Karlsruhe Institute of Technology Campus North

Institute for Neutron Physics and Reactor Technology

KANEXT application note 2011/01

On the use of the modular code system KANEXT for the analysis of potential hazards of inventories of energy producing nuclear fission reactors

with
application to BWR-2 at the Japanese Fukushima Daiichi site

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Chapter 1

Introduction

The recent reactor accident problems with several BWR at the Fukushima site in Japan has initiated an activity in Germany with significant INR participation to prepare an extensive database with information for a broad group of interested people [1]. In this context support for the analysis of burn-up dependent inventories and doses was suggested [2].
The objectives of the before mentioned ad-hoc activity at KIT have been subject of well-coordinated R&D investigations in the former institute FZK/INR. Important activities were:

- Development and application of deterministic multigroup reactor simulation tools. The resulting modular code system KAPROS (see e.g. [3]) is preserved and was recently updated to KANEXT for use on modern computer systems [4]. A special module KARBUS [5] was created for reactor fuel burn-up and depletion.

- Development and application of software for the analysis of spent nuclear fuel. The original code ORIGEN from Oak-Ridge [6] was adapted and improved for the German reactor park. The resulting relatively simple code KORIGEN [7, 8] was utilized extensively and also transferred to German utilities and research center. These codes have a large number of options to calculate consequences during and after irradiation (power generation) of nuclear reactor fuel. E.g. determination of heavy metal and fission product inventories in grams, decay heating in Watt after shut-down, material radio-toxicities in Curie. The current KANEXT modular system contains an adapted version of KORIGEN (KORIGEN2001) with a special interface module KORINT for automatized KORIGEN2001 input preparation from more sophisticated reactor simulations.

- Development and application of software for the analysis of the migration of radio-toxic substances after release from nuclear reactors after accidental events. In the framework of these activities several source terms were created within the KAPROS modular code system. A major activity was the design and realization of the real-time RODOS system. Citation from [9]:
  The development of the Real-time on-line decision support system RODOS is a major item in the area of radiation protection of the European Commission's Framework Programs. In parallel, the German Ministry of Environment, Nature Conservation and Reactor Safety (BMU) financially contributed to the project with emphasis on early emergency response.

- A further FZK/INR contribution was the theoretical and experimental core catcher research.

The detailed analysis of accident behaviour of a commercial BWR is a complex task with several important aspects like:

- Knowledge about fuel element design
- Knowledge about energy production history
- Knowledge about fuel element management
- Knowledge of measured data during the reactor cycles

This specific analysis is a typical task for the reactor operators and the associated engineering companies. On the other hand it is possible to make more generalized analysis for selected reactor conditions and properties. As a general rule, any reactor simulation has to rely on approximations and simplifications. For the more generic investigations the KANEXT modular system with associated stand-alone codes like KORIGEN is a well suited tool. In the next chapters generic investigations for BWR will be discussed.

Chapter 2

Some characteristics of nuclear fuel irradiation

The systematic analysis of nuclear fuel irradiation shows different behaviour of the heavy metals uranium and thorium. The fertile uranium isotope U^{238} is converted by neutron capture to U^{239} and then changing by β-decay with half-life 23.5 minutes to Np^{239}, whereas the only stable thorium isotope Th^{232} is converted to Th^{233} before β-decaying with half-life $\approx$ 28 days to Pa^{233}. During the relatively large half-life of Th^{233} relevant competition between decay and neutron flux density dependent reactions reactions is observed. The figures 6.1 and 6.2 show results from former investigations. We may observe in figure 6.1 that in the case of uranium based fuel, in a LWR the reactivity strongly decreases with normalized burn-up Gigawattdays per tonne initial heavy metal (GWD/TIHM), with only very weak influence of the power rating. In the example in 6.2 for thorium based fuel in a fast reactor system the effects are just opposite.
The results of figure 6.1 are of high interest for generic LWR analysis for uranium based fuel, as applied in all German nuclear power reactor systems. The insensitivity of the reactivity loss curve to the applied power level indicates that the reactor inventories depend mainly on the normalized extracted amount of fission energy from the system under consideration. For the heavy metal inventories the satisfactory prediction of their inventories after reactor irradiation was confirmed by the analysis of an experiment in the German NPP Obrigheim in the 1970-ties (ICE experiment, see e.g. [10, 11]). For fission products, the experimental results from this experiment are less conclusive. The measured fission product isotope ratios show partly excellent agreement, but also large discrepancies. As for the short term impact of nuclear reactor accidents some specific fission products play a dominating role, e.g. Sr^{90}, I^{129}, I^{131}, Cs^{134} and Cs^{137}, additional efforts concerning accuracy estimates for such isotopes are of interest. Here the ongoing preparation of the diploma thesis of K. Kern can give valuable information, especially with respect to the various yield prediction options under consideration, applied to these dangerous isotopes.

Chapter 3

Selected relevant previous work

In this chapter some previous relevant work with KAPROS/KANEXT tools is presented. A main activity concerning nuclear reactor fuel analysis is the utilization of the results of an experiment in the NPP Obrigheim [12]. Moreover, from recent work for depleted PWR and KNK fuel inventory estimates specific KANEXT input files could be utilized.

3.1 Validation of calculation tools for PWR

The results of the experiments in [12] have been analyzed and utilized in various subsequent investigations, e.g. in the references [7], [13], [10], [11]. For the current study the build-up of plutonium as toxic element and the fission products Sr^{90}, I^{129}, I^{131}, Cs^{134} and Cs^{137} are of special interest for short term considerations. The comparison of the plutonium to uranium ratio in the KWO-ICE experiment with the the KANEXT simulation shows good agreement in figure 6.3, indicating that the **prediction of the KANEXT tools is adequate for the plutonium production.** The fission product analysis in the KWO-ICE experiment does not give reliable relevant information for the current study.

3.2 Validation of calculation tools and models for BWR

The systematic review of preserved resources for KAPROS BWR applications reminded a project of the years 1998/99 to create special data libraries for the use of KORIGEN by external customers. For the creation of burn-up dependent libraries for BWR fuel assemblies with gadolinium burnable poison extensive validation work was done. In figure 6.4 a typical result of this validation work is reproduced for an UO2 fuel assembly. **The good agreement between the applied methods allow the application of simple KANEXT cell calculations for generic BWR investigations.**

3.3 Calculation of activities of radionuclides for LWR

In order to extent the burn-up range of discharged nuclear fuel [14], recently table 2-4 of reference [15] has been recalculated for discharge burn-up up till 60 GWD/THIM. As an example a typical result for burn-up level 50 GWD/THIM is shown in table 6.1. For this task a number of specific input files has been created. These files can be used with minor changes for BWR analysis.**After appropriate burn-up cell burn-up calculations, only minor modifications are required in these files to select any combination of isotopes at any burn-up level for arbitrary LWR fuel**.

3.4 Determination of nuclide vectors of spent KNK fuel assemblies

Another related recent activity is the determination of the nuclide inventories in irradiated fuel assemblies of the dismantled KNK-II fast prototype reactor at FZK, still present in the storage area of the closed prototype reprocessing plant Karlsruhe (WAK) [16]. The main outcome of this task are fuel isotope inventory lists of KNK fuel assemblies at specified burn-up levels, see example printout in table 6.2.

Chapter 4

Generic nuclear fuel inventory analysis for BWR

In this chapter a calculation route within the KANEXT modular code system for generic BWR fuel inventory analysis is discussed. The application of simplified deterministic multigroup cell calculations is justified and results for a few selected key scenarios are presented. The preliminary results are the basis for selection of other scenarios of interest. In chapter 5 first estimates for the Japanese Fukushima BWR 2 are analyzed in some detail.

4.1 Discussion of applied models for BWR fuel analysis

A more detailed analysis of the investigations described in chapter 3.2 shows that these calculations are performed for a modern BWR fuel assembly with ”10x10” basic structure. It has to be checked what type of BWR fuel assembly is most representative for the current objectives: e.g. ”7x7”, ”8x8”, ”9x9” or ”10x10” structures.
As a first step the lattice of a ”8x8” BWR fuel assembly from an OECD benchmark activity [17] was compared with the available ”10x10” results. The satisfactory agreement between lattices with ”8x8” structure, representing the first generation of BWR core loadings, and with ”10x10” structure, representing e.g. most existing German BWR core loadings, allowed generic investigations with the ”10x10” lattice. In view of the specific interest for the inventories of the Japanese Fukushima BWR, an additional internet search was performed to obtain relevant inventory information about these reactors. The recent document [18] contains very valuable information for this purpose. For the Fukushima BWR number 2 at the Daiichi site some detailed lattice, power and fuel cycle data is given. The fuel elements have a basic ”9x9” structure. In table 6 a summary is given of the applied parameter for the exploratory investigations, together with the data for the Fukushima BWR-2. In figure 6.5 k_∞ values for the ”8x8”, ”9x9” and ”10x10” lattices are presented. The results in this figure show that the k_∞ curves for the INR 4.2% U^{235} and the OECD 3.8% U^{235} case are close together in the realistic end of life range of 30-40 GWD/THIM burn-up. For a generic judgment of BWR fuel inventories the analysis of the inventories of these cases is also of interest. However, for analysis of the impact of the problems caused by the recent earthquake/tsunami event at the Fukushima site in Japan, the more specific ”9x9” data is more relevant. In the next sections both generic and Fukushima relevant issues will be discussed.

4.2 Determination of activities of radionuclides in a selected BWR model

The results presented in this section are based on cell calculations for the model from the former investigations at FZK/INR: a 10x10 BWR lattice with 4.2% U^{235} enrichment. In order to obtain an estimate for the uncertainties of the predicted fission products currently of interest two options for the fission product handling were applied:

- Use of the current standard file in KANEXT for burn-up fission yields: KORFI4.NDFPS.
- Use of a modified file with burn-up fission yields from an ongoing project to create new consistent fission product data. The file KORFIN.NDFPS is a copy of KORFI4.NDFPS with the fission yields replaced by the data of a JEFF3.11 evaluation (file JEFF311NFY.ASC). In a next step it is foreseen to replace all isotope information on KORFIN.NDFPS by consistent data based on JEFF3.11 evaluations.

In view of the current problems at the Japanese nuclear accident sites the following isotopes are analyzed in more detail: I^{129}, Cs^{134}, Cs^{137} and Sr^{90}. In figure 6.5 the burn-up dependent build-up is showed for these isotopes. All isotopes show increasing concentrations with increasing burn-up.
For a first set of radio-activity estimates of fission products, decay calculations were performed at a burn-up of 40 GWD/THIM of the 10X10 BWR lattice.
Radio-activity of selected isotopes was calculated at 8 time points after termination of the fission process: 1, 3, 7, 14, 21, 28, 35 and 42 days. The first 4 points are in view of the short time decay after the emergency shutdowns after the earthquake, the other points are in view of the discharged fuel in the cooling pool. In the table of figure 6.9 the radio-activity in Curie per tonne initial heavy metal (Ci/THIM) is presented for the two fission product yield libraries. It may be observed that the impact of the different yield libraries is quite moderate.

Chapter 5

Estimates for the Fukushima Daiichi BWR-2 reactor core

5.1 Applied data for Fukushima Daiichi BWR-2 reactor core investigations

On request of participants to the ad-hoc activity [19], mentioned in the introduction, absolute radio-activity estimates in Becquerel were prepared for the Fukushima Daiichi BWR-2. For this purpose a WWW document could be found with relevant valuable information [18]. The specific work for this task was:

- Conversion of standard radio-activity unit Curie in KANEXT/KORIGEN to SI unit Becquerel. Here a straight-forward multiplication factor Curie-to-Becquerel 3.7E+10 may be found in any unit-conversion resource.
- Identification of fuel element details for this reactor. The before mentioned document [18] contains sufficient information for the basic ”9x9” fuel assemblies involved.
- Reliable estimate of the core fuel inventory. Despite extensive survey at KIT Campus North and in the internet resources, up till now no detailed inventory information for this reactor could be found. For this reason, estimates based on available reactor specifications must be used. For the current estimate the following data is applied:
  - Thermal power: 2350 MW. This (rounded) value may be found in several documents for the involved class of BWR.
  - Power rating of the fuel pins. Here the value 160 W/cm was adopted from reference [20].
  - Fuel weight per cm fuel pin. Here, in addition to the lattice geometry parameter, the fuel density must be taken into account. Taking into account information about the actual density of the applied fuel and of the gap between fuel and inner pin, a value of 0.93 is applied for the ratio of the smeared fuel inside the fuel pin to the theoretical density.

On the basis of these approximations the core weight is:
thermal power divided by power rating multiplied with weight per cm fuel.
A value slightly above 100 tonnes of heavy metal is determined in this way (6).

5.2 Inventory estimates for Fukushima Daiichi BWR-2

On the basis of the assumptions of section 5.1 and using standard libraries KANEXT results of absolute values in Becquerel, are obtained and presented in the tables of figure 6.10 for fission products and figure 6.11 for heavy metals. In order to obtain an impression about the uncertainty of the results, two additional calculations were performed:

- Application of a revised KANEXT library with fission product yields, based on JEFF3.11 evaluated data (see section 4.2). In this simulation we may observe the sensitivity of the results to the fission product yields.
- Application of a 350 group library instead of the standard 69 group library. Recently, consistent 69 and 350 multi-group libraries were prepared, starting from the JEFF3.11 evaluated nuclear data library, see also [3]. In this simulation we may observe the sensitivity of the results to the determination of the one-group data for the burnup calculations.

In figure 6.7 the results of these comparisons are presented. The curves for the 69 group cases show a well observable influence by the updating of the fission yields to JEFF3.11 data. The differences for consistent 69 and 350 group calculations are quite small, with a crossing behaviour around 15 GWD/THM.

Chapter 6

Summary

In this note the applicability of the tools of the modular code system KANEXT for analysis of ad hoc problems in nuclear engineering is demonstrated in connection with the impact of the earthquake catastrophe at Japanese nuclear sites. Generic models for BWR lattice simulation could be applied from former investigations at FZK and based on OECD BWR benchmark publications. The LWR burn-up calculations with KANEXT tools are well validated experimentally for the evaluation of the heavy metals in such fuel, e.g. for the plutonium build-up by the validated ratio Pu/U.

Because the validation for the fission product build-up is not really validated, the impact of new fission product yields for KANEXT simulations is investigated. Replacement of fission product yields on the standard KANEXT library by recent JEFF3.11 evaluated data shows only moderate effects.

For important short-term radio-toxic isotopes a few time-dependent tables are presented. In case other isotopic information is of interest, the prepared input files for this note can be applied with minor modifications.

On request of participants to the ad-hoc activity [19], mentioned in the introduction, absolute radio-activity estimates in Becquerel were prepared for the Fukushima Daiichi BWR-2. In order to obtain an impression about uncertainties of the results, these calculations were performed with varying neutron group number and alternative fission yield resources. These investigations show clear, but acceptable, differences.

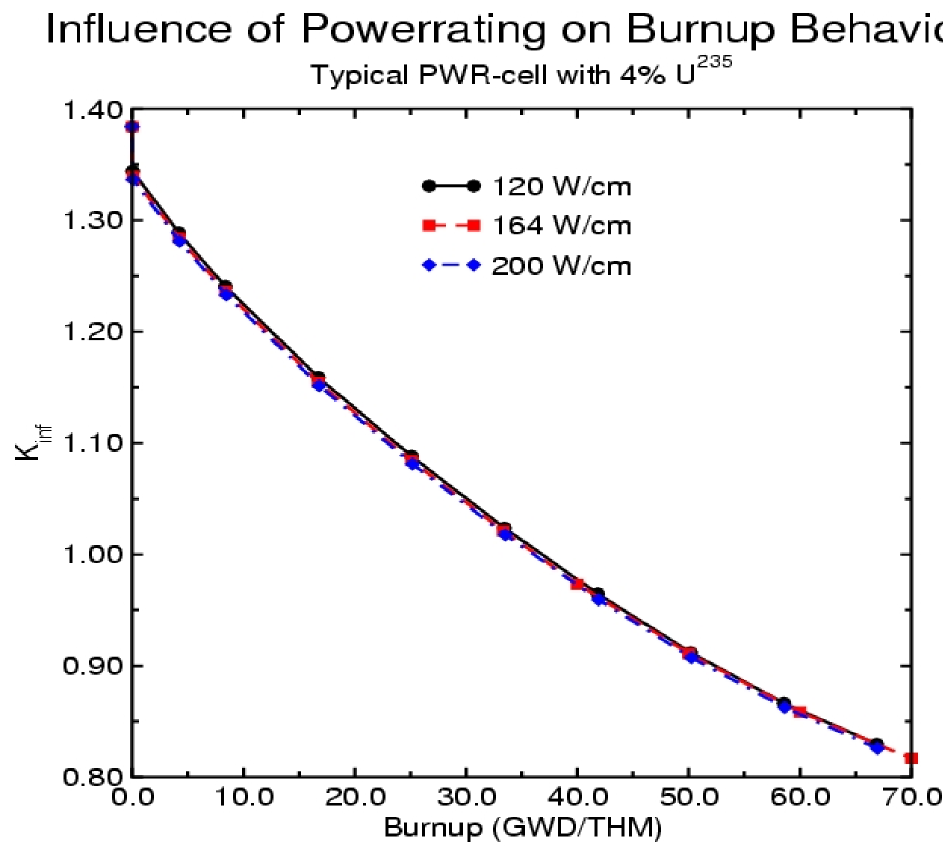

Figure 6.1: Influence of power rating on burn-up reactivity for uranium based fuel

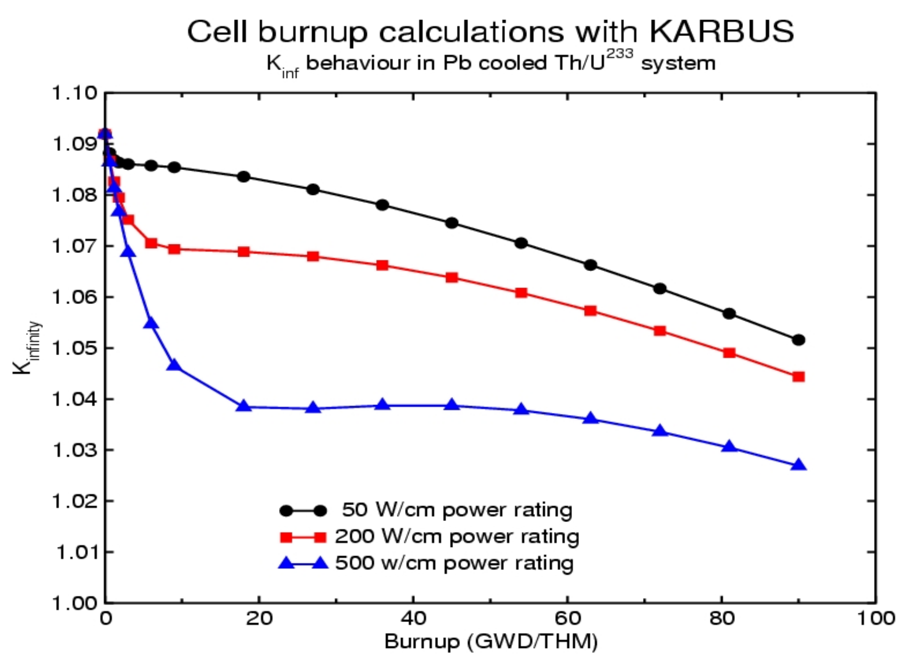

Figure 6.2: Influence of power rating on burn-up reactivity for thorium based fuel

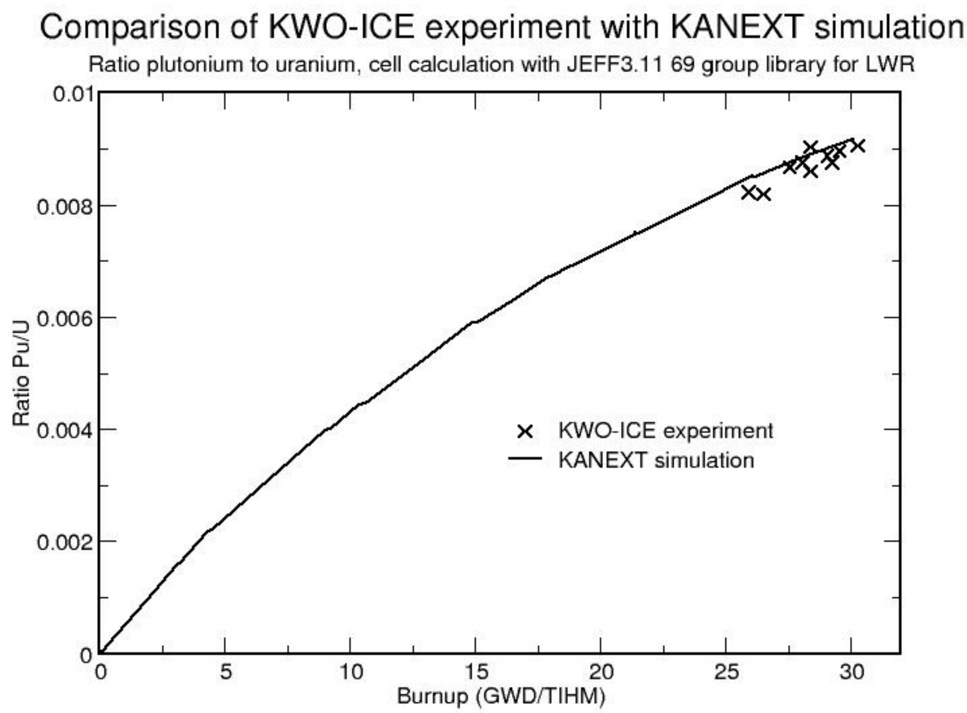

Figure 6.3: Ratio plutonium to uranium during KWO-ICE irradiation

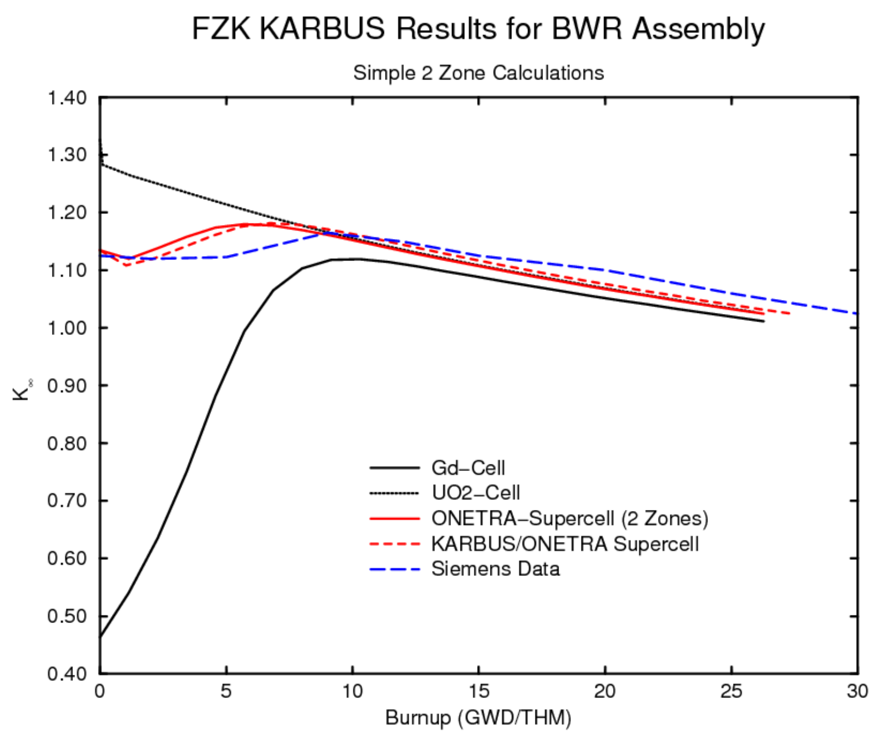

Figure 6.4: Results of validation efforts for BWR assembly in 1999

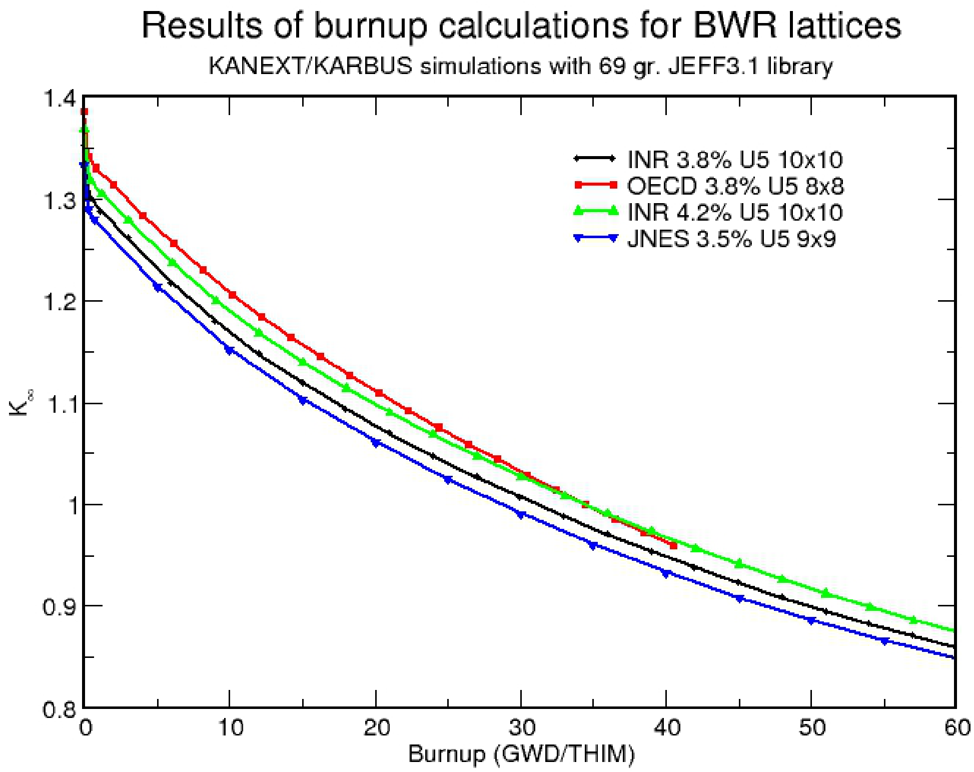

Figure 6.5: Comparison of burn-up of 8x8, 9x9 and 10x10 BWR assemblies

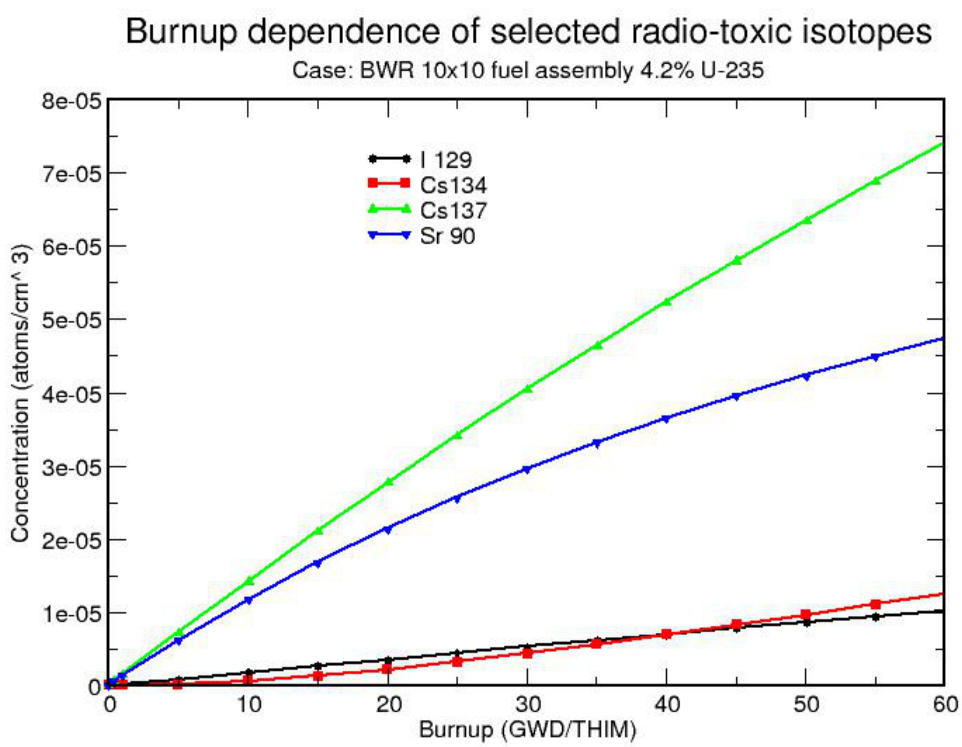

Figure 6.6: Burn-up dependency of selected radio-toxic isotopes in BWR lattices

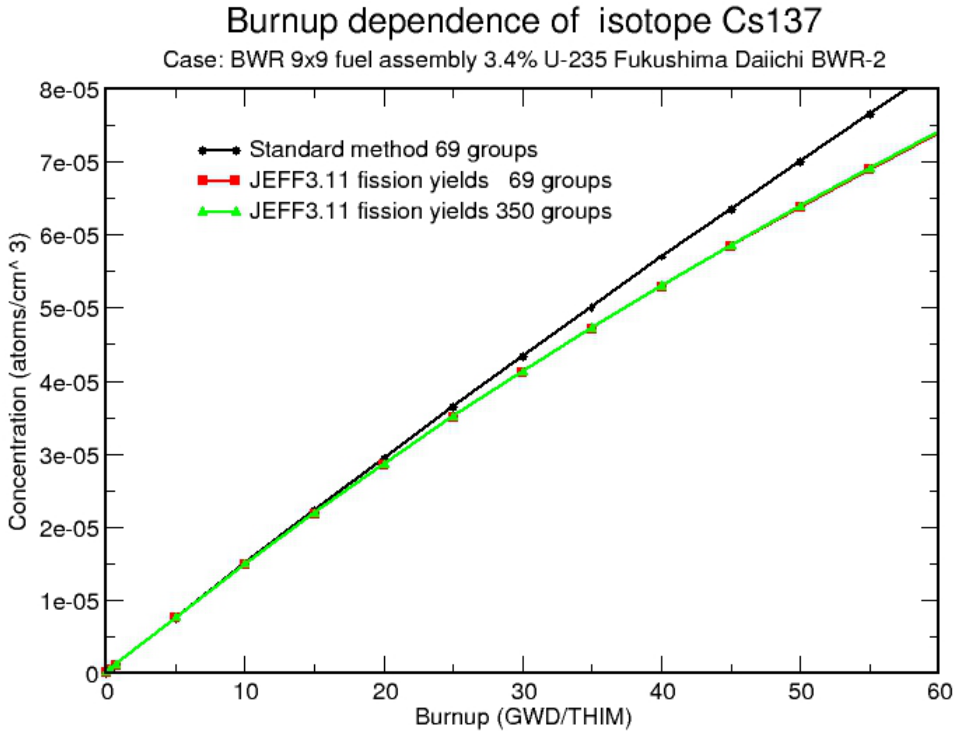

Figure 6.7: Estimated inventory of Cs-137 in Fukushima BWR-2

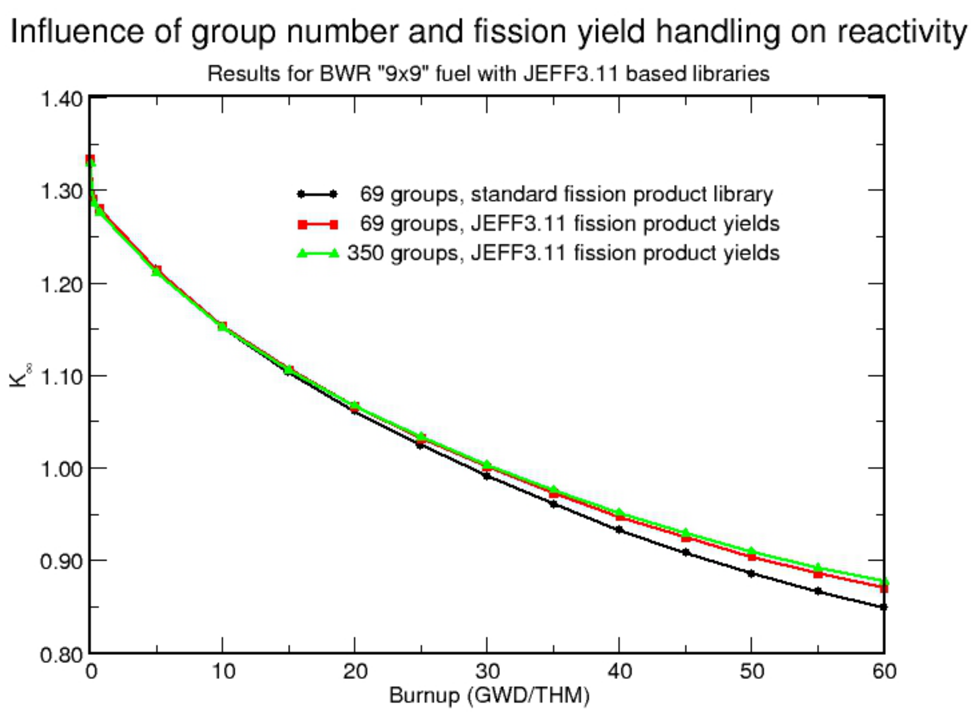

Figure 6.8: Sensitivity results for reactivity of ”9x9” BWR-lattice

Activities of radionuclides in spent PWR fuel

Results of PWR pin cell calculations for 50 GWD/THIM burn-up

Fission products Specific radioactivity (Ci/TIHM)

Isotope	decay	discharge	1 year	3 years	5 years	7 years
’H 3 ’	12.349 y	8.246E+02	7.796E+02	6.968E+02	6.228E+02	5.567E+02
’KR 85 ’	10.720 y	1.507E+04	1.413E+04	1.242E+04	1.091E+04	9.589E+03
’SR 90 ’	29.121 y	1.027E+05	1.003E+05	9.566E+04	9.121E+04	8.697E+04
’Y 90 ’	2.6667 d	1.031E+05	1.003E+05	9.568E+04	9.123E+04	8.699E+04
’ZR 95 ’	63.981 d	1.788E+06	3.422E+04	1.254E+01	4.592E-03	1.682E-06
’NB 95 ’	35.150 d	1.821E+06	7.703E+04	2.783E+01	1.019E-02	3.734E-06
’RU106 ’	1.0080 y	7.838E+05	3.941E+05	9.965E+04	2.520E+04	6.371E+03
’RH106 ’	29.900 s	8.686E+05	3.941E+05	9.965E+04	2.520E+04	6.371E+03
’CS134 ’	2.0619 y	3.096E+05	2.212E+05	1.130E+05	5.768E+04	2.945E+04
’CS137 ’	29.999 y	1.547E+05	1.512E+05	1.444E+05	1.379E+05	1.316E+05
’BA137M ’	2.5517 m	1.469E+05	1.430E+05	1.366E+05	1.304E+05	1.245E+05
’CE144 ’	284.26 d	1.406E+06	5.772E+05	9.726E+04	1.639E+04	2.761E+03
’PR144 ’	17.283 m	1.418E+06	5.772E+05	9.726E+04	1.639E+04	2.761E+03
’PM147 ’	2.6235 y	2.059E+05	1.643E+05	9.689E+04	5.713E+04	3.368E+04
’EU154 ’	8.6001 y	1.201E+04	1.108E+04	9.428E+03	8.025E+03	6.830E+03

Actinides Specific radioactivity (Ci/TIHM

Isotope	decay	discharge	1 year	3 years	5 years	7 years
’U 234 ’	.245+6 y	2.525E-02	4.187E-02	7.587E-02	1.095E-01	1.426E-01
’U 235 ’	.704+9 y	1.900E-02	1.900E-02	1.900E-02	1.900E-02	1.900E-02
’U 236 ’	.234+8 y	3.875E-01	3.875E-01	3.876E-01	3.876E-01	3.876E-01
’U 238 ’	.45+10 y	3.097E-01	3.097E-01	3.097E-01	3.097E-01	3.097E-01
’NP237 ’	.214+7 y	5.066E-01	5.175E-01	5.180E-01	5.188E-01	5.199E-01
’NP239 ’	2.3553 d	2.468E+07	4.064E+01	4.063E+01	4.062E+01	4.062E+01
’PU238 ’	87.744 y	5.617E+03	5.978E+03	5.976E+03	5.886E+03	5.794E+03
’PU239 ’	24064. y	3.791E+02	3.858E+02	3.858E+02	3.858E+02	3.857E+02
’PU240 ’	6537.3 y	6.067E+02	6.074E+02	6.086E+02	6.097E+02	6.108E+02
’PU241 ’	14.399 y	1.871E+05	1.783E+05	1.619E+05	1.471E+05	1.336E+05
’PU242 ’	.387+6 y	3.254E+00	3.254E+00	3.254E+00	3.254E+00	3.254E+00
’AM241 ’	432.23 y	1.647E+02	4.570E+02	9.997E+02	1.491E+03	1.935E+03
’AM243 ’	7380.2 y	4.059E+01	4.064E+01	4.063E+01	4.062E+01	4.062E+01
’CM242 ’	163.19 d	8.949E+04	1.908E+04	8.581E+02	3.864E+01	1.793E+00
’CM244 ’	18.110 y	6.879E+03	6.638E+03	6.149E+03	5.696E+03	5.276E+03

Table 6.1: Activities of radionuclides in spent PWR fuel (example)

FUEL ISOTOPE VECTOR FOR KNK FUEL ELEMENTS STORED AT WAK
BURNUP CALCULATION FOR UNIT-CELL WITH KANEXT/KARBUS CODE
EVALUATION WITH KANEXT MODULE WAKEVA-N

TIME REFERENCE: END OF IRRADIATION IN KNK

BE NAME	NY308	NY305	NY205-IA	NW302	NW306	NU302
BE INFO	MOX TEST	MOX TEST	MOX TEST	UO2 TROM	UO2 TROM	UO2 TRMM
K_I BOL	2.07341	2.07341	2.07341	1.87644	1.87644	1.51965
BU-WAK	38.424	42.678	35.057	26.881	27.411	31.187
BU-CALC	38.400	42.685	35.057	26.914	27.428	31.200
H 1	8.1649E-06	9.0922E-06	7.4475E-06	5.8159E-06	5.9268E-06	5.2822E-03
H 2	4.7004E-11	5.9962E-11	3.8173E-11	2.2699E-11	2.3662E-11	2.7276E-07
H 3	4.4741E-15	6.2413E-15	3.3623E-15	1.2710E-15	1.3474E-15	5.9346E-14
HE 3	6.9040E-17	1.0534E-16	4.8009E-17	1.4316E-17	1.5436E-17	7.1044E-16
H 4	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20
HE 4	4.1609E-06	4.6322E-06	3.7962E-06	3.0560E-06	3.1141E-06	2.8469E-06
HE 6	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20
LI 6	1.5301E-12	1.8869E-12	1.2784E-12	6.6160E-13	6.8665E-13	7.9958E-13
LI 7	5.5988E-14	7.8710E-14	4.1789E-14	1.6227E-14	1.7221E-14	5.3108E-14
LI 8	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20
BE 8	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20
BE 9	2.1249E-09	2.3638E-09	1.9398E-09	1.5072E-09	1.5357E-09	1.7522E-09
BE 10	7.2428E-11	9.0973E-11	5.9601E-11	3.5823E-11	3.7267E-11	3.4613E-11
B 10	1.1375E-17	1.5601E-17	8.6679E-18	4.1173E-18	4.3562E-18	3.5543E-18
BE 11	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20
B 11	3.4155E-13	4.8067E-13	2.5499E-13	1.0688E-13	1.1340E-13	1.0487E-13
B 12	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20
C 12	6.7692E-05	6.7692E-05	6.7693E-05	7.8976E-05	7.8976E-05	7.8976E-05
C 13	3.5294E-06	3.9271E-06	3.2214E-06	2.6523E-06	2.7026E-06	2.2973E-06
N 13	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20
C 14	1.3777E-08	1.7161E-08	1.1416E-08	6.5394E-09	6.7955E-09	5.7846E-09
N 14	6.7022E-13	9.2576E-13	5.0782E-13	2.2451E-13	2.3776E-13	2.2654E-13
N 15	1.3613E-20	1.4067E-20	1.3269E-20	1.2600E-20	1.2652E-20	1.2270E-20
N 16	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20
O 16	1.2291E-02	1.2290E-02	1.2292E-02	1.5257E-02	1.5257E-02	1.1787E-02
O 17	3.5148E-06	3.9076E-06	3.2100E-06	2.6481E-06	2.6981E-06	2.2938E-06
O 18	1.2784E-08	1.5914E-08	1.0600E-08	6.0463E-09	6.2825E-09	5.3105E-09
O 19	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20
F 19	7.2616E-12	8.9435E-12	6.0721E-12	2.2556E-12	2.3415E-12	9.2566E-13
F 20	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20
NE 20	2.0164E-07	2.2438E-07	1.8403E-07	9.3743E-08	9.5520E-08	1.0919E-07
NE 21	1.1331E-10	1.4106E-10	9.4023E-11	3.5206E-11	3.6567E-11	1.3741E-10
NE 22	1.8466E-10	2.2516E-10	1.5550E-10	6.3544E-11	6.5888E-11	2.6728E-10
NA 22	7.9100E-10	8.5439E-10	7.3892E-10	4.0172E-10	4.0788E-10	5.6575E-11
NE 23	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20	1.0000E-20
NA 23	1.1466E-02	1.1466E-02	1.1466E-02	8.7765E-03	8.7765E-03	8.7746E-03

Table 6.2: Top of 1400 lines output file for WAK-KNK-II spent fuel analysis

Results based on standard library KORFI4

Fission products Specific radioactivity (Ci/TIHM)

Isotope	decay	discharge	1. days	3. days	7. days	14. days	21. days	28. days	35. days	42. days
'H 3 '	12.349 y	6.893E+02	6.892E+02	6.891E+02	6.889E+02	6.886E+02	6.886E+02	6.886E+02	6.886E+02	6.886E+02
'KR 85 '	10.720 y	1.421E+04	1.421E+04	1.421E+04	1.420E+04	1.420E+04	1.420E+04	1.420E+04	1.420E+04	1.420E+04
'SR 90 '	29.121 y	1.039E+05	1.039E+05	1.039E+05	1.038E+05	1.038E+05	1.038E+05	1.038E+05	1.038E+05	1.038E+05
'Y 90 '	2.6667 d	1.077E+05	1.068E+05	1.062E+05	1.052E+05	1.045E+05	1.045E+05	1.045E+05	1.045E+05	1.045E+05
'ZR 95 '	63.981 d	1.627E+06	1.610E+06	1.593E+06	1.559E+06	1.509E+06	1.509E+06	1.509E+06	1.509E+06	1.509E+06
'NB 95 '	35.150 d	1.637E+06	1.637E+06	1.636E+06	1.634E+06	1.628E+06	1.628E+06	1.628E+06	1.628E+06	1.628E+06
'RU106 '	1.0080 y	7.877E+05	7.862E+05	7.847E+05	7.818E+05	7.774E+05	7.774E+05	7.774E+05	7.774E+05	7.774E+05
'RH106 '	29.900 s	8.440E+05	7.862E+05	7.847E+05	7.818E+05	7.774E+05	7.774E+05	7.774E+05	7.774E+05	7.774E+05
'I 129 '	.157+8 y	4.256E-02	4.257E-02	4.258E-02	4.259E-02	4.260E-02	4.260E-02	4.260E-02	4.260E-02	4.260E-02
'I 131 '	8.0405 d	8.299E+05	7.728E+05	7.141E+05	6.054E+05	4.691E+05	4.691E+05	4.691E+05	4.691E+05	4.691E+05
'CS134 '	2.0619 y	3.445E+05	3.442E+05	3.439E+05	3.432E+05	3.423E+05	3.423E+05	3.423E+05	3.423E+05	3.423E+05
'CS137 '	29.999 y	1.587E+05	1.587E+05	1.587E+05	1.586E+05	1.586E+05	1.586E+05	1.586E+05	1.586E+05	1.586E+05
'BA137M '	2.5517 m	1.506E+05	1.501E+05	1.501E+05	1.501E+05	1.500E+05	1.500E+05	1.500E+05	1.500E+05	1.500E+05
'CE144 '	284.26 d	1.336E+06	1.333E+06	1.330E+06	1.323E+06	1.314E+06	1.314E+06	1.314E+06	1.314E+06	1.314E+06
'PR144 '	17.283 m	1.346E+06	1.333E+06	1.330E+06	1.323E+06	1.314E+06	1.314E+06	1.314E+06	1.314E+06	1.314E+06
'PM147 '	2.6235 y	1.157E+05	1.161E+05	1.164E+05	1.170E+05	1.178E+05	1.178E+05	1.178E+05	1.178E+05	1.178E+05
'EU154 '	8.6001 y	1.216E+04	1.216E+04	1.216E+04	1.215E+04	1.215E+04	1.215E+04	1.215E+04	1.215E+04	1.215E+04

Results based on standard library KORFI4 with updated JEFF3.11 yields

Fission products Specific radioactivity (Ci/TIHM)

Isotope	decay	discharge	1. days	3. days	7. days	14. days	21. days	28. days	35. days	42. days
'H 3 '	12.349 y	6.265E+02	6.264E+02	6.263E+02	6.261E+02	6.258E+02	6.258E+02	6.258E+02	6.258E+02	6.258E+02
'KR 85 '	10.720 y	1.375E+04	1.375E+04	1.375E+04	1.374E+04	1.374E+04	1.374E+04	1.374E+04	1.374E+04	1.374E+04
'SR 90 '	29.121 y	1.009E+05	1.009E+05	1.009E+05	1.009E+05	1.008E+05	1.008E+05	1.008E+05	1.008E+05	1.008E+05
'Y 90 '	2.6667 d	1.046E+05	1.038E+05	1.031E+05	1.022E+05	1.015E+05	1.015E+05	1.015E+05	1.015E+05	1.015E+05
'ZR 95 '	63.981 d	1.437E+06	1.422E+06	1.407E+06	1.376E+06	1.332E+06	1.332E+06	1.332E+06	1.332E+06	1.332E+06
'NB 95 '	35.150 d	1.455E+06	1.455E+06	1.454E+06	1.452E+06	1.446E+06	1.446E+06	1.446E+06	1.446E+06	1.446E+06
'RU106 '	1.0080 y	5.986E+05	5.974E+05	5.963E+05	5.941E+05	5.907E+05	5.907E+05	5.907E+05	5.907E+05	5.907E+05
'RH106 '	29.900 s	6.428E+05	5.974E+05	5.963E+05	5.941E+05	5.907E+05	5.907E+05	5.907E+05	5.907E+05	5.907E+05
'I 129 '	.157+8 y	3.802E-02	3.804E-02	3.804E-02	3.805E-02	3.806E-02	3.806E-02	3.806E-02	3.806E-02	3.806E-02
'I 131 '	8.0405 d	7.192E+05	6.701E+05	6.194E+05	5.253E+05	4.071E+05	4.071E+05	4.071E+05	4.071E+05	4.071E+05
'CS134 '	2.0619 y	3.249E+05	3.246E+05	3.243E+05	3.237E+05	3.228E+05	3.228E+05	3.228E+05	3.228E+05	3.228E+05
'CS137 '	29.999 y	1.469E+05	1.469E+05	1.469E+05	1.468E+05	1.468E+05	1.468E+05	1.468E+05	1.468E+05	1.468E+05
'BA137M '	2.5517 m	1.395E+05	1.389E+05	1.389E+05	1.389E+05	1.389E+05	1.389E+05	1.389E+05	1.389E+05	1.389E+05
'CE144 '	284.26 d	1.196E+06	1.193E+06	1.190E+06	1.184E+06	1.176E+06	1.176E+06	1.176E+06	1.176E+06	1.176E+06
'PR144 '	17.283 m	1.204E+06	1.193E+06	1.190E+06	1.184E+06	1.176E+06	1.176E+06	1.176E+06	1.176E+06	1.176E+06
'PM147 '	2.6235 y	1.004E+05	1.007E+05	1.010E+05	1.015E+05	1.021E+05	1.021E+05	1.021E+05	1.021E+05	1.021E+05
'EU154 '	8.6001 y	1.085E+04	1.085E+04	1.085E+04	1.084E+04	1.084E+04	1.084E+04	1.084E+04	1.084E+04	1.084E+04

Figure 6.9: Normalized time-dependent radio-activities for selected isotopes

Parameter	lattice-type		
Fuel assembly type	”8x8”	”9x9”	”10x10”
Reference	OECD	JNES	FZK/INR
Lattice pitch (mm)	16.3	14.4	12.95
Fuel pin radius (mm)	5.29	4.89	4.42
Can thickness (mm)	0.86	0.71	0.605
Can material	Zr	Zr	Zr
Water density (g/cm^3)	0.74	0.74	0.74
Ratio to theoretical fuel density	0.886	0.930	0.886
Fuel temperature (K)	1183	1183	1183
Can temperature (K)	879	879	879
Coolant temperature (K)	879	879	879
Fuel	UO_2	UO_2	UO_2
Fuel enrichment (%)	3.8	3.5	4.2
Discharge (EOL) burn-up (GWD/THIM)		≈60	
Mean core burn-up at EOC (GWD/THIM)		≈36	
Number of fuel batches		5	
System thermal power (MW)		2450	
Power rating (W/cm)	200	160	200
Fuel pin weight (g/cm)	7.77	6.75	5.25
Core heavy metal weight (tonnes)		103.4	

Table 6.3: Basic specifications of investigated BWR fuel assemblies.

```
inventory estimate for Fukushima BWR plant 2
thermal power          2450.00 megawatt
power rating            160.00 watt/cm
fuel weight pro cm        6.75 g/cm
fuel weight in core     103.36 tonnes

mean burnup estimate:
mean burnup of discharged fuel assemblies    60.00 gwd/thim
number of fuel batches in core                5
mean burnup of discharged fuel from core     36.00 gwd/thim

Results for BWR 9x9 fuel assembly at  35. GWD/THIM burnup in simulation
```

Fission products — Estimated radioactivity (Bequerel)

Isotope		decay		discharge	1. days	3. days	7. days	14. days	21. days	28. days	35. days	42. days
'H 3	'	12.349	y	2.298E+15	2.298E+15	2.298E+15	2.297E+15	2.296E+15	2.296E+15	2.296E+15	2.296E+15	2.296E+15
'KR 85	'	10.720	y	4.494E+16	4.494E+16	4.490E+16	4.490E+16	4.486E+16	4.486E+16	4.486E+16	4.486E+16	4.486E+16
'SR 90	'	29.121	y	3.370E+17	3.370E+17	3.370E+17	3.369E+17	3.368E+17	3.368E+17	3.368E+17	3.368E+17	3.368E+17
'Y 90	'	2.6667	d	3.460E+17	3.440E+17	3.424E+17	3.401E+17	3.384E+17	3.384E+17	3.384E+17	3.384E+17	3.384E+17
'ZR 95	'	63.981	d	3.802E+18	3.761E+18	3.721E+18	3.641E+18	3.524E+18	3.524E+18	3.524E+18	3.524E+18	3.524E+18
'NB 95	'	35.150	d	3.820E+18	3.820E+18	3.818E+18	3.813E+18	3.799E+18	3.799E+18	3.799E+18	3.799E+18	3.799E+18
'RU106	'	1.0080	y	2.201E+18	2.197E+18	2.192E+18	2.184E+18	2.172E+18	2.172E+18	2.172E+18	2.172E+18	2.172E+18
'RH106	'	29.900	s	2.303E+18	2.197E+18	2.192E+18	2.184E+18	2.172E+18	2.172E+18	2.172E+18	2.172E+18	2.172E+18
'I 129	'	.157+8	y	1.487E+11	1.487E+11	1.487E+11	1.487E+11	1.487E+11	1.487E+11	1.487E+11	1.487E+11	1.487E+11
'I 131	'	8.0405	d	2.002E+18	1.864E+18	1.723E+18	1.461E+18	1.132E+18	1.132E+18	1.132E+18	1.132E+18	1.132E+18
'CS134	'	2.0619	y	1.020E+18	1.019E+18	1.018E+18	1.016E+18	1.014E+18	1.014E+18	1.014E+18	1.014E+18	1.014E+18
'CS137	'	29.999	y	5.358E+17	5.358E+17	5.358E+17	5.354E+17	5.354E+17	5.354E+17	5.354E+17	5.354E+17	5.354E+17
'BA137M	'	2.5517	m	5.079E+17	5.067E+17	5.067E+17	5.067E+17	5.067E+17	5.067E+17	5.067E+17	5.067E+17	5.067E+17
'CE144	'	284.26	d	3.188E+18	3.180E+18	3.173E+18	3.157E+18	3.134E+18	3.134E+18	3.134E+18	3.134E+18	3.134E+18
'PR144	'	17.283	m	3.204E+18	3.180E+18	3.173E+18	3.157E+18	3.134E+18	3.134E+18	3.134E+18	3.134E+18	3.134E+18
'PM147	'	2.6235	y	3.806E+17	3.815E+17	3.823E+17	3.836E+17	3.851E+17	3.851E+17	3.851E+17	3.851E+17	3.851E+17
'EU154	'	8.6001	y	3.985E+16	3.985E+16	3.985E+16	3.981E+16	3.981E+16	3.981E+16	3.981E+16	3.981E+16	3.981E+16

Figure 6.10: Normalized time-dependent radio-activities for selected isotopes

```
inventory estimate for Fukushima BWR plant 2
thermal power        2450.00 megawatt
power rating          160.00 watt/cm
fuel weight pro cm      6.75 g/cm
fuel weight in core   103.36 tonnes

mean burnup estimate:
mean burnup of discharged fuel assemblies    60.00 gwd/thim
number of fuel batches in core                5
mean burnup of discharged fuel from core     36.00 gwd/thim

Results for BWR 9x9 fuel assembly at  35. GWD/THIM burnup in simulation

Actinides                Estimated radioactivity (Bequerel)
Isotope      decay    discharge   1. days   3. days   7. days  14. days  21. days  28. days  35. days  42. days
'U 234   '  .245+6 y  1.522E+11 1.524E+11 1.527E+11 1.531E+11 1.537E+11 1.537E+11 1.537E+11 1.537E+11 1.537E+11
'U 235   '  .704+9 y  5.637E+10 5.637E+10 5.637E+10 5.637E+10 5.637E+10 5.637E+10 5.637E+10 5.637E+10 5.637E+10
'U 236   '  .234+8 y  1.113E+12 1.113E+12 1.113E+12 1.113E+12 1.113E+12 1.113E+12 1.113E+12 1.113E+12 1.113E+12
'U 238   '  .45+10 y  1.194E+12 1.194E+12 1.194E+12 1.194E+12 1.194E+12 1.194E+12 1.194E+12 1.194E+12 1.194E+12
'NP237   '  .214+7 y  2.115E+12 2.118E+12 2.120E+12 2.124E+12 2.129E+12 2.129E+12 2.129E+12 2.129E+12 2.129E+12
'NP239   '  2.3553 d  6.035E+19 4.528E+19 3.373E+19 1.872E+19 7.744E+18 7.744E+18 7.744E+18 7.744E+18 7.744E+18
'PU238   '  87.744 y  2.831E+16 2.835E+16 2.838E+16 2.843E+16 2.847E+16 2.847E+16 2.847E+16 2.847E+16 2.847E+16
'PU239   '  24064. y  1.422E+15 1.426E+15 1.430E+15 1.433E+15 1.436E+15 1.436E+15 1.436E+15 1.436E+15 1.436E+15
'PU240   '  6537.3 y  2.569E+15 2.569E+15 2.569E+15 2.569E+15 2.570E+15 2.570E+15 2.570E+15 2.570E+15 2.570E+15
'PU241   '  14.399 y  7.817E+17 7.817E+17 7.817E+17 7.813E+17 7.813E+17 7.813E+17 7.813E+17 7.813E+17 7.813E+17
'PU242   '  .387+6 y  1.417E+13 1.417E+13 1.417E+13 1.417E+13 1.417E+13 1.417E+13 1.417E+13 1.417E+13 1.417E+13
'AM241   '  432.23 y  1.030E+15 1.033E+15 1.036E+15 1.043E+15 1.053E+15 1.053E+15 1.053E+15 1.053E+15 1.053E+15
'AM243   '  7380.2 y  1.968E+14 1.970E+14 1.970E+14 1.970E+14 1.970E+14 1.970E+14 1.970E+14 1.970E+14 1.970E+14
'CM242   '  163.19 d  4.142E+17 4.138E+17 4.123E+17 4.092E+17 4.038E+17 4.038E+17 4.038E+17 4.038E+17 4.038E+17
'CM244   '  18.110 y  3.943E+16 3.947E+16 3.947E+16 3.947E+16 3.947E+16 3.947E+16 3.947E+16 3.947E+16 3.947E+16
```

Figure 6.11: Normalized time-dependent radio-activities for selected isotopes

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