PHYSOR 2004 -The Physics of Fuel Cycles and Advanced Nuclear Systems: Global Developments
Chicago, Illinois, April 25-29, 2004, on CD-ROM, American Nuclear Society, Lagrange Park, IL. (2004)

Physics and Safety Studies of a Low Conversion Ratio Sodium Cooled Fast Reactor.

J. E. Cahalan, M. A. Smith, R. N. Hill, and F. E. Dunn
Argonne National Laboratory, 9700 South Cass Avenue, Argonne, Illinois 60439

This paper explores the feasibility of a compact fast burner reactor that can achieve a low transuranic conversion ratio. The major design option considered is the reduction of fissile breeding by the removal of fertile material from the fast reactor system. Reductions in the fuel pin diameter and thus fuel loading were employed to remove fertile material. Reactor performance parameters and reactivity coefficients were evaluated for a compact core design with a targeted conversion ratio of 0.25. To assess the safety implications, a detailed transient analysis model was employed using the SAS4A/SASSYS-1 computer code.

A series of calculations was performed to assess the behavior of the reactor and plant in an unprotected loss-of-flow accident (ULOF). A parametric study was also carried out using increasingly conservative modeling assumptions. The computational results show that for nominal, best-estimate analysis assumptions and input data, the low conversion ratio reactor design responds to the ULOF with a very high level of self-protection. Both short-term and long-term quasi-equilibrium reactor conditions predicted in the analysis indicate very large margins of safety.

KEYWORDS: physics and safety studies, sodium cooled fast reactor, self protection, low conversion ratio, TRU destruction, SAS4A, unprotected loss of flow accident.

1. Introduction

The production of actinides above Pu-239 is noticeably repressed in fast spectrum systems when compared to the thermal systems. The main reason for this behavior can be traced to a lower capture/fission ratio in a fast spectrum system. This spectral effect allows the fast spectrum system to operate with better neutron efficiency and produce fewer higher actinides. To date, most fast reactors have been designed for fissile breeding; however, systems can readily be designed to operate as a TRU burner. The transmutation efficiency can be defined quantitatively by the TRU conversion ratio (CR).

$$CR = \frac{TRU\ production\ rate}{TRU\ destruction\ (fission)\ rate}. \tag{1}$$

In traditional fast reactor systems, the focus was placed on loading more fertile material than was needed to operate the reactor thereby allowing for breeding of fissile materials (in particular Pu-239 or U-233). In terms of Eq. (1), these breeder reactors have conversion ratios greater than one (CR>1). For stabilization of the TRU content the conversion ratio is equal to one (CR=1).

In recent design studies [1], the focus has been placed on conversion ratios less than one (CR<1), referred to as fast burner reactors. This revised focus is motivated by the presence of excess weapons material, the existence of significant stockpiles of separated civil plutonium, and the potential improvements to repository management that TRU destruction can provide. Although conventional fast reactor designs (e.g., CRBR, PRISM, EFR) were configured for fissile conservation (CR=1), many of the demonstration reactors have actually been operated in a burner mode by removing the blanket regions; for example, the base FFTF configuration with full MOX loading yielded a CR of 0.4.

For the Advanced Fuel Cycle Initiative (AFCI) mission, the use of high leakage fast burner core designs to target conversion ratios from 0.0 to 0.5 was investigated last year [2]. In that study a simplified transient analysis was performed that indicated favorable passive safety behavior was retained for unprotected (beyond design basis) accidents. The main drawback of that design was a large diameter core (4.5 m) that had significant economic penalties. This paper presents reactor performance data for a compact configuration with a much smaller core diameter (2.7 m).

A 0.25 CR compact core configuration was chosen and a full set of safety-related reactor physics data was generated. Using this data, and assuming an 840 MW plant design taken from previous studies, a series of calculations were performed to assess the performance of the reactor and plant in an unprotected loss-of-flow (ULOF) accident sequence.

In this extremely low probability ($<10^{-6}$ per year) event, it is assumed that all normal coolant pumping power is lost, and the plant protection system fails to operate, so the plant suffers a total loss of coolant flow with failure to scram. The purpose of this study is to determine the potential for the plant to undergo a ULOF transient and avoid the development of severe core damage that might result in uncontrolled release of radiation. The models and assumptions along with the provided reactor physics data and overall plant design were collected into an input model for the SAS4A/SASSYS-1 computer code [3], which was used for the transient simulation of the ULOF accident sequence for BOC and EOC cores.

In Section 2 the physics and safety models used in the design calculations are detailed. In Section 3 the core design is described. In Section 4 the modeling assumptions used in the safety study and a summary of the base case condition are given. Finally, conclusions are drawn from both the reactor physics and safety analysis.

2. Modeling Methodology

The ANL suite of fast reactor analysis codes was used to evaluate reactor operating parameters and reactivity coefficients. Specifically, the MC^2-2, REBUS-3, VARI3D, and DIF3D codes were used. The MC^2-2 code [4] is used to obtain regional group constants for each fuel composition (one for each fuel enrichment zone), based on ENDF-V data. REBUS-3 is a fuel cycle analysis code [5] which couples the DIF3D multigroup neutron flux code system [6] to a multigroup depletion code. In this work the enrichment search option of the REBUS-3 code is used to compute equilibrium cycle compositions for the compact core design for a startup and recycle scenario.

VARI3D is a neutron diffusion perturbation code [7] which also makes use of the DIF3D code system. In this work the VARI3D code is used to compute the reactivity worth coefficients necessary to assess the transient behavior of each reactor design. Some of the reactivity coefficients needed in the safety analysis cannot be computed using VARI3D, consequently, the finite difference or nodal transport option of the DIF3D code was used to perform direct calculations of the eigenvalue changes.

The principal analysis tool used in the safety study is the SAS4A/SASSYS-1 [3] computer code. This code was developed at Argonne National Laboratory and has been employed in safety analyses for a number of U.S. sodium-cooled fast reactor development projects. The modeling techniques used in SAS4A/SASSYS-1 have been validated by comparative analysis of testing performed at TREAT, EBR-II, and FFTF.

In SAS4A/SASSYS-1, the thermal/hydraulic performance of the reactor core is represented with a single-pin model in multiple channels. For the whole-core model, each of the SAS4A/SASSYS-1 channels represents a single, average pin in a subassembly, and several subassemblies are grouped together, so that a single channel may represent all the pins in a number of subassemblies. Pins with similar geometrical dimensions, power, flow, enrichment, burn-up, thermo-physical properties, and performance characteristics (reactivity feedback, mechanical, thermal, fluid dynamics) are grouped for modeling by a single channel. In this way, all of the pins in the reactor are modeled with a multiple channel model. Typical modeling detail ranges from a few to a few dozen channels, depending on the reactor design and the transient phenomena being simulated.

The model in SAS4A/SASSYS-1 for thermal-hydraulic representation of primary and intermediate sodium systems is called PRIMAR-4. This model permits representation of coolant flow and heat transfer effects outside the reactor by simulation with a network of volumes connected with flow paths. The PRIMAR-4 model computes coolant pressures, flow rates, and temperatures in the primary and intermediate heat transport circuits. Components represented by PRIMAR-4 include the inlet and outlet plenums, pipes, pumps, heat exchangers, steam generators, and reactor vessel air cooling systems.

A point kinetics model was employed to calculate the reactor fission power response to the transient reactivity state. At any time, the net reactivity is the sum of a number of individual reactivity feedbacks that are determined by the transient thermal, hydraulic, mechanical, and neutronic state of the reactor. The feedback reactivities considered in this study are fuel Doppler, coolant density, fuel and cladding axial thermal expansion, radial core expansion, and control rod driveline thermal expansion. A decay heat model is integrated with the point kinetics model for the fission power to track shutdown events in sub-critical conditions.

3. Reactor Design Information

The compact core configuration is based upon a hexagonal lattice and consists of 60 high enrichment drivers and 42 low enrichment drivers giving a total of 102 driver assemblies. There are 16 control rod positions and 3 alternate shutdown positions. The diameter of the reactor is 2.76 m (9.1 ft) with an active core diameter of 1.86 m (6 ft) and an active height of 113 cm (44 in). The fuel assembly lattice pitch is 16.14 cm where each assembly consists of 271 fuel pins. For the base design (CR~0.5), the pin outside diameter is 6.7 mm with a pitch to diameter ratio of 1.32.

The modular size reactor has a power rating of 840 MWt with an assumed 85% capacity factor. Pushing the compact reactor to the fast fluence limit of 4×10^{23} n/cm^2, the core can operate on a 6 month cycle using a seven batch loading scheme with an enrichment split of 1.25 (ratio of high to low enrichment). This fluence limit was chosen based on FFTF operational data for the ferrite HT-9 alloy, the material chosen for cladding and subassembly structure. To decrease the conversion ratio, the neutron balance is altered by removing fertile fuel material (decreasing fertile capture). To reduce the amount of fertile fuel, the fuel pin diameter was decreased, thereby reducing the fuel volume fraction and increasing the fuel enrichment. The REBUS-3 code was used to obtain operating parameters for the large range of reactor geometries and fuel pin configurations; the equilibrium cycle performance characteristics are used to identify feasible designs for low conversion ratio applications.

The peak linear power and burnup reactivity swing were found to increase as the conversion ratio was decreased. The peak linear power and burnup reactivity swing represent the two major points of contention which bring into question the feasibility of this design. As a result, a simplified safety analysis was performed on the compact core [2] to ascertain the design limitations. The previous work indicated that none of the designs in Table 1 would behave adversely under several postulated double fault accident scenarios. As a result of this information, the 0.0 CR design was considered, however, the use of a 100% TRU based fuel form is very undesirable. To avoid this problem and help alleviate the problems associated with the higher burnup reactivity swing and peak linear power, a CR of 0.25 was chosen for the final design.

In the final design model both a startup and recycle scenario are studied. In the startup scenario, the "startup" of the reactor using recycled LWR fuel as the enrichment feed is considered since a significant quantity of the recycled LMR would not be present. In the recycle scenario the spent LMR fuel is assumed to be recycled with realistic losses and the recovered TRU is then used as the primary external feed with the recycled LWR feed used as makeup enrichment feed. A recycled uranium contaminant is also considered in the recycled LWR fuel to simulate a more realistic reprocessing scenario.

First considered are the reactor performance characteristics given in Table 1. The major differences between the startup and recycle scenarios can be seen in the zone enrichments. Since the fuel pin diameter was held constant, the enrichment feed effectively changes to an isotopic mix slightly worse than that of the original recycled LWR enrichment [8]. As a consequence, a higher enrichment is needed in the recycle scenario even though the heavy metal inventory doesn't change. Because of the degraded enrichment feed and higher enrichment, the conversion ratio drops from 0.25 to 0.19. Additionally, the change in the enrichment feed and enrichment also impact fissile breeding causing the peak discharge burnup, peak linear power, and burnup reactivity loss to decrease.

Next considered are the whole-core reactivity coefficients given in Table 2 for BOC (EOC values are similar to BOC). Most of the reactivity coefficients are nearly identical between the startup and recycle scenarios. There is, however, a significant decrease in the delayed neutron fraction and increase in the sodium void worth. Both of these are a result of the degraded isotopic fuel content and the increased enrichment. The most important reactivity coefficients to inspect are the large sodium void worth, small Doppler worth, and large control rod driveline expansion. The last of these has a significant impact on the behavior of the reactor in the safety analysis section. The spatial details of the reactivity coefficients which are used in the safety analysis code SAS4A were evaluated by assigning several assemblies with similar compositions and power levels to a thermal-hydraulic channel.

Table 1. Reactor Performance Data for the 0.25 System Point Design.

	Startup	Recycle
Calculated TRU Conversion Ratio	**0.25**	**0.19**
Low Enrichment Zone Enrichment	44	55
High Enrichment Zone Enrichment	56	68
Recycled HM Feed (kg/yr)	0	1158
Recycled Uranium (kg/yr)	0	478
External HM Feed (kg/yr)	1430	269
External Uranium (kg/yr)	701	57
Net TRU consumption rate (kg/yr)	193	211
Average Discharge Burnup (MWd/kg)	177	177
Peak Discharge Burnup (MWd/kg)	321	254
Peak Linear Power (W/cm)	454	433
Peak Fast Fluence (10^{23} n/cm^2)	4.0	3.9
Burnup Reactivity Loss (%dk)	4.3	3.5

Table 2. BOC Reactivity Coefficients for the 0.25 System Point Design.

	Startup	Recycle
TRU Conversion Ratio	0.25	0.19
Beta	2.76E-03	2.57E-03
Prompt Neutron Lifetime	4.35E-07	4.15E-07
Sodium Density Worth (cents/K)	0.10	0.16
Sodium Void Worth ($)	3.83	6.12
Radial Expansion Worth (cents/K)	-0.39	-0.41
Axial Expansion Coefficient	-0.34	-0.36
Ctrl. Rod Driveline Expansion ($/cm)	-0.81	-0.78
Doppler (cents/K)	-0.047	-0.034

4. Safety Analysis

Using the information in Section 3, input data for the SAS4A/SASSYS-1 models was assembled. The channel assignment coincides with the subassembly flow zone assignment, which was selected on the basis of similar subassembly powers within the two enrichment zones.

For the nominal, base case analysis, best-estimate modeling assumptions and data were assumed for all thermal, hydraulic, reactor kinetics, and reactivity feedback models. The initial primary and intermediate coolant pump flow coast-downs were taken as the pump head decays chosen to represent the run-down of an electromagnetic (EM) pump supplemented by an auxiliary energy storage device that can be programmed to supply an electrical voltage following failure of main supply power. This auxiliary power supply smoothes the transition to natural coolant circulation by extending the abrupt flow coast-down that results when power to an EM pump is lost.

ULOF analysis results for the BOC core configuration are presented in Figs. 1 and 2. Figure 1 shows the early histories (500 s) for the reactor power and the flow in channel 1, the hottest subassembly. Figure 2 shows the reactivity feedback histories for the same time period. The initial coolant outlet temperature in the hot subassembly is 505°C, the initial coolant inlet temperature is 360°C, and the initial peak fuel temperature is 664°C (radial pin average). The peak centerline fuel temperature is 793°C, and the peak fuel/cladding interface temperature is 540°C

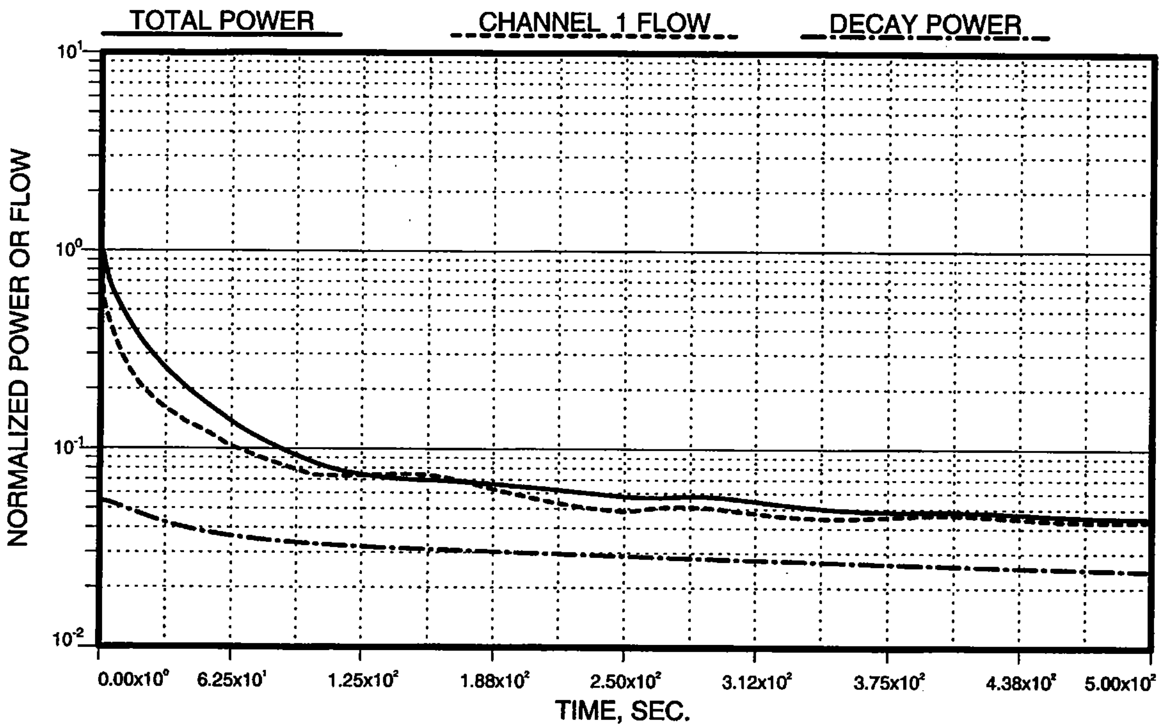

Fig. 1. BOC Base Case Power and Flow History.

The coolant flow in the core drops to around 60% almost instantaneously with the pump power failure, followed by a gradual fall to 10% in the first minute. During the initial flow coast-down, the hot subassembly coolant outlet temperature peaks at 604°C, far below the local boiling point of about 955°C. Although the scram system is assumed not to operate, the inherent reactivity feedbacks combine to give an initial net negative reactivity that reduces reactor power with only a slight lag compared to the flow coast-down.

After the first 90 s of the transient, the flow and power are approaching natural circulation driven only by buoyancy effects. The reactivity effects are caused by the departure from steady-state conditions of the temperatures. During the flow coast-down, fuel axial expansion (prompt with the cladding temperature) and radial core expansion (nearly prompt with the outlet temperature) provide the negative feedbacks, while coolant density feedback is positive. With some delay caused by the lowered coolant velocity and the conduction heat transfer lag, thermal expansion of the control rod driveline inserts negative reactivity. As temperatures fall toward initial values with the power decrease, the net reactivity returns to near zero as the feedbacks drive the reactor system to equilibrate the flow and temperature fields with the available system temperature distribution and heat sinks.

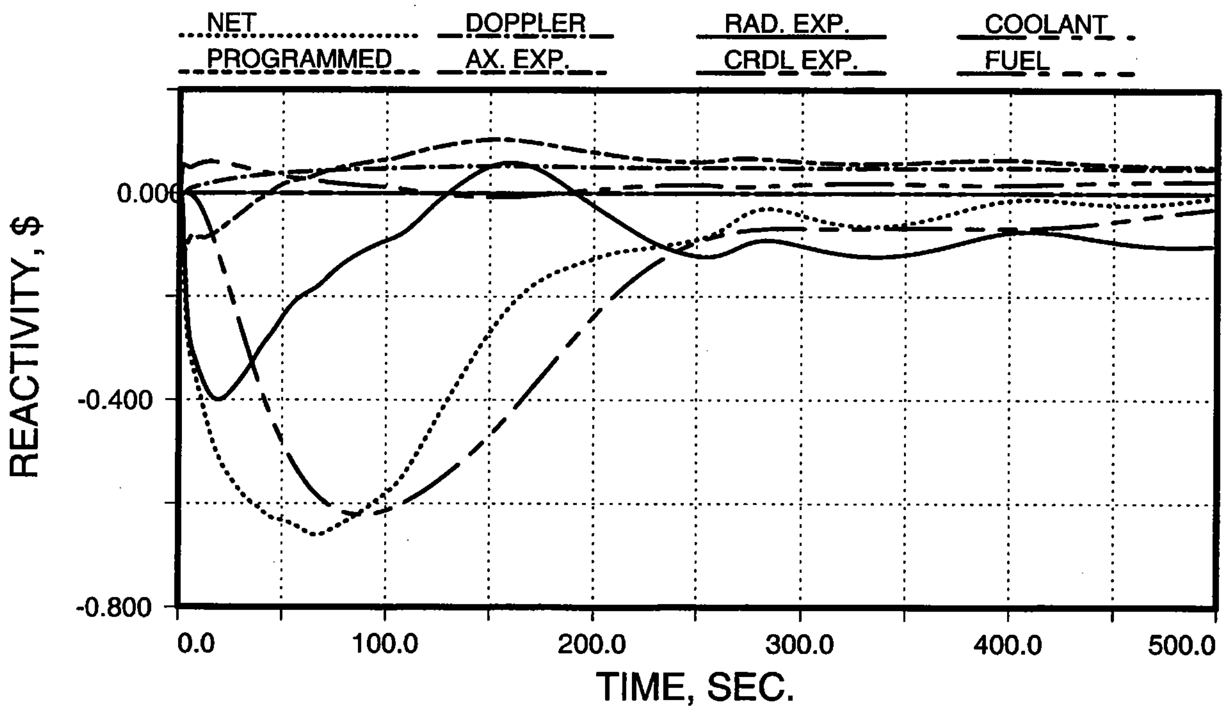

Fig. 2. BOC Base Case Reactivity History.

In the longer term, the system continues to equilibrate slowly as the reactor power falls to decay heat. The net reactivity remains negative until the available heat sink is capable of removing all the decay heat. During this time, radial expansion of the core grid support structure and axial expansion of the fuel and cladding provide negative reactivity feedbacks, while thermal expansion of the reactor vessel (control rod driveline) and coolant density reduction give positive reactivity feedbacks. The fuel Doppler feedback effect is small compared to the other feedbacks. The reactor coolant inlet temperature rises during the first few hours of the transient as the plant temperatures adjust to the heat sink and the reactor fuel decay heat. At the end of the first day, the hot subassembly coolant outlet temperature is 557°C, the heat sink is rejecting 3.45 MW and the decay heat is 4.49 MW, so the system is continuing to heat very slowly. Ultimately, the decay heat will fall to the heat sink capacity, which increases with rising temperature. Subsequently, the system will be cooled as the heat sink removes more energy than is being produced by decay heat. As the system cools, the net reactivity will slowly approach zero, and may temporarily become positive as the fission power increases to make up the difference between decay heat and heat rejection.

For the base case ULOF transient, the BOC core avoids coolant boiling and cladding failure upset conditions by wide margins. The margin to coolant boiling at the top of the core in the hottest subassembly at 24 hours is more than 400°C. The fuel/cladding interface temperature at the top of the core is 557°C This temperature is less than 20°C above the peak initial steady-state interface temperature, so the margin to cladding failure will be essentially the same as the normal operating condition.

ULOF analysis results for the EOC core configuration are similar to that of the BOC case presented. The inherent reactivity feedbacks combine to give an initial net negative reactivity that reduces reactor power with only a slight lag compared to the flow coast-down. For the base case ULOF transient, the EOC core also avoids coolant boiling and cladding failure upset conditions by wide margins. The margin to coolant boiling at the top of the core in the hot subassembly at 24 hours is more than 400°C. Also, the fuel/cladding interface temperature at the top of the core is 556°C. Consequently, no cladding failures would be expected in this transient by a wide margin.

The sensitivity of the best-estimate results to changes in modeling assumptions and input data uncertainties was assessed in parametric analyses. Calculated reactor physics input parameters were changed in the direction of increased conservatism by 20%. Modeling assumptions were changed to reduce and ignore inherent safety reactivity feedback mechanisms. Changes were applied sequentially, so that the effect of each change compounded prior changes. The results of the parametric study show that the passive safety margins are degraded only slightly by incremental changes reflecting data and modeling uncertainties. Significant degradation of passive safety margins occurs only when all of the uncertainties are considered together and the major positive reactivity feedback is increased in the range from 300% to 400%.

5. Conclusions

The feasibility of a compact core for low conversion ratio burning was studied. In this work it has been shown that low conversion ratios can also be obtained using a compact core design, but the compact core has the disadvantage of a shorter operating cycle and increased peak linear power when compared to a high leakage core design of earlier work. A 0.25 conversion ratio configuration of the compact core design was chosen as the system point design for use in a safety analysis study. Both a startup scenario and recycle scenario were investigated.

The safety analysis examined the ability of the reactor design to undergo an extremely low probability accident sequence, an unprotected loss-of-flow (ULOF), without permanent damage. The simulation of the accident was performed using analysis techniques and modeling assumptions validated by data from in-pile testing. The computational results show that for nominal, best-estimate analysis assumptions and input data, the low conversion ratio reactor design responds to the ULOF accident initiator with a very high level of self-protection. Both short-term (first few minutes) transient and long-term (24 hours) quasi-equilibrium reactor conditions predicted in the analysis indicate very large margins to irreversible upsets that could lead to reactor damage and uncontrolled radioactive releases. Margins to coolant boiling are in the range from 350°C in the flow coast-down to 400°C at 24 hours, compared to around 450°C at the normal operating condition. Because temperatures remain near the normal operating range, no fuel-pin cladding failures will occur. The design maintains near-normal safety margins even in an extremely low probability ($<10^{-6}$ per year) accident involving the simultaneous failures of two safety grade systems.

Acknowledgements

This work was supported by the U.S. Department of Energy under Contract number W-31-109-Eng-38.

References

1) R. N. Hill, D. C. Wade, J. R. Liaw, and E. K. Fujita, "Physics Studies of Weapons Plutonium Disposition in the Integral Fast Reactor Closed Fuel Cycle," Nuclear Science and Engineering, **121**, 17 (1995).
2) M. A. Smith, E. E. Morris, and R. N. Hill, "Physics and Safety Studies of Low Conversion Ratio Sodium Cooled Fast Reactors," GLOBAL 2003, Nov. 16-20, New Orleans, LA, 2003.
3) J. E. Cahalan, A. M. Tentner, and E. E. Morris, "Advanced LMR Safety Analysis Capabilities in the SASSYS-1 and SAS4A Computer Codes," Proceedings of the International Topical Meeting on Advanced Reactors Safety, American Nuclear Society, Vol. 2, pp. 1038-1045, Pittsburgh, PA, April 17-21, 1994.
4) H. Henryson II, B. J. Toppel, and C. G. Stenberg, "MC2-2: A Code to Calculated Fast Neutron Spectra and Multigroup Cross Sections," ANL-8144, Argonne National Laboratory (1976).
5) B. J. Toppel, "A User's Guide to the REBUS-3 Fuel Cycle Analysis Capability," ANL-83-2, Argonne National Laboratory (1983).
6) R. D. Lawrence, "The DIF3D Nodal Neutronics Option for Two- and Three-Dimensional Diffusion Theory Calculations in Hexagonal Geometry," ANL-83-1, Argonne National Laboratory (1983).
7) C. H. Adams, "Specifications for VARI3D—A Multidimensional Reactor Design Sensitivity Code," FRA-TM-74, Argonne National Laboratory (1975).
8) R. N. Hill, et. al. "Multiple Tier Fuel Cycle Studies for Waste Transmutation," ICONE 10, April 14-18, Arlington, VA, 2002.