Nuclear Engineering and Design 239 (2009) 2646–2653

Contents lists available at ScienceDirect

Nuclear Engineering and Design

journal homepage: www.elsevier.com/locate/nucengdes

Decay heat in fast reactors with transuranic fuels

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ARTICLE INFO

ABSTRACT

In this work, we performed an evaluation of decay heat power of advanced, fast spectrum, lead and molten salt-cooled reactors, with flexible conversion ratio. The decay heat power was calculated using the BGCore computer code, which explicitly tracks over 1700 isotopes in the fuel throughout its burnup and subsequent decay. In the first stage, the capability of the BGCore code to accurately predict the decay heat power was verified by performing a benchmark calculation for a typical UO_2 fuel in a Pressurized Water Reactor environment against the (ANSI/ANS-5.1-2005, "Decay Heat Power in Light Water Reactors," American National Standard) standard. Very good agreement (within 5%) between the two methods was obtained. Once BGCore calculation capabilities were verified, we calculated decay power for fast reactors with different coolants and conversion ratios, for which no standard procedure is currently available. Notable differences were observed for the decay power of the advanced reactor as compared with the conventional UO_2 LWR. The importance of the observed differences was demonstrated by performing a simulation of a Station Blackout transient with the RELAP5 computer code for a lead-cooled fast reactor. The simulation was performed twice: using the code-default ANS-79 decay heat curve and using the curve calculated specifically for the studied core by BGCore code. The differences in the decay heat power resulted in failure to meet maximum cladding temperature limit criteria by ~100 °C in the latter case, while in the transient simulation with the ANS-79 decay heat curve, all safety limits were satisfied. The results of this study show that the design of new reactor safety systems must be based on decay power curves specific to each individual case in order to assure the desired performance of these systems.

1. Introduction

An accurate prediction of total decay heat and its time dependence is essential for determining the heat removal requirements after reactor shutdown, during spent fuel transportation and storage, and in the reactor safety analyses for various accidents.

The amount of the decay heat generated by UO_2 fuel after normal or emergency shutdown in existing Light Water Reactors (LWRs) can be predicted with a high degree of accuracy on the basis of computer simulations and extensive experimental data. This experience resulted in establishment of a standardized calculation procedure developed by the American Nuclear Society panel of experts for evaluation of the decay heat power in Light Water Reactors (ANSI/ANS, 2005).

A number of well known computer codes such as ORIGEN (Croff, 1983), CINDER (Wilson et al., 1998), FISPIN (Burstall, 1979; Clarke, 1972), RIBD (Thayer and Lurie, 1982) and several others have been used successfully for prediction of nuclide inventories during fuel depletion and decay. In all of the codes, neutron reaction cross-sections required for the solution of the Bateman equations are provided externally as libraries generated using other codes designated for this purpose. Once the nuclide concentrations as a function of time are known, the decay heat following reactor shutdown can be calculated by multiplying activities of the nuclides by their respective recoverable energy per decay values and summing over the entire nuclide vector. Numerous studies have shown good agreement between such summation calculations and decay heat values obtained experimentally (see for example Gauld, 2006). The ANS Standard, mentioned earlier, has been developed in part based on the summation calculations performed with different nuclide inventory calculation codes.

However, the amount of decay heat in future generations of nuclear reactors or in existing reactors operating with innovative fuel cycles has much greater uncertainty for a number of reasons:

- Alternatives to UO_2 fuel compositions have different fission product (FP) yields because they originate from different fissionable nuclides.
- Fission product yields are different in fast and thermal neutron spectra.

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doi:10.1016/j.nucengdes.2009.07.010

- Existing computer codes, capable of predicting the decay power, have, in some cases, outdated databases for the FP decay constants, FP yields, and recoverable decay energy Q-values.
- The choice of nuclides included in the existing computer codes is based either on the nuclides' neutronic importance during irradiation or their out-of-core characteristics (radiotoxicity, decay power etc.) but never both.
- Only limited experimental data exists to confirm the computer simulation predictions.

The ANS Standard considers only 4 fissionable nuclides (U235, U238, Pu239, and Pu241). Except for U238, the fissions are assumed to be induced by thermal neutrons. In the advanced reactors, however, actinides (other than the above four) may contribute a significant fraction to the total power and, thus, generate a non-negligible fraction of fission products. Each actinide has a unique distribution of fission product yields and value of energy released per fission.

Moreover, the fission yield distribution for any actinide depends on the energy of the fission-inducing neutrons. Therefore, the concentration and variety of fission products can be different for reactor systems operated at the same nominal power but with different fuel composition and neutron energy spectrum.

The BGCore reactor analysis system, recently developed at Ben-Gurion University, allows detailed calculations of in-core fuel composition and post-irradiation fuel characteristics, including decay heat, for all existing reactor types and those under development. In contrast to the conventional approach of tracking only the most neutronically important nuclide densities, in the BGCore system, the calculations are performed for the entire set of over 1700 nuclides during all stages of fuel depletion and subsequent decay.

In this paper, we present a brief description of the BGcore system, as well as verification of BGCore system decay heat calculation capabilities. In addition, we calculate the decay heat produced by various actinide bearing fuels in lead and liquid salt-cooled fast spectrum reactors with flexible conversion ratio (Nikiforova et al., 2009; Petroski et al., 2009) using the BGCore computer code. The obtained results are compared with the standard UO_2 fuel decay power curve and subsequently used for simulation of a representative lead-cooled reactor transient in order to assess the importance of using detailed, design specific decay curves.

2. Decay heat calculation methodology

The BGCore reactor analysis system was developed for calculating in-core fuel composition and spent fuel characteristics following discharge. It couples the Monte Carlo neutron transport code MCNP4C (Briesmeister, 2000) with the SARAF (Fridman et al., 2008b) burnup and decay module. The SARAF module was independently developed at Ben-Gurion University. The module can be used in a stand-alone mode similarly to the well known ORIGEN2 code (Croff, 1983). In the BGCore system, the SARAF module receives the relevant data from MCNP, executes the depletion time step, and passes the updated fuel composition back for the next MCNP time step.

The BGCore system is written entirely in the MATLAB programming environment. This greatly simplifies the source code, making it more transparent, efficient, and less error prone. In addition, the MATLAB data can be stored in a standard, fast and easy-access, platform-independent binary format which is also easy to visualize.

Similar to the ORIGEN code, SARAF uses the matrix exponential method to solve a set of first order differential equations representing the evolution of each nuclide concentration with time. However, no asymptotic approximations are used. That is, the main calculation matrix always includes all isotopes available from the SARAF library. The execution time for the depletion step is notably longer than that of the ORIGEN code for similar input. However, it is still negligible compared to the Monte Carlo step.

The actual values of neutron fluxes necessary for performing the depletion are calculated by normalization to the system power using the following procedure. Relative power in each burnable region is calculated as

$$P_i = \frac{\sum_j \Phi_i^{\mathrm{MCNP}} \Sigma_{ij} E_{ij} V_i}{\sum_i \sum_j \Phi_i^{\mathrm{MCNP}} \Sigma_{ij} E_{ij} V_i} \quad (1)$$

where $\Phi^{\mathrm{MCNP}}{}_i$ is the one-group flux in burnable region i as calculated by MCNP i.e. normalized per fission source neutron, Σ_{ij} is the fission cross-section of nuclide j in burnable region i, E_{ij} is the recoverable energy per fission of nuclide j in burnable region i, and V_i is the volume of region i.

The flux multiplication factor (FMF) is calculated as

$$\mathrm{FMF} = \frac{P_{total}}{\sum_i \sum_j \Phi_i^{\mathrm{MCNP}} \Sigma_{ij} E_{ij} V_i} \quad (2)$$

where P_{total}—total system power in Watts.

Then, the average neutron flux in each burnable region (including the regions with no Heavy Metal) is just:

$$\Phi_i = \Phi_i^{\mathrm{MCNP}} \times \mathrm{FMF} \quad (3)$$

The depletion calculations are executed for each burnable region with the calculated real neutron flux values in two stages, known as the Predictor–Corrector (P–C) algorithm (Kang and Mosteller, 1983). The P–C procedure is required to reduce the error introduced by the fact that the depletion calculation is performed using the beginning of timestep values of the flux and cross-sections, which, in reality, may change significantly during the timestep.

In general, point depletion codes such as CINDER or ORIGEN track thousands of isotopes. These isotopes require reaction cross-section data, which is strongly dependent on the nature of the studied system. Therefore, the reaction cross-sections must be obtained either from the Monte Carlo simulation or from a problem-dependent cross-section library provided with the depletion code. The use of problem-dependent libraries may lead to significant errors, while collecting information explicitly for each isotope from the Monte Carlo run would result in prohibitively long computation times. A combination of the two approaches is typically adopted in Monte Carlo–Burnup coupling codes such as MOCUP (Moore et al., 1995), Monteburns (Poston and Trellue, 1998), and MCODE (Xu et al., 2002). In these codes, the cross-sections of the most neutronically important isotopes are calculated by MCNP, while the data for the rest of the isotopes is taken from a reactor-type-dependent ORIGEN library.

The BGCore approach for generation of one-group cross-sections takes advantage of the fact that dividing the neutron flux tally into multiple energy bins has practically no effect on the MCNP execution time. The following calculation procedure is therefore adopted. A fine group spectrum (currently 50,000 lethargy points) is tallied in each burnable region by MCNP and passed on to the SARAF module. The one-group cross-sections are then calculated in a separate subroutine using pre-generated multi-group cross-section sets and the fine group neutron spectrum obtained from MCNP. This approach was originally proposed by Haeck et al. (2005) and implemented in the ALEPH code.

The multi-group cross-section library is generated using the NJOY (MacFarlane and Muir, 1994) computer code for the same 50,000 group structure as the neutron flux tallied by MCNP. The library contains the data for all isotopes and relevant neutronic reaction types included in the most recent JEFF-3.1 evaluated data

file. The multi-group cross-section data is generated at several temperatures relevant for the most common reactor applications.

This procedure dramatically reduces the computation time of MCNP. A reduction in execution time by a factor of 3–10 between the standard and multi-group approaches was observed depending on the nature and complexity of the problem. An additional advantage of the method is the fact that the one-group cross-sections are calculated for all available isotopes and, in principle, for all available reaction types, without any increase in computation time. As mentioned earlier, in conventional coupling methods, the calculated cross-section data is limited only to the most neutronically important nuclides and reactions, while the rest of the data is taken from a standard reactor-type-dependent library.

Considerable effort was made to ensure that the multi-group approach gives one-group cross-section values identical to those obtained with the conventional direct reaction rate tally approach. It was found that for nuclides with complex resonance structure, which are present in the fuel at high concentrations, increasing the number of energy groups does not reduce the error in one-group cross-section below ~1%, especially at low fuel temperatures. The error originates almost exclusively from the unresolved resonance energy region. This is due to the probabilistic treatment of unresolved resonances in MCNP, which correctly predicts the average value of reaction rates but in principle cannot provide the fine structure of the neutron flux. As a result, the self-shielding effect in the unresolved resonance energy region is not accounted for correctly. The error introduced is not statistical in nature but systematic, since the shielded cross-section is always smaller than the infinite dilution one.

In order to overcome this inaccuracy, the multi-group approach was extended by introducing the background cross-section (σ_0) tabulation into the calculation scheme (Fridman et al., 2008a). A series of multi-group cross-section sets is generated for selected isotopes with significant resonance cross-sections for several values of σ_0. The background cross-section is automatically calculated by the code for each calculation case using a simplified formulation described in Fridman et al. (2008a) and then, used to extract (through linear interpolation) the appropriate multi-group cross-section set for a specific resonance isotope which is further used to obtain the "shielded" one-group cross-section value. Introduction of such an extension reduces the difference between directly tallied and collapsed from multi-group cross-sections to well below 1%, while still taking advantage of the fast MCNP execution.

A number of benchmark cases of BGCore against well established and verified, state of the art computer codes for thermal and fast spectrum lattices were performed. Excellent agreement in prediction of most important parameters was observed (Fridman et al., 2008b).

The SARAF data library required for execution of BGCore is also based on JEFF-3.1 evaluated data files. These include fast and thermal fission product yields for over 30 fissionable actinides, decay constants, atomic masses, energy-dependent decay reaction branching ratios and recoverable energy per decay for decay heat calculations.

Currently, 1743 isotopes are tracked in the SARAF code. The isotopes incorporated into the SARAF library include all the nuclides that have evaluated neutron reaction cross-sections in the JEFF-3.1 file, with their respective decay chains as well as all the nuclides with available fission yield data, also with their decay chains.

The fact that all of the isotopes, and not just the most neutronically important ones, are tracked throughout all depletion steps, allows calculation of post-irradiation fuel characteristics such as, activity, radiotoxicity, and decay heat with accuracy determined solely by the availability of the basic data and the data uncertainty.

3. Decay heat calculation methodology benchmark

In the safety analysis of LWRs, the decay heat power as a function of time elapsed after shutdown is calculated routinely using the ANS Standard (ANSI/ANS, 2005) procedure developed specifically for this purpose. The ANS Standard methodology is considered highly reliable. It was developed based on an extensive experimental database as well as summation calculations with a number of computer codes.

In order to ensure that BGCore can correctly predict the decay heat power, a benchmark calculation was performed for a standard PWR UO_2 fuel enriched to 4.2% and irradiated for 1350 EFPD at constant specific power of 37.7 W/gHM. The burnup calculations were performed on a fuel assembly lattice level using typical PWR geometry and operating conditions. The soluble boron concentration in the coolant was assumed to be zero. Although the presence of soluble boron would, in general, have some effect on the concentration of fissile nuclides as well as fission products, the effect can be considered as minor in this case because, as explained later, both ANS Standard and BGCore decay heat calculations were consistently based on the same data generated during the burnup run performed with BGCore code.

In the first stage of the analysis, BGCore was used for both fuel burnup and subsequent decay calculations, so that the decay heat power was obtained directly from the code. In addition, the relative contribution of different fissionable nuclides to total power during fuel burnup was also calculated with BGCore in the same burnup calculation (Table 1). The fractional fissions data was subsequently used as an input for the Standard ANS decay heat calculation procedure. Then, the decay heat calculated directly with BGCore was compared with that calculated with the ANS Standard methodology.

According to the ANS Standard, fissions in only 4 nuclides contribute to the reactor power. The decay heat resulting from fission events in each of these nuclides as a function of time is described via summation of 23 exponential terms. The effect of neutron capture in fission products is taken into account through an additional multiplier provided in either tabular form or as semi-empirical formulas. The contribution of actinides is limited to the decay of U239 and Np239 and calculated explicitly. The heat resulting from the decay of other actinides is not accounted for.

The results obtained with BGCore agreed with ANS Standard predictions within 5% up to 10^8 s after shutdown (Fig. 1), which is within the 2σ uncertainty band of the ANS Standard.

Fig. 2 shows integrated decay heat power calculated for a 3400 MW_{th} PWR reactor core with BGCore and using ANS Standard data. Here, again, the obtained agreement between the results of

Table 1
Initial data for decay power calculation with ANS Standard methodology.

	Fractional fissions			
Irradiation time (days)	U235	U238	Pu239	Pu241
0	0.94	0.06	0.00	0.00
100	0.89	0.06	0.05	0.00
200	0.81	0.06	0.12	0.00
300	0.75	0.07	0.18	0.01
400	0.69	0.07	0.22	0.02
500	0.64	0.07	0.26	0.03
600	0.59	0.08	0.30	0.04
700	0.55	0.08	0.33	0.05
800	0.51	0.08	0.36	0.06
900	0.47	0.08	0.38	0.07
1000	0.43	0.08	0.40	0.08
1100	0.40	0.09	0.42	0.10
1200	0.36	0.09	0.44	0.11
1300	0.33	0.09	0.46	0.12
Q-value (MeV/fission)	202.2	205.5	211.2	213.7

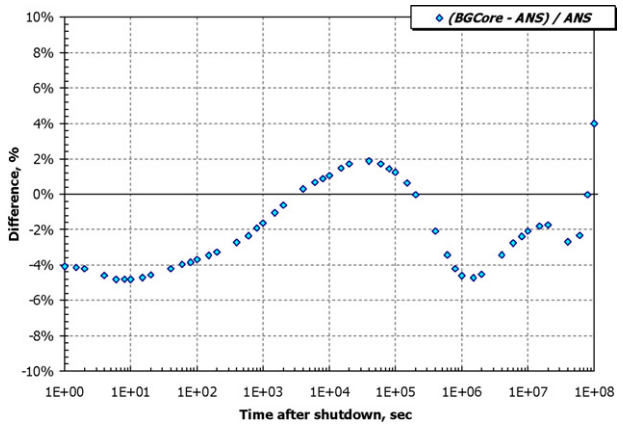

Fig. 1. Percent difference between BGCore and ANS Standard decay heat calculation (PWR unit cell).

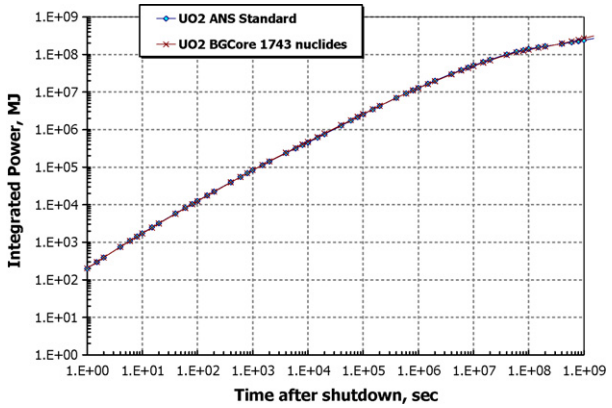

Fig. 2. Integrated decay power for 3400 MW_{th} PWR core.

the two codes is excellent, suggesting that BGCore can be reliably used for decay heat calculations.

Once the capability of BGCore code to predict the decay heat after reactor shutdown was verified, the code was used to calculate the decay heat power of the reactor designs studied in the Flexible Conversion Ratio Fast Reactors Evaluation project (Todreas et al., 2009).

4. Decay heat curves for flexible conversion ratio cores

In this section, we present the results of decay heat calculations for two different fast reactor designs with flexible conversion ratio. Brief description of the cases considered is presented in Table 2. The core nuclear designs used in this analysis are described in detail in Shwageraus and Hejzlar (2008) and Shwageraus and Hejzlar (2009) for the lead-cooled and liquid salt-cooled core designs respectively. The considered reactors can operate interchangeably either

Table 2
List of calculated cases.

Case designation	Fuel	Reactor type
PWR UO_2	UO_2, 4.2% enrichment	Typical PWR
Lead, CR = 1	Metallic Zr–U–TRU, 16.7 wt.% TRU enrichment	Lead-cooled fast reactor, CR = 1
Lead, CR = 0	Metallic Zr–TRU, 34 wt.% TRU enrichment	Lead-cooled fast reactor, CR = 0
Salt, CR = 1	Metallic Zr–U–TRU, 15.7 wt.% TRU enrichment	Liquid salt-cooled fast reactor, CR = 1
Salt, CR = 0	Metallic Zr–TRU, 30 wt.% TRU enrichment	Liquid salt-cooled fast reactor, CR = 0

Table 3
Initial TRU isotopic vector.

Isotope ID (ZZ AAA)		Wt.%
93	237	6.64
94	238	2.75
94	239	48.65
94	240	22.98
94	241	6.93
94	242	5.03
95	241	4.65
95	242m	1.47
95	243	0.02
96	243	0.0050
96	244	0.4960
96	245	0.0380
96	246	0.0060

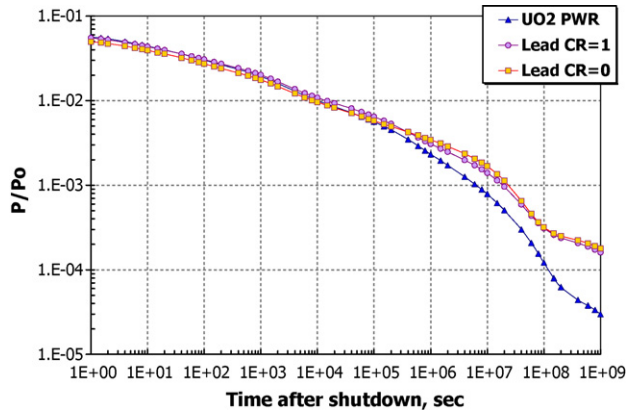

Fig. 3. Decay heat power for UO_2 LWR and lead-cooled CR1 cores (Log–Log scale).

as actinide burners with zero conversion ratio (CR = 0) or in a self-sustainable fuel cycle (CR = 1). The initial Trans Uranium (TRU) isotopes composition used as fissile component in all the fast reactor fuel cases presented in Table 2 corresponded to that of a typical LWR spent fuel with initial enrichment of 4.2%, discharge burnup of 50 MWd/kg, and 10 years of cooling following the fuel discharge. The TRU isotopic vector is shown in Table 3. In the decay heat calculations of the fast reactor fuel cases, it was conservatively assumed that the shutdown occurs at the end of irradiation cycle: after 1200 Effective Full Power Days (EFPD) for the CR = 1 cases and 550 EFPD for the CR = 0 cases.

The results of the calculations are presented in Figs. 3–6. Figs. 3 and 5 compare decay heat of the lead-cooled and salt-cooled reactor designs respectively with that of a typical PWR on a Log–Log

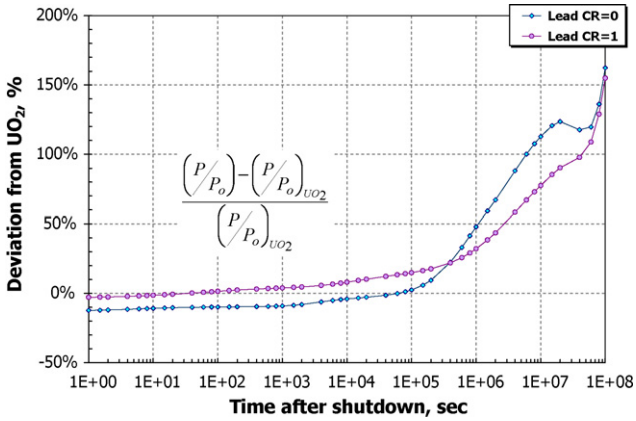

Fig. 4. Difference between P/P_o values for UO_2 LWR and lead-cooled cores.

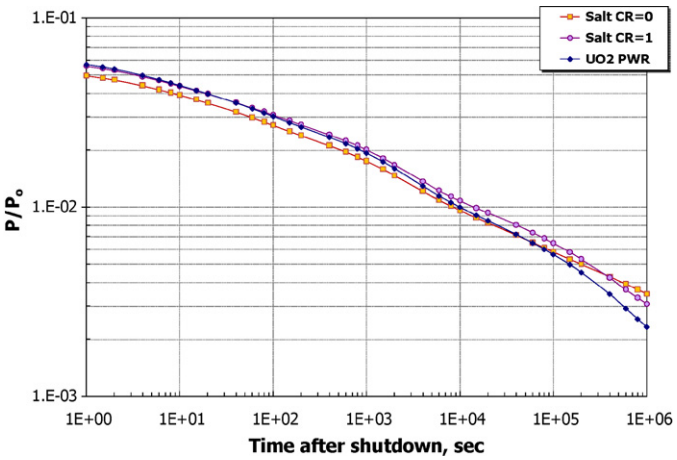

Fig. 5. Decay heat power for UO_2 LWR and salt-cooled cores (Log–Log scale).

scale, while Figs. 4 and 6 present relative deviation of the decay heat for the respective core designs from the typical PWR UO_2 fuel values. The figures plot deviation of the decay to nominal power ratio relative to the UO_2 LWR case according to the following formula:

$$\frac{[P/P_o]_{\text{Case}} - [P/P_o]_{UO_2}}{[P/P_o]_{UO_2}} \times 100\% \quad (4)$$

The CR = 1 lead-cooled core has slightly lower decay power than in the PWR case up to about 100 s after shutdown. After 100 s, the CR = 1 decay power becomes larger than that of the LWR and the difference increases with time and reaches as much as 30% at 10^6 s (~300 h).

In the case of the CR = 0 lead-cooled core, the LWR decay power remains higher for about 20 h. However, from that point on, the LWR heat decreases with time at a much faster rate than that of the CR = 0 case, leading to about 50% higher power for the CR = 0 case at about 300 h. For the first 140 h after shutdown, the CR = 0 core decay heat is generally smaller than that of CR = 1 core by up to 10%.

As in the lead-cooled design, the salt-cooled CR = 1 core decay heat becomes considerably higher than that of the PWR after about 100 s after shutdown. The lead- and salt-cooled CR = 1 core designs have very similar decay heat because initial fuel composition, burnup, and neutron spectra are comparable for both cores (Hejzlar et al., 2009).

Decay heat of the CR = 0 salt-cooled core is lower than that of the CR = 1 core by about 10%. The decay heat values of the two designs become roughly equal at about 4×10^5 s after shutdown.

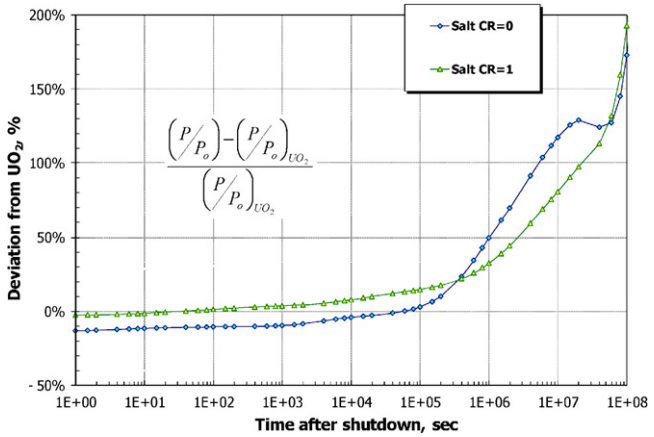

Fig. 6. Difference between P/P_o values for UO_2 LWR and salt-cooled cores.

The observed trends are similar to the lead-cooled designs. As compared with the UO_2 PWR case, the CR = 0 salt-cooled core has lower decay heat by a few percent up to about 10^5 s. After that, the CR = 0 core decay heat becomes larger and the difference reaches a factor of 2 at about 6×10^6 s.

The performed calculations clearly demonstrate the importance of using decay heat data derived from an appropriate reactor model, which takes into account fuel composition and neutron spectrum. Therefore, the results of the decay heat calculations for the CR = 1 and CR = 0 lead- and salt-cooled reactor cores were subsequently used as input data for safety analysis of the respective reactors with the RELAP code performed in the framework of the Flexible Conversion Ratio Fast Reactors Evaluation project (Nikiforova et al.,2009; Petroski et al., 2009).

5. Impact of decay heat on safety analyses

This section presents an example of a transient analysis which demonstrates the importance of accurate decay heat power modeling. An Unprotected Station Blackout (SBO) transient of the lead-cooled fast reactor with unity conversion ratio was chosen for this purpose. This is motivated by the fact that in the course of the lead-cooled reactor design, the aforementioned transient scenario was found to be the most restrictive with respect to decay heat removal capabilities, while preserving integrity of the core.

The transient was modeled with RELAP5-3D (2005)—a thermal hydraulic code with capabilities to simulate the transient response of various reactor systems with different coolants and working fluids, including supercritical CO_2 and lead–bismuth. Since lead–bismuth has properties very close to those of lead, the lead–bismuth property file was used in the simulations. The studied lead-cooled reactor is coupled to a power conversion system (PCS), with supercritical CO_2 as working fluid, through intermediate heat exchangers (IHX) integrated into the reactor vessel. The details of the RELAP5-3D model are discussed in an accompanying paper (Nikiforova et al., 2009).

The SBO transient simulations were performed twice: using the ANS-79 Standard decay heat curve built into RELAP5, then, repeated using BGCore data generated for the specific reactor core. The adequacy of the safety system design to remove the decay heat, while satisfying all the safety limits imposed on the system components, was compared for the two simulations.

In an unprotected SBO accident, the core fission power is determined only by inherent negative reactivity feedbacks, whereas decay heat is removed mainly through the Reactor Vessel Auxiliary Cooling System (RVACS). The Power Conversion System (PCS) is isolated by automatic valves to prevent turbine overspeed. The supercritical CO_2 working fluid is redirected into a heat exchanger immersed in a large water tank located outside the reactor containment. This Passive Secondary Auxiliary Cooling System (PSACS), supplemental to RVACS, is required in order to assure that the peak cladding temperature remains below the failure limit at all times during this unprotected accident. Once the power conversion system is isolated, the IHX bypass valves open to prevent flow stagnation in the loop.

Each of the four PCS loops is connected to an independent PSACS as illustrated schematically in Fig. 7.

The design and performance of the lead-cooled reactor decay heat removal in various accident scenarios is described in more detail in the accompanying paper by Nikiforova et al. (2009).

The unprotected SBO accident assumes simultaneous occurrence of the following events:

- Loss of forced circulation as reactor coolant pumps trip due to loss of AC.
- Loss of AC causes loss of PCS precooler pumps.

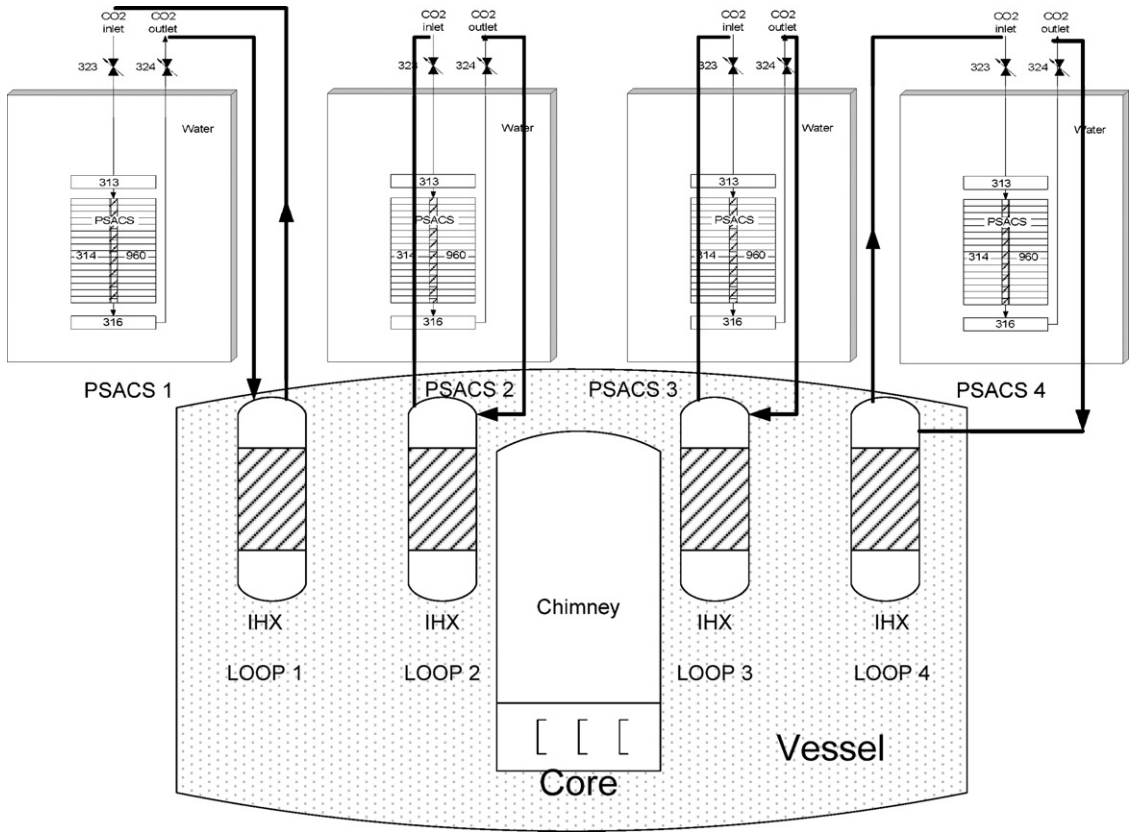

Fig. 7. Relative layout of the reactor and PSACSs as modeled in RELAP5.

- Isolation of generators from the grid due to loss of load.
- Failure of the reactor to scram and shut down due to negative reactivity feedback.

The core simulation in RELAP5 is based on a point kinetics model. The model uses core-average fluid conditions, power-squared weighting factors, and reactivity feedback coefficients to model total reactivity for the kinetics calculations of the core total power (RELAP5, 2005).

Current regulations (10 CFR 50.63) require a SBO mitigation strategy of up to 8 h. Future reactor designs such as AP1000 and ESBWR have 72 h SBO mitigation strategies, allowing for the elimination of safety-related emergency diesel generators. Seventy-two hours is a time typically considered for sufficient recovery of onsite or off-site AC power and was therefore adopted for our analysis.

An additional assumption was that operation of only two trains should be sufficient for satisfactory performance during the station blackout accident.

In the initial analysis, the ANS-79 Standard decay heat curve available in RELAP5-3D was used. The ANS-79 decay heat curve includes only four fissionable nuclides: U235, U238, Pu239, and Pu241. However, as shown in the previous section, the FCR reactor cores operate in a fast neutron spectrum and have significant TRU loading, including Pu, Am, Cm, and Np isotopes. This uniquely determines the amount and variety of the fission products generated, resulting in significant differences in decay heat, as demonstrated in Figs. 3 and 4.

Fig. 8 compares the decay heat generated by the CR = 1 and CR = 0 lead-cooled TRU loaded cores with that of the ANS-79 Standard included in RELAP5. As can be observed, the CR = 1 core is the limiting case due to its higher decay heat generation rate. Moreover, it has less favorable reactivity feedback coefficients in comparison with the CR = 0 core (Shwageraus and Hejzlar, 2008).

The response of the FCR CR = 1 reactor to the station blackout accident with the ANS-79 Standard decay heat curve and with the BGCore-generated decay heat curve is presented in Fig. 9. The cladding temperature initially decreases because both the RVACS and the PSACS remove heat at higher rate than is generated. Once the water in the PSACS tanks boils off, the PSACS is isolated and the peak cladding temperature begins to rise until the decay heat generation is balanced by the heat removal through the RVACS. As can be observed from Fig. 9, evaporation of water in the PSACS tanks occurs about an hour earlier (point of maximum PCT) in the case of the BGCore-generated decay heat curve, and the cladding temperature decrease in the initial stage of the transient is less pronounced. Furthermore, in the simulation based on the BGCore decay heat data, the peak cladding temperature increases after the PSACS isolation occurs at a much faster rate and results in exceed-

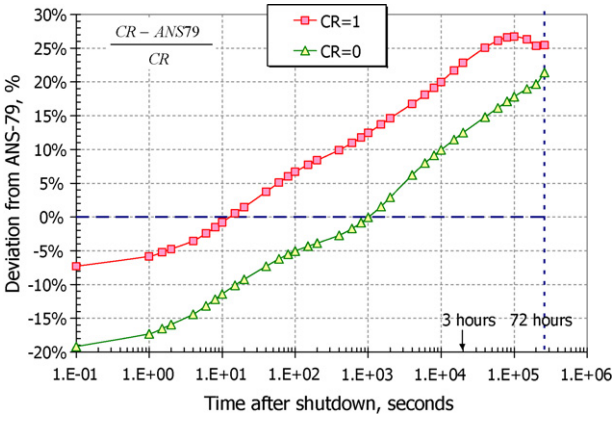

Fig. 8. Decay heat generation rate deviation (%) from the ANS-79 decay heat curve.

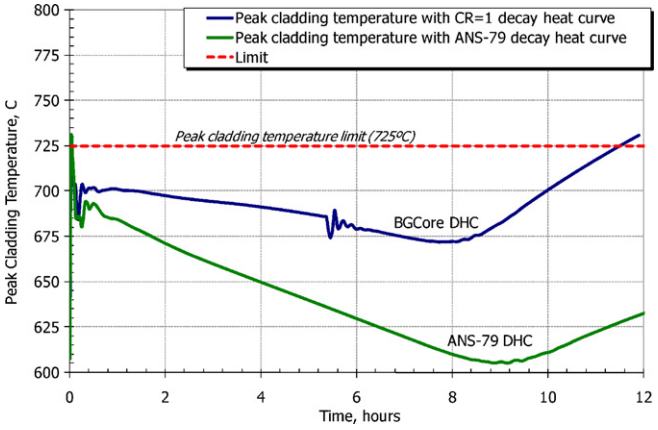

Fig. 9. Comparison of peak cladding temperature for CR = 1 core with different decay heat curves.

ing the cladding temperature limit about 12 h after initiation of the transient. The simulation based on ANS-79 data, on the other hand, suggests that the performance of the safety systems in this accident is satisfactory.

As a result of adopting the BGCore decay heat data, the PSACS required significant design change to cope with the considerable increase in the amount of the decay heat generated. First, the increase in the net amount of the decay heat generated required additional water in the PSACS water tank. Second, the PSACS heat exchanger size was adjusted to obtain the desired heat removal rate. Additional discussion related to mitigation of Unprotected SBO accidents in the lead-cooled fast reactor with flexible conversion ratio can be found in Nikiforova et al. (2009).

6. Summary and conclusions

Dissipation of decay heat following normal reactor shutdown or in case of an accident is one of the major safety concerns of any reactor. A standard procedure for calculation of the decay heat developed and verified by the American Nuclear Society Standards Committee is routinely used in the accident analysis of existing Light Water Reactors. The ANS Standard methodology relies on a number of simplifying assumptions, which make this a procedure applicable only to the current generation of LWRs with conventional UO_2 fuel. The use of the ANS Standard decay heat calculation methodology for the analysis and safety systems design of advanced reactors with non-conventional fuel is much less reliable. In this paper, we evaluate the extent of this uncertainty and present an example, which illustrates the importance of accurate decay heat prediction in the design of advanced reactors. A lead-cooled fast reactor with flexible conversion ratio was used as a representative advanced reactor design.

Detailed decay heat calculations of the lead-cooled fast reactor were performed with the BGCore reactor analysis system, recently developed at Ben-Gurion University. The code couples a Monte Carlo neutron transport solver with a burnup module in a unique way that allows accurate tracking of over 1700 nuclides during irradiation and following shutdown as well as calculation of post-irradiation fuel characteristics, including decay heat power.

Initially, we performed a benchmark exercise, which tested decay heat calculation capabilities of the BGCore system by comparing BGCore results against the most recent edition of the ANS Standard for calculation of UO_2 fuel decay heat in LWRs. Very good agreement between BGCore and ANS Standard methodology results was obtained.

Then, we used the BGCore system for calculation of the decay heat generated by the lead-cooled fast reactor and the liquid salt-cooled fast reactor, each with zero and unity conversion ratios. The results of the calculations were compared with those of typical PWR UO_2 fuel. The results show that decay heat produced by the lead- and liquid salt-cooled reactors differs substantially from that of the PWR fuel, suggesting that the ANS Standard decay heat curve cannot be used as a basis for safety systems design of these reactors.

Finally, the decay heat data generated by BGCore was used for modeling the lead-cooled reactor response to an Unprotected Station Blackout accident. The results of this simulation were compared with those obtained from the same model but using ANS-79 Standard decay heat data. The differences in the decay heat data resulted in the requirement for significant design changes in the reactor safety systems.

In summary, the calculations performed clearly demonstrate the importance of using decay heat data derived from an appropriate reactor model, which takes into account fuel composition and neutron spectrum. The use of the ANS Standard LWR decay heat data for the safety analysis of advanced reactors may result in poor design choices, either compromising reactor safety or leading to overdesign of decay heat removal systems, and leading to unnecessary associated cost increases.

Acknowledgments

The support provided by the U.S. DOE Nuclear Energy Research Initiative (NERI) for this project on flexible conversion ratio fast reactors is gratefully acknowledged. The authors acknowledge the contribution of Anna Nikiforova at MIT who performed transient analysis calculations with the RELAP5-3D code, Emil Fridman at BGU who assisted in development and benchmarking of the BGCore code, and Michael Driscoll at MIT who guided the overall analysis

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