FINDINGS OF THE OECD/NEA STUDY ON BURNUP CREDIT

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ABSTRACT

Findings from a four year study by an international benchmarking group in the comparison of computational methods for evaluating burnup credit in criticality safety analyses are presented in this paper. Approximately 20 participants from 11 countries have provided results for most problems. Four detailed benchmark problems for Pressurized Water Reactor (PWR) fuel have been completed and are summarized in this paper. Preliminary results from current work addressing burnup credit for Boiling Water Reactor (BWR) fuel will also be discussed as well as planned activities for additional benchmarks including Mixed-Oxide (MOX) fuels, subcritical benchmarks, international databases, and other activities.

INTRODUCTION

The Organization for Economic Cooperation and Development's Nuclear Energy Agency (OECD/NEA) has sponsored a criticality safety benchmark group for more than a decade. The group has addressed criticality safety issues of storage, dissolution and transportation of nuclear materials. In 1991, the benchmark group elected to pursue a study of burnup credit criticality benchmarks.

Burnup credit is a term that applies to the reduction in reactivity of burned nuclear fuel due to the change in composition during irradiation. Conventional reactor codes and data used for in-core physics calculations can be used to evaluate the criticality state of burned light water reactor (LWR) fuel. However, these codes involve complicated models and have large computational and data requirements.

In reactor applications, these detailed analyses are required for the efficient operation of specific reactors. In away-from-reactor applications such as the design of casks (flasks) for the transportation of spent nuclear fuel, the candidate fuel for use in the cask may come from any reactor and it is desirable that the design bound as much of the existing and expected fuel inventories as safely possible. In other words, for reactor operations the objective is to most effectively use very specific fuel in a specific application and for away-from-reactor applications the objective is a general design for a wide variety of fuel.

Traditionally, established away-from-reactor codes (MCNP, KENO, WIMS, APOLLO, etc.) have been used for applications such as the design of storage and transportation (S/T) casks. In this type of analysis, the fuel is usually assumed to be at its full initial enrichment to provide a large safety margin for criticality safety analyses. The incentives for pursuing burnup credit over the current fresh fuel approach are widely recognized.[1,2] The approach can extend enrichment limitations for existing S/T containers, and may contribute to the development of higher capacity S/T systems that would result in fewer fuel shipments and therefore decreased risk to the public. There is also potential application to criticality safety in dissolvers for fuel reprocessing as well as for timely and efficient transport to and from reprocessing facilities.

However, before such an approach can be approved by licensing agencies, it would be necessary to demonstrate that the available criticality safety calculational tools are appropriate for application to burned fuel systems and that a reasonable safety margin can be established.[2,3] Towards this end, this paper describes the suite of burnup credit criticality benchmarks that was established by the OECD/NEA Burnup Credit Criticality Benchmark Group. The benchmarks have been selected to allow a comparison of results among participants using a wide variety of calculational tools and nuclear data sets. The nature of the burnup credit problem requires that the capability to calculate both spent fuel composition and reactivity be demonstrated. The benchmark problems were selected to investigate code performance over a variety of physics issues associated with burnup credit: relative performance of fission products and actinides with respect to the multiplication factor (k) for pressurized water reactors (PWRs); trends in k and isotopic composition with burnup and enrichment for PWRs; effects of axially distributed burnup in PWRs; effects for boiling water reactors (BWRs); and effects for mixed oxide (MOX) fuels. It is important to note that the focus of the working group is the comparison of the results submitted by each participant to assess the capability of commonly used code systems, not to quantify the physical phenomena investigated in the comparisons or to make recommendations for licensing action. Participants used a wide variety of codes and methods based on transport and diffusion theory, using Sn, nodal and Monte Carlo techniques. Nuclear data (both cross section and decay data) were taken from a variety of sources - multiple versions of the Evaluated Nuclear Data Files (ENDF/B), the Japan Evaluated Nuclear Data Libraries (JENDL) and the Joint Evaluated Files (JEF). Both multigroup and continuous energy cross section data were used in the study. Table I is a summary of the benchmark problems addressed by the group noting both the primary objective and current status of each.

The following section provides a brief description of each of the benchmark problems and a summary of results. Since the objective of the benchmark group has been, thus far, to assess code capabilities, the results are most often presented as the standard deviation (σ) among participants. The group has not attempted to make a safety case for licensing nor has there been an attempt to provide bounding values on the observed trends or physical phenomena (e.g., the effect of axially distributed burnup). Specific or suspected sources of discrepancies are discussed. Based on 2σ results, some areas for future study are identified.

TABLE I

Summary of Benchmark Problems Addressed by OECD/NEA Criticality Safety Benchmark Group

Benchmark	Primary Objective	Status
Phase I-A	Examine effects of 7 major actinides and 15 major fission products for an infinite array of PWR rods. Isotopic composition specified at 3.6 wt. % U-235 at 0, 30, 40 GWd/MTU and at 1 and 5 yr cooled.	Completed 13 cases Ref. 5
Phase I-B	Compare computed nuclide concentrations for depletion in a simple PWR pin-cell model, comparison to actual measurements at 3 burnups (27.34, 37.12, and 44.34 GWd/MTU). Comparisons made for 12 major actinides and 15 fission products for each burnup case.	Completed 3 cases Ref. 6
Phase II-A	Examine effect of axially distributed burnup in an array of PWR pins as a function of initial enrichment, burnup and cooling time. Effects of fission products independently examined. Isotopic compositions specified.	Completed 26 cases Ref. 7
Phase II-B	Repeat study of Phase II-A in a 3-D geometry representative of a conceptual burnup credit transportation container. Isotopic compositions specified.	Completed 5 cases
Phase III-A	Investigate the effects of moderator void distribution in addition to burnup profile, initial enrichment, burnup and cooling time sensitivities for an array of BWR pins. Isotopic compositions specified.	Preliminary results for 22 cases
Phase III-B	Compare computed nuclide concentrations for depletion in a BWR pin-cell model.	Draft specifications
Phase IV	Investigate burnup credit for MOX spent fuel	Proposed
Phase V	Investigate burnup credit in subcritical systems	Draft specifications

BENCHMARK PROBLEMS

Phase I-A: Multiplication Factors- PWR Infinite Lattice Studies (1D)

This benchmark consists of 13 cases. Each case is an eigenvalue calculation of a simple infinite lattice of PWR fuel rods. The investigated parameters were burnup, cooling time and combinations of nuclides in the fuel region. The groupings of nuclides include four subgroups: major actinides (U-234, 235, 236, and 238; Pu-239, 240 and 241); minor actinides (Pu-238 and 242; Am-241 and 243; Np-237); major fission products (Mo-95; Tc-99; Ru-101; Rh-103; Ag-109; Cs-133; Sm-147, 149, 150, 151 and 152; Nd-143 and 145; Eu-153; and Gd-155) and minor fission products (all others available to participant). The fuel compositions for each case by nuclide were provided as part of the problem specification[4] so that the results could be focused on the calculation of (impacts on) the multiplication factor. In total, 25 sets of results were submitted from 19 institutes in 11 countries. The detailed results are presented in Ref. 5.

Phase I-A is perhaps the most detailed of the benchmark problems in terms of types of data collected and analyzed. Participants were asked to provide the following: codes used, nuclear data libraries, and energy grouping of libraries (group structure or continuous energy); calculated multiplication factor; neutron spectrum in water; neutron spectrum in fuel; absorption rates for all major and minor actinides, major fission products and oxygen; and production rates and neutrons per fission for all major and minor actinides.

Results - Multiplication Factors: Only 17 of the 25 participants providing solutions were able to execute the problem as specified. Some participants had difficulty incorporating the specified compositions and/or did not have cross section data for all the major fission products. Approximately 23 of the participants were able to successfully complete the actinide-only cases. The results presented in Table II are the average of the results of 17 participants.

TABLE II

Results of Phase I-A: Average Multiplication Factor, k (2σ), Ref. 5

Nuclides Set	Fresh Fuel	30 GWd/MTU* 1 yr cooled	40 GWd/MTU* 1 yr cooled	30 GWd/MTU* 5 yr cooled	40 GWd/MTU* 5 yr cooled
All Actinides All Fiss Prod.	1.4378(0.0175)	1.1080(0.0194)		1.0758(0.0185)	
All Actinides No Fiss Prod.		1.2456(0.0107)	1.1885(0.0110)	1.2284(0.0109)	1.1657(0.0099)
Maj Actinides No Fiss Prod.		1.2635(0.0108)	1.2566(0.0109)		
All Actinides Maj Fiss Prod.		1.1402(0.0169)	1.0638(0.0170)	1.1123(0.0164)	1.0240(0.0156)

*Burnup is given in gigawatt days per metric ton initial uranium

An examination of the results in Table II suggest that the largest component of uncertainty originates from the minor fission products as indicated by the larger 2σ values in the cases of "All Fission Products". For all other cases, including those with "Major Fission Products", the 2σ values are smaller than for the case of fresh fuel. The agreement among participants for the "No Fission Product" cases is significantly better than the fresh fuel and fission product cases. No trends in the standard deviation among participants were observed with either burnup or cooling time. Trends in the multiplication factors with burnup and cooling time were as expected; k decreases as both burnup and cooling time increase. The larger 2σ value for the fresh fuel case was expected based on known biases which decrease with fuel depletion.[5,8]

Results - Neutron Spectra: Fourteen participants provided neutron spectra in both the fuel and water. The number of energy groups varied from 27 to 247 and the maximum energy boundaries vary from 20 MeV to 8.2 MeV. Results based on continuous energy data were converted for mutual comparison. The spectra were in quite good agreement. The effects of Pu resonances were clearly seen

at approximately 0.3 eV and 1.0 eV in the fuel region and smaller effects at these energies were observed in the moderator region.[5,9]

Results - Reaction Rates: Seventeen participants supplied the requested reaction rate data. Both the absorption rates and production rates were normalized to unity for comparison. A comparison of absorption rates revealed differences of 0.4 - 0.7% of the total absorption rate for U-238, U-235 and Pu-239. The production rates for these nuclides revealed observed differences among participants of 0.6 to 0.8% of the total production rate. Differences were also observed in the calculated values of neutrons per fission for these nuclides, however there were some discrepancies among participants in the definition of this parameter so the results are not conclusive. Smaller differences in absorption rates (less than 0.1% of the total absorption rate) were observed for Pu-240, Pu-241, Gd-155, Nd-143, Rh-103, Sm-149, Sm-151 and Tc-99.

Phase I-B: Spent Fuel Compositions, PWR Fuel

The purpose of this calculational benchmark problem was to compare computed nuclide concentrations for depletion in a simple pin-cell model.[10,11] The detailed problem description and results are given in Ref. 6. This benchmark consists of three cases, each with a different burnup. The specific power and boron concentrations for each cycle and cumulative burnup were given in the problem description. Initial isotopic compositions for both the fuel and the moderator were given. Participants were requested to report calculated compositions for the 12 actinides and 15 fission products named in Phase I-A. A total of 21 sets of results were submitted by 16 organizations from 11 countries.

Given that the objective is to ultimately calculate the reactivity of spent fuel, the significance of the differences in nuclide concentrations should be examined from this perspective. As an example of relative importance in the evaluation of multiplication factor, the change in reactivity associated with a change in concentration equal to the observed standard deviation among participants was evaluated.

Table III is a list of nuclides with a standard deviation greater than 10% or a change in reactivity greater than 0.01%Δk (10 pcm) per % change in concentration (%ΔN). A large standard deviation indicates poor agreement in the calculation of the inventory of a given nuclide. Unlike Phase I-A, trends in the standard deviation with burnup are evident in this study. For many nuclides this trend is relatively small, however the trend of increasing standard deviation with increasing burnup appears to be significant for U-235. A list of nuclides for which further study and comparison of additional information (such as fission product yield data, thermal cross sections, etc.) would be warranted is as follows: Pu-239, Gd-155, U-235, Pu-241, Pu-240, Sm-151, and Sm-149, as these have the largest integral effect on k. Of these nuclides, only Gd-155 and Sm-149 exceed both the 10% standard deviation and a Δk/%ΔN of 0.01%.

Phase II-A: Multiplication Factors-Distributed Burnup Studies (2D)

The configuration considered in this benchmark problem was a laterally infinite array of PWR fuel assemblies with the following characteristics: initial enrichment equal to 3.6 wt % or 4.5 wt %; fuel radius equal to 0.412 cm and array pitch equal to 1.33 cm which leads to a moderation ratio V_{mod}/V_{ox} = 2.0; different burnups were considered (0, 10, 30 or 50 GWd/MTU) and two cooling times, 1 or 5 years; axially, a symmetrical configuration was adopted including 9 fuel regions (total height = 365.7 cm); and an upper and lower plug and water reflector (30 cm). Specific isotopic compositions were specified for each fuel region and conditions. Cases were analyzed for the axially distributed burnup as well as a uniform burnup assumption equal to the assembly average burnup. The axial burnup profiles used were symmetric about the midplane. As in Phase I-A, the effect of major actinides

and fission products were also investigated. Participants were asked to provide calculated multiplication factors and fission densities by axial zone for three cases. In total, 22 results for the 26 configurations were calculated by 18 different participants from 10 countries.

Details of the problem specification and results for this benchmark are presented in Ref. 7. The average multiplication factors and 2σ values for the 26 cases are summarized in Table IV.

TABLE III

Summary of Phase I-B Results (Ref. 6)

Nuclide	Case A (27.35 GWd/MTU)[a]		Case B (37.12 GWd/MTU)		Case C (44.34 GWd/MTU)	
	σ[b]	Δk[c] (x100)	σ	Δk (x100)	σ	Δk (x100)
U-235	2.98	0.4410	6.01	0.6485	8.12	0.6285
Pu-238	15.68	0.0329	14.80	0.0562	13.86	0.0679
Pu-239	5.16	0.7085	6.08	1.0611	7.12	1.3962
Pu-240	3.95	0.2054	4.27	0.2404	5.27	0.2772
Pu-241	6.45	0.2219	5.97	0.3248	6.86	0.4583
Am-243	11.31	0.0079	10.41	0.0198	10.40	0.0302
Ag-109	11.03	0.0143	10.61	0.0191	10.21	0.0214
Sm-149	14.14	0.1386	15.01	0.1471	15.61	0.1499
Sm-150	5.30	0.0090	7.07	0.0177	8.50	0.0255
Sm-151	22.41	0.1502	21.72	0.1434	22.31	0.1539
Sm-152	7.20	0.0331	9.01	0.0469	9.68	0.0503
Gd-155	33.45	0.5252	33.28	0.8120	32.97	0.9792

[a] Burnup is given in gigawatt days per metric ton uranium.
[b] The standard deviation among participant results.
[c] Represents an example of the change in multiplication factor times 100 from a one σ change in isotopic composition. The quantity is given as a positive value since the change in composition may be +/-.

Results - Comparison of Multiplication Factors: No significant trends in the agreement among participants (2σ values) were observed with initial enrichment or burnup. As in Phase I-A, the inclusion of fission products results in a greater deviation among participants (larger 2σ values). No clear trends were observed with the inclusion of the axially distributed burnup, although cases with both high burnup (greater than 10 GWd/MTU) and with fission products have some indications of increasing 2σ when axially distributed burnup is considered. At higher burnup (50 GWd/MTU with and without fission products) there is a suggestion of a trend in 2σ with cooling time. A comparison of multiplication factors from this benchmark with corresponding cases in Phase I-A indicate that the axial leakage is small. Overall, the most interesting result in this benchmark is that the largest discrepancy (2σ) among participants is still seen for the fresh fuel cases.

Results - End Effect: The "end effect" was defined as the difference in the multiplication factors between the corresponding cases with and without an axial burnup distribution. Tendencies were observed in the multiplication factors that indicate an increase in end effect with increasing burnup. It is very important to note that the end effect is calculated as the difference of two close values and,

therefore, has large calculated standard deviations, from 25% to greater than 100% of the value calculated for the end effect (in most cases approximately 75%). Although these tendencies are believed to be representative in general, the effects of both neutron leakage and axial asymmetry of material composition (which was not considered here) may make a considerable difference in the magnitude of the end effect.

Results - Fission Density: The fission density data provided by the participants was found to be in relatively good agreement. The data illustrate the importance of the end regions, approximately 70% of the total fissions occurred in the upper 40 cm of the fuel (representing approximately 22% of the total fuel volume). Therefore, adequate modeling and convergence at the fuel ends are essential to obtain reliable eigenvalues for highly irradiated spent fuel systems.

TABLE IV

Summary of Phase II-A Results, Average Multiplication Factor (Ref. 7)

Case	Initial Enrichment	Burnup GWd/MTU	Cooling time(yr)	Fission Products	Burnup Profile	k(2σ)
1	3.6 wt %	Fresh	N/A	N/A	N/A	1.4335 (0.0217)
2	3.6 wt %	10	1	Yes	Yes	1.3053 (0.0161)
3	3.6 wt %	10	1	Yes	No	1.3126 (0.0159)
4	3.6 wt %	10	1	No	Yes	1.3607 (0.0175)
5	3.6 wt %	10	1	No	No	1.3665 (0.0174)
6	3.6 wt %	30	1	Yes	Yes	1.1360 (0.0155)
7	3.6 wt %	30	1	Yes	No	1.1358 (0.0138)
8	3.6 wt %	30	1	No	Yes	1.2339 (0.0129)
9	3.6 wt %	30	1	No	No	1.2419 (0.0119)
10	3.6 wt %	30	5	Yes	Yes	1.1160 (0.0144)
11	3.6 wt %	30	5	Yes	No	1.1062 (0.0136)
12	3.6 wt %	30	5	No	Yes	1.2176 (0.0119)
13	3.6 wt %	30	5	No	No	1.2256 (0.0113)
14	4.5 wt %	Fresh	N/A	N/A	N/A	1.4783 (0.0232)
15	4.5 wt %	30	1	Yes	Yes	1.1996 (0.0151)
16	4.5 wt %	30	1	Yes	No	1.2025 (0.0161)
17	4.5 wt %	30	1	No	Yes	1.2972 (0.0145)
18	4.5 wt %	30	1	No	No	1.3064 (0.0139)
19	4.5 wt %	50	1	Yes	Yes	1.0838 (0.0175)
20	4.5 wt %	50	1	Yes	No	1.0584 (0.0136)
21	4.5 wt %	50	1	No	Yes	1.1999 (0.0121)
22	4.5 wt %	50	1	No	No	1.1983 (0.0116)
23	4.5 wt %	50	5	Yes	Yes	1.0543 (0.0156)
24	4.5 wt %	50	5	Yes	No	1.0123 (0.0135)
25	4.5 wt %	50	5	No	Yes	1.1800 (0.0104)
26	4.5 wt %	50	5	No	No	1.1734 (0.0096)

Phase II-B: Multiplication Factors-Distributed Burnup Studies (3D)

In this benchmark problem, a realistic configuration of 21 PWR spent fuel assemblies in a stainless steel transport cask was evaluated. A borated stainless steel basket centered in the flask separates the

assemblies. The basket (5x5 array with the 4 corner positions removed) was fully flooded with water. The main characteristics of the fuel assembly are: 17x17 array (289 rods, no guide tubes), water moderated cells with pitch equal to 1.25984 cm; initial fuel enrichment equal to 4.5 wt %; fuel radius equal to 0.4096, fuel rod ID= 0.41785 cm and OD = 0.475 cm which lead to a moderation ratio V_{mod}/V_{ox}=1.67; as in Phase II-A, the fuel was divided axially into 9 symmetrical zones; burnups of 0, 30 and 50 GWd/MTU and 5 years cooling were used; and the fuel compositions were as specified Phase II-A. Cases were analyzed for the axially distributed burnup as well as a uniform burnup assumption equal to the average burnup. Fourteen participants from 7 different countries submitted partial or complete results (k-eff and fission densities) for the 9 cases specified.

This benchmark is in the final stages of study by the working group and the detailed results are expected to be published as an OECD/NEA report in late 1996. Table V is a summary of the current status of this benchmark. Note that there could be additional submissions/corrections before the final results are published and, therefore, Table V should be considered preliminary.

TABLE V

Preliminary of Phase II-B Results - Average Multiplication Factors

Case	Initial Enrichment	Burnup GWd/MTU	Cooling time(yr)	Fission Products	Burnup Profile	k (2σ)
1	4.5 wt %	Fresh	N/A	N/A	N/A	1.1256 (0.0155)
2	4.5 wt %	30	5	Yes	No	0.8934 (0.0065)
3	4.5 wt %	30	5	No	No	0.9714 (0.0099)
4	4.5 wt %	30	5	Yes	Yes	0.8949 (0.0087)
5	4.5 wt %	30	5	No	Yes	0.9640 (0.0106)
6	4.5 wt %	50	5	Yes	No	0.7641 (0.0042)
7	4.5 wt %	50	5	No	No	0.8735 (0.0065)
8	4.5 wt %	50	5	Yes	Yes	0.7929 (0.0058)
9	4.5 wt %	50	5	No	Yes	0.8781 (0.0077)

Significant differences in the multiplication factors observed for this benchmark relative to Phase II-A are due to differences in the configuration (radially finite, borated stainless basket and stainless steel reflector) and differences in the moderation ratio. There are also significant differences in the calculated standard deviations, which are systematically lower than the corresponding Phase II-A cases. In this benchmark, the trend previously observed indicating increasing dispersion among participant results (higher values of 2σ) for cases including fission products is reversed. In Phase II-B the results with fission products have *smaller* 2σ values than those cases with no fission products. Consistent with earlier results the highest value of 2σ is for the fresh fuel case. Overall, the agreement among participants is better for Phase II-B than in the Phase II-A benchmark.

Phase III: Proposed BWR Studies

Phase III-A: This benchmark problem was developed to evaluate the criticality safety of spent boiling water reactor (BWR) fuel in storage facilities or transportation casks. The main features of BWRs important in criticality analyses that differ substantially from PWRs are the moderator void distribution in the core and the complicated composition of a fuel assembly. In BWRs, the moderator void volume fraction is about 70% near the top region of the core and nearly zero near the bottom of the core. The core average void fraction is approximately 40%. A BWR fuel assembly consists of many kinds of fuel rods whose initial enrichments are different from each other. Some fuel rods contain Gd, which is a strong neutron absorber. BWR assemblies also have a large water rod located at their center. For this benchmark problem, the assembly geometry was simplified such that the composition of all the fuel rods in an assembly is considered to be the same. The water rod, cladding, channel box, end plugs and gas plenum are all modeled per the specification. Isotopic compositions for the fuel and water are also given. Twenty-two cases were proposed where burnup varies from 0 to 40 GWd/MTU, fission products are included in some cases, an axial burnup distribution is considered in some cases, an axial void distribution is used in some cases, 40% and 70% uniform void cases are considered and cooling times of 1 and 5 years are specified. Participants were asked to provide calculated multiplication factors and fractional fission densities for five cases.

Preliminary results indicate that the largest differences among participants are for the 70% uniform void cases (other than the fresh fuel case). In these cases, the neutron energy spectrum is harder and the Pu production rate is high compared to the 40% cases and the cases with an axially distributed void fraction. The detailed results are not presented in this paper because of their preliminary nature (not yet reviewed by the working group). It is expected that the detailed results for this benchmark problem will be published with an OECD/NEA designation early in 1997.

Phase III-B: This benchmark was developed to investigate the ability of evaluation tools to calculate the isotopic composition of irradiated BWR fuel. Unlike the problem specification for Phase III-A, the geometry of the BWR fuel assembly was not simplified for this benchmark. The fuel assembly consists of fuel rods at 5 different initial enrichments and with and without Gd. The initial isotopic composition of each rod and explicit geometry descriptions were specified. As in the Phase III-A specification, the void fraction is varied, cases are evaluated at 0, 40 and 70% uniform void fractions. Number densities for the 12 actinides and 15 fission products of Phase I-A are requested for each of 9 fuel pins in a 1/8 assembly model. The average composition of each of the 5 fuel rod types and assembly average compositions are requested. The calculated burnup for each of the 9 fuel pins is also requested. Participants are also asked to provide neutron multiplication factors for burnups of: 0, 0.2, 10, peak burnup, 20, 30, 40, 50 GWd/MTU for each of the three void fraction cases.

The working group is expected to begin evaluating this benchmark problem in late 1996. Normally, the evaluation period for a benchmark problem is 18 to 24 months from acceptance of a specification to publishing results.

ADDITIONAL STUDIES

Spent Fuel Isotopic Composition Database

Reference 12 discusses a database of LWR spent fuel assay data that has been compiled. This database system, SFCOMPO,[13] contains data collected from 13 LWRs, including 7 PWRs and 6 BWRs in Europe, the USA and Japan. Over the past year, axial burnup profiles from 2 Japanese reactors have been added to the database. The database will be maintained by adding new data as they become

available, revising old data as necessary, and providing recommendations for criticality evaluations. This database is unique and provides a valuable resource for the evaluation of burnup credit.

Criticality Benchmark Experiments

There have been several activities that involve experiments that are applicable to burnup credit.

Exponential Experiments in the Tank Typed Critical Assembly (TCA) of JAERI:[14] Reference 15 describes an experimental technique which has been applied to 2 PWR spent fuel assemblies stored in a pool after post irradiation examination. The technique measures the exponential decay factor in the axial direction, which is one of the eigenvalues representing the degree of subcriticality. The measured results are in good agreement with calculations based on a 4-group neutron diffusion model. The effective multiplication factor of the assembly is estimated from the decay factor. The estimated multiplication factors were found to be in good agreement with MCNP analyses. Chemical assay data are also available for these assemblies.

International CERES Experimental Program: The CERES program was designed for the validation of cross section data and inventory predictions for actinides and fission products important to the study of burnup credit.[2] The principal participants were from France and Great Britain with some United States involvement in the latter stages of the program. CERES was part of an extensive experimental program developed at CEA Cadarache involving oscillation experiments in the MINERVE reactor.[16] The British experiments[17,18] were performed in the DIMPLE reactor at the United Kingdom Atomic Energy Agency (UKAEA) in Winfrith and were also supported by British Nuclear Fuels Limited (BNFL). These experiments provide data for the validation of major fission product cross sections, reactivity worths of UO_2 samples enriched with separated fission products, spent fuel worths as well as chemical assay data for spent fuel.

Fission Product Experiments by the Institut de Protection et de Sãreté Nucléaire (IPSN): CEA/IPSN has initiated an experimental program[19] to provide benchmark data for 6 fission products; Rh-103, Cs-133, Nd-143, Sm-149, Sm-152, and Gd-155. The critical experiments use a tank assembly and with the fission products in solution. Results for Sm-149 have been published[19] indicating reactivity worths of Sm in the assembly at 2000-6000 pcm. The benchmarks results indicate the Sm-149 cross section at energies less than 1 < eV is well qualified. The experimental program is ongoing.

Spent Fuel Safety Experiment (SFSX): The United States has proposed a set of experiments, Spent Fuel Safety Experiment (SFSX),[20] to provide integral benchmarks for validating spent fuel reactivity. The assembly height is approximately 30 cm which should allow independent study of the "end effect." The SFSX critical assembly is a fuel replacement experiment designed to measure the critical array size for three fuel configurations, fresh fuel, spent fuel center region, spent fuel and regions. The spent fuel to be used in the experiment is from a US PWR (CE14x14) that has been carefully analyzed including measured fuel composition and burnup. This experiment is in the proposal stage.

Effect of Nuclide Radial Distribution in LWR Pins

The effect of the nuclide radial distribution inside a LWR spent fuel rod was investigated in an extreme burnup case, 63 GWd/MTU. The calculated results were compared with experimental nuclide profiles obtained from a 17x17 PWR assembly irradiated for 5 cycles. Both fission products and actinides were investigated. The reactivity calculations of cooled LWR assemblies indicated negligible reactivity worth due to radial nuclide distributions, about 30 pcm. These results permit the

recommendation of burnup credit calculation with averaged nuclide concentrations and one mesh point in the fuel rod.[21]

FUTURE WORK

The benchmark group is continuing to pursue studies with BWR fuel and to initiate studies with MOX fuel (proposed Phase IV benchmark). Other proposals being evaluated by the working group include pursuing an international criticality handbook, a benchmark calculation for basic minimum critical values, a subcritical benchmark problem (proposed Phase V benchmark) and an international database for axial burnup profiles.

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