Safety Analysis of Na and Pb-Bi Coolants in Response to Beam Instabilities

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Abstract

A comparative safety study has been performed on sodium vs. lead/bismuth as coolant for accelerator-driven systems. Transient studies are performed for a beam overpower event. We examine a fuel type of recent interest in the research on minor actinide burners, i.e. uranium-free oxide fuel. A strong positive void coefficient is calculated for both sodium and lead/bismuth. This is attributed to the high fraction of americium in the fuel. It is shown that the lead/bismuth-cooled reactor features twice the grace time with respect to fuel or cladding damage compared to the sodium-cooled reactor of comparable core size and power rating. This accounts to the difference in void reactivity contribution and to the low boiling point of sodium. For improved safety features the general objective is to reduce the coolant void reactivity effect. An important safety issue is the high void worth that could possibly drive the system to prompt criticality.

Introduction

Both sodium and lead/bismuth are considered as coolant candidates in accelerator-driven systems. At RIT a global safety study of accelerator driven systems is performed to investigate neutronic and transient characteristics of lead/bismuth vs. sodium as primary coolant and the performance of oxide, nitride, and metallic fuels for various accident initiators and core sizes. In the present analysis we benchmark the two coolants for oxide fuel in the response to a sudden beam excursion. This type of accident initiator is unique to accelerator-driven systems and is open to considerable question. One of the most questionable items is the nature of the initiating circumstances; for example; what is the maximum beam load change that could possibly occur and at what speed can this transition materialize? The outcome will depend strongly on the details of these conditions as well as on the time over which the beam remains on. In the following paper, the beam is presumed to double in strength in an instant and remain on for an unspecified time. The extreme nature of this assumption is subject to debate. However, the analysis of accidents that appear incredible is an important part of the design of a safe reactor. Much can be learned from simulated severe accidents. The purpose of the present paper is to measure the strengths and weaknesses of two particular coolants, independent of probability, or even possibility, of occurrence.

Model and assumptions

The benchmark is performed using a common design, set of assumptions, and computational methods. The continuous energy MCNP simulation code is applied to the neutronics analysis. A three-dimensional pin-by-pin model is defined. Oxide fuel is adopted being diluted with zirconium dioxide. In order to flatten the power distribution, the core is subdivided into two regions with varying content of ZrO_2. We have adopted a Pu to TRU ratio of 40% at BOL since this composition minimizes reactivity losses over a large number burnup cycles [1]. The Pu/TRU ratio is kept constant. The plutonium isotopic vector corresponds to the discharge from spent MOX fuel (5% ^{238}Pu, 38% ^{239}Pu, 30% ^{240}Pu, 13% ^{241}Pu, and 14% ^{242}Pu). The americium composition consists of two thirds ^{241}Am and one third of ^{243}Am. The analysis aimed at increasing the core diameter through an increase in pin pitch while holding the pin diameter and core height constant. Pitch-to-diameter ratios are varied in the range from P/D=1.25 to 2.25 (constant D=8 mm). To compensate the reactivity loss when P/D is increased the fraction of ZrO_2 is adjusted (from core average of 30% at P/D=2.25 to 70% at P/D=1.25) in order to preserve k_{eff}=0.97. A summary of design parameters is presented in Table 1.

Transient analysis is performed with the aid of the SAS4A safety code [2]. A primary heat transport system is defined and represented by the core, primary pumps, the shell side of the heat exchangers, connecting piping, and compressible pool volumes with cover-gas surfaces. Coolant passage through the core is modelled by a single thermal and hydraulics channel. The feedwater system is assumed to remove heat at 100% for all time. Thus, when the power increases above nominal, there will be a mismatch in heat production and heat removal and the net effect is core inlet temperature rising with time. The point kinetics approximation is used for calculating transient power. A value of β_{eff} equal to 0.20% is assumed, a representative value for a minor actinide burner. The coolant flow rate in a lead/bismuth-cooled reactor is limited by erosion/corrosion damage of structural material. At present the flow rate of lead/bismuth is taken to be 2.5 m/s. No such limitation exists for the sodium-cooled reactor where the main concern in the past has been to limit pumping power requirement. For that reason a sodium flow rate of 5 m/s is adopted. Transient response is calculated assuming intact core geometry; i.e., fuel pins and coolant channels are well defined, precluding the possibility for insertion of large reactivity values by core compaction. Temporal and spatial void distributions are calculated. Reactivity feedbacks are modelled by coolant density changes and an assumed Doppler constant of Tdk/dT=-38 pcm. As will be seen, the Doppler coefficient has negligible influence on the operational behavior. The void reactivity coefficient and the prompt neutron lifetime are determined from static neutronic analysis, as discussed in the next section. In a preliminary study, a uniform void coefficient is used. Structural reactivity feedback phenomena (e.g. radial and axial core expansion) have been excluded considering the low responsiveness of a source-driven system to reactivity changes [3]. Under the present conditions, structural expansion introduces reactivity changes that are small with respect to the void effect. It is recognized, however, that such reactivity feedback effects may affect the calculated performance values.

Table 1. Design parameters

Characteristic	Value
Core power	800 MWth
Average linear power	16 kW/m
Core coolant inlet temperature	573 K (Pb/Bi and Na)
Coolant flow rate	2.5 m/s (Pb/Bi) and 5.0 m/s (Na)
Fuel composition	$(Pu_{0.6}Am_{0.4})O_2 + ZrO_2$
Fuel porosity	10 %
Core height	1.0 m
Fission gas plenum height	1.50 m
Outer fuel radius	3.45 mm
Inner cladding radius	3.50 mm
Outer cladding radius	4.00 mm
P/D	Varied from 1.25 to 2.25
Doppler constant (Tdk/dT)	-38 pcm
k_eff (eigenvalue)	0.97
β_eff	0.20 %

Neutronics analysis

Following coolant voiding there is hardening of the spectrum caused by a decrease in neutron scattering. Removal of coolant also results in higher neutron leakage. Hardening of the neutron spectrum and increased neutron leakage are the two dominating physical phenomena contributing to the void reactivity effect. In general, hardening of the spectrum leads to a positive reactivity component due to an increase in the number of neutrons released per neutron absorbed in the fuel while increased leakage gives rise to a reactivity loss since more neutrons may escape the core. The void reactivity effect has been calculated for the present system and is illustrated in Figure 1. The void coefficient is expressed as a function of pitch-to-diameter ratio. The void coefficient is obtained by calculating the k-eigenvalue at a given density and then performing a second calculation but with a density corresponding to a temperature increase of 200 degrees Celsius. The density is changed uniformly over the core and the upper plenum. Figure 1 indicates that the negative reactivity effect associated with increased neutron leakage is not sufficient to offset the positive reactivity contribution of a harder spectrum. The spectrum effect becomes more positive as P/D increases. As a result, the void reactivity coefficient becomes increasingly positive at higher P/D. It is observed that both coolants possess a significant positive void reactivity coefficient. However, the void coefficient tends to be more positive for sodium because of higher moderating power and an influential scattering resonance in ^{23}Na at 3 keV. In the energy region above 100 keV, the fission-to-capture ratio for ^{241}Am rises more rapidly than for ^{239}Pu. For that reason, the void coefficient becomes more positive if the fraction of americium is increased and the fraction of plutonium is correspondingly decreased.

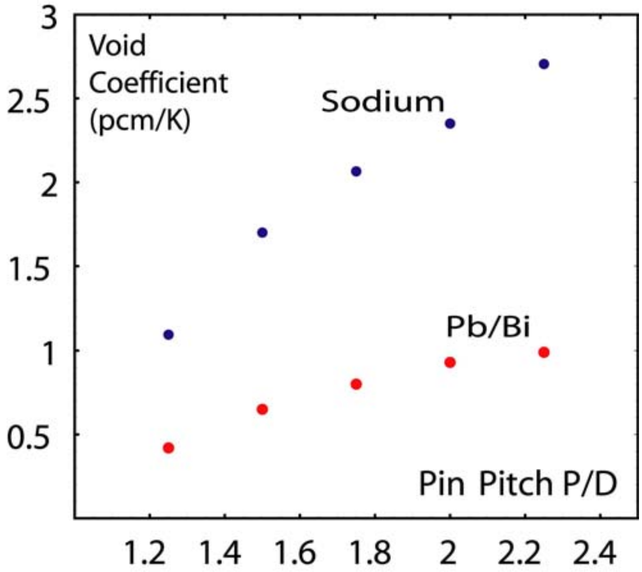

Figure 1. Void reactivity coefficient [pcm/K]

The prompt neutron lifetime was calculated using MCNP. As expected, the prompt neutron lifetime increases with increasing P/D, corresponding to a softer spectrum and longer distance travelled by neutrons up to their point of absorption. Note that the average neutron lifetime in the lead/bismuth-cooled core exceeds 1μs for high pin pitches (P/D>2.0).

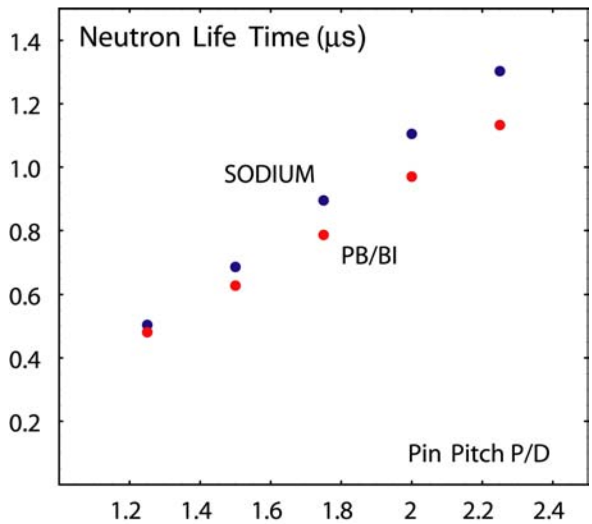

Figure 2. Prompt neutron lifetimes

Failure criteria

In order to predict core damage a set of failure criteria has been postulated, those are listed in Table 2. Several difficulties exist in attempting to provide failure criteria for the existing system. The principal difficulty is the uncertainty in the operating performance of the fuel and structural materials. Chemical and mechanical interactions between the fuel, cladding, and coolant, as well as irradiation performance, etc. are not well known. Validation of failure criteria will require the availability of experimental test data. Nonetheless, preliminary safety margins can be established as a first estimate to envelop worst-case conditions. The fuel is assumed to be stable up to the melting point, which is a reasonable assumption for substoichiometric oxide fuel. The fuel melting point as well as thermophysical properties vary with the stoichiometry. Present fuel properties correspond to an oxygen-to-metal ratio of 1.93. The failure temperature is based on the melting point of PuO_2 [4] and AmO_2 [5] together with the melting point of diluent ZrO_2, applying Vegard's law. The maximum cladding temperature is constrained by mechanical considerations. The primary cladding loading is the internal gas pressure; fuel-cladding mechanical interaction is neglected. We have assumed a maximum internal pin pressure of 10 MPa in steady-state as a result of pressure build-up by the continuous release of fission gases. Under transient conditions the pressure may increase even further causing an increase in the loading of the cladding. Simultaneously, the cladding loses its strength at elevated temperatures. The cladding failure temperature is determined from correlations based on the calculated hoop stress and the failure temperature measured in cladding burst tests (20% cold-worked type 316 austenitic stainless steel) [6]. The transient burst temperature is representative for fast transients where the temperature is rapidly increasing until the cladding fails, providing less time for creep-type deformation.

Table 2. List of failure temperatures

Failure mechanism	Failure temperature	Comment
Melting of oxide fuel	2886 K	$0.11(Pu_{0.6}Am_{0.4})O_2 + 0.89ZrO_2$
Cladding burst temperature	1333 K	20% CW SS316, 5.56 °C/sec, hoop stress 100 MPa.

Transient analysis

Transient response has been examined for an unprotected transient overpower (UTOP) event. It is assumed that the intensity of the external neutron source is *promptly increased by twice the initial value*. Reactor shutdown is disregarded. It is possible to imagine that a control system failure or simply inadvertent operation of the accelerator could lead to an accidental increase of beam power. However, it is important to acknowledge the highly hypothetical nature of the accident under discussion.

Transient power is displayed in Figure 3. For the case displayed the pitch-to-diameter ratio is 1.50. The magnitude of the initial burst is the same, independent of the coolant. The steady-state power will multiply by a factor of S/S_0 if the source strength is stepped from S_0 to S. The speed of the transition is determined by the prompt period. Delayed neutrons do not appreciably slow the response. Following the prompt jump, the power changes as a result of reactivity feedbacks. Coolant void reactivity feedbacks contribute to the course of the accident by adding reactivity. The small negative reactivity feedback associated with the Doppler effect does not influence the course of the accident. Differences in transient behaviour between lead/bismuth and sodium result primarily from the difference in boiling point and void reactivity effect. Coolant density changes provide modest changes in reactivity compared to the full void reactivity effect, which may introduce significant positive reactivity values. This causes the reactivity insertion rate to be considerable larger in the sodium-cooled core. Void generation, and thus positive reactivity insertion, is abrupt in the vicinity when boiling starts. Sodium boiling begins at the core outlet and develops axially downward. In the sodium-cooled core, the void effect adds enough reactivity to bring the reactor to a prompt critical state, with possible severe safety consequences. Prompt critical conditions are established about 400 seconds after accident initiation. Large positive reactivity insertions are potentially possible due to lead/bismuth voiding as well. However, it is seen that the high boiling temperature for lead/bismuth (1943 K) compared to sodium (1154 K) makes voiding less probable even though there are other ways of voiding the coolant besides boiling, i.e. large scale steam generator failure or possibly sudden gas release from ruptured pins. Voiding could possibly occur in severe loss of coolant accidents, such as tank rupture, however this must be regarded as extremely unlikely. It should be recognized that structural damage most likely occurs before boiling is encountered in a lead/bismuth-cooled reactor.

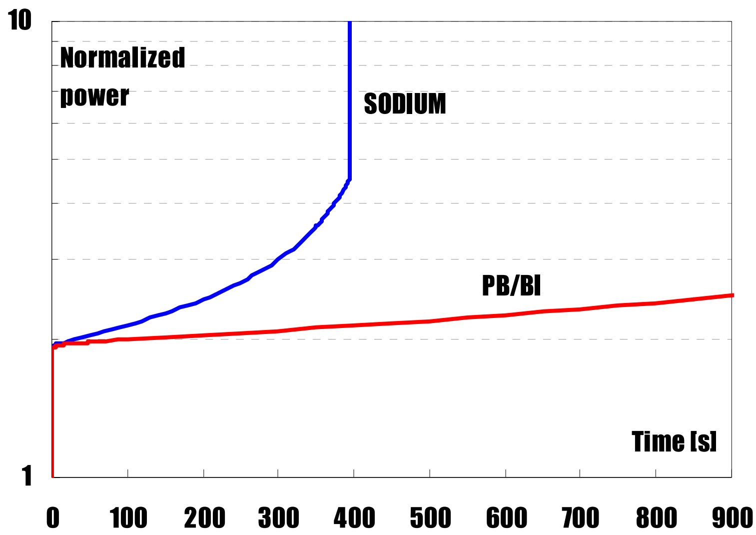

Figure 3. Normalized reactor power. P/D=1.50.

In Figure 4 peak fuel and cladding temperatures are shown for the case P/D=1.50. Since no time is required for heat transport, the fuel suffers a rapid temperature rise. The amount of beam input determines whether there is immediate fuel damage or not. Subsequent heat-up occurs as a result of positive feedback from voiding and insufficient heat removal capability. The steam generators are assumed to remove heat at a rate of nominal power, resulting in increasing core inlet temperature as the transient proceeds. Sharp fuel temperature increase is calculated in the sodium case, as a result of a significant void reactivity insertion. The failure criterion for the fuel is exceeded in 200 seconds and the cladding is expected to reach its burst temperature in 350 seconds. The fuel fails prior to the initiation of sodium boiling (~350 sec) and this might disable the reactor before boiling and prompt criticality occurs. However, it is difficult to determine the consequences of fuel melting. In the sodium case, cladding failure is predicted to occur by burnout. Cladding failure occurs simultaneously with sodium boiling. It is recognized that a substantial change in the nature of the accident may occur at the onset of fuel or cladding damage. Therefore, extrapolation beyond the actual failure points is subject to considerable uncertainty.

The assumption of constant heat rejection rate is conservative. At higher power levels, it is likely there will be some increase in the heat removal above 100% rather than removal at the nominal rate. Taking this into account would yield less pessimistic results. In reality, the feedwater system would try to maintain the correct coolant temperature returning to the core, and if it is not able to do so, the feedwater system would trip and issue reactor shutdown. It should be recognized that for both coolants considered the grace period is in the order of several minutes, which, in principle, provides considerable time for a well-designed safety system to act.

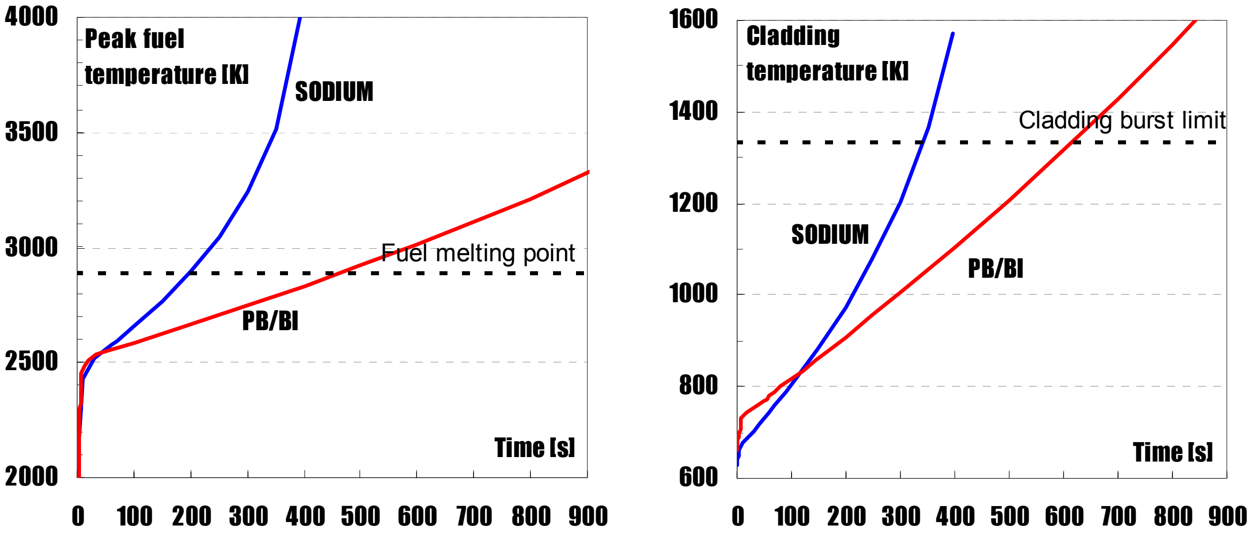

Figure 4. Peak fuel temperature (left) and peak cladding temperature (right). P/D=1.50.

The calculation is repeated for a range of pitch-to-diameter ratios. In Figure 5, the grace period is calculated for different P/D's. The grace period is measured in seconds. The grace time decreases somewhat at large pitches, a consequence of higher void coefficient for larger P/D. From the very basis of the assumptions, the choice of coolant does not change the inevitability of reaching a failure point; the timing of failure is different, however. The Pb/Bi cooled core features twice the grace time compared to the sodium-cooled core with the same P/D and power rating. The calculation revealed a small margin to prompt criticality at large pitches (sodium case). It was found that rapid sodium vaporization and expulsion occurred at the onset of boiling. Prompt criticality could possibly occur in less than 1 sec (P/D>1.50) once sodium boiling is initiated.

Oxide fuel temperatures are sensitive to linear power ratings. The allowable linear power is limited by the melting point. The low thermal conductivity of oxide fuel is compensated somewhat by a high melting point. Figure 6 illustrates the sensitivity of grace time on linear power. The calculation was performed for P/D=1.50. It should be recognized that different power ratings correspond to different core total powers in Figure 6. The number of fuel pins is fixed while the steady-state linear power is varied. In the reference case the linear power is 16 kW/m corresponding to a total reactor power of 800 MWth. The mode of failure differs; fuel failure dominates at high linear powers while cladding failure supersedes as the mode of failure at low linear power (<14 kW/m). The grace period provides an indication of the time available for a safety system to act. It was found that the safety performance of oxide fuel deteriorates rapidly with increasing pin power rating. At high linear power immediate fuel damage may occur, providing little time for a protection system to respond. It is possible to extend the grace period by derating the oxide fuel, but it has some obvious penalties.

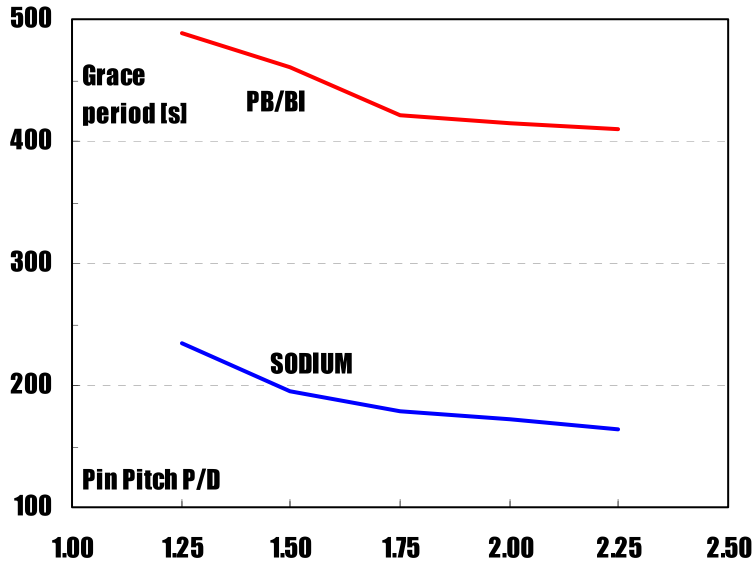

Figure 5. Grace period as a function of pin pitch. Linear power=16 kW/m.

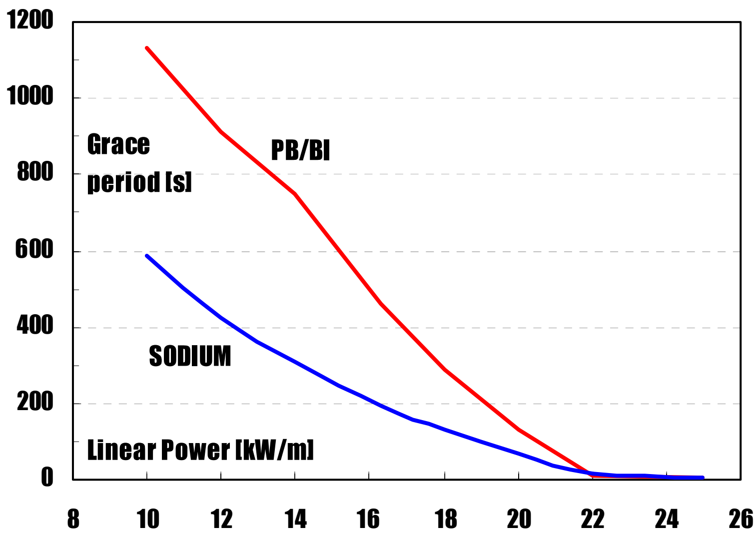

Figure 6. Grace period as a function of linear power rating. P/D=1.50.

The characteristics that have the greatest effect in the present analysis are the differences in boiling point and void coefficient. While the boiling temperature is fixed, the void coefficient can change significantly with design parameters. The void reactivity effect is the result of several physical phenomena and various methods have been proposed for reducing the void worth by design [7]. One possible way of void worth reduction is to reduce the pin size. The net result is shown in Figure 7, where the void worth of lead/bismuth and sodium, respectively, is calculated as a function of P/D. The coolant void worth is determined by removing all coolant from the core and the upper plenum. The results suggest that a significant reduction in the void worth is achievable using smaller pin diameter. Reducing the void worth is an essential design objective. Large values of the void worth may present a difficulty in the licensing of minor actinide burners because of the risk for severe damage to the plant and public safety.

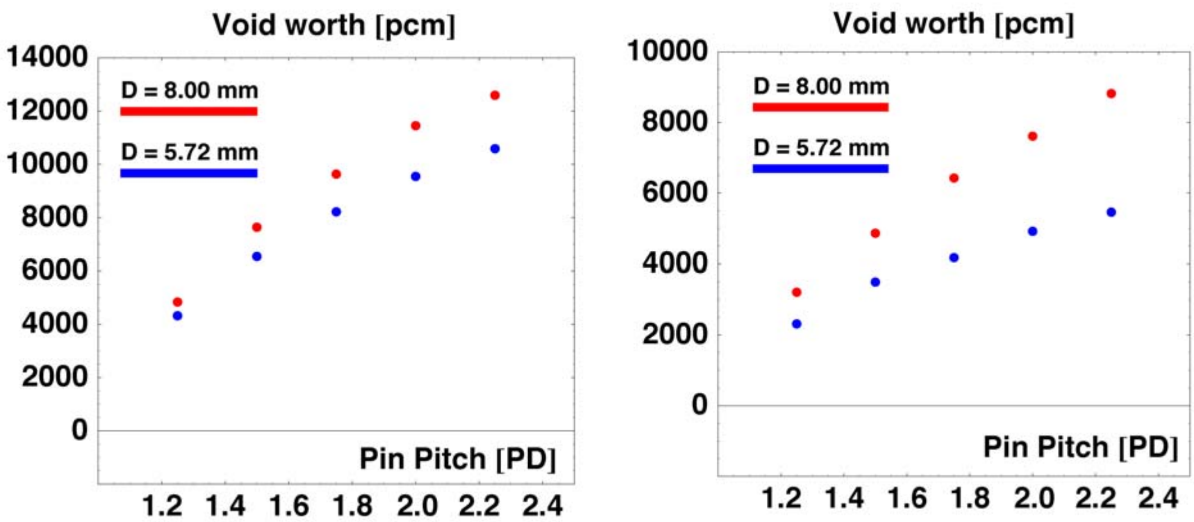

Figure 7. Sodium (left) and lead/bismuth (right) void worth as a function of P/D. Pin diameter is a parameter.

Summary

Comparison was made of the safety performance of sodium vs. lead/bismuth as primary coolant in a minor actinide burner reactor. The systems were benchmarked for oxide fuel. Neutronic investigations were made on the void reactivity effect for a range of pitch-to-diameter ratios. Transient behavior for a beam overpower event and the time-to-failure were compared.

A strong positive void coefficient was found for both sodium and lead/bismuth. The considerable void effect is attributed to a high fraction of americium (60%) in the fuel. It was found that void reactivity insertion rates increases with P/D. In response to the particular accident under discussion, the Pb/Bi-cooled core featured twice the grace time compared to the sodium-cooled core. The essential difference is attributed to the difference in boiling point and void reactivity contribution. An important safety issue is the high void worth that could possibly drive the system to prompt criticality. The problem is the result of the present fuel composition and it exists in both the sodium-cooled reactor and the lead/bismuth-cooled reactor. To some degree, this may be counter-balanced with proper core design, e.g. smaller pitch and pin diameter. For improved safety features, however, the general objective is to reduce the coolant void reactivity effect. The sodium-cooled core was found to have a smaller safety margin to prompt criticality. The high boiling temperature of lead/bismuth makes voiding less probable. The low effective thermal conductivity of oxide fuel results in high fuel temperatures and imposes constraints on the allowable linear power. Derating the oxide fuel could enhance the safety performance, but it has some obvious penalties.

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