WHC-SA--0952

DE91 004551

Performance Analysis of a Mixed Nitride Fuel System for an Advanced Liquid Metal Reactor

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Date Published
November 1990

To Be Presented at
American Nuclear Society
Winter Meeting
Washington, D.C.
November 11-16, 1990

Prepared for the U.S. Department of Energy
Assistant Secretary for Nuclear Energy

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Hanford Operations and Engineering Contractor for the
U.S. Department of Energy under Contract DE-AC06-87RL10930

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PERFORMANCE ANALYSIS OF A MIXED NITRIDE FUEL SYSTEM FOR AN ADVANCED LIQUID METAL REACTOR

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ABSTRACT

The conceptual development and analysis of a proposed mixed nitride driver and blanket fuel system for a prototypic advanced liquid metal reactor design has been performed. As a first step, an intensive literature survey was completed on the development and testing of nitride fuel systems. Based on the results of this survey, prototypic mixed nitride fuel and blanket pins were designed and analyzed using the SIEX computer code. The analysis predicted that the nitride fuel consistently operated at peak temperatures and cladding strain levels that compared quite favorably with competing fuel designs. These results, along with data available in the literature on nitride fuel performance, indicate that a nitride fuel system should offer enhanced capabilities for advanced liquid metal reactors.

1. INTRODUCTION

Based on years of experience and intense development, oxide fuel (UO_2) has been the reference fuel system for liquid metal reactors (LMRs) throughout the world. Nonetheless, as part of the development of advanced LMRs, enhanced performance and passive safety have prompted a renewed interest in alternate fuel designs. For example, the U. S. has baselined a metal fuel system for advanced LMR studies. Gradually, however, a concensus is forming within the international LMR community that nitride fuels, such as mixed nitride [(U,Pu)N] and mononitride (UN), offer an attractive option for development in the future. As a result, nitride fuel systems have experienced a resurgence of interest of late. A wide variety of tests with nitride fuels are currently being conducted throughout the world. With its combination of favorable neutronic, physical, and mechanical characteristics; chemical compatibility with liquid metal coolants and relative ease in reprocessing (similar to mixed oxide, $(U,Pu)O_2$), nitride fuels exhibit the potential for greatly enhancing the performance of proposed advanced LMR systems.

This project began by surveying the available literature on the performance of nitride fuels and current experimental programs. Based on this review, a nitride fuel and blanket design was developed for the ongoing Advanced Liquid Metal Reactor (ALMR) design effort supported by the U.S. Department of Energy.[1,2] A performance analysis was conducted using a revised version of the SIEX computer code,[3] which now includes performance models for metal- and nitride-fueled pins. As a basis of comparison, the ALMR reference ternary metal fuel system (U-Pu-Zr) was also evaluated.

1.1 Background

In 1984 the U.S. adopted a UN fuel system for its space reactor program and has initiated an aggressive irradiation testing program in both Experimental Breeder Reactor-II (EBR-II) and the Fast Flux Test Facility (FFTF).[4] Previous U.S. work on nitride fuels can be found in the literature with notable presentations at the Advanced Liquid Metal Fast Breeder Reactor Fuels meeting held in Tuscon in 1977 and at the International Conference on Fast Breeder Reactor Fuel Performance held in Monterey in 1979. The European Community is also investigating nitride fuels and in 1982 nitride fuel was adopted for their alternate fuels development program.[5] Fabrication and characterization studies of (U,Pu)N fuel are currently underway at Cadarache by the French Commissariat à l'Energie Atomique (CEA). Fabrication studies are also being supported by the Paul-Sherrer-Institute (PSI) in Switzerland and the Institute for Transuranium Elements (ITE) in Germany.[6,7] The PSI is involved in the development of a gelation process for the production of (U,Pu)N microspheres and pellets while the ITE is concentrating on a direct pressing process following a carbothermic reduction/nitriding of $(U,Pu)O_2$ fuel.[8] Two irradiation tests of CEA-ITE (U,Pu)N pins are underway in the Phènix reactor. A third irradiation test using PSI fabricated pins is planned. Additionally, other irradiation tests of nitride fuels have been conducted in the high flux materials testing reactor at Petten in the Netherlands.

The Soviet Union has also gained much experience with nitride fuels by irradiating an entire core load of UN in the BR-10 reactor.[9] The test reached a burnup of 8.3 at. % prior to completion. A number of tests using uranium carbide, mixed carbide, and carbonitride fuels were also irradiated to over 10 at. % burnup in the BOR-60 reactor. Japan is investigating the development of nitride fuels in support of their liquid metal fast breeder reactor program as well.[10] The program is being conducted by the Japan Atomic Energy Research Institute (JAERI). Research has been initiated into (U,Pu)N fabrication methods that stress compatibility with conventional fabrication and reprocessing technology.

2. FUEL SYSTEM DESIGN AND ANALYSIS

The comparative analysis of the two fuel systems, (U,Pu)N and U-Pu-Zr, was based on a one-to-one replacement of the reference ALMR metal fuel and blanket pins with the proposed nitride fuel and blanket pins. The values for the various fuel pin parameters such as linear heat rate, pin diameter, and plenum volume were taken from available literature. Other data used to characterize the ALMR design such as thermal-hydraulic and neutronic information were taken from a combination of past SIEX analyses and modified as needed to match the parameters in the literature.

Based on the data available in the literature, the following design criteria were designated for the nitride fuel system:

- restrict the steady-state, peak fuel temperature to less than 1200 °C
- maintain the steady-state, peak cladding temperature to less than 1000 °C
- use a value of 0.5% ΔD/D per at. % burnup for fuel pellet swelling
- use 10% fission gas release
- use either a sodium or helium bond between the fuel and cladding
- assume inexorable fuel swelling
- assume no fuel/cladding chemical interaction (FCCI)
- designate either significant fuel cladding contact or a cladding strain limit as end-of-life
- use an 80% fuel-smeared density.

Using the above criteria, prototypic (U,Pu)N fuel and UN blanket pins were designed. After preliminary evaluation it was determined that a sodium-bonded pin using HT9 cladding would best meet the design criteria. A fuel shroud tube was not included since it was felt that low pin powers and temperatures would produce very little fuel cracking and that it would add unnecessary complexity to the conceptual design study. A list of the design parameters for the nitride fuel and blanket pins is shown in Tables 1 and 2. Data for the reference metal fuel system is shown for comparison.

Table 1. Nitride and Metal Driver Fuel Designs.

	U-26 Pu-10 Zr	(U,28.5 Pu)N
Pin Length (m)	4.267	4.267
Pin O. D. (mm)	6.680	6.680
Cladding Material	HT9	HT9
Cladding Thickness (mm)	0.508	0.457
Pellet Diameter (mm)	4.902	5.283
Gap (mm)	0.762	0.483
Pellet Density (%TD)	100	95
Smeared Density (%TD)	75	80
Wire Wrap Diameter (mm)	1.295	1.295
Wire Wrap Pitch (m)	0.305	0.305
Bond	Na	Na
Pins Per Assembly	331	331

Table 2. Nitride and Metal Blanket Fuel Designs.

	U-10 Zr	UN
Pin Length (m)	4.267	4.267
Pin O. D. (mm)	12.090	12.090
Cladding Material	HT9	HT9
Cladding Thickness (mm)	0.559	0.508
Pellet Diameter (mm)	10.117	10.668
Gap (mm)	0.856	0.406
Pellet Density (%TD)	100	97
Smeared Density (%TD)	85	90
Wire Wrap Diameter (mm)	0.813	0.813
Wire Wrap Pitch (m)	0.305	0.305
Bond	Na	Na
Pins Per Assembly	127	127

The burnout rate used for both the metal and nitride fuel pins was a constant 1.5% per 1.0 at. % burnup. This value was used over the normal three-cycle irradiation time for the driver fuel. Approximating the burnout rate for the blanket pins, however, was not as straightforward. Choosing the proper rate was complicated by the fact that the ALMR has two blanket regions broken into three zones and that during their five-cycle lifetime most of the blanket pins are shuffled every cycle throughout the core. Because data were not provided in the literature that sufficiently described the burnout rate, it was determined that a "bounding" analysis should be used in order to conservatively envelope the operating conditions of the blanket pins. This was done by approximating the power-time profile to match the general shape of the temperature history for the peak blanket pin by interpolating between the peak and final cladding midwall temperatures. The resultant power profile was loaded into SIEX and the coolant flow rates were adjusted until the proper cladding midwall temperatures were reached.

Another modification adopted in the input models was the reduction of the effective plenum length to account for the displacement of the sodium bond from the fuel/cladding gap. This modification increased the conservativity of the cladding strain predictions by accounting for the fact that as the fuel swells outward during irradiation, it forces the sodium within the gap into the plenum. This results in higher cladding pressure loadings because the fission gas is released into a smaller plenum volume.

3. COMPUTATIONAL RESULTS

The results of the SIEX calculations for the nitride and metal fuel and blanket designs have been tabulated and include data for the peak cladding midwall, peak fuel surface, and peak fuel centerline temperatures as well as the peak cladding strain for the peak fuel and blanket pins. Figure 1 illustrates the profiles predicted by SIEX for the cladding midwall and fuel surface temperatures for both the peak nitride and metal fuel pins. The peak cladding temperatures for the nitride and metal pin were maintained well below the targeted maximum temperature of 1000 °C.

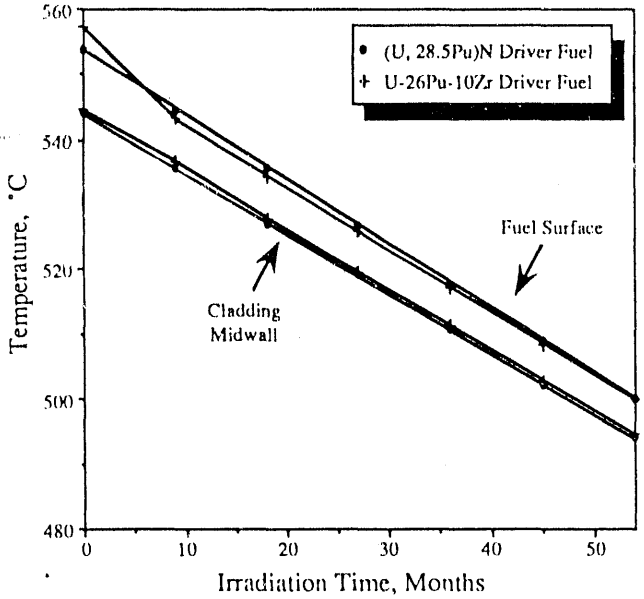

Figure 1. Peak Cladding Midwall and Fuel Surface Temperatures for (U,28.5Pu)N and U-26Pu-10Zr Driver Fuel.

The predicted results for the peak fuel centerline temperatures are shown in Figure 2. This plot indicates that the peak temperatures in the nitride fuel were consistently lower than those predicted for the metal fuel. The maximum peak temperature exhibited by the nitride fuel was 658 °C, some 542 °C lower than the targeted maximum peak temperature of 1200 °C.

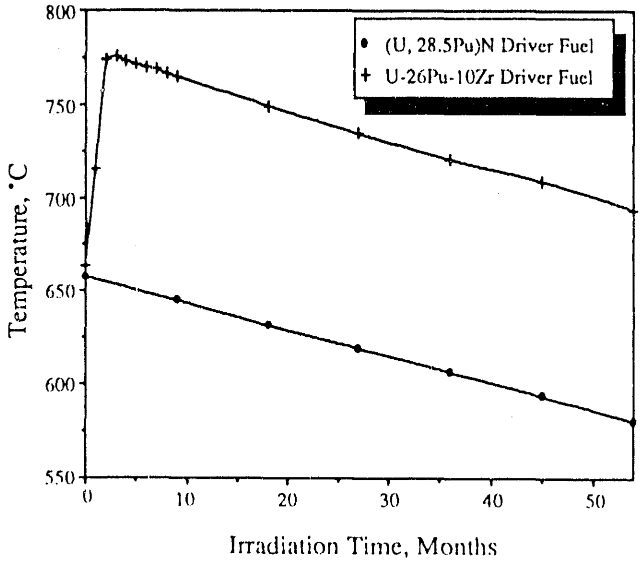

Figure 2. Peak Fuel Centerline Temperatures for (U,28.5Pu)N and U-26Pu-10Zr Driver Fuel.

The SIEX-predicted metal fuel temperatures were somewhat higher than those available in the literature. This was especially true at higher burnups. The discrepancy was most likely due to differences in the modeling of the degradation of the thermal conductivity of metal fuel. Initially, metal fuels have a very high thermal conductivity, typically much greater than that of ceramic fuels. However, at low burnups a large volumetric expansion has been observed.[11, 12] This leads to the degradation of the thermal conductivity due to porosity formation. This degradation reduces the thermal conductivity of metal fuels to approximately 50% of their pre-irradiated value. The difference between the SIEX-predicted temperatures and those available in the literature was apparently due to the fact that in other models it is assumed that once gap closure occurs, sodium logging of the interconnected porosity within the metal fuel increases its thermal conductivity. This assumption was not made in the SIEX models for either the metal or nitride fuel system.

A more detailed look at the nitride and metal fuels' behavior is shown in Figure 3 where the peak and beginning of life (BOL) fuel radial temperature profiles are plotted. This figure illustrates that the nitride fuel pin had a flatter temperature profile than the metal fuel pin indicating a lower temperature gradient across the (U,Pu)N pin. It should also be noted that the BOL profile for the nitride pin also represented its peak temperatures whereas for the metal fuel, the peak temperatures occurred at approximately 90 equivalent full-power days after initial irradiation. This effect can be attributed to the degradation of the thermal conductivity of the metal fuel as discussed earlier.

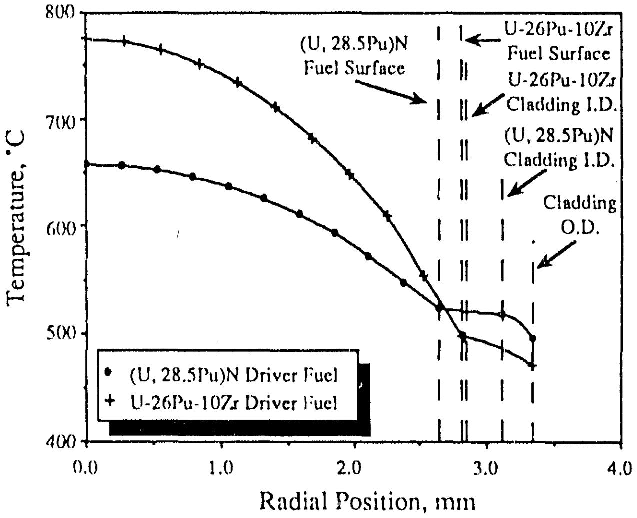

Figure 3. Peak RadialTemperature Profiles for (U,28.5Pu)N and U-26Pu-10Zr Driver Fuel.

Figure 4 depicts the peak cladding strain as a function of time for the peak fuel pin. For the HT9 cladding designated in the design, a 2% maximum cladding strain limit was set. Although both fuels successfully operated well below the strain limit, the nitride fuel exhibited much better performance throughout the irradiation history than the metal fuel (approximately an order of magnitude lower strain levels). Calculations indicated that the (U,Pu)N pin could operate beyond 20 at. % burnup without reaching the 2% cladding strain limit.

Figure 5 illustrates the nominal data for the peak cladding midwall and peak fuel surface temperatures for the peak blanket pins for the nitride and metal fuel systems (UN and U-Zr, respectively). In this case the performance of each fuel type was very similar.

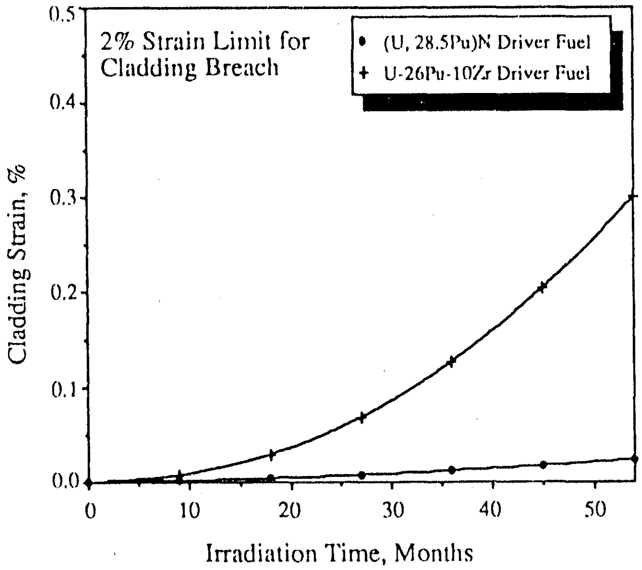

Figure 4. Peak Cladding Strains for (U, 28.5Pu)N and U-26Pu-10Zr Driver Fuel.

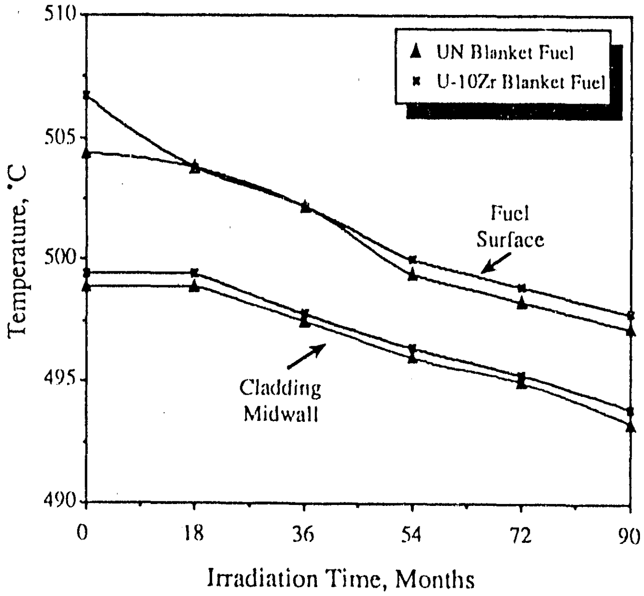

Figure 5. Peak Cladding Midwall and Fuel Surface Temperatures for UN and U-10Zr Blanket Fuel.

Figure 6 shows the predicted peak centerline temperature profiles for the peak blanket pin. The peak cladding strains for the peak blanket pin are illustrated in Figure 7. As in the driver fuel case, both the UN and U-Zr blanket pins operated well below the 2% cladding strain limit with the UN blanket pin consistently exhibiting lower cladding strains (again by approximately an order of magnitude).

Overall, the (U,Pu)N driver fuel was predicted to operate at lower temperatures and cladding strains than the U-Pu-Zr driver fuel. The performance of both blanket pins was quite comparable with the exception of cladding strain.

As with the nitride driver fuel, the UN blanket pin had predicted strain levels approximately an order of magnitude lower than the U-Zr blanket pin.

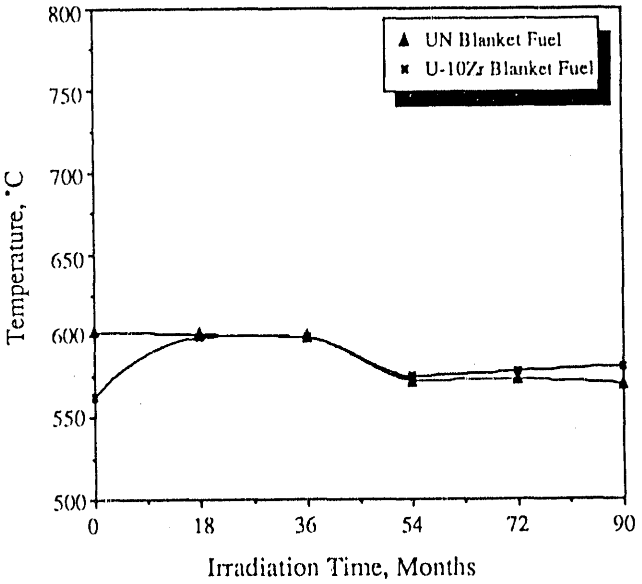

Figure 6. Peak Fuel CenterlineTemperatures for UN and U-10Zr Blanket Fuel.

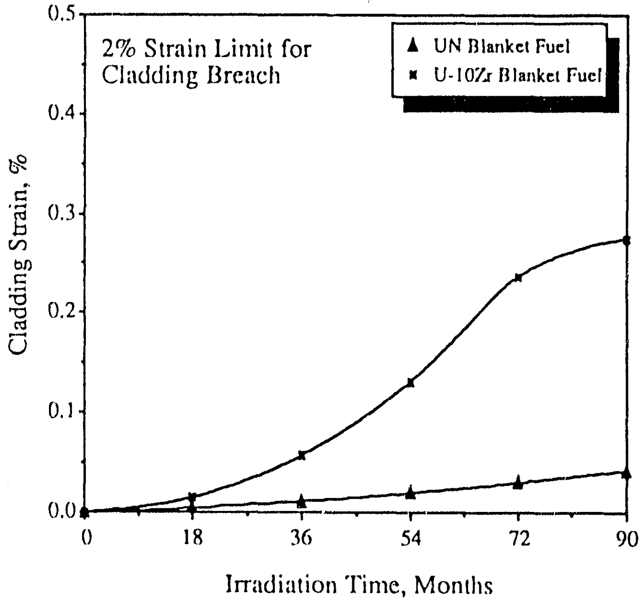

Figure 7. Peak Cladding Strains for UN and U-10Zr Blanket Fuel.

A simplified visual comparison of the operational levels for both driver fuel systems relative to their respective limits for FCCI, melting temperature, and cladding strain (from gas pressure loading) is provided in Figure 8. Although the fractions indicated for temperature based limits will vary dependent on the temperature limits chosen, the conclusion that the (U,Pu)N has greater predicted operating margins remains constant.

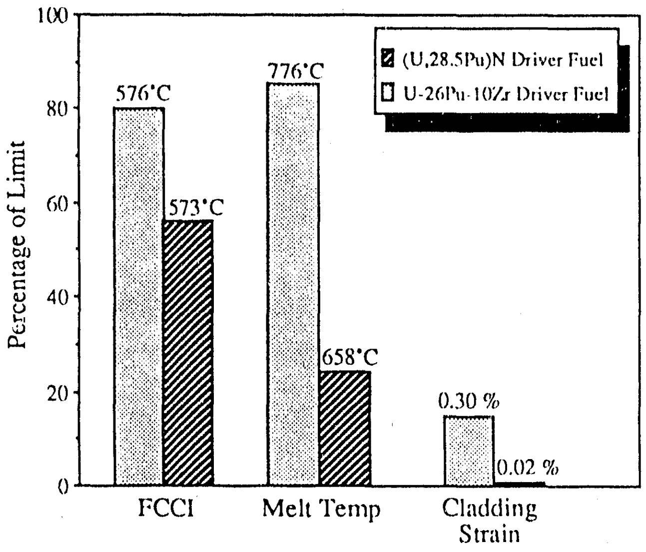

Figure 8. Comparison of Operational Levels Relative to Operational Limits for the (U, 28.5Pu)N and U-26Pu-10Zr Fuel Systems.

4. ADDITIONAL ANALYSES

Several other studies have been conducted to evaluate the performance of nitride fuels with competing fuel systems for prototypic LMRs. A series of analyses by Padilla et al.[13] compared the performance of several competing fuel designs in two beyond design basis events (BDBEs) for the FFTF. Using the SASSYS/SAS4A accident analysis code, four core designs using $(U,Pu)O_2$, U-Zr, UO_2 and (U,Pu)N for the FFTF were evaluated for the loss of flow without scram (LOFWOS) and transient overpower without scram (TOPWOS) BDBEs. The pin designs used for the (U,Pu)N and U-Zr fuel analyses were comparable to those used in the SIEX analysis. Figures 9 and 10 illustrate the predicted peak coolant temperatures for the LOFWOS and TOPWOS in FFTF, respectively. A comparison of these graphs indicates that the (U,Pu)N has the best combined performance of the four fuel systems. This conclusion was also reached by the authors when considering all aspects of the study. Cited characteristics that enhanced the performance of (U,Pu)N included high FCCI thresholds, good thermal performance, and relatively low sodium void coefficients.

5. CONCLUSIONS

In general, the temperatures and cladding strains predicted by the revised version of the SIEX fuel pin performance code agreed well with the literature data. The results indicated that the (U,Pu)N pins had predicted performances equal to or better than the U-Pu-Zr fuel in every facet of the comparison. The nitride fuel consistently operated at lower peak temperatures and lower cladding strain levels than the metal fuel. It also provided larger operating margins due to a combination of superior thermal performance and higher sustainable operational thresholds such as for FCCI and melting temperature.

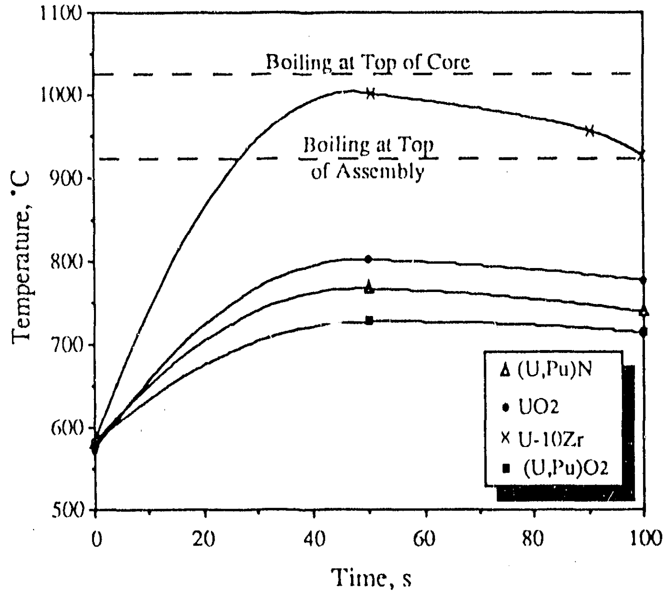

Figure 9. Peak Coolant Temperatures for the Transient Overpower Without Scram. (adapted from Ref. 13)

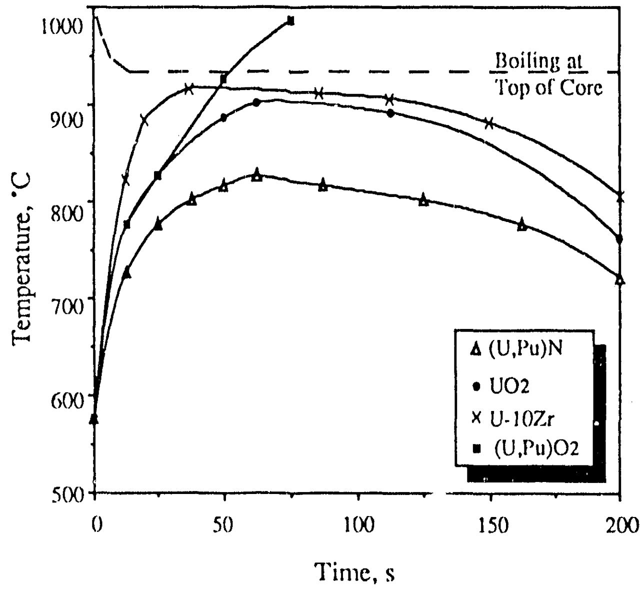

Figure 10. Peak Coolant Temperatures for the Loss of Flow Without Scram. (adapted from Ref. 13)

Based on the literature review, the results of the SIEX study and the additional analyses by Padilla et al., it appears that the nitride fuel and blanket system offers the potential for a wide variety of advantages in performance over competing fuel designs and represents an attractive option as an alternate or primary fuel for advanced LMRs. These advantages can deliver enhanced performance and/or increases in passive safety by providing greater operational margins that aid in the mitigation of various BDBEs and to the overall flexibility of the design of future advanced LMRs.

ACKNOWLEDGMENTS

This work was performed under the auspices of the U. S. Department of Energy.

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