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Thermal Analysis of a Uranium Silicide Miniplate Irradiation Experiment

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ABSTRACT

This paper outlines the thermal analysis for the irradiation of high-density uranium-silicide (U_3Si_2 dispersed in an aluminum matrix and clad in aluminum) booster fuel for a Boosted Fast Flux Loop designed to provide fast neutron flux test capability in the Advanced Test Reactor. The purpose of this experiment (designated as Gas Test Loop-1 [GTL-1]) is two-fold: (1) to assess the adequacy of the U_3Si_2/Al dispersion fuel and the aluminum alloy 6061 cladding, and (2) to verify stability of the fuel cladding boehmite pre-treatment at nominal power levels in the 430 to 615 W/cm^2 (2.63 to 3.76 Btu/s·in^2) range. The GTL-1 experiment relies on a difficult balance between achieving a high heat flux, yet keeping fuel centerline temperature below a specified maximum value throughout an entire operating cycle of the reactor. A detailed finite element model was constructed to calculate temperatures and heat flux levels and to reveal which experiment parameters place constraints on reactor operations. The inclusion of machining tolerances in the numerical model has a large effect on heat transfer. Analyses were performed to determine the bounding lobe power level at which the experiment could be safely irradiated, yet still provide meaningful data under nominal operating conditions. Then, simulations were conducted for nominal and bounding lobe power levels under steady-state and transient accident conditions with the experiment in the reactor. Reactivity changes due to a loss of commercial power with pump coast-down to emergency flow or a standard in-pile tube pump discharge break were evaluated. The elapsed time after reactor shutdown when the emergency pumps can be turned off and the experiment can be adequately cooled by natural convection cooling was determined using a system thermal hydraulic model. A canal draining scenario was investigated to determine the necessary additional cooling time in water prior to experiment handling and removal.

KEYWORDS

Uranium silicide fuel, irradiation experiment, fast neutron flux

1. INTRODUCTION

A Gas Test Loop (GTL) system (now known as the Boosted Fast Flux Loop [BFFL]) is being developed to provide a high intensity fast-flux irradiation environment for testing fuels and materials for advanced concept nuclear reactors. The GTL system is designed for operation in the northwest test lobe of the Advanced Test Reactor (ATR), a high power density/high neutron flux research reactor operating in the United States. The Technical and Functional Requirements for the BFFL stipulate a minimum neutron flux intensity (10^{15} n/cm^2·s) and fast to thermal neutron ratio (>15) for the test environment.[1] Incorporation of booster fuel within the test lobe is necessary to achieve these neutron flux requirements. U_3Si_2 is an appropriate choice for the booster fuel meat since the fuel exhibits low parasitic neutron absorption and good irradiation and chemical stability.[2]

Although silicide fuels are in current use elsewhere, the fuel meat for this application is thicker and the heat fluxes higher than previous experience. The inclusion of the booster fuel requires that the fuel be qualified for safe operation in the ATR. Previous experiments conducted in the BR2 reactor of SCK•CEN in Mol, Belgium exhibited failure of aluminum clad U_3Si_2 fuel plates exposed to high heat fluxes.[3] In that instance, fuel plate failure was attributed to the accelerated corrosion of the cladding. The corrosion layer has a much lower thermal conductivity than the cladding, which can lead to high fuel centerline temperatures. To avoid this problem in ATR, driver fuel is pre-treated by the application of a stable, protective oxide layer (boehmite) formed by autoclaving the fuel with a water solution at an appropriate temperature, pressure and pH.[4] Although the Belgian fuel plates were not pre-treated, concerns over the buildup of this low thermal conductivity corrosion layer during reactor operation prompted conducting in-pile tests for representative samples of the proposed booster fuel.

The purpose of the miniplate experiment (designated as GTL-1) is to demonstrate acceptable performance of 25% enriched, high density (4.8 g-U/cm^3) U_3Si_2-Al dispersion fuel plates clad with aluminum alloy 6061 pre-treated with a stable oxide film. The experiment's U-235 enrichment is 25% with a total mass of approximately 28 grams. The GTL-1 miniplate experiment is designed for a 56 day irradiation cycle in the South Flux Trap (SFT) position in the ATR (Figure 1). The set of 16 miniplates are designed to operate at several different power levels depending upon their axial location relative to the core centerline, encompassing the 450 W/cm^2 nominal heat flux level for GTL booster fuel operation, as well as higher levels. Miniplate specimens will be exposed to nominal plate peak power levels in the 430 to 615 W/cm^2 (2.63 to 3.76 Btu/s·in^2) range. The higher power levels are included in the experiment to explore a margin of safety above the nominal operating power level with respect to corrosion and other radiation damage effects.

2. NUMERICAL MODEL

2.1 Finite-Element Model Description

The GTL-1 irradiation experiment configuration was modeled using ABAQUS/CAE for construction of the finite element models and ABAQUS/Standard for the steady-state and transient temperatures and heat fluxes. The assembly is comprised of the miniplates, rails, capsules, spacers, basket, adapter and coolant channels. The model includes all four capsules of the test train assembly within the active core, along with the top, upper and bottom spacers that are used to position the capsules axially with the basket.

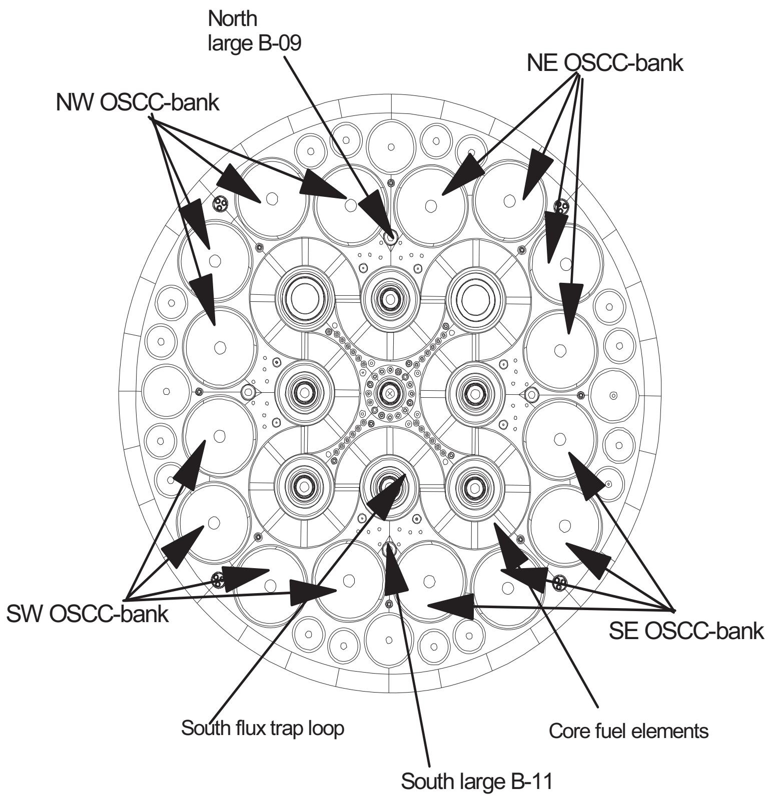


Figure 1. Illustration of SFT position within the ATR.

The adapter is used to fill the space between the outer diameter of the basket and the inner diameter of the chopped dummy in-pile tube (IPT) in the SFT. Primary coolant water flows through the test train, a 40 mil water annulus between the basket and adapter, and a 37 mil water annulus between the adapter and IPT. Small nubs on the outside diameter of the basket and adapter maintain the spacing in the adjacent water annuli. The total length of the portion of the assembly modeled is 57.835 inches (1.469 m). A cross-section and vertical arrangement of the assembly are shown in Figures 2 and 3.

The miniplates are housed in four interconnecting flow-through capsules. There are four miniplates in each capsule arranged vertically in a 2×2 array. The miniplates are 1.0 inch (2.5 cm) wide and 4.0 inches (10.2 cm) long (Figure 4). The fuel meat thickness of all plates is 0.04 inches (0.102 cm), the cladding is 0.03 inches on either side of the fuel, which yields a total plate thickness of 0.1 inches (0.254 cm). A 6061 aluminum border surrounds the 3.4 inch × 0.72 inch fuel region (Figure 4). Inside of each capsule, the two columns of two miniplates are arranged end-to-end, with no spacing between the plates in the flow direction (i.e., the bottom end of the upper plate rests against the top end of the lower plate), measuring 8.0 inches (20.3 cm) in length. There is a 0.385 inch (0.98 cm) axial separation between fuel plates in adjacent capsules.[5] There are three water coolant channels approximately 0.14 inches (0.36 cm) thick and 0.888 inches (2.26 cm) wide in contact with the fuel plates (see Figure 5).

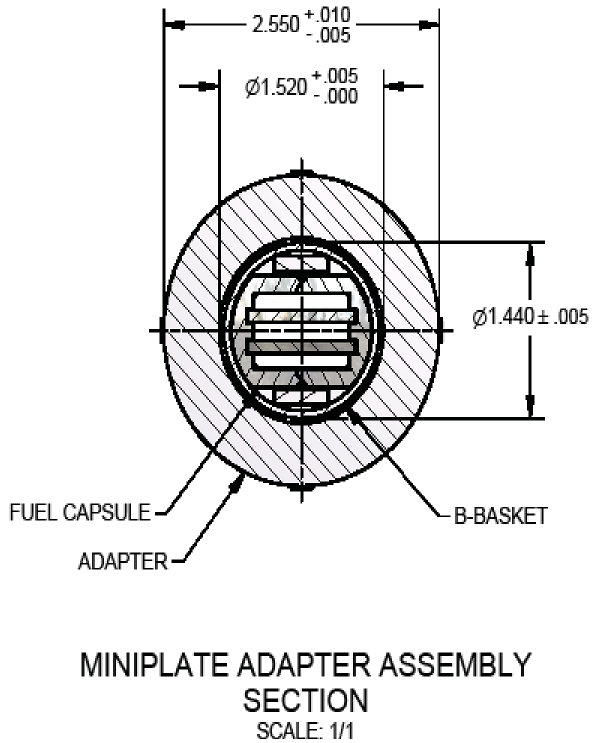
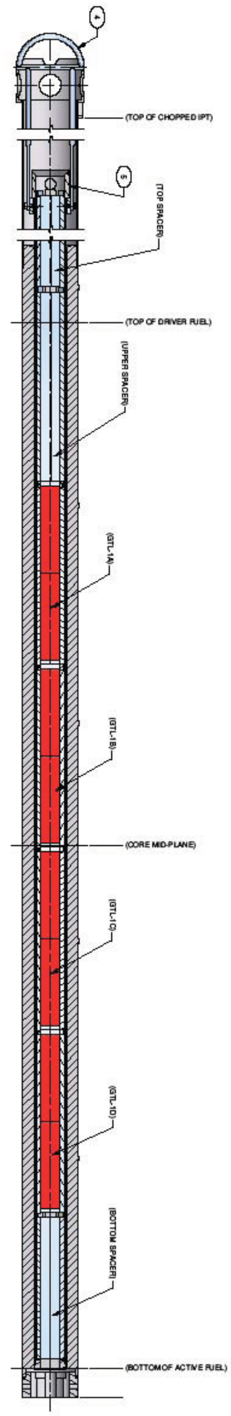


Figure 2. Cross-section of irradiation test train assembly.



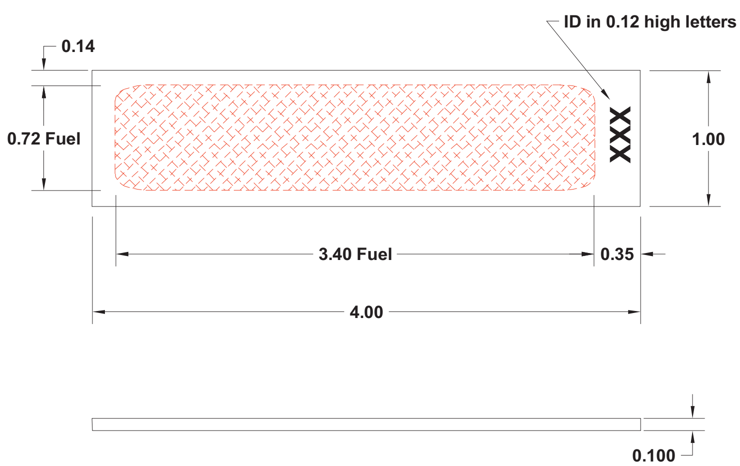


Figure 4. Miniplate design dimensions (inches).

Figure 3. Vertical arrangement of irradiation test train assembly.

Thirteen of the 16 miniplates were pre-treated with a surface boehmite layer (78.7 microinches or 2.0 µm thick) formed in an autoclave using water with a pH of 7.8. The remaining three miniplates were pre-treated with a thinner, denser boehmite layer (19.7 microinches or 0.5 µm thick) deposited using a similar procedure with water at a pH of 5.7.[6] Figure 6 shows the arrangement and numbering convention of the miniplates within the four capsules. The faces of the miniplates will be oriented such that they are parallel to a radial line from the core center through the center of the SFT position.

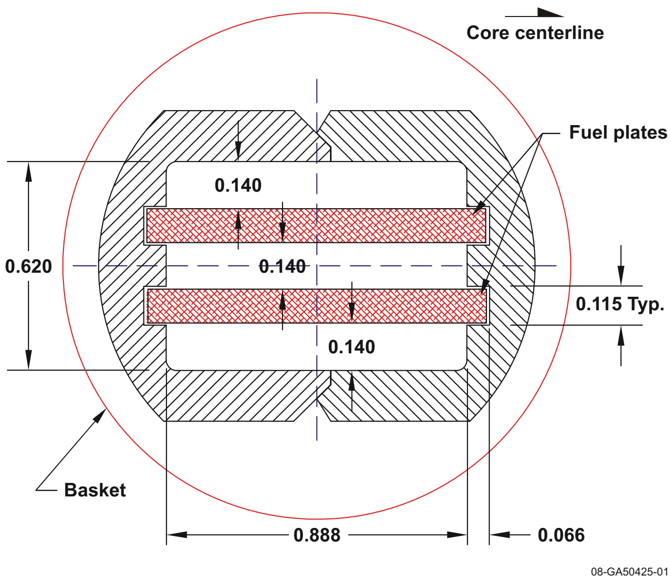


Figure 5. Flow through capsule cross-section (dimensions in inches).

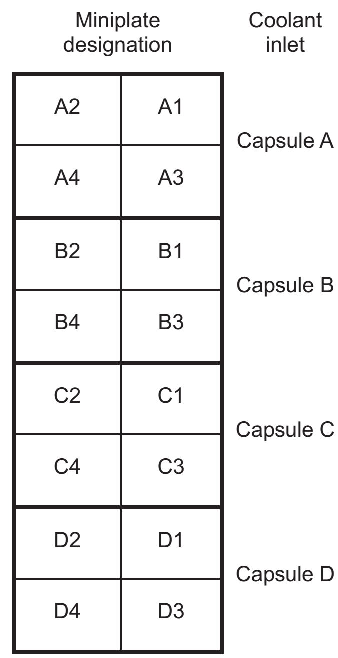


Figure 6. Capsule and miniplate arrangement.

The finite element model geometry was constructed in ABAQUS CAE version 6.7-3[7] using shells and 3D solids. The model contains 199094 nodes and 174381 elements. Structured meshing was used throughout the model. The analysis was performed on a SGI Altix ICE 8200 distributed memory cluster with 512 Intel Xeon quad core 2.66 GHz processors running SUSE Linux Enterprise Server 10.

2.1 Analysis and Calculations

A series of preliminary simulations were run to determine the bounding power level at which the experiment could be safely operated. It was concluded that the fuel temperature is the limiting parameter. As a result, thermal analyses were performed for the GTL-1 miniplate irradiation experiment in the SFT position at a nominal South lobe source power of 26.7 MW and at an analyzed bounding South source power of 35.6 MW. The six cases listed in Table 1 were analyzed.

Coolant flow enters from the top with an inlet temperature of 125°F (52°C). Flowing water travels through the test train, the annulus between the basket and the adapter, and the annulus between the adapter and the IPT. Stagnant water is assumed to be present in the small gaps ("ears") where the edges of the fuel plates fit into the capsules and the region surrounding the rails between the outside of the capsules and the inside of the basket. Heat transfer coefficients for turbulent forced convection to the cooling water were obtained from the Colburn correlation using the film temperature method to account for fluid property variation. The loss coefficients and resulting velocity and mass flux (47.8 ft/s, 20.45 lbm/s-in^2) across the miniplates were obtained from a hydraulic flow test.[8] The velocity of cooling water flowing through the annular gaps was computed from a hydraulic analysis employing the Bernoulli equation with the inlet conditions (125°F and 360 psi) and pressure drop (77 psi) for two-pump operation. The calculated total flow through the experiment assembly is 85.2 gpm.

The thickness of the oxide layer is assumed to be 0.5 mils (12.7 μm) uniformly over the entire miniplate. This assumption is reasonable considering that the maximum thickness of the applied pre-film layer for the GTL miniplates is 0.079 mil (2 μm). This stable, protective hydroxide coating (a crystalline, non-porous gamma-alumina hydrate or boehmite) is produced by autoclaving the fuel with a water solution at an appropriate temperature, pressure and pH. The literature indicates that this type of very thin, uniformly adherent boehmite layer formed by pre-treatment is more stable and much less likely to grow or spall than a hydroxide layer formed during irradiation.[4]

Table 1. Thermal analysis cases analyzed.

Case	Description	Condition
1	Steady-state reactor operation at nominal cycle power (26.7 MW) and Δp=77 psi with as-built miniplate average uranium density and various oxide thicknesses	Steady-state during reactor operation
2	Steady-state reactor operation with safety factors (35.6 MW) and as-built miniplate maximum uranium density	
3	A 1.5 second reactor transient following loss of commercial power with pump coast-down to emergency flow[9]	Transient during reactor operation
4	Reactivity changes resulting from a loss of commercial power with pump coastdown or a standard in-pile tube (SIPT) pump discharge break	
5	Experiment horizontal in air after removal from the reactor	Steady-state after shutdown
6	Natural convection cooling in reactor position after normal reactor shutdown	

The mesh Monte Carlo N-Particle Transport Code (MCNP) neutronic analysis[10] is based upon uranium silicide fuel with a 25 wt% U-235 enrichment.[6] The neutronics analysis provided average heat loads for each of the 16 fueled miniplates. The plate average heat loads were multiplied by peaking factors over the 11×11×1 mesh over the 3.4 in. × 0.72 in. fuel region on each miniplate. The highest heat loads are located along the edges of the fuel. Analyses were completed for the GTL-1 test train with fuel plate heating and structural component gamma/neutron heating based on the physics analyses reported by Chang and Jewel[10] for a South source power of 24.3 MW. These results were linearly scaled to 26.7 MW for the nominal cases and 35.6 MW for the safety cases. The gamma heating of aluminum components and coolant water is calculated based upon the axial cosine power distribution.

3. RESULTS

3.1 Steady-State Analysis Results (Cases #1 and #2)

Tables 3 and 4 list the computed results for steady-state reactor operation at a nominal lobe power of 26.7 MW. Table 3 lists the calculated peak temperatures in the coolant, oxide, clad and fuel and Table 4 lists the peak heat flux at the oxide surfaces for three different oxide thicknesses. The maximum coolant temperature occurs in the water located in the “ear” region, which is conservatively modeled in ABAQUS as stagnant. For the safety case at a bounding South lobe power of 35.6 MW, the main channel water temperature reaches a maximum of 200.2°F (93.4°C) in the stagnant “ear” region and 158.2°F (70.1°C) at the location of highest heat flux (Capsule B). The coolant outlet bulk temperatures are 183.2°F (84.0°C) for the main water channel, 150.9°F (66.1°C) for the annulus between the basket and adapter, and 154.5°F (68.1°C) for the annulus between the adapter and IPT. The computed peak fuel centerline temperature for the safety case is below the maximum acceptable fuel temperature of 842.0°F (450.0°C).[5]

3.2 Transient Analysis Results (Cases #3 and #4)

This section presents results for steady-state reactor operation followed by either: (1) a loss of commercial power accident with primary coolant pump coast-down to emergency flow (Condition 2) (Case #3), or (2) Condition 4 power excursion initiated by a standard in-pile tube (SIPT) pump discharge break[11] (Case #4).

For the pump coastdown conditions, the reactor power safety factors, reduced coolant velocity, and forced convection heat transfer coefficients were calculated and applied. The DNBR (ratio of critical heat flux to the maximum heat flux) is the ratio of the critical heat flux calculated using an empirical correlation derived from critical heat flux data[12] to the calculated heat flux at the oxide surface. The fuel is aluminum clad plate fuel, thus the correlation is applicable for this evaluation. The flow instability ratio (FIR) is the ratio of the critical coolant temperature rise (i.e., boiling) to the actual coolant temperature rise. Flow instability occurs when the coolant enthalpy reaches saturation. Safety limits (SAR-153[13]) were evaluated for the worst case.

Table 3. Computed peak temperatures for 26.7 MW nominal lobe power.

Power: 26.7 MW	Computed Nominal Peak Temperature					
Component	oxide=0.02 mil		oxide=0.079 mil		oxide=0.984 mil	
	°F	°C	°F	°C	°F	°C
Coolant Max	168.6	75.9	169.6	76.4	174.8	79.3
Coolant Outlet	161.0	71.7	161.0	71.7	161.0	71.7
Oxide	295.0	146.1	298.4	148.0	352.0	177.8
Cladding	318.3	159.1	325.4	163.0	435.7	224.3
Fuel	519.2	270.7	525.8	274.3	621.4	327.4

Table 4. Computed peak oxide heat flux for various oxide thicknesses for 26.7 MW nominal lobe power.

Power: 26.7 MW	**Computed Nominal Peak Heat Flux**						
		oxide=0.02 mil		**oxide=0.079 mil**		**oxide=0.984 mil**	
Capsule	**Plate**	**Btu/(s-in^2)**	**W/cm^2**	**Btu/(s-in^2)**	**W/cm^2**	**Btu/(s-in^2)**	**W/cm^2**
A	**1**	2.625	429	2.625	429	2.59	424
	2	2.616	428	2.616	428	2.581	422
	3	3.190	522	3.191	522	3.147	515
	4	3.215	526	3.215	526	3.171	519
B	**1**	3.581	586	3.582	586	3.537	578
	2	3.588	587	3.589	587	3.544	580
	3	3.764	616	3.764	616	3.711	607
	4	3.760	615	3.76	615	3.707	606
C	**1**	3.738	611	3.738	611	3.704	606
	2	3.752	614	3.753	614	3.718	608
	3	3.674	601	3.675	601	3.632	594
	4	3.669	600	3.669	600	3.626	593
D	**1**	3.440	563	3.441	563	3.395	555
	2	3.436	562	3.437	562	3.392	555
	3	3.075	503	3.073	503	3.039	497
	4	3.075	503	3.073	503	3.039	497

For the pump coastdown scenario, the peak fuel-clad interface temperature reaches 605.0°F (318.3°C) and has a DNBR of 2.7 and a FIR of 3.1. This accident scenario satisfies the minimum SAR requirements of 2.0 for the FIR and DNBR. The peak fuel-clad interface temperature is significantly below the solidus temperature of 1079.7°F (582°C). The peak fuel centerline temperature for the hottest fuel plate is 875.0°F (468.3°C) at t=0.9278 seconds into the transient. According to Hayes,[5] short-term fuel temperatures up to (932°F) 500°C are permitted without adverse effects to the fuel. These results are summarized in Table 5. Figures 7 and 8 show the time history of the computed maximum fuel and clad temperatures for the two transient accident scenarios. The computed coolant outlet temperature for the three water channels during the transient are shown in Figures 9 and 10 for the pump coastdown and the SIPT pump discharge break, respectively. Contour plots of the ABAQUS results are shown in Figures 11 and 12. The direction of coolant flow is from right to left in these figures. Figure 11 shows the temperature distribution at the fuel plate centerline. As expected, the fuel meat is hotter than the surrounding aluminum border region on each plate. The plates located closest to core centerline are the hottest, with plate B3 exhibiting the maximum fuel centerline temperature. Figure 12 shows the heat flux distribution on the oxide layer of the miniplates, which is the highest for the plates closest to the core centerline. The surface in contact with the coolant has the highest heat flux.

Table 5. Summary of pertinent safety parameters for pump coastdown transient.

Parameter	Value
DNBR	2.7
FIR	3.1
Maximum Oxide Heat Flux	6.026 Btu/in2-s (985.5 W/cm2) @ t=0.9278 s, Plate B3
Coolant Temperature at Location of Peak Heat Flux (used to calculate DNBR)	166.9°F (74.9°C) @ t=0. 9278 s
Maximum Coolant Temperature (used to calculate FIR)	216°F (102.2°C) @ t=0.9278 s
Peak Fuel-Clad Interface Temperature	605.0°F (318.3°C) @ t=0.9278 s, Plate C2
Peak Fuel Centerline Temperature	875.0°F (468.3°C) @ t=0.9278 s, Plate C2

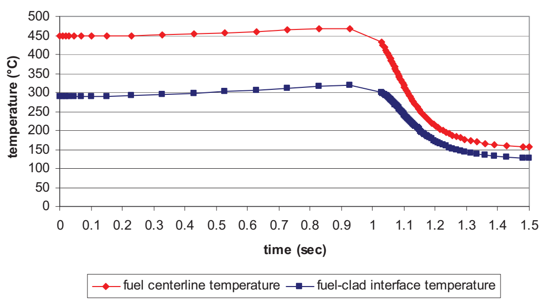


Figure 7. Transient response of maximum fuel and cladding (°C) temperature during pump coastdown.

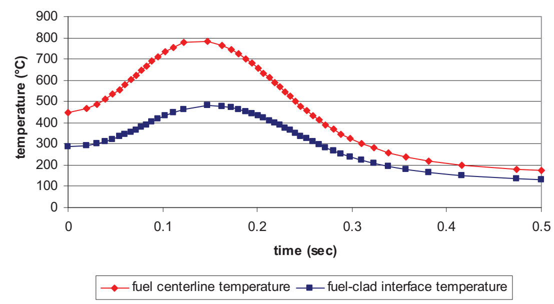


Figure 8. Transient response of maximum fuel and cladding (°C) during SIPT discharge break.

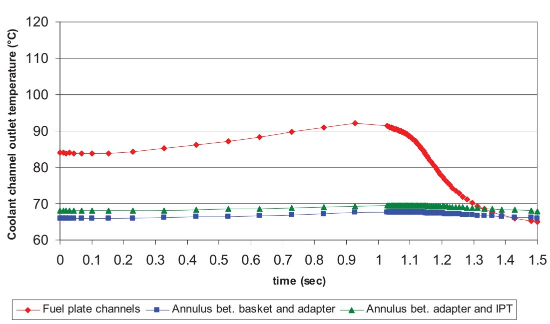


Figure 9. Transient response of maximum coolant water temperature (°F) during pump coastdown.

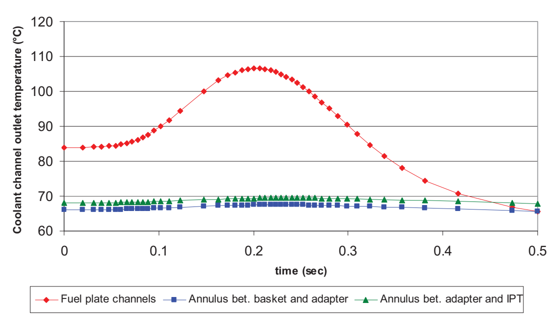


Figure 10. Transient response of maximum coolant water temperature (°F) during SIPT discharge break.

Figure 11. Temperature distribution (°F) at the fuel centerline.

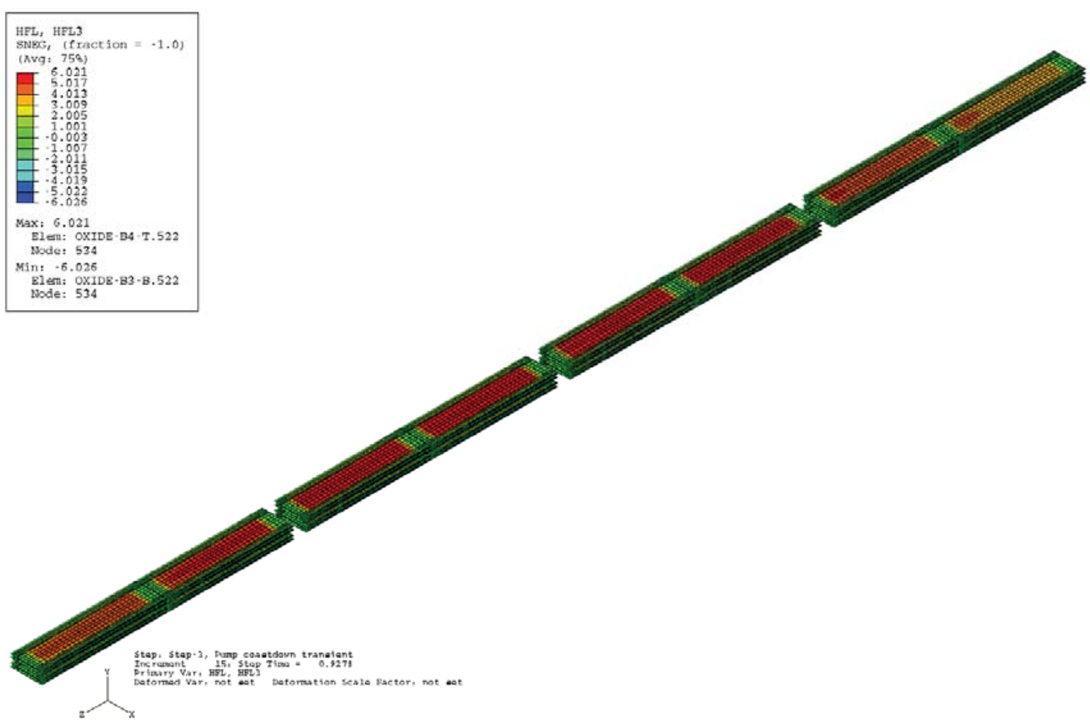


Figure 12. Heat flux at oxide layer (Btu/s·in^2).

For the reactivity insertion accident scenarios, the transient response of the maximum coolant water temperature was calculated using ABAQUS. The increased coolant water temperature feeds back to increase the flux trap reactivity. The ΔTeff used for the reactivity feedback and cascading feedback calculations was obtained by area-weighting the change in water outlet temperature from initiation of the transient by the cross-sectional area of the three coolant channels. From Chang and Jewell[10], the maximum linear reactivity increase ($) as a function of SFT water temperature (°F) for the tests occurs with the fueled GTL-1 experiment during a typical power split weighted toward the south lobes. The applicable reactivity perturbation due to water temperature and density changes for mean coolant temperatures between 160 and 195 °F is 0.00033 $/°F. For a loss of commercial power with pump coast-down (Case #3), the calculated reactivity insertion is 0.0026 $ and the maximum reactivity insertion rate is 0.0028 $/s. For a SIPT pump discharge break (Case #4), the maximum reactivity insertion is 0.0063 $ and the maximum reactivity insertion rate is 0.0313 $/s.

3.3 Analysis Results after Reactor Shutdown (Cases #5 and #6)

An ABAQUS finite-element analysis was completed for the basket containing the experiment horizontal in air after a cool down interval of 21 hours after scram (Case #5). This scenario represents a hypothetical ATR canal draining accident. The maximum cladding temperature of 1034.2 °F (556.8°C) occurs in Plate C2. This temperature is below the cladding solidus temperature of 1079°F (582°C). This demonstrates that at the end of the 56 day cycle, the experiment can be handled 21 hours after shutdown and there are no special requirements for canal storage.

A thermal hydraulic analysis to assess the effectiveness of natural convection cooling 13 hours after reactor shutdown (Case #6) was performed using RELAP5/MOD3 version 3.2.1.2[14]. RELAP5 results for natural convection in the reactor core position show a peak coolant temperature of 174.9°F (79.4°C) after the experiment has been cooled for 13 hours via forced convection. The results confirm that the experiment is cool enough to be handled in water at any time after 13 hours of forced convection cooling.

4. SUMMARY

Results of the thermal analysis indicate that the performance of the GTL-1 experiment is adequate to satisfy the objectives of the experiment and meet ATR safety requirements. The experiment can withstand sufficiently high heat flux levels to assess corrosion and other radiation damage effects, yet keep the fuel centerline temperature below recommended operating limits. The analyses reveal the experiment parameters that place constraints on reactor operations. From the simulations, it was concluded that fuel centerline temperature is the parameter that limits the maximum power level at which the experiment can be safely irradiated. To keep the peak steady-state fuel centerline temperature below 842.0°F (450.0°C), the maximum indicated lobe power level must be at or below 30.7 MW. Transient reactor response was evaluated for: (1) a loss of commercial power with pump coast-down, or (2) a SIPT pump discharge break. The DNBR and FIR for the pump coastdown scenario exceeds the minimum required value of 2.0 with a margin of safety, experiment temperatures are acceptable, and the flux trap reactivity change during a reactivity insertion accident is bounded by the accident analyses prescribed in the Upgraded Final Safety Analysis Report[13]. After reactor shutdown, 13 hours of forced convection cooling is required before the experiment can be cooled only by natural convection of the primary coolant without bulk boiling occurring. The limiting parameter for experiment handling is that the temperature of the fuel-cladding interface remain below the solidus temperature

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