Contract No:

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Disclaimer:

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INTRODUCTION

The Savannah River Site's H-Canyon facility has been tasked with accelerating the disposition of used research reactor fuels. The fuels are primarily aluminum clad uranium oxide, uranium cermet, or uranium-aluminum alloy. Enrichments, prior to irradiation, range from 19 wt.% up to 93 wt.%. These fuels include Material Test Reactor (MTR) type fuels as well as High Flux Isotope Reactor cores. This paper deals only with the former.

MTR type fuels are dissolved in nitric acid in a large volume vessel containing several thousand liters of nitric acid. Individual fuel assemblies are small, typically 2 to 4 feet in length. They are thus bundled together to optimize storage space and dissolution. A bundle is an aluminum tube typically containing 3 to 5 MTR type fuel assemblies and is approximately 11 feet long.

These MTR bundles are fed to the nitric acid dissolver in a controlled geometry that provides spacing for criticality safety during dissolution. This spacing device, called a well-insert, contains 10 locations for fuel bundles. The insert was originally designed for the processing of SRS reactor fuels which contained less than 3 kg of highly enriched uranium, were initially less than 93 wt.% enriched, and had received significant burnup. The bounding MTR fuel bundle is assumed to have 3.2 kg of 93 wt% HEU. The lattice design is also significantly more reactive than the SRS reactor fuel design.

The consequence of this bounding bundle assumption was that no more than five of the wells could be charged with fuel for dissolution at any one time. An insert plug was made as a passive control feature that blocked fuel from being inserted into five of the wells, in an alternating pattern to maximize separation between the fuels. The result was that when a shipment of 12 bundles would arrive at the facility, they would have to be dissolved in three sub-batches of 3, 4, and 5 bundles each.

Recent analytical work was able to show through advanced, though homogenized, modeling that it was possible to safely dissolved six of the bounding bundles at one time. This is a significant improvement, allowing for two sub-batches of 6 bundles each during dissolution. An insert plug that blocks only four of the ten wells was installed accordingly.

This paper investigates the possibility of loading all ten wells with MTR-type fuel bundles in order to maximize the utilization of the dissolver. All calculations are performed in SCALE 6.1 using the ENDF-VII 238 group cross section library (Ref. 1) and using nominal chemical parameters publicly available.

METHODOLOGY

Differences from Current Analytical Approach

The current analytical approach to criticality safety for MTR-type fuel dissolution in the H-Canyon dissolvers is highly conservative and simplified (Ref. 2). Instead of modeling the entire detail of the dissolver, it is simulated as a right circular cylinder of stainless steel with no internal components. A thick, flat concrete slab is modeled as the base. The well insert supporting structure is ignored save for the stainless steel tubes themselves and their spacing. All of this heterogeneity is replaced with additional bulk fissile solution which would otherwise not physically be present.

The largest source of conservatism comes from how the fuel is modeled. The fuel is simulated as a slug of 150 g U/L uranyl nitrate solution, compared to the bulk solution of 4.0 to 20 g U/L. This value is based on an SRNL mixing study which analyzed the formation of high concentration layers around the dissolving uranium in a static beaker. The 150 g U/L values is approximately 4 times what was observed. The canyon dissolvers are not static. The acid is boiling during dissolution and the uranium disperses into the bulk solution effectively immediately. The modeled 150 g U/L solution, assumed to be 93.5 wt.% enriched and drives up the k-effective value of the system. Fig. 1 represents the dissolver under the current analysis.

This study replaces the high concentration solution with an actual discrete model of a bounding bundle type (Fig. 2). Discrete plates are modeled with a gap between each. The fuel cross section is modeled after the more reactive fuel in the SRNS used fuel inventory. To make the fuel type bounding, the plate spacing is adjusted, the enrichment fixed at 93.5 wt.%, burnup is not credited, the aluminum content is reduced to maximize the cross-sectional uranium content, and the fuel form extended for the entire 11-foot length of the bundle. This results in a total fissile mass of ~3.5 kg HEU while in reality no more than ~3.2 kg can physically be present.

For additional conservatism, the interstitial moderator is retained as the 150 g U/L solution. It is conceivable that acid flow and bubble formation would be minimal between the plates as opposed to on their surface.

This study also models details of the dissolver (Fig. 3). The full detail of the insert and inset column are modeled. The sloped bottom of the tank as well as the air gap between

the floor and tank are modeled as built. The only conservatism is that the heating coils in the tank are not modeled. The level of the bulk solution is adjusted to conservatively account for this assumption.

Determination of Safe Multiplication Factor

An in-house validation for the SCALE 6.1 KENO-VI code for HEU solution systems was performed. A conservative k_{SAFE} assumed for this work was 0.9564 when gadolinium poison is added to the solution and 0.9664 when no gadolinium is present. The difference is due to taking a applicability penalty when considering the benchmark data for gadolinium poisoned HEU solution. These limits were compared against the calculated k-effective plus two times the Monte Carlo relative uncertainty.

Mirroring of Current Analyses' Conditions

The normal condition was assumed to be all ten wells loaded with the bounding discrete fuel assembly as described above. Consistent with the current analysis, normal condition cases with and without 0.15 g/L gadolinium nitrate poison and with and without 5M excess nitric acid were modeled. The final nitric acid is to prevent precipitation of aluminum, plutonium, and other dissolved solids, maintain a reasonable dissolution rate throughout the batch operation, and were at one point to facilitate solvent extraction with minimal chemical adjustment. The normal condition (or base case) model also considered these conservative assumptions:

- Bulk solution concentration of 4.0 gU/L
- 500 g of undissolved uranium solids in an optimally moderated hemisphere below the wells
- Placement of the wells marginally closer to account for manufacturing tolerances

Credible abnormal and some beyond credible abnormal conditions from the current analysis were also simulated. These included:

- Movement of the wells to contact to similar beyond credible severe damage (note the dissolver insert is a credited design feature and designed not to have this type of failure)
- Increase of the enrichment to 100 wt.% U-235
- Decrease of the aluminum in the fuel meat to 50%, 25%, and 0% of its nominal value to account for inconsistent dissolution and severe manufacturing errors (which would have been rejected before or during irradiation)
- Increasing the bulk solution from a nominal 4 gU/L to 4.8 (credible over concentration), 10 gU/L, and 20 gU/L

All abnormal conditions were repeated in 4-case sets both with and without gadolinium poison and with and without 5M excess nitric acid.

All changes in solution and fuel meat composition were entered as atom densities which were manually computed.

RESULTS

Table I provides the results of the base case - or normal condition models. Table II presents the results of the credible abnormal cases and beyond credible abnormal branch cases in sets.

The only cases that did not meet k_{SAFE} were at high concentrations with no gadolinium and/or at physical minimums with no gadolinium. While incorrect neutron poison is a credible upset, deformation of the well insert to the extent that all wells are physically touching is beyond the design basis. The well insert is a credited safety class design feature. The high concentrations are a credible scenario but would represent loss of volume control. It is not credible that this would be concurrent with incorrect poison.

The detailed "10 well loading" model and current homogenous "6 well loading" can be compared. Maintaining the bulk solution at 4.8 g/L stays below k_{SAFE} for both models at 0.839 and 0.941 respectively. Increasing to 10.0 g/L surpasses it for both models where they have a similar k_{eff} of 0.987 and 0.986 for the detailed and homogenous model respectively. This implies the bulk solution starts to drive the multiplication at a concentration greater than 4.8 g/L. Further investigation of this effect on the detailed model may be warranted.

CONCLUSIONS

From the standpoint of criticality safety for the dissolver, loading of up to all ten wells should be achievable. The only cases that did not meet k_{SAFE} were those of beyond the credible abnormal condition of the deformation of the well insert. Using or deforming a well insert is considered not credible as it is a quality assured credited design feature.

This is appealing in the potential for throughput but has limitations that need further investigation. Dissolution of aluminum-based reactor fuel carries with it the non-nuclear concern of hydrogen generation. Hydrogen gas is purged from the system as it is generated. Hydrogen generation rates for all ten wells loaded with high aluminum fuels may be too high for the purge system to handle. Logistics is the other issue. Bundles currently arrive at H-Canyon in batches of 12. To operate efficiently the transport would need to be increased to 20 bundles, or in-canyon storage would have to maintain a semi-continuous inventory of 60 bundles for processing.

ACKNOWLEDGEMENTS

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While the co-authors currently hail from different companies, it should be noted this was not a multi-company work. At the time the work was conducted, all co-authors worked for Savannah River Nuclear Solutions.

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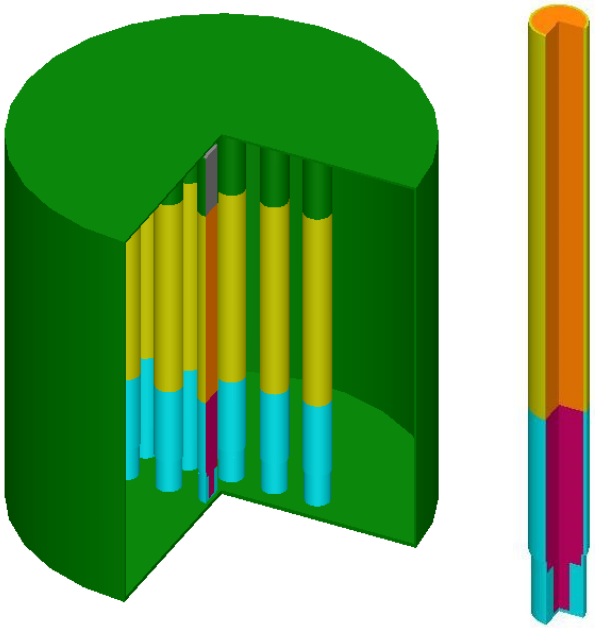

Fig. 1: Current analytical model of H-Canyon Dissolver

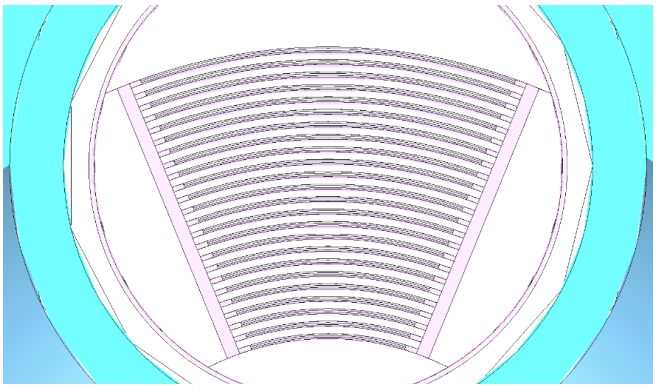

Fig. 2: Discrete fuel bundle cross section.

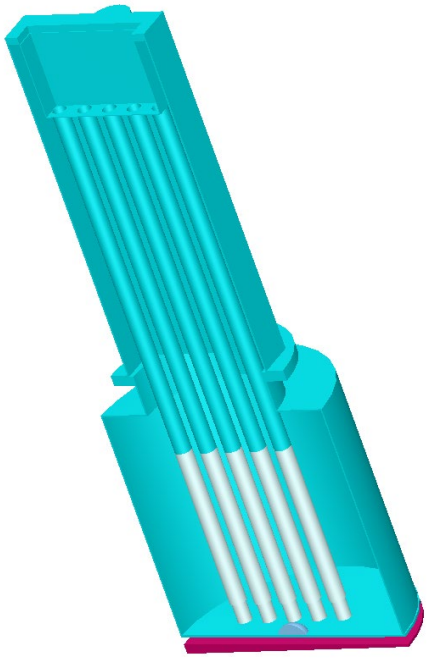

Fig. 3. Detailed Dissolver Model.

Table I: Base Model with Undissolved Solids

Model -Detailed	**$k_{eff+2\sigma}$**
10-well loading	0.810
Gadolinium Poisoned	0.503
5 M Excess Acid	0.749
Gadolinium Poison and Excess Acid	0.507

Table II-A: 100 wt% enrichment, 4.0 gU/L Bulk Solution

Model	$k_{eff+2\sigma}$
Base Case	0.816
Gadolinium Poisoned	0.504
5 M Excess Acid	0.756
Gadolinium and Excess Acid	0.507

Table II-B: 100 Wt% Enrichment And 50% Reduction In Aluminum, 4.0 gU/L Bulk Solution

Model	$k_{eff+2\sigma}$
Base Case	0.824
Gadolinium Poisoned	0.504
5 M Excess Acid	0.760
Gadolinium 5M Excess Acid	0.508

Table II-C: 100 Wt% Enrichment And 75% Reduction In Aluminum, 4.0 gU/L Bulk Solution

Model	$k_{eff+2\sigma}$
Base Case	0.839
Gadolinium Poisoned	0.510
5 M Excess Acid	0.780
Gadolinium and Excess Acid	0.507

Table II-D: 100 Wt% Enrichment, No Aluminum, 4.0 gU/L Bulk Solution

Model	$k_{eff+2\sigma}$
Base Case	0.821
Gadolinium Poisoned	0.525
5 M Excess Acid	0.823
Gadolinium and Excess Acid	0.507

Table II-E: 100 Wt% Enrichment, 4.8 gU/L bulk solution

Model	$k_{eff+2\sigma}$
Base Case	0.839
Gadolinium Poisoned	0.512
5 M Excess Acid	0.779
Gadolinium and Excess Acid	0.516

Table II-F: 100 Wt% Enrichment, 10.0 gU/L Bulk Solution

Model	$k_{eff+2\sigma}$
Base Case	0.987
Gadolinium Poisoned	0.513
5 M Excess Acid	0.930
Gadolinium and Excess Acid	0.516

Table II-G: 100 Wt% Enrichment, 20.0 gU/L Bulk Solution

Model	$k_{eff+2\sigma}$
Base Case	1.232
Gadolinium Poisoned	0.515
5 M Excess Acid	1.179
Gadolinium and Excess Acid	0.519

Table II-H: Well Separation at Physical Minimums, 4.0 gU/L bulk Solution

Model	$k_{eff+2\sigma}$
Base Case with	1.015
Gadolinium Poisoned	0.506
5 M Excess Acid	0.961
Gadolinium and Excess Acid	0.511