	0.00	0.0	-0.01	0.00	0.01
V: ZrO2:	17	0.23	0.01	0.00	0.00	0.09	0.06	0.00	0.2	-0.07	-0.02	0.07
V: ZrU:	38	0.27	0.02	0.01	0.03	0.47	0.08	0.06	0.7	-0.11	-0.21	0.34
Total	411	4.02	0.21	0.14	0.20	2.04	0.83	0.56	3.98	-1.17	-1.39	1.42

† Although this table shows no credit for process improvement, some rate improvements would be possible were fuel rebundled.

*All estimates are high spot (-30%+50%): M=Medium risk; H=High risk; P=Potentially reactive, and V= Very High Risk.

7.0 Conclusions

The ABD Program is designed to accelerate the deinventory of L-Basin and accelerate the SNF disposition mission. Spent fuel will be dissolved in H-Canyon with no recovery of Uranium-235 (U-235). The dissolver solutions will be temporarily stored, then transferred to the tank farms and immobilized in the Defense Waste Processing Facility. ABD accelerates basin closure, significantly reduces programmatic risk and greatly reduces the lifecycle budget requirements for the site by eliminating the need for a SNF drying and packaging project. The ABD approach represents a significant change to the clean-up approach for the Savannah River Site. The ABD Program requires changes in how H-Canyon is configured and operated.

The ABD program will require development of a strategy for the dispositioning the Non-Aluminum Spent Nuclear Fuel (NASNF), The purpose of this assessment is to bin the inventory into fuel types and evaluate the feasibility, risks and impacts associated with electrolytic dissolution of NASNF at SRS. L-Basin NASNF contains approximately 20 MTHM with approximately 700 kg fissile content. The 395 containers are configured as 371 bundles, 23 oversized cans, and 1 bucket. Through this assessment, the NASNF inventory is put into five categories based on risk:

1. Flowsheet proven (UO_2) – 92 containers
2. Some demonstration needed (MOX & PuO_2) – 34 containers
3. Difficult to fully dissolve (ThO_2, BeO) – 104 containers
4. Potentially hazardous (e.g. UZr) – 103 containers
5. Very high dissolving risk or high U losses in electrolytic dissolver – 62 containers

These five categories are addressed, as appropriate, based on the 10 risks identified as:

1. Physical size of NASNF bundles requires unproven “stray currents” for electrolytic dissolution
2. Impact of bundle bottom dissolution on fuel dropping, buoyancy, and protrusions
3. Sludge behavior and removal following Zr-clad dissolutions
4. Other potential concerns related to ZrO_2 sludge properties, reactivity, and disposition
5. Dissolution and corrosion impacts for refractory oxide pellet fuels (NASNF categories 2&3)
6. UZr fuel hazardous acid reactions (NASNF category 4)
7. Understanding very high risk NASNF and alternative approaches (NASNF category 5)
8. H_2 and autocatalytic reactions due to metal fines
9. Oversized items lack of material characterization and may need repackaging
10. Extensive processing time and potentially unacceptable corrosion rates due to material configuration and bundling

Risks 1 – 4 are common to all NASNF processing and must be addressed even for proven technologies (UO2 core fuels). The issue is not whether the material could successfully be processed, but what is the potential impact associated with the risk. Understanding the risks through further research and demonstration can reduce or eliminate them. Risks 5 – 10 are unique to one or two fuel categories and have specific impacts associated with each risk and fuel type.

Table 7-1 provides a high-spot impact analysis of the various risks assuming success (e.g., the risk is not whether stray current works but how long it takes), limited repackaging, and the electrolytic dissolution rates previously demonstrated. The incremental time impact associated with each risk was estimated along with the potential corrosion for each scope element. The estimated impacts are high spot with potential outcomes 30% less or 50% more.

Table 7-1. Impact Assessment (years of incremental time).

	Baseline w/o repackaging						Expected Risk Impact				Opportunities		Net	
Fuel Group	Number of ED Bundles	Heavy Metal MT	Aluminum MT	Zirconium MT	Stainless Steel MT	Baseline ED Time (yrs)	Common Risks 1-4 (yrs)	Core Processing Risks 5-7 (yrs)	Current Bundling Impac (yrs)	Total w/current bundling (yrs)	Gain Through Rebundling: risk 10 (yrs)	Potential process gain w/more risk (yrs)	Net Impact w/changes (yrs)	Est. 304L Corrosion mils w/o rebundling
UO_2 core	92	3.6	1.4	1.7	0.7	0.9	0.06	0.00	0.22	0.27	-0.26	0.00	0.01	0
Refractory Core	147	9.0	2.4	0.4	4.1	1.4	0.24	0.87	0.46	1.58	-1.04	-0.22	0.32	36
UZr metal Core	110	7.6	2.0	2.3	0.0	1.2	0.17	0.60	0.51	1.28	-0.83	-0.15	0.30	122
Highest risk	62	0.2	0.8	1.3	0.1	0.5	0.07	0.57	0.21	0.85	-0.43	-0.14	0.28	22
Total	411	20.3	6.6	5.7	4.9	4.0	0.55	2.04	1.39	3.98	-2.56	-0.51	0.90	179

ED= Electrolytic Dissolver; MT= Metric Ton; Zr includes Nb; SS includes Hast. & other: Estimates are High Spot (-33% +50%)

A brief review of major impacts (only canyon impacts shown) and opportunities (note that estimated time impacts are preliminary high-spot estimates with substantial uncertainty):

- The impact from longer stray current dissolution, potential physical concerns, and removing ZrO_2 sludge slows historical dissolution rates.
- Dissolving the non-UO_2 corestakes substantially longer (approximately 2 years) than dissolving UO_2 (which is concurrent with electrolytic dissolution).
- The electrolytic dissolution rate is lower for Al-bundled fuel filling only one well. Many bundles (due to NCS, pebble volume, or chemical safety) are limited to one well per charge unless rebundled. The heart of the problem is that the current bundling was optimized for storage and not for processing.
- Rebundling to 4.75" bundles (mostly SS) to avoid the above ED losses and allow two or more wells to be charged per dissolution, which would allow more efficient operation, could regain most of the losses.
- Further study, process optimization, and taking greater corrosion risk could possibly cut processing time by a third, but to avoid double counting, a one-quarter allowance was assumed.
- Even with these changes, the processing time would exceed the four-year allowance by almost a year.
- Impacts not shown are characterization and repackaging risks in L Area, largely driven by the damaged UZr fuel.

There may be the potential to identify alternative disposition pathways for some of the problematic NASNF. This would require DOE support and has unique risks not described here. However, pursuing alternative disposition paths instead of trying to electrolytically dissolve all of the NASNF could simplify the approach, reduce processing risks, and reduce the L Basin lifecycle. However, this study is focused on the baseline approach of NASNF electrolytic dissolution and does not evaluate potential alternative disposition strategies. The ABD program will evaluate alternative disposition options in early FY21. The fuels offering the greatest benefit for alternative disposition are the high ZrO_2 fuel (17 bundles) and the damaged UZr fuel (39 bundles). These two fuels would only require a few canisters and would be the least difficult to put into dry storage assuming use of the Hanford MCO, since the failed UZr fuels are very similar to the N-Reactor fuel, with a well-defined and accepted drying and disposal pathway. Further, the metallic core sludge present in the damaged UZr fuel likely poses the greatest processing, characterization, and repackaging risk as discussed in Section 3.2 of this report. Compared to the baseline approach of electrolytic dissolution of the NASNF, other disposition options may offer programmatic advantages (e.g., cost or schedule savings) that could be evaluated in an evaluation of alternatives.

These conclusions assume that corrosion can be managed. The high spot model (see Appendix B for details) estimates that 304L corrosion (not counting electrolytic dissolution corrosion, which is low) would be approximately 180 (-50%+100%) mils if all these materials were processed. Niobium corrosion would be similarly challenging. Repackaging to allow two or more assemblies rather than just one to be dissolved at the same time reduces both processing time and corrosion by up to a factor of two; hence, rebundling is essential for both rate and corrosion considerations.

8.0 References

Abfah (2000): Abrefah, J., “Drying Results of K-Basin Damaged/Corroded SNF Internal Sludge and Surface Coating,” September 2000; See also Abrefah, J., “Examination of the Surface Coatings Removed from K-East Basin Fuel Elements,” PNNL-11932, September 2000.

Abrefah (1994): Abrefah, J. *et al*., Characterization Plan for Hanford Spent Nuclear Fuel,” PNL-10210 Rev. 0, December 1994.

Akobori & Shiratori (1994): Akabori, M and Shiratori, T, Mitsuo. “Dissolution of ThO_2 Based Oxides in Nitric Acid Solutions at Elevated Temperatures,” *Journal of Nuclear Science and Technology*, 31:6, 539-545, June 1994.

Allison (1960): Allison, W. W., *et al*., “A Preliminary Study of the Pyrophoric Properties of Zirconium…,” WAPD-TM-14, September' 1956.December 1960.

Askew (2020): Askew, N. M., “Potential Hydrogen Issues with Transport and Electrolytic Dissolution of Spent Nuclear Fuel,” draft February 6, 2020.

Baker (1996): Baker, M. M., *et al*., Trans. Faraday Soc., 62, 2513 (1996).

Baryshnikov (1983): Baryshnikov, N. V., *et al*., “Solubility of Lanthanum Fluorides in Nitric Acid Solutions of Aluminum, Trans. from *Zhournal Neorganicheskoi Khimii*, 28:1303-1308 (1983)

Ball and Duncan (1998): Ball, D. E. and D. R. Duncan, 1998, Fuel Surface Area, HNF-SD-SNF-CN-017, Rev. 3, DE&S Hanford, Inc., Richland, Washington, 1998.

Beary (1968): Beary, M. M. *et al*., “K Basin Spent Nuclear Fuel Sludge Treatment Alternatives Study, Volume II-Technical Options,” PNL-10398 Vol.11, January 1995.

Blanco (1962): Blanco. R. E., *et al*., “Aqueous Processing of Thorium Fuels,” ORNL-3219, May 1962.

Bond (1958) Bond, W. D., “Dissolution of Sintered Thorium-Uranium Oxide Fuel in Nitric Acid-Fluoride Solutions,” ORNL-2519, November 4, 1958.

Bull (1972): Bull, H., “Handling Insoluble Materials in the Electrolytic Dissolver,” DPST-71-202, January 11, 1972.

Bull and Koonce (1970): Bull, H. and Koonce, J. E., “Performance of an Electrolytic Dissolver at SRS,” November 29, 1970.

Burch (1979): Burch, W. D., *et al*., “Consolidated Fuel Reprocessing Program Progress Report for Period January 1 to March 31, 1979,” ORNL/ TM-6836, June 1979.

Carlsen (2005): Carlsen, B., “Damaged Spent Nuclear Fuel at U.S. DOE Facilities, Experience and Lessons Learned,” INL/EXT-05-00760, November 2005.

Caracciolo (1964a) -Caracciolo, V. P. to Webster, D. S., “Electrolytic Dissolver: Appendix,” D. S., February 11, 1964.

Caracciolo (1964b) -Caracciolo, V. P., “Electrolytic Processing of Power Fuels,” DPST-64-303, June 9, 1964.

Chafee (2004): Chaffee, G.A., “K Basins Safety Analysis Report,” HNF-SD-WM-SAR-062, Rev.81, Fluor Hanford Inc., Richland, Washington, 2004.

Clark and Kibbey (1958): Clark, W. B. and Kibbey, A. H., “Hydrofluoric Acid Decladding of Zirconium-Clad Power Reactor Fuel,” ORNL-2460, October 15, 1958.

Cooper (1984): Cooper, T. D, *et al*., “Review of Zirconium-Zircaloy Pyrophoricity,” RHO-RE-ST-31 P, November 1984.

Culler (1964): Culler, F. L., *et al*., “Chemical Technology Divisions Annual Progress Report,” ORNL-3627, May 1964

Das & Bharadwaj (2013): Das, D. and Bharadwaj, S. R., "Thoria Based Nuclear Fuels," Springer, 2013.

Delegard and Schmidt (2008)- Delegard, C. H. and Schmidt, A. J., “Uranium Metal Reaction Behavior in Water, Sludge, and Grout Matrices,” PNNL-17815, September 2008.

DOE (1998): DOE, “Supplement Analysis of Environmental Effects · of Changes in OOE's Preferred Alternative for Management of Spent Nuclear Fuel from the K Basins at the Hanford Site, Richland, Washington,” DOE/EIS0245/SA1, July 1998.

DP-1500 (1979): Hyder, M. L., *et al*., Processing of Irradiated Enriched Uranium Fuels at the Savannah River Plant, Savannah River Site, DP-1500, April 1979.

DP-318 (1958): Occhipinti, E. S., *et al*., “Processing of Power Reactor Fuels – 2nd Quarterly Progress Report,” DP-318, 9/1958

DP-319 (1958): Occhipinti, E. S., *et al*., “Processing of Power Reactor Fuels - 3th Quarterly Progress Report,” DP-319, 12/1958

DP-439 (1959): Occhipinti, E. S., *et al*., “Processing of Power Reactor Fuels - 8th Quarterly Progress Report,” DP-439, 11/1959

DP-479 (1960): Meyer, L. H., *et al*., “Processing of Power Reactor Fuels - 9th Quarterly Progress Report,” DP-479, 3/1960

DP-491 (1960): Occhipinti, E. S., *et al*., “Processing of Power Reactor Fuels - 10th Quarterly Progress Report,” DP-491, 10/1960

DP-546 (1961): Occhipinti, E. S., *et al*. Processing of Power Reactor Fuels - 12th Quarterly Progress Report,” DP-546, 1/1961

DP-574 (1961): Rust, G. D., *et al*., “Processing of Power Reactor Fuels - 13th Quarterly Progress Report,” DP-574, 4/1961

DP-647(1961): Clark, A. T., *et al*., “Electrolytic Dissolution of Power Reactor Fuels in Nitric Acid, Savannah River Laboratory,” DP-647, 1961.

DP-706 (1962): Rust, F.G., *et al*., “Processing of Power Reactor Fuels - 16th Quarterly Progress Report,” DP-706, 2/1962.

DP-740 (1962): Clark, A. T., *et al*., “Processing of Power Reactor Fuels - 18th Quarterly Progress Report,” DP-740, 6/1962

DP-824 (1963): Caraccioli, V. P., *et al*., “Processing of Power Reactor Fuels - 21st Quarterly Progress Report,” DP-824, 3/1963

DP-839 (1963): Kishbaugh, A.A., *et al*., “Processing of Power Reactor Fuels - 22nd Quarterly Progress Report,” DP-839, 4/1963

DP-896 (1964): Caracciolo, V. P. *et al*., “Electrolytic Dissolver for Power Fuels, Savannah River Laboratory,” DP-896, 1964.”

Duncan (2005): Duncan, D.R., “Spent Nuclear Fuel Product Specification,” HNF-SD-SNF-OCD-001, Rev. 7., Fluor Hanford, Inc., Richland, Washington, February 23, 2005.

Dyer (2019a): Dyer, W. G., “Extracted TA and Monthly Report Information from 1969 to 1980 electrolytic Dissolver Operation in H Canyon,” SRNS-RP-2019-00278, July 10, 2019.

Dyer (2019b): Dyer, W. G., “H Canyon Electrolytic Dissolver Performance,” SRNS-TR-2019-0107 April 1969 to February 1980,” July 10, 2019.

Ebner (1996)- Ener, M. A., “The Potential Pyrophoricity of BMI-SPEC and Aluminum Plate Spent Fuels Retrieved from Underwater Storage,” INEL-96/0235, 1996.

El-Mahdy (1996): El-Mahdy, G. A. *et al*., “Effect of halide ions on the formation and dissolution behavior of zirconium oxide,” *Effect of halide ions on the formation and dissolution behavior of Zirconium Oxide.”* Vol. 286, pp. 289-294, 1996.

Ferris and Bradley (1965): Ferris, L. M. and Bradley, M. J., “Reactions of the Uranium Carbides with Nitric Acid1,” Journal of the American Chemical Society / 87:8 / April 20, 1965.

Fisher (2015): Fischer, D.L. *et al*., *"*Damaged fuel storage and recovery — A case study," *Management of Spent Fuel from Nuclear Power Reactors, Proceedings of an International Conference*, 31 May–4 June 2010, Vienna, Austria, IAEA, 2015.

Forstner (1968): Forstner, J. L. To Martin, R. I., "Electrolytic Dissolver Limits -Fist Campaign," March 25, 1969.

Forstner (1978): Forstner, J. L., "Nuclear Criticality Safety Analysis 78-7 Dissolution of Miscellaneous Fully-enriched UO2 Fuel and Scrap in the Electrolytic Dissolver," DPSPU-78-272-186, November 9, 1978.

Forstner to Zeyfang (1973): Forstner, J. F. to Zeyfang, R. W., Nuclear Safety of the Zirconium Sludge in the Dissolution of VBWR Fuel," January 14,1974.

Gens (1958): Gens, T. A., "Explosive Reactions During Reprocessing of Reactor Fuels Containing Uranium and Zirconium or Niobium," ORNL-58-11-31, November 13, 1958.

Gens (1962): Gens, T. A., "The Chemistry of Niobium in Processing Nuclear Fuels," ORNL-4311, February 5, 1962.

Gens and Baird (1959), Gens, T. A., and Baird, "Modified Zirflex Process for Dissolution of Zirconium and Niobium-Bearing Nuclear Fuels In Aqueous Fluoride Solutions," ORNL-2713, December 22, 1959.

Gwinner (2020): Gwinner, B., *et al*., "Corrosion of Zirconium in the Context of Spent Nuclear Fuel Reprocessing," CEA-02491643, February 2020.

Harmon (1976): Harmon, H. D, Penetrate-Leach Dissolution of Zirconium-clad uranium Fuels, DP-1424, October 1976.

Hart (1979): Hart, P. F., et a.l, ThO2-Based Pellet Duels -Their Properties, Methods of Fabrication, and Irradiation Performance, PNL-3064, September 1979.

Hutton (1975): Hutton. A. E., "Dissolution of HTGR Fuel", GA-A13278, September 1975.

Hyder (1966): Hyder, L. *et al*., "Dissolution of Thorium Oxide," DP-1044, July 1966.

Johnson (1996) -Johnson, A. B, "Desludging of N Reactor Fuel Canisters: Analysis, Test, and Data Requirements," PNL-10618, January 1996.

Johnson (2003): Johnson, P.G., "Safety Analysis Report for Packaging (On-Site) Multi-Canister Overpack Cask," HNF-SD-TP-SARP-017, Rev. 3C, Waste Management Federal Services, Inc., Richland, Washington, 2003.

Johnson and Crowe (2015): Johnson, M. E. and Crowe, R. D., "Sludge Treatment Project-Engineered Container Retrieval and Transfer System -Thermal and Gas Analyses for Sludge Transport and Storage Container (STSC) Storage at T Plant," PRC-STP-00241, Rev. 4, December 2015.

Jordon (2012): Jordan, J. M. and Lee, S. Y., "Thermal Analysis of SRS 70 Ton Cask." M-CLC-A-00386, Rev. 2., 2012.

Knollmeyer (2006): Knollmeyer, P., et al., "Progress with K Basin Sludge Retrieval, Stabilization, and Packaging at Hanford Nuclear Site," 2006.

Kranzlein (1959): Kranzlein, P. M., "Corrosion of Stainless Steel in Thorex Process Solutions," DP-429, December 1959.

Kranzlein (1960): Kranzlein, P. M., "Corrosion of Stainless Steel in HNO_3-HF Solutions," DP-486, July 1960.

Kullen (1977): Kullen, B. J., *et al*., "Management of Waste Cladding Hulls: Part II: An Assessment of Zirconium Pyrophoricity and Recommendations for Waste Hull Handling," ANL-77-63, November 1977.

Larsen (1954): Larsen, R. P. *et al.*, "A Study of the Explosive Properties of Uranium-Zirconium Alloys," ANL-5135, July 1954.

Laurinat (2020a): Laurinat, J. E. to Moore, E. N., email, February 25, 2020.

Laurinat (2020b): Laurinat, J. E., "Evaluation of Minimum Fluoride Addition for the Electrolytic Dissolution of Zirconium Metal with Other Metals and Alloys," SRNL-L3210-2020-00002, February 11, 2020).

Leggett and Kemper (1967): Leggett RD, and RS Kemper, "Status of Metallic Uranium Fuel Technology*,"* BNWL-CC-1444 Part 2, Battelle-Northwest, Richland, WA., 1967.

Loscoe (2000): Loscoe, P.G., “Transitioning Metallic Uranium Spent Nuclear Fuel from Wet to Dry Storage,” WM’00 Conference, February 27 – March 2, 2000, Tucson, AZ.

Mainmoni (1980): Maimoni, A., “Vapor Liquid Equilibrium for Nitric Acid-Water and Plutonium Nitrate-Nitric Acid-Water Solutions,” UCID-18148, January 1980.

Martin and Field (1958): Martin and Field, “Reactions of Zirconium and Zirconium-based Alloys with Nitric Acid and Hydrofluoric Acid: Hazardous Reactions, Part II- Dissolution Rates,” AERA C/R 2692, United Kingdom Atomic Energy Authority, December 1958.

McCormack (2014): McCormack, R., “Multi-Canister Overpack Presented to: Nuclear Waste Technical Review Board,” August 2014.

Meichle (02000): Meichle, R. H., “Spent Nuclear Fuel Project Technical Databook, Volume 2, Sludge,” HNF-SD-SNF-TI-015, Rev. 30, Vol. 2, March 2020.

Melinger (2004)- Mellinger, G. B., *et al*., “Disposition Options for Hanford Site K-Basin Spent Nuclear Fuel Sludge,” PNNL-14729, January 2004.

Mikeska and Bennison (1999): Mikeska, K. R and Bennison, S. J., “Corrosion of Alumina in Aqueous Hydrofluoric Acid,” Journal of the American Ceramic Society—Vol. 82, No. 12, December 1999.

Moore (1957): Moore, R. L, *et al*., “Nitric Acid Dissolution of Thorium”, Industrial and Engineering Chemistry, Vol. 49, No. 5 · May 1957.

Nayak & Sehgal (2019), Nayak & Sehgal, “Thorium for the Future: Select Papers THEC15,” *Springer*, 2019

Occhipinti and Owen (1964): Occhipinti, E. S. and Owen, J. H., “Dissolution of Zirconium Alloy Fuels in 309SCB Stainless Steel Vessels,” DP-872, June 1964.

Ondrejcin (1980): Ondrejcin, R. S., “Corrosion of High Ni-Cr Alloys and Type 304-L Stainless Steel in HNO_3-HF,” DP-1550, April 1980.

Pahl (1996): Pahl, R. G., *et. al.,* “Technical Assessment of Continued Wet Storage of EBR Fuel” *1996 DOE Spent Nuclear Fuel and Fissile materials Management Conference,* July 16-20, 1996.

Pahl (1998): Pahl, R. G., ”Characterization of Degraded EBR-11 Fuel from the ICPP-603 Basin: National Spent Nuclear Fuel Program FY 1999 Final Report,” ANL-00/9, April 2000.

Pahl (2000): Pahl, R. G., *et. al.,* “Degradation of EBR-II Driver Fuel During Wet Storage” *ANS 4th Annual Topical Meeting*, 2000.

Pajunen (1997) Pajunen, L. “Cold Vacuum Drying Residual Free Water Test Description: Cold Vacuum Drying Residual Free Water Test Description,” HNF-1851, December 23, 1997.

Pajunen (1998): Pajunen, A.L., “Uranium Hydride in Multi-Canister Overpacks,” HNF-3372, October 1998.

Perkins (1973): Perkins, “Electrolytic Dissolution of Zircaloy Clad Fuels,” DPST-73-393, July 9, 1973.

Perkins (1974): Perkins, “Metal Solubilities in Electrolytic Dissolver Solution,” DPST-74-344, May 3, 1974.

Phillips and Huber (1968): Phillips, J. F. and Huber, H. D., “A Process for Dissolution of Alumina-Clad Thoria Fuel Elements,” BNWL-240-July 1968.

Pickett (1988) – Pickett, C., “List of Stainless-steel Clad Type Fuels Processed in 221-H,” 04/22/1988.

Piepho (2001): Piepho, M. G., “Temperatures of Mk-lV Fuel with Low Decay Heat Power during Vacuum Drying,” SNF-9001, Rev 1, Fluor Federal Services, Richland, Washington, 2001.

Piepho and Crowe (1998): H.G. Piepho, R.D. Crowe, “Thermal Analysis of Cold Vacuum Drying of Spent Nuclear Fuel,” HNF-SD-SNF-CN-023, Rev. 0A, July 20, 1998.

Pierce (2008): Pierce, R. A., *et al*., “Flowsheet Studies for the Dissolution of High-Fired Plutonium Dioxide in the Presence of Gadolinium,” WSRC-STI-2007-00673, Rev. 1, July 2008.

Pierce and Olson (2018): Pierce, R. A. and Olson, L. C., “Demonstration of Electrolytic Dissolution Using Actual and Simulated Fuel Elements,” SRNL-TR-2018-00129, Revision 2, March 2018.

Pierce (2003): Pierce, R. A., “Uranium Metal Dissolution in the Presence of Fluoride and Boron,” WSRC-TR-2003-00500, November 2003.

Pitner (1998): Pitner, A. L,. “Summary Assessment of Fuel Damage Distributions in K Basins, Rev. 0., HNF-2586, April 21, 1998.

Plys (1998): Plys, M. G., and D. R. Duncan, “Simulation of Normal and Off-Normal Multi-Canister Overpack Behavior” HNF-2256, Rev 2, DE&S Hanford, Richland, Washington, 1998.

Plys (2000): Plys, M. G., and D. R. Duncan, “MCO Internal Gas Composition and Pressure During Interim Storage,” HNF-SD-SNF-TI-040, Rev. 3B, Fluor Hanford, Inc., Richland, Washington, 2000,

Plys (2004): Plys, M. G., “MCO Internal Gas Composition and Pressure During Interim Storage,” HNF-SD-SNF-TI-040, Rev. 4, Fluor Hanford, Inc., Richland, Washington, 2004.

Prajapati (2014): R.R. Prajapati *et al*., “Dissolution kinetics of zirconium dioxide in nitric acid,” *Desalination and Water Treatment* 52 (2014) 490–497.

PSE (1996): “Preliminary Safety Evaluation for the Spent Nuclear Fuel Project’s Cold Vacuum Drying System,” WHC-SD-SNF-PSE-003 Revision 0, May 1996. See also HNF-3553, Volume 3, Annex B, *Cold Vacuum Drying Facility Final Safety Analysis Report.*

Randall (2018): Randall, S. N., “Spent Nuclear Fuel Project Technical Databook,” HNF-SD-SNF-TI-015, Vol. 1, Rev.9, January 2018.

Reilly (1998): Reilly, M. A., HNF-SD-TI-015, Rev. 6, October 23, 1998.Roberts (1965): Roberts J. T., *et al*., “Reprocessing Methods and and Costs for Selected Thorium Bearing Reactor Fuel Types,” May 1965.

Rudisill and Crowder (2005): T.S. Rudisill and M. L. Crowder, “Dissolution of FB-Line Residues Containing Beryllium Metal,” WSRC-TR-2005-00042, Revision 1, October 2005.

Russell (1967): Russell, E. R. *et al.,* “A Sol-Gel Thorium Oxide with Improved Dissolving Characteristics,” *Nucl Sci Eng.* 30, 20-24(1967).

Sautman (2013): Sautman, M. *et al*., “Storage Conditions of Reactive Metal Fuel in L-Basin at the Savannah River Site,” DNFSB/TECH-38, January 2013.

Schelor (2004): Schelor, J. L. *et al*., “Gas Behavior in large Diameter Containers (LDCs) During and Following Loading with North Loadout Pit Sludge,” SNF-22059, Rev, 0), July 2004.

Schmidt (2004): Schmidt, A. J., ., *et al*., “Segregation of Uranium Metal from KBasin Sludge: Results from Vendor Testing,” PNNL-14845, September 2004.

Schmidt and Delegard (2002): Schmidt, A. J., and C. H. Delegard, “Assessment of K Basin Sludge Volume Expansion Resulting from Uranium Corrosion During Storage. PNNL-13786, Pacific Northwest National Laboratory, Richland, WA., 2002.

Schmidt and Delegard (2003): Schmidt, A. J., and Delegard, C. H., “Updated Volumetric Expansion Factors for K Basin Sludge During Storage,” PNNL-14228, March 2003.

Schmidt and Elmore (2004): Schmidt, A. J. and Elmore, M. R., “Interim Report on Uranium Metal Segregation Testing,” PNNL-14731, April 2004.

Sexton (2004): Sexton, R. A., “Calculation of Bounding Particulate and Water Content from Unwashed K East Fuel (OCRWM),” SNF-21047, Rev. 0, Fluor Hanford, Inc., Richland, Washington, 2004.

Shehee and Pierce (2015): Shehee, T. C and Pierce, R.A., "Kernel Dissolution Studies," SRNL-STI-2015-00048, Rev. 1, June 2015

Sinkov (2011): Sinkov, S. I., *et al*., “Mitigation of Hydrogen Gas Generation from the Reaction of Uranium Metal with Water in K Basin Sludge and Sludge Waste Forms,” PNNL-20455, June 2011**.**

Sloughter (1999a): Sloughter, J. P., 1999a, “Estimates of Particulate Mass in Multi-Canister Overpacks,” HNF-1527, Rev. 3, Numatec Hanford Corp., Richland, Washington, 1999.

Sloughter (1999b): Sloughter, J. P., 1999b, “Uranium Metal Water Reactions in the Integrated Water Treatment System Issue Closure Package,” SNF-4424, Rev. 0, Numatec Hanford Corp., Richland. 1999.

Smith (1960): Smith, P. W., “The Zirflex Process Terminal Development Program,” HW-65979, August 20, 1960.

Sreenivasulu (2017): Sreenivasulu, B, *et al.*, “Dissolution and characterization studies on U–Zr and U–Pu–Zr alloys in nitric acid medium,” Journal of Radioanalytical and Nuclear Chemistry (2017) 311:789–800.

Stevenson and Paige (1968): Stevenson, C. E and Paige, D. M. “Research and Development on Aqueous Processing,” in Reactor and Fuel Processing Technology, Vol. 11. No. 1. Winter 1967-1968. Pp. 93-94.

Stitt (2015)- Stitt, C. A, *et al.*, “An Investigation on the Persistence of Uranium Hydride during Storage of Simulant Nuclear Waste Packages,” PLOS One (July 2015).

Sutter (2009): Sutter, H. G., *et al.,* “K Basins Sludge Treatment Project Phase 1 Technology Readiness Assessment Report,” November 2009.

Swanson (1985): J. L. Swanson, et al., “Laboratory Studies of Shear/leach Processing of Zircaloy Clad Metallic Uranium Reactor Fuel,” PNNL-5708, December 1985.

Swenson (1996): Swenson, C. E., “Performance Specification for the Spent Nuclear Fuel Canister Storage Building,” WHC-S-0425, Rev. 2, Westinghouse Hanford Company, Richland, Washington, 1996.

Swenson and Crowe (2010): Swenson, J. A. and Crowe, R. D.," PRC-STP-00187, "Sludge Treatment Project KOP Disposition - Thermal and Gas Analysis for the Cold Vacuum Drying Facility," March 2010.

TA 2-683 (1969), Swindell, C. W., “Electrolytic Dissolution and Subsequent Processing of Off-Site Fuels,” DPSOX 7313, March 25, 1969.

TA 2-683B (1970), Koonce, J. E. and Zeyfang, R. W., “Electrolytic Dissolution and Subsequent Processing of Off-Site Fuels,” DPSOX 7648, August 31, 1970.

TA 2-683D (1971), Norton, E. R., “Electrolytic Dissolution and Subsequent Processing of Off-Site Fuels,” DPSOX 7847, July 12, 1971.

TA 2-683E (1972), Norton, E. R., “Electrolytic Dissolution and Subsequent Processing of Off-Site Fuels,” DPSOX 7973, February 11, 1972

TA 2-776B (1972), Zeyfang, R. W., “Low-Enriched Uranium Fuel Processing,” DPSOX 8131, December 8, 1972

TA 2-776C (1973), Zeyfang, R. W., “Low-Enriched Uranium Fuel Processing,” DPSOX 8303, October 23, 1973 .

TA 2-827 (1974), Zeyfang, R. W. and Frank. W. L., “VBWR Fuel Processing,” DPSOX 8343, January 23, 1974.

TA 2-834A (1975), Zeyfang, R. W. and Frank. W. L., “Low-Enriched UO2 Fuel Processing”, DPSOX 8636, July 14, 1975.

TA 2-972 (1979), Owen, J. E., “VBWR Fuel Dissolving,” DPSOX 9246, December 18, 1979.

Tallent and Mailen (1977): Tallent, O. K. and Mailen, J. C., "Study of the Dissolution of Refractory PuO_2 in Nitric - Hydrofluoric Acid Dissolvents at 100°C," Nucl. Technol. 32, 167-175, 1977.

Triay (2010): Triay, I. R., “Sludge Treatment Project – Preliminary Project Execution Plan,” June 3, 2010.Truong (2019).

Truong, T. T., “Status Report on FCA Plate Electrolytic Dissolution Studies,” SRNL-STI-2019-00620, Revision 1, December 2019.

Uriarte. A. L and Rainey (1965): Uriarte. A. L and Rainey R H, “Dissolution of High Density UO_2, PuO_2, and UO_2-PuO_2 Pellets in Inorganic Acid,” ORNL-3695, April 1965.

Vander Wall (1959): E. M. Vander Wall. *et al*., “Concentrated Nitric and Dilute Hydrofluoric Acid Mixtures in Dissolution of Zirconium Metal,” *Industrial and Engineering Chemistry* 51.1 (1959) 51-54.

Vander Wall (1960): Vander Wall, E. M., *et al*., “Dissolution of Zirconium and Zircaloy-2 in Nitric Acid Mixtures at reflux Temperatures,” IDO-14497, April 13, 1960. Note: This report with higher rates is assumed to replace earlier January 1959 report with lower rates.

Warren (1962): K. S. Warren *et al*., “Dissolution of BeO- and Al_2O_3- Base Reactor Fuel Elements,” ORNL-3220, January 30, 1962.

Wilson (1972): Wilson, T. R. to Hair, G. H., “Off-Site Fuels- Processing Classification,” December 4, 1972.

Woods (2020): Woods, M. L. to R. G. Brown, “Disposition of Existing H Canyon Electrolytic Dissolver NDE Inspection,” SRNS-El 120-2020-00003, Rev. 1, February 25, 2020.

Wyman (1999): Wyman, H. A., “Estimates of Particulate Mass for an MCO Containing Mark IA Fuel,” HNF-4465, Rev 0, COGEMA Engineering Corporation, Richland, Washington, 1999.

Appendix A. Cost and Schedule

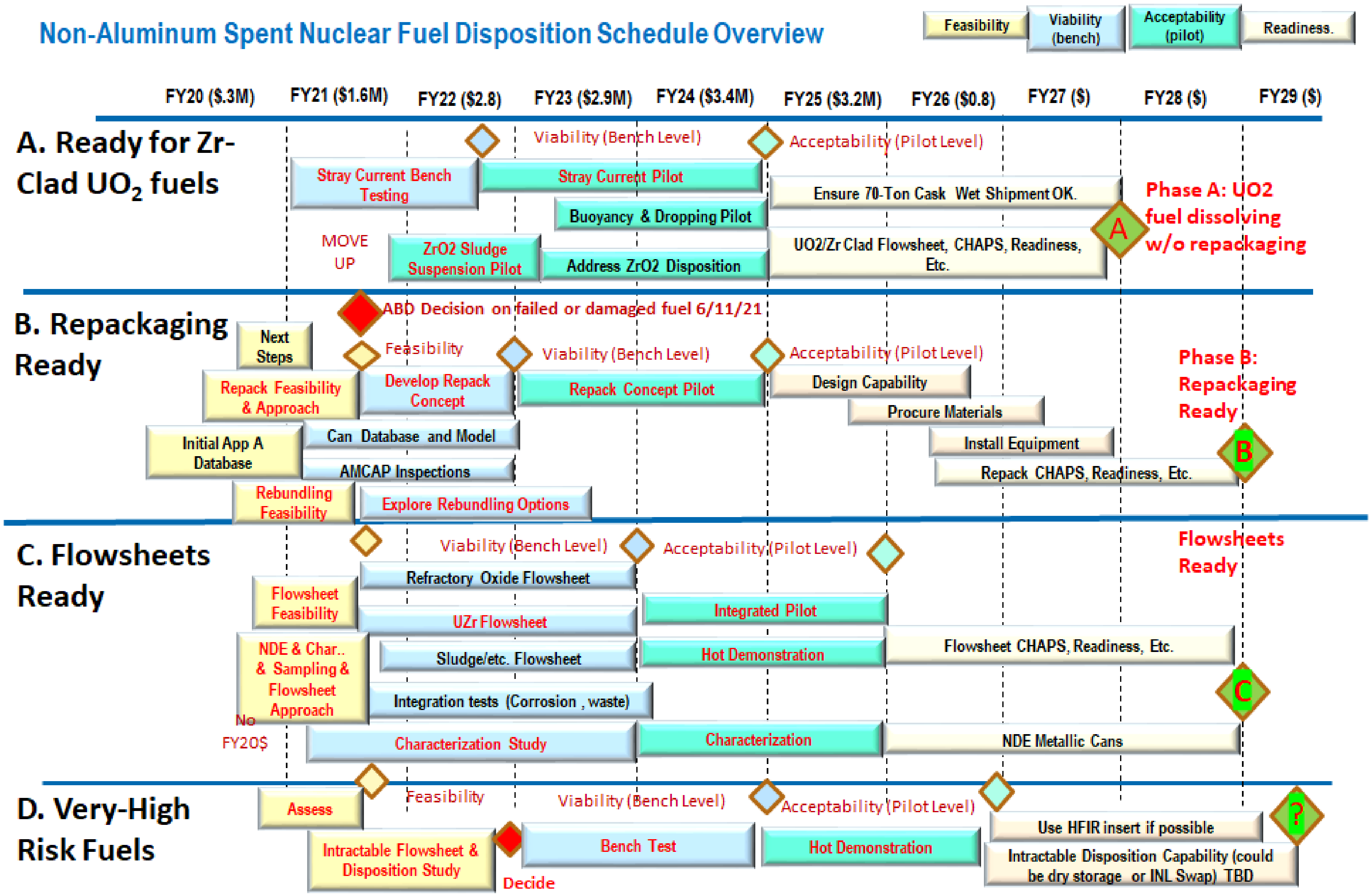

Figure A-1. Development schedule to support the ABD program

Table A-1. The tasks, initial approximate cost, and spend-out projections ($M)

Task	Task Description	FY21	FY22	FY23	FY24	FY25	FY26	Total
1a	Demonstrate bench scale stray current dissolution ($0.1M).	$70	$15	$15				$100
1b	Demonstrate stray current dissolution on a pilot, prototypical scale ($3M).	$100	$1,000	$1,150	$750			$3,000
2a	Demonstrate physical considerations (e.g., damage to dissolver from fuel dropping, floating cans, and protrusions) during pilot demonstration for stray current dissolution. (Cost included in Task 1b)							
2b	Create a CAD model of the dissolver to better understand fuel fit issues and a physical model to evaluate impacts of different fuel configurations, gravity/dropping, etc. ($0.2M)		$200					
3	Conduct pilot testing with a full-scale tank bottom to ensure existing spray sweeps remove solids. The test should address the impact of corrosive degradation of the spray nozzles to ensure that sludge can be removed even with corroded nozzles ($0.5M).	$200	$200	$100				$500
4	Address other Zr processing limiting risks related to sludge include potential higher uranium losses to sludge, potential buildup of sludge in basket, sludge disposal concerns, and the need to understand and address different properties for sludge from UZr processing.	See below tasks						
4a	Develop model for sludge transfer of various sludges (mixed, higher density U-ZrO_2 particles, UZr double oxides) to the tank farm and demonstrate with cold pilot if needed ($0.2M).			$200				$200
4b	Develop conceptual approach and assess feasibility (including MC&A, S&S, and transportation concerns) for solid waste disposal of sludge ($0.2M).			$200				$200
4c	Evaluate all Zr clad oxide fuel types for potential UZr_3 at oxide-Zr cladding interface ($0.1M).				$100			$100
4d	Assess potential for higher uranium losses to sludge during zirconium dissolution ($0.2M).		$200					$200
4e	Assess potential for zirconium oxide build-up in the dissolver basket as part of the pilot demonstration ($0.2M).		$200					$200
5	Model, develop, and demonstrate flowsheets for MOX, ThO2, and High BeO/UO2 fuels, and perform corrosion studies for all fuel types.	See below tasks						
5a	Review literature, assess, and refine processing rate and corrosion estimates for MOX and refractory fuels to support mid-FY21 Non-intact decision ($0.1M).	$50						$50
5b	Complete Lab scale flowsheet studies using simulated pellet to define fluoride levels, processing time, sludge cleanout, and evaluate other concerns ($0.5M).		$150	$200	$150			$500

Task	Task Description	FY21	FY22	FY23	FY24	FY25	FY26	Total
5c	Perform Lab/bench corrosion studies (F^- impact on Nb, Pt, and SS parts) over entire range of bounding conditions for all fuel types. ($0.15M) [Applies to risk 6 & 7 also.]		$50	$100				$150
5d	Model and assess corrosion for all fuel types ($0.2M).		$100	$100				$200
5e	Assess viability of NCS for failed fuel and for refractory fuels containing ThO_2 and PuO_2, which are likely to not remain in the basket and could potentially be transferred out of the dissolver to other tanks. ($ not included since NCS is a plant function)							
6	Modelling and Lab/bench scale studies to refine and validate a UZr flowsheet and address K-Gd-F precipitation, process safety, uranium losses to sludge, etc.	See below tasks						
6a	Review literature, assess, and refine processing rate and corrosion estimates for UZr fuels to support mid-FY21 Non-intact decision ($0.1M).	$100						$100
6b	Perform a modeling and repackaging assessment to develop the optimal UZr approach. Options to address include (1) to decide if repackaging UZr material to one assembly per bundle is warranted and (2) to identify an optimal fluoride strategy that could allow two current bundles per charge. Identify any process feedback information needed for safety ($0.2M).	$50	$150					$200
6c	Lab/bench scale studies to refine and validate UZr flowsheet and address K-Gd-F precipitation, process safety, uranium losses to sludge, etc. ($0.35M).	$50	$150	$150				$350
6d	Demonstrate with irradiated fuel UZr fuel dissolution rate, minimal losses, sludge impact, and stable fluoride interactions ($1.5M).			$100	$550	$850		$1,500
7	Flowsheet development or alternative approaches for high risk fuels [RP]. See subtasks for $.	See below tasks						
7a	Review literature and assess feasibility of using the same flowsheet for high Zr, ZrO2, and BeO as for UZr and define alternatives and make recommendations for ABD Non-Intact Decision. ($0.1M)	$100						$100
7b	Develop flowsheets for high risk fuel disposition (~1 year, $600k) [RP]		$50	$100	$200	$250		$600
7c	Demonstrate new flowsheets for high risks fuels with irradiated fuels (irradiated 1.5 years, $2,000K.) [RP]			$100	$500	$800	$600	$2,000
7d	Alternative disposition options should be considered. (EM)		$100					$100
8	Characterize fuel, with supporting databases and modeling, to avoid potential H2 generation concerns and autocatalytic reactions with metal fines.							

Task	Task Description	FY21	FY22	FY23	FY24	FY25	FY26	Total
8a	Review literature, assess, and refine processing rate and corrosion estimates for characterization and NDE feasibility to support mid-FY21 ABD Non-intact decision ($0.05 M).	$50						$50
8b	Develop a model (building on the Hanford approach) to identify and assess characterization needs (e.g., allowable mass of sludge per bundle) to ensure safe handling and processing. Hanford, Loscoe (2000), estimates that given the variables and uncertainty involved to obtain knowledge with absolute certainty could have required millions of samples. Hence, modeling and parametric bounding of the key variables with conservative limits was the only practical solution. Include resin, filters, and sludge ($0.2M).	$50	$150					$200
8c	Develop a characterization and sampling plan with 95% confidence in results and with minimal sampling as done at Hanford per Abrefah (1994) ($0.1M).	$50	$50					$100
8d	Develop and provide QA for supporting databases that map every container to a ready for processing state and provide reliable information for process studies ($0.2M).	$50	$100	$50				$200
8e	Obtain and analyze samples where needed to validate that Hanford data are bounding ($1.1M).		$100	$100	$300	$300	$300	$1,100
8f	Develop and implement NDE technology to determine whether canned fuels have transitioned into compounds that are not compatible with dissolution process flowsheet assumptions (sludges, oxides, and combinations, etc.) ($0.5M).	$50	$100	$200	$150			$500
8g	Develop basis for wet shipment and explore strategies to optimize cask such as a new rack for HFIR cask ($0.2M)				$50	$50	$100	$200
8h	Identify and define pathway for sludge disposition in excess of limit. Pending study, assume canning in filter vented containers containing <3.5 (TBD) kg sludge for disposition in the dissolver ($0.3M).		$50	$100	$100	$50		$300
9	Develop repackaging strategy, approach, and any needed new technology, processes, and instruments for repackaging damaged fuel.	See below tasks						
9a	Identify repackaging requirements for each fuel, define new containers needed, and develop and QA supporting database to manage effort ($0.15M).	$50	$50	$50				$150
9b	Assess options and develop a conceptual approach for repackaging damaged fuel bundles and removing and collecting sludge without excessively contaminating basin ($0.3M).	$100	$150	$50				$300
9c	Develop conceptual approach for removing rods from assemblies too large to fit into the electrolytic dissolver ($0.25M).		$100	$100	$50			$250

Task	Task Description	FY21	FY22	FY23	FY24	FY25	FY26	Total
9d	Develop approach for integrating characterization (developed in task 8) and addressing MC&A requirements, including development of needed instruments ($0.3M).			$100	$100	$100		$300
9e	Develop options and demonstrate feasibility for in situ passivation of sludge using hydrothermal treatment (see Beary (1968)) and for disposal packaging ($0.25M).		$50	$100	$100			$250
10	Perform modeling and optimization studies, integrated with lab, bench, and pilot studies, to establish and then validate the optimal process approach. Optimization is vital for timely completion and to reduce corrosion (which is unacceptable with current bundling approach) and involves highly complex tradeoffs.	See below tasks						
10a	Prepare a report to support ABD non-intact fuel decision. Assess overall NASNF processing viability and rebundling options, accounting for rates, corrosion, and rebundling difficulties ($0.2M).	$200						
10b	Develop electrolytic and chemical model for balancing rates and corrosion ($0.2M).		$200					
10c	Perform an integrated pilot demonstration to support and validate key model assumptions for the key processes involved ($1.5M).			$250	$750	$500		$1,500
11	Management Support and Oversight (15%)	$198	$550	$542	$578	$435	$150	$2,453
	Total Cost by Year	$1,518	$4,215	$4,157	$4,428	$3,335	$1,150	$18,203

Appendix B. Corrosion Rate Model

Corrosion rate model:
Corrosion rate data were derived from Cole (1974), Ondrejcin (1980), Occhipinti and Owen (1964), Kranzlein (1959), and Kranzlein (1960), with the latter two being given greatest weight since they were most complete and includes data for boiling and at 60 °C. From these, rate equations versus acid molarity at boiling were developed and compared to those for 60 °C. Simple least square fits were used to interpolate the uncomplex corrosion rate for a given acid (between 6-12 M) and fluoride (0.01 to 0.5M). A generalized temperature correction curve was derived from Ondrejcin (1980) and DP-574, but optimized to best fit temperatures between 60 °C and 113 °C (BP @ 10 M). The values are corrected for complexing species like aluminum, zirconium, and thorium approximations to match an equivalent Al rate fitting data from Figures 5-7 from Ondrejcin (1980) for 10M HNO_3 with 0.1M HF. In general, an Al/F ratio of 0.5 and 1, respectively, result in a drop in corrosion of 50% and 80% for aluminum. Simple multiples were used to correlate Zr and Th to Al to fit the same equation. These corrections are simplistic and do not account for the complex interactions of temperature and concentration and the Al/F ratio. The aim was simply to fit empirical results provided by Onrejcin at 95 °C. The model is intended only to provide a conservative order-of-magnitude estimate. The rate in mils per day was then multiplied times the days of actual operation. The resultant question is as follows:

Corrosion 304L in mils per day=((1169.5*Initial_Acid_Molar-3671.4)* Dissolving_Fluoride_Molarity+
6.9045*Initial_Acid_Molar-29.489)*(0.000762*Dissolving_Temperature_oC^3-0.15*
Dissolving_Temperature_oC^2+9.91*Dissolving_Temperature_oC-212)/(0.000762*113^3-
0.15*113^2+9.91*113-212)*IF(Al_F_ratio_Dissolving<=1,1-0.9*Al_F_ratio_Dissolving^0.83,0.1
*(1/(Al_F_ratio_Dissolving+0.5)))/365.

Time Impact Model:
The historic electrolytic dissolution rates were assumed to be for SS, Al, and Zr, respectively 50, 30, and 35 kg/day. A 16% factor from Bull and Koonce was applied to account for batching and transfer time. A simplistic approach assuming 40 operating days per year, seven days a week with 65% attainment would imply the completion in 4 years were no adjustments made or important differences or were no consideration given to the chemical dissolution time. The incremental impacts were estimated in fractions of days for the various needed adjustments for considerations identified in this report. For each fuel type, the assumed number of wells charged and charges per batch were approximated to fit the allowed limits, SS M, Zr M, pebble volume, and concentration (<4 g/l assumed). The methodology is highly simplistic and should be seen as a first pass approximation for comparative purposes only.

Appendix C. Fuels and Packaging

Group Category	Fuel	Outer container	Package Description	Process Tag	Damage Category	MD Fuel Cladding	EDC Count	Current Count	Al to ED kg	SS to ED kg	Zr to ED kg	HM to ED kg
H: BeO:	GCRE	L-Bundle	Al 5.0" L Bundle w/ Al 4.75" Al LISC with Al 4.0"x41 Outer Can with SS 2.0" GCRE Scrap Can w/multiple HX pins (0.25x31")	UO2-BeO	broke	Hastelloy X	1	1	8.9	4.1	0.0	1.0
			Al 5.0" L Bundle w/2 Al 4.75" Al LISC with Al 4.0"x41 Outer Can with SS 2.0" GCRE Scrap Can w/multiple HX pins (0.25x31")	UO2-BeO	broke	Hastelloy X	2	2	20.6	15.9	0.0	3.3
			Al 5.0" L Bundle w/2 Al 4.75" Al LISC with each with an assembly (2.24x34.25")	UO2-BeO	intact	Hastelloy X	1	1	14.5	5.0	0.0	1.7
			Al 5.0" L Bundle w/Al 4.75" SISCs w/4 assemblies (2.24x34.25")	UO2-BeO	intact	Hastelloy X	1	1	21.5	10.0	0.0	3.5
			Al 5.0" L Bundle w/Al 4.75" SISCs w/6 assemblies (2.24x34.25")	UO2-BeO	intact	Hastelloy X	31	31	673.5	311.8	0.0	109.0
H: BeO: Total							36	36	739.0	346.7	0.0	118.5
H: ThO2:	DRESDEN	Dresden 3 assy encap can (#10)	Assume Al EBWR 5.0 Sq Tube for Removed PF-4 Assembly /36 SS rods ~0.48Dx117.937 each; now in Dresden Can#10 (11.875" SS)	ThO2-UO2	broke	SS	1	1	15.0	28.6	0.0	88.1
			Assume Al EBWR 5.0 Sq Tube for Removed PF-5 Assembly /36 SS rods ~0.489Dx117.937 each; now in Dresden Can#10 (11.875" SS)	ThO2-UO2	broke	SS	1	0	15.0	32.9	0.0	86.3
			Assume Al EBWR 5.0 Sq Tube for Removed PF-6 Assembly /36 SS rods ~0.489Dx117.937 each; now in Dresden Can#10 (11.875" SS)	ThO2-UO2	broke	SS	1	0	15.0	34.4	0.0	86.3
		Dresden 3 assy encap can (#27)	Assume Al EBWR 5.0 Sq Tube for Removed PF-1 Assembly /36 SS rods ~0.489Dx117.937 each; now in Dresden Can#27 (11.875" SS)	ThO2-UO2	broke	SS	1	1	15.0	28.6	0.0	86.0
			Assume Al EBWR 5.0 Sq Tube for Removed PF-7 Assembly /36 SS rods ~0.489Dx117.937 each; now in Dresden Can#27 (11.875" SS)	ThO2-UO2	broke	SS	1	0	15.0	31.3	0.0	94.1
			Assume Al EBWR 5.0 Sq Tube for RemovedPF-2 Assembly /36 SS rods ~0.489Dx117.937 each; now in Dresden Can#27 (11.875" SS)	ThO2-UO2	broke	SS	1	0	15.0	28.6	0.0	90.0
		Dresden 3 assy encap can (#28)	Assume Al EBWR 5.0 Sq Tube for removed D-2 Assembly /36 SS rods ~0.441Dx116.656 each; now in Dresden Can#28 (11.875" SS)	ThO2-UO2	broke	SS	1	1	15.0	39.6	0.0	96.9
			Assume Al EBWR 5.0 Sq Tube for Removed PF-3 Assembly /36 SS rods ~0.489Dx117.937 each; now in Dresden Can#28 (11.875" SS)	ThO2-UO2	broke	SS	1	0	15.0	28.3	0.0	88.5
		Dresden Single Assy Encap Can	Assume Al EBWR 5.0 Sq Tube for Removed 4x4x131 SS FDC w/fuel segments	ThO2-UO2	suspect	SS	1	1	15.0	183.6	0.0	12.0
			New Q (presuMed: Mox-SS 4.75") SS tube w/<=36 rods each from 88 tube bundle	ThO2-UO2	suspect	SS	6	2	90.0	479.1	0.0	322.0

Group Category	Fuel	Outer container	Package Description	Process Tag	Damage Category	MD Fuel Cladding	EDC Count	Current Count	Al to ED kg	SS to ED kg	Zr to ED kg	HM to ED kg
		EBWR Square Tube Bundle	Al EBWR 5.0 Sq Tube w/4.25x4.25x130.63" assembly w/34 rods (0.441x116.66")	ThO2-UO2	intact	SS	23	23	345.0	1166.3	0.0	1429.0
			Al EBWR 5.0 Sq Tube w/4.25x4.25x130.63" assembly w/34 rods (0.441x130.63)	ThO2-UO2	intact	SS	1	1	15.0	51.0	0.0	64.0
	ERR	GP Tube	Al 5.0" GP Tube (4.875"x14+') w/125 rods (0.452x65.5")	ThO2-UO2	broke	SS	3	3	22.5	68.7	0.0	289.1
					suspect	SS	34	34	255.0	1127.1	0.0	4707.7
			Al 5.0" GP Tube w/125 rods (0.452x65.5")	ThO2-UO2	suspect	SS	1	1	7.5	6.4	0.0	26.5
						H: ThO2: Total	77	68	870.0	3334.4	0.0	7566.3
L: UO2:	CVTR	L-Bundle	5" Al L-Bundle w/13 intact tubes (1.245 cm x 260.1 cm)	UO2	intact	SS	0	0	0.0	1.1	0.0	3.7
						Zr-4	2	2	15.0	0.4	15.4	47.8
			5" Al L-Bundle w/8 intact tubes (1.245 cm x 260.1 cm)	UO2	intact	Zr-4	1	1	7.5	0.1	5.1	15.9
	EBWR	EBWR FF Bundle	SS EBWR 4.25 Sq Tube w/4.25 X 4.25X 128.25 assembly w/8 plates (0.25 X 4.812 X 36.5)	UO2-SS	intact	SS	4	4	66.4	239.2	0.0	1.7
		EBWR Square Tube Bundle	Al EBWR 5.0 Sq Tube w/2 assemblies (3.75x3.75x62")	UO2	intact	Zr-2	53	53	971.4	151.4	1510.6	2780.9
		L-Bundle	Al 5.0" L Bundle w/16 rods (0.426 X 54.656")	UO2	intact	Zr-2	1	1	7.5	0.4	3.6	11.6
			Al 5.0" L Bundle w/17 rods (0.426 X 54.656")	UO2	intact	Zr-2	1	1	7.5	0.4	3.8	12.4
	GCRE	L-Bundle	Al 5.0" L Bundle w/2 Al 4.75" Al LISC with each with plates (1.31x32.6")	UO2-SS	intact	SS	1	1	18.1	3.7	0.0	0.7
	HWCTR	JCG Can	Al 5.0" B Can w/ None---Assembly 2.05x114	UO2	broke	Zr-4	1	1	10.0	0.0	4.7	20.9
		L-Bundle	5" Al L-Bundle w/10 intact tubes (3.66x13 inches)	UO2	intact	Zr-2	1	1	7.5	0.0	20.9	39.1
			5" Al L-Bundle w/11 intact tubes of which 7 are (2.05x13.74 inches) and 4 are (2.52x13,74 inches)"	UO2	intact	Zr-2/Zr-4	1	1	7.5	0.0	11.5	31.1
			Al 5.0" GP Tube w/ Al 3" A Can w/cut pieces	UO2	broke	Zr-4	1	1	23.6	0.0	4.0	21.2
			Al 5.0" L Bundle w/ 13 Al 4.5" K Can w/cut pieces	UO2	broke	Zr-4/Zr-2	3	3	71.5	0.0	25.6	71.6
			Al 5.0" L Bundle w/ 2 Assemblies- 2.05x114	UO2	intact	Zr-4	2	2	15.0	0.0	4.2	80.7
			Al 5.0" L Bundle w/ 56 CANSU rods and one Al 4.5" K Can w/cut pieces	UO2	broke	Zr-2	0	0	0.0	0.0	0.0	4.1
					intact	Zr-2	1	1	7.5	0.0	7.2	38.7
			Al 5.0" L Bundle w/ 7 tubes (2.28x22 inches)	UO2	intact	SS	1	1	7.5	5.1	0.0	32.3
						Zr-2	1	1	7.5	0.0	8.1	50.3
			Al 5.0" L Bundle w/ 9 tubes (2.52x13.74 inches)	UO2	intact	Zr-2	1	1	7.5	0.0	9.3	29.3
			Al 5.0" L Bundle w/ 4 Al 4.5" K Can w/cut pieces	UO2	broke	Zr-4	1	1	12.9	0.0	0.8	12.4
			Al 5.0" L Bundle w/ SS 4.125" A Can w/cut pieces of oxide fuel from 14 cans	UO2	broke	Zr-2/Zr-4/SS	1	1	7.5	20.3	7.1	43.4

Group Category	Fuel	Outer container	Package Description	Process Tag	Damage Category	MD Fuel Cladding	EDC Count	Current Count	Al to ED kg	SS to ED kg	Zr to ED kg	HM to ED kg
			Al 5.0" L Bundle w/ SS 4.125" A Can w/cut pieces of oxide fuels from 14 cans	UO2	broke	Zr-2	1	1	7.5	20.3	10.3	38.2
			Al 5.0" L Bundle w/ SS 4.125" A Can w/cut pieces of oxide fuels from 12 cans	UO2	broke	Zr-2/Zr-4/SS	1	1	7.5	22.4	4.0	41.8
			Al 5.0" L Bundle w/ SS 4.125" A Can w/cut pieces of oxide fuels from 14 cans	UO2	broke	Zr-2/Zr-4/Zr	1	1	7.5	20.3	7.7	30.2
			Al 5.0" L Bundle w/ SS 4.125" A Can w/cut pieces of oxide fuels from 13 K cans	UO2	broke	Zr-2/Zr-4	1	1	7.5	20.3	9.6	34.6
	SAXTON	L-Bundle	Al 5.0" L Bundle w/ 3.62" SS can w/cut pieces	UO2	broke	Zr	1	1	7.5	32.1	4.5	18.9
			Al 5.0" L Bundle w/ 4.5" SS cans w/total of 40 rods	UO2	intact	Zr-4 and SS	2	2	15.0	64.7	3.4	35.5
			Al 5.0" L Bundle w/ 4.5" SS cans w/total of 43 rods	UO2	intact	Zr-4 and SS	1	1	7.5	32.0	3.0	18.4
		OSC	Repack into Q (presuMed: Mox-SS 4.75" SS) can to include 51 rods (0..391x39.051) (6 are failed) removed from 5.5x5.5x50.25" assembly now in 8" oversized can	UO2	broke	SS/Zr-4	1	1	23.7	11.5	5.1	22.0
	SHIPPING PORT	Bucket	New Q (presuMed: Mox-SS 4.75") SS can for disassembled clusters with a total of 121 pins (in various clusters) w/each pin (0.41x10")	UO2	intact	Zr-2	1	1	12.0	0.0	6.2	17.0
	SPERT	L-Bundle	Al 5.0" L Bundle w/ 1 SS 3.5" SPERT Cans w/cut pieces	UO2	broke	Zr-2	1	1	7.5	14.5	1.0	5.3
			Al 5.0" L Bundle w/ 2 SS 3.5" SPERT Cans w/cut pieces	UO2	broke	Zr-2	1	1	7.5	29.0	2.8	4.9
	VBWR	L-Bundle	Al 5.0" L Bundle w/ 6 Al X cans with SEB (3.75x.4375x53") each with two having CP-1 plates (3"x.14" x (one @15" and one @16.5" long)) and four having CP-2 plats (2.95x0.155x15.875)	UO2	suspect	Zr	1	1	11.0	0.0	1.0	0.5
						Zr-2	0	0	7.0	0.0	0.8	1.6
		L-Bundle w/SEB	Al 5.0" L Bundle w/ Al rod loose transfer can (labelled BWL-Zirc 3"x51.375") w/ 2 intact rods (0.563x37.94")	UO2	broke	Zr-2	1	1	10.2	0.0	0.2	1.9
L: UO2: Total							92	92	1417.9	689.1	1701.7	3620.9
M: MOX:	ANL-MXOX	L-Bundle	5" Al L-Bundle w/4.75"Al SISC w/8 SS clad intact pins and w/4.75"Al SISC w/assembly	UO2-PuO2	intact	SS	1	1	10.7	0.3	0.0	0.6
	B&W	L-Bundle	5" Al L-Bundle w/SS 2.875" B&W can w/cut pin fragments(dry)	UO2-PuO2	debris	SS	1	1	7.5	7.6	0.0	0.1
	EBWR	EBWR Square Tube Bundle	Al EBWR 5.0 Sq Tube w/1 Al 3.75" Square PMOX Can (3.75x3.75x63.875") w/54 rods (0.426x58.25")	UO2-PuO2	intact	Zr-2	1	1	23.8	0.8	12.3	39.5
			Al EBWR 5.0 Sq Tube w/2 Al 3.75" Square PMOX Can (3.75 X 3.75 X 63.875) w/ 54 rods (0.426x58.25") eaxh	UO2-PuO2	intact	Zr-2	4	4	112.7	4.4	60.8	196.0
			Al EBWR 5.0 Sq Tube w/2 Al 3.75" Square PMOX Can (3.75x3.75x63.875") w/54 rods (0.426x58.25")	UO2-PuO2	intact	Zr-2	9	9	293.0	14.5	217.0	696.9

Group Category	Fuel	Outer container	Package Description	Process Tag	Damage Category	MD Fuel Cladding	EDC Count	Current Count	Al to ED kg	SS to ED kg	Zr to ED kg	HM to ED kg
	GCRE	L-Bundle	5" Al L-Bundle w/4.75"Al SISC w/8 SS clad intact pins and w/4.75"Al SISC w/assembly	UO2-BeO	suspect	Hastelloy X	0	0	0.0	2.2	0.0	0.7
	HB Robinson	L-Bundle	Al 5.0" L Bundle w/2 Al 4.5" P Cans and 1 SS 4.0" BW-1 Can	UO2	broke	Zr-4	0	1	11.0	2.5	0.0	0.5
	HWCTR	L-Bundle	Al 5.0" L Bundle w/2 Al 4.5" P Cans and 1 SS 4.0" BW-1 Can	PuO2	broke	Zr-4	1	0	29.5	0.0	0.0	0.0
				UO2-PuO2	broke	SS	0	0	0.0	1.8	0.0	0.5
	LMFBR (HEDL-ENR)	L-Bundle	Q (presuMed: Mox-SS 4.75") SS can w/ repackged 18 welded sealed tubes 0.5"Dx38" Tall w/63 total pins cut pieces from SS 5.625x40.75" HEDL Overpack	PuO2-UO2	broke	SS	1	1	16.8	10.4	0.0	2.7
	LWR	L-Bundle	Al 5.0" L Bundle w/ 2 Al 3.75" G Cans w/cut pieces	UO2-PuO2	broke	Zr-4/SS	1	1	46.2	0.0	0.0	3.7
			Al 5.0" L Bundle w/ 3 Al 3.75" G Cans w/cut pieces	UO2-PuO2	broke	Zr-4/SS	1	1	65.6	0.0	0.0	9.2
	SAXTON	L-Bundle	Al 5.0" L Bundle w/ 1 Al 1" Saxton can and w2 Al 4.5" P Cans w/cut pieces	UO2	broke	Zr-2/Zr-4	0	0	16.8	5.3	0.7	2.4
				UO2-PuO2	broke	Zr-4	1	1	25.0	5.2	0.6	2.3
			Al 5.0" L Bundle w/ 3.62" SS can w/cut pieces	UO2-PuO2	broke	Zr-2/Zr-4	1	1	7.5	23.3	2.4	10.0
			Al 5.0" L Bundle w/ 4.5" SS cans w/total of 46 rdos	UO2-PuO2	intact	Zr-4	1	1	7.5	26.9	5.7	20.7
			Al 5.0" L Bundle w/ 4.5" SS cans w/total of 63 rods	UO2-PuO2	intact	Zr-4	1	1	7.5	27.1	7.8	28.9
			Al 5.0" L Bundle w/ 4.5" SS cans w/total of 70 rods	UO2-PuO2	intact	Zr-4	1	1	7.5	27.8	8.7	33.0
			Al 5.0" L Bundle w/ 4.5" SS cans w/total of 71 rods	UO2-PuO2	intact	Zr-4	1	1	7.5	27.2	9.3	33.0
			Al 5.0" L Bundle w/ 4.5" SS cans w/total of 73 rods	UO2-PuO2	intact	Zr-4 and SS	1	1	7.5	27.2	8.7	33.7
			Al 5.0" L Bundle w/ 4.5" SS cans w/total of 75 rods	UO2-PuO2	intact	Zr-4 and SS	1	1	7.5	27.9	9.2	35.1
			Al 5.0" L Bundle w/ 4.5" SS cans w/total of 80 rods	UO2-PuO2	intact	Zr-4	1	1	7.5	28.0	9.9	34.9
			Al 5.0" L Bundle w/ 4.5" SS cans w/total of 81 rods	UO2-PuO2	intact	Zr-4 and SS	1	1	7.5	27.6	9.4	38.1
			Al 5.0" L Bundle w/SS 4.0" external can SS 3.0" internal can Battelle-Saxton w/rod	UO2-PuO2	suspect	Zr-4	1	1	7.5	54.2	0.1	0.5
		OSC	Repack into Q (presuMed: Mox-SS 4.75") SS can to include 33 rods (0..391x39.051) (9 are failed) removed from 5.5x5.5x50.25" assembly now in 8" oversized can	UO2-PuO2	broke	Zr-4	1	1	18.0	7.8	8.0	15.2
			Repack into Q (presuMed: Mox-SS 4.75") SS can to include 32 rods (0.391x39.051) (20 are failed) removed from 5.5x5.5x50.25" assembly now in 8" oversized can	UO2-PuO2	broke	Zr-4	1	1	22.2	7.3	4.3	14.6
	SRE	L-Bundle	Al 5.0" L Bundle w/Aluminum SRE Can (welded) w/ Al 3.5x110.25" can (AI-15) w/ rods (multiple 0.5 and 0.798"D w/various lengths 1.5-6")	UC	intact	SS	1	1	16.9	13.4	0.0	44.3

Group Category	Fuel	Outer container	Package Description	Process Tag	Damage Category	MD Fuel Cladding	EDC Count	Current Count	Al to ED kg	SS to ED kg	Zr to ED kg	HM to ED kg
M: MOX: Total							34	34	790.5	380.7	375.1	1297.3
P: UZr:	EBWR	EBWR Square Tube Bundle	Al EBWR 5.0 Sq Tube w/1 assemblies (3.75x3.75x58")	U-5Zr-Nb	intact metal	Zr-2	2	2	30.0	0.0	35.2	112.5
			Al EBWR 5.0 Sq Tube w/2 assemblies (3.75x3.75x58")	U-5Zr-Nb	intact metal	Zr-2	59	59	885.0	0.0	1914.8	5705.4
			Al EBWR 5.0 Sq Tube w/2 assemblies (3.75x3.75x62")	U-5Zr-Nb	intact metal	Zr-2	4	4	60.0	5.8	93.3	219.9
				UO2	intact	Zr-2	0	0	6.3	1.5	14.5	26.6
	HWCTR	L-Bundle	Al 5.0" L Bundle w/ 1 Assemblies- one 2.05x114	U metal w/<=2% Zr	intact metal	Zr-2	1	1	7.5	5.4	4.4	23.8
			Al 5.0" L Bundle w/ 14 tubes (1.69x11.26 inches)	U metal w/<=2% Zr	intact metal	Zr-2	1	1	7.5	0.0	4.2	40.4
			Al 5.0" L Bundle w/ 2 Assemblies- one 2.05x114 and one 2.07x118	U metal w/<=2% Zr	intact metal	Zr-4	0	0	0.0	0.0	4.4	23.9
				UO2	intact	Zr-4	1	1	7.5	0.0	0.0	20.9
			Al 5.0" L Bundle w/ w/assembly (2.05 x132 inches)	U metal w/<=2% Zr	intact metal	Zr-2	1	1	7.5	0.0	5.3	49.6
		TFEN Can	TFEN 5.0" Can w/ w/assembly (2.95 x 175 inches)	U metal w/<=2% Zr	intact metal	Zr	2	2	15.0	0.0	26.9	138.1
P: UZr: Total							71	71	1026.3	12.6	2103.0	6361.1
P: UZr: Defectiv	EBWR	EBWR Square Tube Bundle	Al EBWR 5.0 Sq Tube w/Al 4.25" EBWR Sq Can (4.25 X 4.25 X 72.5) w/1 plates (0.276x3.625x53")	U-5Zr-Nb	m.rubble	Zr-2	1	1	31.5	0.0	2.6	9.4
			Al EBWR 5.0 Sq Tube w/Al 4.25" EBWR Sq Can (4.25 X 4.25 X 72.5) w/47 plates (0.212x3.625x55 in 47 cut pieces)	U-5Zr-Nb	rubble	Zr-2	1	1	24.1	0.0	12.9	38.4
			Al EBWR 5.0 Sq Tube w/Al 4.25" EBWR Sq Can (4.25 X 4.25 X 72.5) w/5 plates (0.276x3.625x53")	U-5Zr-Nb	m.rubble	Zr-2	2	2	44.8	0.0	23.8	81.0
	HWCTR	HWCTR OS Can Bottom w/lid	Unknown Sludge Now in A5	U metal w/<=2% Zr	sludge	n/a	4	1	40.0	0.0	0.0	14.2
		JCG	Al 5.0" B Can w/ Al 4.0" FEC B can w/cut pieces	U metal w/<=2% Zr	h.rubble	Zr-4	5	5	112.5	0.0	37.3	172.4
					rubble	Zr-4	1	1	22.5	0.0	7.5	34.5
			Al 5.0" B Can w/ Al 4.0" FEC w/cut pieces	U metal w/<=2% Zr	rubble	Zr-4	1	1	10.0	0.0	7.5	34.5
		L-Bundle	Al 5.0" GP Tube w/ Al 3" A Can w/cut pieces	U metal w/<=2% Zr	rubble	Zr-2	1	1	23.6	0.0	7.1	16.4

Group Category	Fuel	Outer container	Package Description	Process Tag	Damage Category	MD Fuel Cladding	EDC Count	Current Count	Al to ED kg	SS to ED kg	Zr to ED kg	HM to ED kg
						Zr-4	1	1	23.6	0.0	4.5	17.2
			Al 5.0" L Bundle w/ 13 Al 4.5" K Can w/cut pieces	U metal w/<=2% Zr	h.rubble	Zr-4	0	0	4.1	0.0	0.9	6.7
			Al 5.0" L Bundle w/ 10 Al 4.5" K Can w/ cut pieces	U metal w/<=2% Zr	h.rubble	Zr-2	1	1	19.8	0.0	3.7	29.4
						Zr-4	0	0	1.4	0.0	0.2	1.5
			Al 5.0" L Bundle w/ 13 Al 4.5" K Can w/ cut pieces	U metal w/<=2% Zr	h.rubble	Zr-2	1	1	22.5	0.0	7.4	33.3
						Zr-4	0	0	2.7	0.0	0.4	3.3
			Al 5.0" L Bundle w/ 2 Al 4" E Canand 7 K cans with cut pieces	U metal w/<=2% Zr	h.rubble	Zr-2	0	0	10.9	0.0	2.8	27.8
					rubble	Zr-4	1	1	11.9	0.0	1.1	2.2
			Al 5.0" L Bundle w/ 3 Al 4.5" P Can w/ cut pieces	U metal w/<=2% Zr	rubble	Al	1	1	41.3	0.0	0.0	21.8
			Al 5.0" L Bundle w/ Al 4.5" K Can w/cut pieces	U metal w/<=2% Zr	h.rubble	Zr	1	1	8.9	0.0	2.0	0.0
			Al 5.0" L Bundle w/ SS 4.125" A Can w/cut pieces of metallic core fuels from 3 cans	U metal w/<=2% Zr	rubble	Zr-2/Zr-4	1	1	7.5	20.3	1.1	6.3
		OSC	New Q Can for Removed Al 3.125" FEC (Z1) w/ cut or failed pieces- likley rubble Removed from Al 8.0" Oversize Can	U metal w/<=2% Zr	rubble	Zr-2	1	1	23.5	0.0	1.6	25.1
			New Q Can for Removed Al 3.125" FEC (Z13) w/cut pieces Removed from Al 8.0" Oversize Can	U metal w/<=2% Zr	h.rubble	Zr-2	1	1	23.5	0.0	3.2	48.0
			New Q Can for Removed Al 4.0" FEC (Z15) w/ cut piecesRemoved from Al 8.0" Oversize Can	U metal w/<=2% Zr	h.rubble	Zr-2	1	1	31.7	0.0	12.6	47.9
			Q Can for Removed 2 assemblies (one 2.05x132.165 inches and one 2.05x118 inches) w/ Removed from Al 8.0" Oversize Can	U metal w/<=2% Zr	m.rubble	Zr-2	1	1	12.0	0.0	5.0	48.1
						Zr-4	0	0	0.0	0.0	4.9	24.2
			Removed Al 5.0" FEC w/Cans Inside w/13 SPR cans (screw top unknown design) and 2 Al 4.5 inch K-CAns w/cut pieces Removed from Al 8.0" Oversize Can	U metal w/<=2% Zr	h.rubble	(blank)	1	1	27.7	0.0	5.2	36.6
				UO2	broke	Zr-2	0	0	2.7	0.0	1.4	5.3
			Removed Al 5.0" FEC w/Cans Inside w/14 K-cans (4.5" can w/cut pieces) Removed from Al 8.0" Oversize Can	U metal w/<=2% Zr	h.rubble	Zr-2	0	0	12.3	0.0	2.3	30.3
				UO2	broke	Zr-2	0	0	6.8	0.0	4.0	21.6
			Removed Al 5.0" FEC w/Cans Inside w/15 K-cans (4.5" can w/cut pieces) Removed from Al 8.0" Oversize Can	U metal w/<=2% Zr	h.rubble	Al	0	0	4.1	0.0	0.5	14.2
						Zr-2	2	1	57.7	0.0	4.5	52.0

Group Category	Fuel	Outer container	Package Description	Process Tag	Damage Category	MD Fuel Cladding	EDC Count	Current Count	Al to ED kg	SS to ED kg	Zr to ED kg	HM to ED kg
						Zr-4	0	0	4.1	0.0	1.3	9.0
				UO2	broke	Zr-2	1	1	42.3	0.0	5.9	23.6
			Removed Al 5.0" FEC w/Cans Inside w/17 Al screw top slug cans w/cut slugs and w/8 K-cans (4.5" can w/cut pieces) and 4 X cans (3" w/cut slug) Removed from Al 8.0" Oversize Can	U metal w/<=2% Zr	h.rubble	Zr-2	0	0	9.5	0.0	2.2	26.8
					m.rubble	Zr-2	1	1	38.7	0.0	4.8	23.6
				UO2	broke	Zr-2	0	0	1.4	0.0	0.4	1.9
			Removed Al 5.0" FEC w/Cans Inside w/28 Al screw top slug cans w/cut slugs and w/3 K-cans (w/cut pieces) Removed from Al 8.0" Oversize Can	U metal w/<=2% Zr	h.rubble	Zr-2	0	0	4.1	0.0	1.6	7.5
					m.rubble	Zr-2	1	1	42.1	0.0	6.5	32.0
			Removed Al 5.0" FEC w/Cans Inside w/3 Al screw top slug cans w/cut slugs and w/13 K-cans (w/cut pieces) Removed from Al 8.0" Oversize Can	U metal w/<=2% Zr	h.rubble	Zr-2	0	0	8.2	0.0	2.5	25.5
						Zr-2/Zr-4	0	0	1.4	0.0	0.7	7.5
					m.rubble	Zr-2	0	0	1.4	0.0	0.7	3.4
				UO2	broke	Zr-2	1	1	36.8	0.0	3.1	16.4
			New Q (presuMed: Mox-SS 4.75 inch) SS can for Removed Al 4.5" K Can (PK48) w/ Removed from Al 8.0" Oversize Can	U metal w/<=2% Zr	h.rubble	Zr-2	1	0	13.4	0.0	0.3	3.7
			New Q (PresuMed: Mox-SS 4.75 inch) can for Removed Al 3.125" FEC (Z14) w/ cut or failed pieces- likley rubble Removed from Al 8.0" Oversize Can	U metal w/<=2% Zr	rubble	Zr-2	1	0	23.5	0.0	6.1	30.1
			New Q (presuMed: Mox-SS 4.75 inch) SS Removed Al 4" TWIT Can (4" Sch. 40 Pipe) w/ 3 assemblies (each 1.03 x 118 inches) Removed from Al 8.0" Oversize Can	U metal w/<=2% Zr	m.rubble	Zr-2	1	1	30.5	0.0	6.0	31.6
			New Q (presuMed: Mox-SS 4.75 inch) SS can for Removed Al 3.125" FEC (failed Z10) w/ Cut/Failed in HWCTR OS A5, reduced to oxide rubbleRemoved from Al 8.0" Oversize Can	U metal w/<=2% Zr	rubble	Zr-4	1	1	28.2	0.0	5.0	24.2
			New Q (presuMed: Mox-SS 4.75 inch) SS can die Removed Al 3.125" FEC (failed Z2) w/ Cut/Failed in HWCTR OS A5, reduced to oxide rubbleRemoved from Al 8.0" Oversize Can	U metal w/<=2% Zr	rubble	Zr-2	1	0	24.1	0.0	3.0	32.4
P: UZr:Defective Total							39	32	975.0	20.3	218.0	1202.8
V: BeO:	HTRE	L-Bundle	5" Al L-bundle w/4.75DLISC w/4.25"D Welded Can w/ unclad fuel inside	UO2-BeO	intact	none	7	7	96.8	11.6	0.0	4.0
V: BeO: Total							7	7	96.8	11.6	0.0	4.0

Group Category	Fuel	Outer container	Package Description	Process Tag	Damage Category	MD Fuel Cladding	EDC Count	Current Count	Al to ED kg	SS to ED kg	Zr to ED kg	HM to ED kg
V: ZrO2:	EBWR	EBWR Square Tube Bundle	Al EBWR 5.0 Sq Tube w/ Al 4.25" EBWR Sq Can (4.25 X 4.25 X 72.5) w/45 rods (0.376x51")	ZrO2-CaO-UO2	broke	Zr-2	1	1	22.4	0.0	16.9	0.8
			Al EBWR 5.0 Sq Tube w/1 assembly (3.56x3.56x60.63") and w/1 assembly (3.75x3.75x62"	UO2	intact	Zr-2	1	1	15.0	1.5	14.5	26.8
				ZrO2-CaO-UO2	intact	Zr-2	0	0	0.0	2.5	20.0	0.9
			Al EBWR 5.0 Sq Tube w/2 assemblies (3.56x3.56x60.63")	ZrO2-CaO-UO2	intact	Zr-2	5	5	75.0	25.4	199.6	9.0
			Al EBWR 5.0 Sq Tube w/2 assemblies (3.56x3.69x60.63")	ZrO2-CaO-UO2	intact	Zr-2	10	10	150.0	50.7	399.3	18.2
V: ZrO2: Total							17	17	262.4	80.1	650.3	55.7
V: ZrU:	HWCTR	GP Tube	Al 5.0" L Bundle w/ 2 Assemblies (2.3x117 inch)	Zr-9.3U	intact	Zr-2/Zr-4	22	22	165.0	0.0	589.2	34.0
		L-Bundle	Al 5.0" GP Tube w/ Al 3" A Can w/cut pieces	Zr-9.3U	broke	Zr-2	1	1	23.6	0.0	10.5	0.6
			Al 5.0" GP Tube w/ Al 3" A Can(RBOF Storage Can)	Zr-9.3U	broke	Zr-2	2	2	38.0	0.0	21.1	1.3
						Zr-2/Zr-4	1	1	22.8	0.0	7.1	0.4
			Al 5.0" L Bundle w/ 4Al 4.5" K Can w/cut pieces	Zr-9.3U	broke	Zr-2/Zr-4	1	1	13.4	0.0	4.2	0.2
			Al 5.0" L Bundle w/ 2 Al 4.5" K Can w/cut pieces	U metal w/<=2% Zr	h.rubble	Zr	2	2	17.7	0.0	8.0	0.1
				Zr-9.3U	broke	Zr-2	1	1	13.7	0.0	2.6	0.1
						Zr-2/Zr-4	0	0	3.7	0.0	1.5	0.1
			Al 5.0" L Bundle w/ 2 Al 4.5" K Can w/cut pieces and 1 tube (2.52x13.74 inches)	Zr-9.3U	broke	Zr-2	1	1	11.0	0.0	1.6	0.1
			Al 5.0" L Bundle w/ 2 None w/cut pieces and 1 tube (2.52x13.74 inches)	UO2	intact	Zr-2	0	0	0.0	0.0	1.0	3.3
			Al 5.0" L Bundle w/ 3 Al 4.5" K Can w/cut pieces	Zr-9.3U	broke	Zr-2	2	2	25.1	0.0	4.7	0.2
			Al 5.0" L Bundle w/ 4 Al 4.5" K Can w/cut pieces	Zr-9.3U	broke	Zr-2/Zr-4	1	1	13.9	0.0	3.9	0.2
			Al 5.0" L Bundle w/ Al 3" Z20 Can w/cut pieces	Zr-9.3U	broke	Zr-2	1	1	35.9	0.0	10.1	0.5
			Al 5.0" L Bundle w/ Assembly (2.3x118 inch)	Zr-9.3U	intact	Zr-4	1	1	7.5	0.0	13.2	0.9
			Al 5.0" L Bundle w/ assembly (2.56x118 inches)	ThU	intact metal	Zr-2	2	2	15.0	0.0	14.4	79.1
V: ZrV: Total							38	38	406.4	0.0	693.0	121.2
Grand Total							411	395	6584	4875	5741	20348