International Workshop on Advances in Applications of Burnup Credit for Spent Fuel Storage, Transport, Reprocessing, and Disposition

Hotel Hesperia Córdoba
Córdoba, Spain, 27-30 October, 2009

Organized by the Nuclear Safety Council of Spain (CSN) in cooperation with the International Atomic Energy Agency (IAEA)

Co-sponsored by ENUSA Industrias Avanzadas S.A. **and** Empresa Nacional de Residuos Radiactivos S.A. **(ENRESA)**

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Workshop Registration: Monday 26th from 19:00 to 21:00
Tuesday 27th from 8:30 to 13:00

Welcome cocktail*: Monday 26th at 19:30
Hotel Hesperia

Workshop dinner*: Wednesday 28th at 21:00
Restaurante "Bandolero", c/ Torrijos, 6

* Welcome cocktail and workshop dinner are sponsored by ENUSA Industrias Avanzadas y Empresa Nacional de Residuos (ENRESA)

OBJECTIVES OF THE WORKSHOP

The most common assumption used in criticality safety analysis of spent nuclear fuel from power reactors is that the irradiated fuel has the same reactivity as the unburned fuel. This approach is usually known as the "fresh fuel assumption" and results in a significant conservatism in the calculated value of the system's reactivity. Modern calculation methods have made possible taking credit for the reactivity reduction associated with the fuel burnup process, hence reducing the analysis conservatism while maintaining an adequate criticality safety margin.

Spent fuel management is a common and costly activity for all operators of nuclear power plants, which involves different operational safety risks. An accepted possibility to achieve a reduction in fuel cycle costs while diminishing the risks associated to the different processes is to implement burnup credit in spent fuel management systems. In fact, in many countries, burnup credit is already applied to transport systems, wet and dry storage facilities, and components of reprocessing plants. For disposal of spent fuel and reprocessing of some advanced fuel designs, burnup credit is considered to be important for viable schemes.

In 1997, the IAEA initiated a task to monitor the implementation of burnup credit in spent fuel management systems, to provide a forum to exchange information, to discuss the matter and to gather and disseminate information on the status of national practices of burnup credit (BUC) implementation in the Member States. The IAEA started this active programme with an advisory meeting in 1997 (resulting in TECDOC-1013, 1998), followed by major meetings on BUC held in Vienna in 2000 (TECDOC-1241, 2001), Madrid in 2002 (TECDOC-1378, 2003), and London in 2005 (TECDOC-1547, 2007). Moreover, the Agency has contributed to the organization of BUC training courses held in different countries.

The Nuclear Energy Agency of the Organization for Economic Cooperation and Development (OCDE/NEA) has also performed an extensive collection of activities since it created the Expert Group on Burnup Credit in 1991. This group was set up to examine BUC as applied to criticality safety in the transportation, storage, and treatment of spent fuel for a wide range of fuel types, including UOX and MOX fuels for PWR, BWR, and VVER. The major assignments of the expert group include carrying out international comparison exercises and benchmarks to assess the ability of code systems to predict the reactivity of spent nuclear fuel systems, including comparison with experimental data as available; investigating the physics and predictability of burn-up credit based on the specification and comparison of calculational benchmark problems; and publishing the results for the benefit of the criticality safety community, so that the work may be used to help establish suitable safety margins.

The CSN has since the beginning supported both the IAEA and NEA in their cooperative efforts to improve the knowledge of the physics underlying in BUC criticality safety analysis and to maintain adequate safety margins in its implementation. Given the time elapsed since the last major BUC meeting was held, the CSN, in cooperation with IAEA, is organizing an International Workshop on Advances in Applications of Burnup Credit for Spent Fuel Storage, Transport, Reprocessing and Disposition.

The annual meeting of the Expert Group on Burnup Credit of the OCDE/NEA will be held back to back with the international workshop.

The objective of the workshop is to identify the benefits that accrue from recent improvements of the burnup credit analysis methodologies; to discuss and analyze the implications of applying improved burnup credit methodologies, focusing on both the safety-related and operational aspects; and to foster the exchange of international experience in licensing and implementation of burnup credit applications.

BUC WORKSHOP AGENDA

Tuesday October 27th

08:30-09:00

OPENING OF THE WORKSHOP
Welcome address José M. Conde (CSN)
Description of the Technical Agenda Jens-Christian Neuber (AREVA)
General Announcements and Practicalities Consuelo Alejano (CSN)

SESSION I: INTERNATIONAL AND NATIONAL REPORTS **09:00-15:30**
Co-Chairs: J.-C. Neuber (AREVA), J. M. Conde (CSN)

1.1	"BUC activities performed by IAEA" Zvonko Lovasic (IAEA)	09:00-09:30
1.2	"Lessons Learned from International Investigations of Burnup Credit Criticality" M. C. Brady-Raap (PNNL), J. Gulliford (NNL), L. Markova (NRI), H. Okuno (JAEA), G. O'Connor (UK DfT), P. H. Liem (NAIS)	09:30-10:00
1.3	"Activities of the OECD/NEA Expert Group on Assay Data for Spent Nuclear Fuel" Ian C. Gauld (ORNL), Y. Rugama (NEA)	10:00-10:30

Coffee break 10:30-11:00

1.4	"Burnup Credit Development and Implementation in the Slovak Republic" Juraj Vaclav (NRA SR)	11:00-11:30
1.5	"Applying Burn-up Credit Technology for Spent Fuel Storage in China" Guoshun You, Qingfu Zhu, Chuanwen Hu (CIAE)	11:30-12:00
1.6	"National program for BUC application in India" A. K. Pandey,(NPCIL)	12:00-12:30
1.7	"Research Activities of JAEA Considering the Future Needs of Japan in Burnup Credit Implementation" H. Okuno, K. Tonoike, K. Suyama, K. Ohkubo, G. Uchiyama (JAEA)	12:30-13:00

Lunch break 13:00-14:30

1.8	"Review of Technical Studies in the United States in Support of Burnup Credit Regulatory Guidance" J. C. Wagner, C. V. Parks, D. E. Mueller, I. C. Gauld (ORNL)	14:30-15:00
1.9	"Latest Studies Related to the Use of Burnup Credit in France" L. Jutier, I. Ortíz de Echevarría and S. Evo (IRSN), E. Guillou and J. Jaunet (AREVA), A. Bonne (EDF)	15:00-15:30

SESSION II: CALCULATION METHODS
II.A Codes and methods **15:30-17:45**
Co-Chairs: S. Evo (IRSN), John C. Wagner (ORNL)

	Chair overview of the status and the main challenges in burnup credit depletion calculation	15:30-15:40
2.1	"Enhancements to the Burnup Credit Criticality Safety Analysis Sequence in SCALE" G. Radulescu, Ian C. Gauld (ORNL)	15:40-16:05

Coffee break 16:05-16:30

2.2	"Assessment of the MCNP+ACAB Code System for Burnup Credit Analysis" N. García-Herranz, O. Cabellos, J. Sanz (UPM)	16:30-16:55
2.3	"Convergence issues in Best-estimate Monte Carlo Depletion Calculations" Y. Richet, W. Haeck, B. Cochet and J. Miss (IRSN), O. Jacquet (Consultant)	16:55-17:20
2.4	"Peak Reactivity Characterization and Isotopic Inventory Calculations for BWR Criticality Applications" C. Casado, J. Sabater, J. F. Serrano (ENUSA)	17:20-17:45

Wednesday October 28th

II.B Nuclear and assay data **08:30-10:15**
Co-Chairs: A. Santamarina (CEA), R. D. McKnight (ANL)

	Chair overview of the main improvements in nuclear data with respect to burnup credit nuclides and on the availability of assay data	08:30-08:45
2.5	"Correction Factors Derived from French Experiments with the Recent JEFF3.1.1 Library for PWR-UOx BUC Applications" C. Riffard, A. Santamarina, and L. San Felice (CEA), J.-F. Thro (AREVA NC)	08:45-09:15
2.6	"Study of Burnup Reactivity and Isotopic Inventory in REBUS Program" T. Yamamoto and Y. Ando (JAEA), K. Sakurada and Y. Hayashi (Toshiba)	09:15-09:45
2.7	"PWR and BWR Fuel Assay Data Measurements" C. Alejano and J.M. Conde (CSN), M. Quecedo and M. Lloret (ENUSA), P. Zuloaga and F. J. Fernández (ENRESA), J. A. Gago (ENDESA)	09:45-10:15

Coffee break 10:15-10:45

II.C Sensitivity/Uncertainty analyses **10:45-13:00**
Co-Chairs: J.-C. Neuber (AREVA), T. Ivanova (IRSN)

	Chair overview of the objectives, key concepts, and methods	10:45-11:00
2.8	"Sensitivity/uncertainty Analysis applied to the Phase VII Benchmark" O. Cabellos, B. Cabellos and N. García-Herranz (UPM), J. Sanz (UNED), P. Ortego (SEA)	11:00-11:30
2.9	"Use of French Fission Product Experiments for Burnup Credit Validation" T.Ivanova, N. Leclaire, E. Létang (IRSN), J.-F. Thro (AREVA NC)	11:30-12:00
2.10	"General hierarchical Bayesian procedure for calculating the bias and the a posteriori uncertainty of neutron multiplication factors" J.-C. Neuber, A. Hoefer (AREVA)	12:00-12:30
2.11	"Usage of TSUNAMI in a Hierarchical Bayesian Procedure for Calculating the Bias and the a posteriori Uncertainty of keff" J.-C. Neuber, A. Hoefer (AREVA)	12:30-13:00

Lunch break 13:00-14:30

II.D Verification and validation **14:30-17:45**
Co-Chairs: M. Brady-Raap (PNNL), Ian C. Gauld (ORNL)

	Chair overview of the available data and methods	14:30-14:45
2.12	"Determination of a Depletion Uncertainty from Fuel Management Experience" D. B. Lancaster (NuclearConsultants.com), C. T. Rombough (CTR)	14:45-15:15
2.13	"Sufficiency of Available MOX Experiments for Criticality calculation validation of VVER Burnup Credit Application" G. Hordosy (KFKI), S. Patai Szabo (ANANDOR)	15:15-15:45
2.14	"Evaluation of Fission Product Critical Experiments and Associated Biases for Burnup Credit Validation" D. E. Mueller, B. T. Rearden, D. A. Reed (ORNL)	15:45-16:15

Coffee break 16:15-16:45

2.15	"Regulatory Perspective on Computer Code Validation for Burnup Credit Criticality Analyses for Spent Nuclear Fuel Transportation Packages" M. Rahimi, Z. Li, M. Call (USNRC)	16:45-17:15
2.16	"SCALE Validation Experience using an Expanded Isotopic Assay Database for Spent Nuclear Fuel" I. C. Gauld, G. Radulescu, G. Ilas (ORNL)	17:15-17:45

Thursday October 29th

SESSION III: APPLICATIONS AND IMPLEMENTATION **08:30-17:30**
Co-Chairs: J. Gulliford (NNL), A. Barto (USNRC)

	Chair overview of reactor operations impacting the spent fuel reactivity and the burnup distribution	08:30-08:45
3.1	"Regulatory Perspective on Confirmatory Burnup Measurements for Burnup Credit in Spent Nuclear Fuel Transportation Packages " A. Barto, N. Jordan (USNRC)	08:45-09:15
3.2	"Fuel Burnup Plant Records: Generation and Accuracy" M. Aissa (USNRC)	09:15-09:45
3.3	"Recommended Bounding Axial Burnup Profiles In BUC Applications From Actual Burnup Measurement Of French PWR Assemblies" C. Riffard and A. Santamarina (CEA), J.-F. Thro (AREVA-NC), F. Lavaud (EdF)	09:45-10:15

Coffee break 10:15-10:45

3.4	"Inventory Prediction and BUC Calculations Related to MEU/LEU IRT Fuels of LVR-15 Research" L. Markova, F. Havluj, M. Marek (NRI)	10:45-11:15
3.5	"Fuel Depletion Calculation in MTR-LEU NUR Reactor" Z. Fodil (COMENA)	11:15-11:45
3.6	"Burnup Credit in the Swedish Interim Storage Facility (CLAB)" L. Agrenius (Agrenius Ingenjörsbyra AB)	12:15-12:45

Lunch break 12:45-14:30

3.7	"Lessons Learnt from OECD/NEA Phase II-C through Phase II-E Benchmarks" J.-C. Neuber (AREVA)	14:30-15:00
3.8	"Review of results for the OECD/NEA Phase VII Benchmark: Study of the Spent Fuel Compositions for Long Term disposal" G. Radulescu, J. Wagner (ORNL)	15:00-15:30
3.9	"Burnup Credit Approach for the Proposed United States Repository at Yucca Mountain" J. M. Scaglione, J. C. Wagner (ORNL)	15:30-16:00

Coffee break 16:00-16:30

3.10	"Regulatory Issues for Final Disposal" J.-C. Neuber (AREVA)	16:30-17:00
3.11	"Burnup Credit in the Canister for Final Disposal of Spent Nuclear Fuel" L. Agrenius (Agrenius Ingenjörsbyra AB)	17:00-17:30

Friday October 30th

PANEL DISCUSSIONS **09:00-12:30**

Panel I: Code developments and nuclear data **Chair: A. Santamarina (CEA)** Participants: S. Evo (IRSN) J. C. Wagner (ORNL) R. D. McKnight (ANL) Y. Rugama (NEA)	**09:00-10:00**

Panel II: Sensitivity/uncertainty analysis, verification and validation **Chair: J.-C. Neuber (AREVA)** Participants: T. Ivanova (IRSN) D. Mueller (ORNL) O. Cabellos (UPM) Ian C. Gauld (ORNL) A. Vasiliev (PSI)	**10:00-11:00**

Coffee break 11:00-11:30

Panel III: Applications and implementation **Chair: J. Gulliford (NNL)** Participants: M. Brady-Raap (PNNL) A. Barto (NRC) R. Kilger (GRS) J. Vaclav (NRA SR) J. F. Serrano (ENUSA)	**11:30-12:30**

CLOSING SESSION **12:30-13:00**

WORKSHOP ABSTRACTS

SESSION I: INTERNATIONAL AND NATIONAL REPORTS

1.1

IAEA Activities in Spent Fuel Management

Zvonko Lovasic (IAEA)

1.2

Lessons Learned from International Investigations of Burnup Credit Criticality

OEDC/NEA Working Party on Nuclear Criticality Safety
Expert Group on Burnup Credit Criticality (EGBUC)

Michaele Brady Raap, Chairman EGBUC
Pacific Northwest National Laboratory
Richland, WA USA

Jim Gulliford (UK National Nuclear Laboratory)
Ludmila Markova (NRI, Czech Republic)
Hiroshi Okuno (Japan Atomic Energy Agency)
Greg O'Connor (UK Department for Transport)
Peng Hong Liem (NAIS Co., Japan)

This summary report defines, discusses and makes recommendations related to the physics and analysis of burnup credit criticality on the basis of the combined experience of the OECD/NEA Expert Group on Burnup Credit Criticality (EGBUC) members over the past 15 years. The report emphasizes the relevance of EGBUC benchmark evaluations and comparisons in deriving these conclusions. This report addresses systems containing irradiated Light Water Reactors (LWRs) including Pressurized Water Reactors (PWRs), Boiling Water Reactors (BWRs) and water-cooled, water-moderated energy reactors VVER (Russia-design PWRs). Studies for PWRs included the use of Mixed-Oxide (MOX) fuels.

The purpose of this presentation is to assess and document the conclusions of the EGBUC members and their experience regarding the importance of various parameters for the implementation of burnup credit as a criticality safety strategy in establishing the safety basis/argument for nuclear material storage, transportation and reprocessing of irradiated LWR (PWR, BWR and MOX) fuel. This report also attempts to establish practical rules and identify applicable tools when appropriate. Lessons learned regarding any inconvenience or problem encountered in the experience of performing the international comparison problems will be presented and discussed.

The presentation will summarize the activities of the EGBUC, highlight the current findings by reactor and fuel types and address the future goals of the group. Current studies directed at the evaluation of burnup credit in a geologic repository timeframe and environment will be described.

1.3

ACTIVITIES OF THE OECD/NEA EXPERT GROUP ON ASSAY DATA FOR SPENT NUCLEAR FUEL

Ian Gauld and Y. Rugama[1]

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[1]OECD Nuclear Energy Agency
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[1] Managed by UT-Battelle, LLC, for the U.S. Department of Energy under contract No. DE-AC05-00OR22725.

Abstract – Management of spent nuclear fuel is a key issue for many NEA member countries. As interim storage facilities in many countries reach their design capacities, the need to optimize spent fuel storage is becoming an increasingly important issue to managing fuel cycle costs. In nuclear criticality safety, the decision of many countries to advance burnup credit as part of their licensing strategy has heightened recent interest in measurement data that are needed to validate computer code calculations for a burnup credit methodology. The importance of measured isotopic assay data from irradiated fuel experiments to validate computer code predictions of spent fuel composition used in safety-related studies has long been recognized by members of the OECD/NEA/NSC/WPNCS (Working Party on Nuclear Criticality Safety). Under the auspices of the WPNCS, an Expert Group on assay data has been formed to help coordinate isotopic assay data activities and facilitate international collaboration between NEA member countries developing or implementing burnup credit methodologies. An important contribution of the WPNCS in this area has been the electronic database for publicly available spent nuclear fuel composition data, SFCOMPO. The objectives of the Expert Group on assay data include (1) expanding the SFCOMPO experimental database of spent fuel isotopic measurements, (2) making the data accessible through the SFCOMPO website, (3) sharing best practices on radiochemical analysis methods, (4) indentifying input data and modeling requirements, and (5) evaluating uncertainties associated with the measurements and deficiencies in documented design and reactor operating history information. This paper discusses recent activities of the Expert Group, focusing on the planned expansion of the database of spent fuel measurements, SFCOMPO, a description of the new fuel samples and measurement data they provide, and future activities.

1.4

Burnup Credit Development and Implementation in the Slovak Republic

Vladimir Chrapciak, VUJE
Juraj Vaclav, Nuclear Regulatory Authority of the Slovak Republic (NRA, SR)

Nuclear Regulatory Authority of the Slovak Republic (UJD) has prepared various research tasks under the R&D program. The Division of Nuclear Materials has executed a task of the burnup credit (BUC) application in the criticality calculation of the VVER-440 fuel assemblies in cooperation with Nuclear Power Plants Research Institute (VUJE). The task was divided into two parts. VUJE performed first task in 2005 through 2007.

The following subtasks have been addressed under this research task:

Verification of SCALE 5.0 calculation system

The aim was to verify applicability of the latest version of the SCALE 5.0 calculation system to the VVER-440 spent fuel storage and transport. It consists of the SCALE 5.0 system testing during the calculations of criticality nuclide composition and residual heat of the VVER-440 fuel and verification of the system applicability by means of the results comparisons with the ones of the numerical models.

Methodology of the BUC for the VVER-440 fuel

The aim was to develop appropriate methodology of the BUC application for the VVER-440 fuel. It consists of the proposal of the calculation analyses range in order to ensure sufficient subcriticality during the VVER-440 spent fuel storage and transport.

Application of the BUC for the dry storage conditions of the VVER-440 fuel

This task demonstrated that when the burn-up consequences are partially taken into account, it significantly decreases requirements on the VVER-440 spent fuel storage under dry conditions. The task puts an emphasis on the BUC analysis for the dry storage of the VVER-440 spent fuel. The results will serve for validation of the basic parameters of the Mochovce dry store.

Application of the BUC for the wet storage conditions of the VVER-440 fuel

The aim was to examine possibilities of the VVER-440 spent fuel storage and transport with higher original enrichment in the existing storage and transport facilities. It consists of the analysis of the possibility to transport and store the VVER-440 spent fuel with original enrichment up to 5% U235 in the existing C-30 transport container with T-12 or KZ48 casks and in the at-reactor spent fuel storage pools.

Under those subtasks we have developed methodology for BUC utilization, taking into account actinides only, and we have validated the SCALE 5.0 system as a tool for VVER-440 fuel.

The second part of the project will also include fissile products. This subtask started in 2008 and will be finished in 2010.

In order to have validated results three Slovak organizations (VUJE, JAVYS, UJD) have joined an international consortium focused on further investigation of nuclide composition of VVER-440 spent fuel within the framework of project ISTC #3958. Having these results we will continue the verification of the SCALE 5.1 and 6 systems for nuclide composition calculations. The UJD will prepare a guide on BUC application in Slovakia.
The BUC will be necessary for the licensing of the new fuel with enrichment of 4.87% ^{235}U in at reactor pool and in basket KZ-48.

1.5

Applying Burnup Credit Technology for Spent Fuel Storage in China

Guoshun You, Quingfu Zhu, Chuanwen Hu
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Since 1990s, the concept of Burn-up Credit was introduced into China, a lot of work has been done in CIAE. From the year of 2002, a technical co-operation project with IAEA including a series of training course on Burn-up Credit technology has been conducted. The nuclear authorities of China have approved applying BUC in some areas of spent fuel managements. At present time BUC technology has been applied in spent fuel storage in China. The standards and regulations on BUC are under being developed.
In this paper, the analysis method on BUC technology what we used is presented. The activities on applying Burn-up Credit technology for spent fuel storage are introduced. The problems we faced with and the research programs we will conduct are also introduced in this paper. The BUC technology will play an important role on spent fuel management in China in future.

1.6

National Program for BUC Application in India

(A.K. Pandey, NPCIL)

1.7

Research Activities of JAEA Considering the Future Needs of Japan in Burnup Credit Implementation

Hiroshi Okuno, Kotaro Tonoike, Kenya Suyama*, Kyoshi Ohkubo, Gunzo Uchiyama
Japan Atomic Energy Agency (JAEA)
* Present Affiliation: Ministry of Education, Culture, Sports, Science & Technology (MEXT)

In the Rokkasho Reprocessing Plant, actinide-only burnup credit is implemented for storage and dissolving processes of spent nuclear fuels. Inclusion of fission products in burnup credit would be the next step to follow. Nevertheless, burnup credit is not implemented for dry/wet storage or transport of spent nuclear fuels in Japan, except gadolinium credit which has been implemented even for storage and transport of spent BWR fuels. In the criticality safety evaluation of spent nuclear fuels for disposal, implementation of burnup credit should be crucial; however, to reprocess all the spent fuels has been the national policy of Japan, which has delayed our investigation into this field.
For inclusion of fission products in burnup credit, a series of critical experiments were performed that measured reactivity worth of fission product elements using a heterogeneous core at the Static Criticality Experiment Facility (STACY) of the Japan Atomic Energy Agency (JAEA). The relation with a similar experiment was discussed that involved gadolinium dissolved in the uranium nitrate solution, which was also performed at the STACY.
Development of a new version of SWAT code, named SWAT-3.1, was made that allowed the users to utilize not only SRAC code system and MVP code, both of which were developed by the former Japan Atomic Energy Institute (JAERI), but also the world-widely popular Monte Carlo neutron transport code, MCNP5. This version was applied to the OECD/NEA Burnup Credit Phase IIIB Benchmarks and the results were compared with the results of other burnup calculation codes to verify that the SWAT-3.1 would give reasonable results.
According to a suggestion of the Atomic Energy Commission of Japan, the JAEA has been collecting information on the direct disposal of spent fuel. Especially for the criticality issues of the direct disposal, two divisions of the JAEA, i.e., the Nuclear Safety Research Center and the Geological Isolation Research & Development Directorate have started collaboration in doing this.
“A Guide Introducing Burnup Credit, Preliminary Version” was published as a report from JAERI in 2001. Efforts are underway for translating the original Japanese report into English as well as for updating its texts and data.

1.8

REVIEW OF TECHNICAL STUDIES IN THE UNITED STATES IN SUPPORT OF BURNUP CREDIT REGULATORY GUIDANCE

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Abstract - Taking credit for the reduction in reactivity associated with fuel depletion can enable more cost-effective, higher-density storage, transport, disposal and reprocessing of spent nuclear fuel (SNF) while maintaining a sufficient subcritical margin to establish an adequate safety basis. Consequently, there continues to be considerable interest in the United States, as well as internationally, in the increased use of burnup credit in SNF operations, particularly related to storage, transport and disposal of commercial SNF. This interest has motivated numerous technical studies related to the application of burnup credit, both domestically and internationally, as well as the design of SNF storage, transport and disposal systems that rely on burnup credit for maintaining subcriticality. Responding to industry requests and needs, the U.S. Nuclear Regulatory Commission (NRC) initiated a burnup credit research program in 1999, with support from the Oak Ridge National Laboratory (ORNL), to develop regulatory guidance and the supporting technical bases for allowing and expanding the use of burnup credit in pressurized-water reactor SNF storage and transport applications. Although this NRC research program has not been continuous during the past ten years, considerable progress has been achieved in many key areas in terms of increased understanding of relevant phenomena and issues, availability of relevant information and data, and subsequently updated regulatory guidance. This paper will review the technical studies performed by ORNL for the U.S. NRC burnup credit research program. Examples of topics that will be addressed include: reactivity effects associated with reactor operating characteristics, fuel assembly characteristics, burnable absorbers, control rods, spatial burnup distributions, cooling time and assembly misloading; methods and data for validation of isotopic composition predictions; methods and data for validation of criticality calculations; and operational issues and data related to assembly burnup confirmation. The objective of this paper is to provide a summary of the work and significant accomplishments, with references to the technical reports and publications for complete details, that will be a useful resource to others in the burnup credit community.

1.9

Latest Studies Related to the Use of Burnup Credit in France

L. Jutier[a], I. Ortiz de Echevarria[a3], S. Evo[a],
E. Guillou[b], J. Jaunet[c], F. Lavaud[d], A. Bonne[d]

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[d] EDF R&D, 1 avenue du Général de Gaulle, BP 408, 92141 Clamart, France

In France, criticality safety analysis for nuclear fuel cycle facilities and transport casks usually consider fresh fuel. In some cases, however, limited to PWR UOX fuel, Burnup Credit is taken into account with some pessimistic hypothesis (only actinides are considered and the value of burnup used in the studies is equal to the mean burnup in the 50-least-irradiated centimetres) [1]. As the UOX fuel initial enrichment and the storage needs for spent fuel increase, operators have wished to develop a less penalizing method to implement Burnup Credit in criticality studies, by taking into consideration some fission products (the most neutron absorbent, stable or with a long half-life and not volatile) and a suitable bounding axial burnup profile.

[2] Managed by UT-Battelle, LLC, for the U.S. Department of Energy under contract No. DE-AC05-00OR22725.
[3] Author presenting the paper.

In this context, a working group composed of the main French nuclear companies, CEA and IRSN was formed to study the conservatism of all steps of the process to take fuel burnup into account in the criticality studies considering fission products and an axial burnup profile. These steps are: the definition of the axial profile of burnup in the studies, the depletion calculations, the criticality calculations (particularly regarding the knowledge of the cross-sections of the isotopes that are being taken into account). This paper proposes to give an outlook of the present state of knowledge of the working group.
To date, the discussions of the working group have focused on PWR UOX fuels. The early studies are presented in reference [2]. The latest ones mainly concern: the determination of a conservative axial profile for most of the profiles already measured, the determination of correction factors for the isotopic composition in Burnup Credit applications, and the validation study of depletion code performed with a Monte Carlo code.
In addition, studies on Burnup Credit for PWR MOX fuels are now under way. One of the main difficulties of a Burnup Credit implementation for PWR MOX fuel is the wide range of parameters compared to PWR UOX fuel (initial plutonium composition, plutonium content, uranium composition, presence of ^{241}Am, zoning of assemblies). Contrary to PWR UOX fuel, for PWR MOX fuels, a fresh fuel with a conservative isotopic vector (of plutonium) will not inevitably lead to the most reactive fuel after irradiation. Therefore, to simplify MOX burnup implementation, it was necessary to set up a method to determine, for a given ratio of Pu/(U+Pu), a bounding plutonium vector for the fresh fuel that gives, after irradiation, the most reactive fuel, whatever the irradiation history. At present, the studies on PWR MOX fuels concern the determination of a conservative inventory of the irradiated fuel.
Moreover, in the framework of optimization in some parts of the fuel cycle, namely reprocessing plants, Burnup Credit implementation for BWR UOX fuels is being investigated.
Finally, this paper gives results of criticality calculation for storage configurations and transport casks obtained when applying the different conservatisms studied by the working group.

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SESSION II: CALCULATION METHODS

II.A Codes and methods

2.1 ENHANCEMENTS TO THE BURNUP CREDIT CRITICALITY SAFETY ANALYSIS SEQUENCE IN SCALE

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Abstract - SCALE (Standardized Computer Analyses for Licensing Evaluations) is a system of computer codes and automated calculational sequences developed at Oak Ridge National Laboratory for criticality safety, shielding, and reactor analyses. The SCALE sequence to perform automated burnup credit criticality calculations for spent nuclear fuel systems is STARBUCS (Standardized Analysis of Reactivity for Burnup Credit using SCALE). STARBUCS uses the capabilities of the SCALE driver to automate depletion and Monte Carlo criticality calculations using the ORIGEN isotope generation and depletion code and either the CSAS5 (KENO V.a) or CSAS6 (KENO-VI) criticality safety sequence, respectively. The STARBUCS input options allow analysts or reviewers to investigate the impact on criticality safety of various assumptions related to burnup credit calculation methodology for spent fuel in transport or storage casks. Options are provided to model the axial and horizontal variations of the burnup within a spent fuel

[4] Managed by UT-Battelle, LLC, for the U.S. Department of Energy under contract No. DE-AC05-00OR22725.

assembly, to select the actinide and fission product nuclides for spent fuel composition calculations, and to adjust the predicted inventories to account for isotopic composition bias and bias uncertainty. The following STARBUCS enhancements are available with the SCALE 6 release: the capability of performing burnup loading curve search, the ability to use either multi-group or continuous energy Monte Carlo criticality calculations, reduced number of iterations required to achieve eigenvalue convergence, reduced output options, and the option of saving the input files created for the criticality calculations for use in subsequent calculations such as cross-section sensitivity and uncertainty analyses with SCALE/TSUNAMI. For burnup loading curve analyses, STARBUCS performs iterative burnup credit criticality calculations using an initial fuel enrichment search to determine combinations of fuel enrichment and burnup that yield a user-specified critical limit within convergence tolerance criteria. The paper presents a review of the STARBUCS features available in SCALE 6, STARBUCS applications that illustrate the impact of various modeling assumptions on burnup loading curves, and a comparison between STARBUCS calculations using SCALE multi-group and continuous energy cross-section libraries in terms of computer time and k_{eff} results for various spent fuel configurations.

2.2

Assessment of the MCNP+ACAB code system for burnup credit analyses

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An automated tool for depletion and sensitivity/uncertainty analysis in BUC calculations is presented. This system combines the Monte Carlo neutron transport code MCNP and the inventory code ACAB as a suitable tool for high burnup calculations.
The potential impact of nuclear data uncertainties on some response parameters (decay heat, neutron emission, radiotoxicity, keff, ...) may be large, but only very few codes can treat this effect. The uncertainty analysis methodology implemented in the ACAB code, including both the sensitivity-uncertainty method and the uncertainty analysis by the Monte Carlo technique, enables to assess the impact of neutron cross section uncertainties on the inventory and other inventory-related responses in high burnup applications.
A well referenced high burnup pin-cell benchmark exercise is used to test the MCNP–ACAB performance in inventory predictions. In addition, the potential of our system, including the uncertainty capability, is demonstrated. It is proved that the inclusion of ACAB in the system allows obtaining results that are at least as reliable as those obtained using other inventory codes. We estimate the errors due to activation cross section uncertainties in the prediction of the isotopic content up to the high-burnup spent fuel regime. The most relevant uncertainties are highlighted, and some of the most contributing cross sections to those uncertainties are identified.

2.3

Convergence issues in Best-estimate Monte Carlo Depletion Calculations

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Source convergence issues in Monte Carlo criticality simulation were previously seen as a difficult research problem, fortunately having a marginal impact on real world criticality licensing, which focuses mainly on k-effective estimation. Applied in depletion calculations, Monte Carlo source convergence is now a key point for the reliability and usability of this type of application. As a first approach, the VESTA depletion interface is used to simulate isotopes composition histories on a parametric ill-conditioned benchmark of infinite lattice of PWR fuel pins modeled with MORET5 Monte Carlo code. Besides initial random seeds, several simulation parameters (neutrons population size, number of batches, time sampling and geometrical binning) are compared in term of simulation cost versus time compositions discrepancy. The probability density of resulting compositions obtained on such “crude” Monte Carlo propagation of simulation noise due to Boltzmann equation Monte Carlo solving process tends to arise numerical

instability which lead to false neutronic oscillations similar to (physically true) Xenon effect. The density function observed are investigated to suggest some explanations based on Monte Carlo neutron simulation characteristics and convergence issues.

2.4

PEAK REACTIVITY CHARACTERIZATION AND ISOTOPIC INVENTORY CALCULATIONS FOR BWR CRITICALITY APPLICATIONS

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For BWR criticality calculations supporting spent fuel storage racks design, the most reactive fuel assembly to be stored, at its maximum reactivity point (peak reactivity) must be used.
Determination of the most reactive fuel assembly should consider the distribution (as a function of burnup) of fissile content, burnable absorbers and fission product inventories in the fuel assembly.
BWR fuel designs are highly heterogeneous both radial and axially, with increasing (in current designs) number of different lattices with specific enrichment and gadolinium configurations. Moreover, the isotopic evolution (and hence the reactivity) of the BWR fuel shall also depend on the local core conditions, mainly void fraction and control rod insertion, during its life in the core that, in turn, change with burnup. All those characteristics make not an easy task, nor practical for production purposes, the BWR fuel characterization for criticality calculations.
This paper will present a simple conservative model for peak reactivity definition and isotopic inventory calculation for BWR criticality applications. The relevant parameters analyzed to define this model include:

- Core conditions (void fraction and control blades insertion)
- Gadolinium distribution
- Axial enrichment distribution
- Radial enrichment distribution
- Radial isotopic distribution
- Axial burnup and isotopic distribution

As a result of the performed sensitivity analysis, a practical (i.e. simple and conservative) method to define the fuel bundle material model to be used in criticality calculations is defined. Results of the sensitivity calculations shall be included in the paper, as well as different comparisons of the result produced by the model vs. those obtained with best-estimate approaches, that confirm the conservatism of the proposed model.

II.B Nuclear and assay data

2.5

Correction Factors derived from French experiments with the recent JEFF3.1.1 library for PWR-UOx BUC applications

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The concept of taking credit for the reduction of the reactivity of nuclear spent fuel due to their burnup is referred to as “Burnup Credit” (BUC). Allowing reactivity credit for spent nuclear fuels (SNF) offers many economic incentives. The increasing ^{235}U enrichment and need for fuel transport and storage point out the interest for BUC methods. A recent and rigorous methodology[1] was developed by the CEA and AREVA-NC, carrying out the French BUC calculation route based on the code systems DARWIN[2] and CRISTAL[3]. It is accounting of:

1. 15 poisoning FPs, stable and non-gazeous, in addition to the actinides;
2. Conservative hypotheses for the depletion calculations;
3. The qualification of the spent fuel inventory obtained with DARWIN and of the reactivity worth of BUC nuclides calculated with CRISTAL;
4. A bounding axial profile of assembly BU.

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The impact from recent improvements of BUC nuclide evaluation in the JEFF3.1.1 library[4] and on the validation of the French BUC code systems are mainly described in this paper. Namely, a set of conservative Correction Factors (CF), to be applied to the isotopic concentrations obtained from the depletion calculations, was established for each BUC nuclide involved in PWR-UOx Burnup Credit. These CF are one of the key of the French BUC method proposed to guarantee the conservatism of the fuel reactivity in safety-criticality calculations. They allow the conservatism of the isotopic concentrations with regards to the calculation - experiment bias of the code DARWIN as well as the conservatism of the BUC nuclide worth in CRISTAL. These conservative CF are derived from two kinds of experimental programs[5,6]. These programs involve two kinds of experiments, chemical analyses and microprobe measurements of PWR irradiated fuel pins on one hand, and reactivity worth measurements of the BUC nuclides in the MINERVE reactor on the second hand.
The recent qualification results of spent fuel inventory and reactivity worth calculations using the recommended libraries and code versions for fuel cycle applications, show many improvement of the bias trends. Therefore, new CF for PWR-UOx BUC application are calculated and presented, including a comparison with the previous ones based on JEF2.2[7]. The reduction of the biases (C-E)/E between the calculated isotopic concentrations and the experimental values for some important BUC nuclides (^{242}Pu, ^{153}Eu, ^{155}Gd) are especially highlighted, leading to a reduction of the penalty on the keff, calculated in the case of a PWR-UOx pool storage.

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2.6

Study of Burnup Reactivity and Isotopic Inventories in REBUS Program

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The REBUS program has a unique feature, that is, direct determination of the burnup reactivity of UO2 and MOX fuel bundles, which were fabricated from irradiated fuel assemblies discharged from LWRs in Europe. The program consists of core physics experiment at a LWR critical test facility VENUS, nondestructive measurement of the burnups of irradiated fuel rods by gamma-ray spectroscopy, and radiochemical isotopic analysis of pellet samples. In the critical experiment, a core tank was loaded with a 27x27 square lattice core that consists of a 7x7 cell test bundle in the core center, and 3.3 and 4.0 wt% enriched UO2 fuel rods surrounding the test bundle as a driver region. Five types of test bundles were included in the experiment: (1) fresh BR3 (PWR) MOX fuel, (2) irradiated BR3 MOX fuel, (3) irradiated BWR MOX fuel, (4) fresh PWR UO2 fuel, and (5) irradiated PWR UO2 fuel. While measured burnup reactivity values were well reproduced by core calculations, discrepancies between the calculated and measured inventories were observed for both irradiated UO2 and MOX fuel. The present authors have indicated that the biases in inventory and reactivity calculations compensate each other, which makes the total biases of the burnup reactivity small for the BR3 MOX fuel. In the present study, the ratios of calculated to measured inventories (C/Es) for each nuclide were utilized to estimate probable inventories for the PWR UO2 and BWR MOX fuel bundles, and burnup reactivity was analyzed with the probable

inventories. From the results, the biases in inventory and reactivity calculations for burnup reactivity were discussed.

2.7

PWR and BWR Fuel Assay Data Measurements

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In the framework of a high burnup fuel demonstration program, rods with an enrichment of 4.5% ^{235}U were operated to a rod average burnup of about 70 MWd/kgU in the Spanish Vandellós 2 pressurised water reactor. The rods were sent to hot cells and used for different research projects. This paper describes the isotopic composition measurements performed on samples of those rods, with burnup values ranging from 40 MWd/kgU up to 75 MWd/kgU. The main results obtained and their comparison with values calculated using different methods are presented.
In 2008, a project to measure the isotopic composition of irradiated BWR fuel was started. The mother rod was irradiated in the Swedish Forsmark-3 boiling water reactor, and the fuel samples selected have a burnup ranging from 40 to 55 MWd/kgU. The paper describes the scope of the project and its current status.

II.C Sensitivity/Uncertainty analyses

2.8

Sensitivity/uncertainty Analysis Applied to the Phase VII Benchmark.

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In November 2008, a Phase VII Benchmark entitled “UO2 Fuel: Study of spent fuel compositions for long-term disposal” was proposed by the Expert Group on Burnup Credit of the NEA-OECD. The main objective of this benchmark is to study the ability of relevant computer codes and associated nuclear data to predict spent fuel isotopic compositions and corresponding keff values in a cask configuration over the time duration relevant to spent nuclear fuel disposal, out to 1,000,000 years. The results of this exercise are expected to show differences in international nuclear data sets (decay/branching data to predict the isotopic inventory along decay time and cross sections to predict keff) and computing tools (inventory and transport codes).
For decay calculations, and in order to accomplish this objective at the same time increasing the understanding and confidence in our ability to predict keff for timeframes relevant to long-term disposal, we have performed a decay sensitivity analysis assessing: i) impact of different nuclear decay libraries, ii) importance of numerical solvers to predict the inventory, iii) study of the main pathways for the formation of relevant nuclides, iv) sensitivity to decay data for the formation of relevant nuclides, v) overall uncertainty analysis by a general Monte Carlo procedure.
For criticality (keff) calculations, a representative PWR cask model is selected. MCNP and KENO, with different nuclear data libraries, were used to compare the predicted keff. To take advantage of the previous decay sensitivity analysis, the sensitivity profiles to the isotopic composition have also been calculated. This permits the identification of the most contributing cross sections and the most relevant uncertainties in this problem.

2.9

USE OF FISSION PRODUCT EXPERIMENTS FOR BURNUP CREDIT VALIDATION

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Criticality safety analyses traditionally assume the reactivity reduction in spent fuel due to production of actinide neutron absorbers but not fission products (FPs). To compensate for the lack of integral experiments with FPs, a series of 145 critical experiments, referred to as the FP experimental program, was conducted at Apparatus B research facility in Valduc (CEA, France). The experiments were performed with U(4.738 wt% $^{235}U)O_2$ or HTC rod arrays in moderating solution. The following key fission products encountered in the solutions either individually or as mixtures were studied: ^{103}Rh, ^{133}Cs, ^{nat}Nd, ^{149}Sm, ^{152}Sm, and ^{155}Gd.
A criticality validation tool based on generalized linear least-squares (GLLS) method (sometimes referred to as "adjustment") is under development in the IRSN. Modifications to the method have been recently implemented to differentiate bias due to the FPs from bias due to other materials. Two approaches can be used for the bias differentiation. One of them consists in applying the GLLSM not to k_{eff} but to reactivity of test material (FP). It is briefly described in Ref. 1. Another approach applies the method to k_{eff} in a few steps. In the first step, cross-section data for major actinides and moderator are corrected using results of replacing experiments without the FPs. Then, the "adjusted" major cross-section-covariance matrices and corrections to cross sections are involved in a procedure to "adjust" those for the FPs. A complete vector of corrections to cross sections and covariance matrix is then used to project to the application system bias due to the FPs available in the experiments. This paper presents performance of the second approach.
The FP experimental series provides unique data for this study because measurements were performed with and without the test FP on nearly the same array. This makes the experimental uncertainties strongly correlated for the pairs of experiments and allows magnifying the FP effect by applying the modified "adjustment" method.

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2.10

General hierarchical Bayesian procedures for calculating the bias and the a posteriori uncertainty of neutron multiplication factors

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In a Criticality Safety Analysis (CSA) uncertainties have to be treated on different hierarchical levels, where uncertainties on a lower level have an impact on uncertainties on a higher level. E. g., in a Burn-up Credit (BUC) CSA uncertainties in the parameters and the outcomes of post-irradiation experiments evaluated in order to estimate the isotopic biases in a predicted Spent Nuclear Fuel (SNF) isotopic inventory, which are due to the depletion code applied, result in uncertainties in the predicted SNF isotopic concentrations. These uncertainties affect the uncertainty in the neutron multiplication factor of any system containing the SNF. It is, therefore, preferable to use Bayesian hierarchical models making it possible to follow the impacts of the uncertainties through all the levels.
Uncertainties in a given set of parameters **x** are generally taken into account by treating **x** as a random vector defined by some probability distribution

$$F(\mathbf{x} \in R \mid \mathbf{\Theta}) = \int_R d\mathbf{x}\ p(\mathbf{x} \mid \mathbf{\Theta}).$$

$p(\mathbf{x}|\mathbf{\Theta})$ denotes the related probability density function, where $\mathbf{\Theta}$ represents the set of parameters characterizing the distribution model adequate to **x**. In general, the values of $\mathbf{\Theta}$ are unknown. They have therefore to be estimated from the evaluation of sampled data $\mathbf{D}_x$ of **x**. The model parameters $\mathbf{\Theta}$ can hence be regarded as random variables as it is done in Bayesian statistics. The Bayesian approach allows to perform random draws, by means of Monte Carlo techniques, of model parameters $\mathbf{\Theta}$ and parameters **x** on different hierarchical levels. The respective random draws can then be used as an input to the next higher level.

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The hierarchical Bayesian procedures required for estimating an adequate upper tolerance limit of the sum ($k_{eff} + \Delta k_B$) of the neutron multiplication factor k_{eff} of an SNF system of interest and the bias Δk_B in k_{eff} related to this system due to the employed criticality calculation code will be described, and it is shown how these procedures consider all the uncertainties in the parameters characterizing the SNF system and the uncertainties in the nuclear data involved.

2.11

Usage of TSUNAMI in a hierarchical Bayesian procedure for calculating the bias and the a posteriori uncertainty of k_{eff}

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In the paper which will be presented under the title "General hierarchical Bayesian procedures for calculating the bias and the a posteriori uncertainty of neutron multiplication factors" the uncertainties in the nuclear data involved in the criticality safety analysis of a Nuclear Fuel System (NFS) are considered in the most general way possible. This way cannot be realized at present through lack of information about variances and covariances in the basic nuclear data. However, multi-group-based covariance matrices are available in the SCALE system [1] for usage with the TSUNAMI module sequences of this system. It will be shown how results from a TSUNAMI analysis of an NFS of interest can be integrated in a hierarchical Bayesian procedure for calculating the bias and the a posteriori uncertainty of the neutron multiplication factor of the NFS.

Reference:

[1] SCALE: "A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation", Version 6, RSICC, CCC-750/01

II.D Verification and validation

2.12

Determination of a Depletion Uncertainty From Fuel Management Experience

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Historically the US NRC has allowed spent fuel pools to use 5% of the delta k of depletion as the depletion uncertainty. Recently, there have been requests from the NRC to show the basis of the uncertainty. This paper shows that commercial reactor fuel management experience can be used to justify an uncertainty of less than 2% of the delta k of depletion.

A commercial vendor and its customers have analyzed over 200 cycles of commercial power operation with current generation codes. The mean bias in the startup hot zero power ppm is 3 ppm, which is only 0.0003 in k. The standard deviation about the mean of these 200 startups is 13 ppm or 0.0013 in k. The mean volume averaged core burnup of these startups is 12.2 GWD/MTU. The range of the volume averaged core burnup of any of these 200 cases is 10.7 to 14.7 GWD/MTU. No trend with burnup has been observed.

As well as the startup predictions, this same vendor has documented (in-house) its capability to predict the HFP end of cycle reactivity with these same codes for 189 cycles. The mean end of cycle bias in reactivity is 0.0007 in k. The standard deviation about this condition is 0.0024. The mean volume averaged core burnup of these measurements is 21.3 GWD/MTU. The core average burnups ranged from 17.1 to 29.3 GWD/MTU. Again no significant trend has been observed with burnup.

Even though differences were observed, these differences are due to a combination of factors, such as the measurement uncertainty, errors in modeling, differences between the actual core conditions and the modeled conditions, and finally, errors in the delta-k due to depletion. For simplicity, it will be conservatively assumed that all the difference between the measured ppm and the predicted ppm is due to errors in the depletion. This is a very conservative assumption since there has been no observed trend with burnup, which implies that most, if not all, of the differences are due to factors that have nothing to do with burnup.

The delta k from enriched UO_2 to the average startup burnup of 12.2 GWD/MTU is 0.117 in k. Using data from the startup measurements and assuming that all of the differences are due to errors in the depletion, the bias is then 0.0003/0.117 or 0.3% of the delta k of depletion. The uncertainty in this bias is 0.0013/0.117 or 1.1% of the delta k of depletion. The same analysis can be done using the end of cycle reactivity measurements. Here the delta k from enriched UO_2 to the average end of cycle burnup of 21.3 GWD/MTU is 0.207 in k. The burnup bias from the end of cycle results is –0.0007/0.207 or 0.3% of the delta k of depletion. The uncertainty in this bias is 0.0024/0.207 or 1.2% of the delta k of depletion. The bias and uncertainties from both the startup predictions and the end of cycle predictions are very similar. To be conservative the higher bias and uncertainty is used. It is concluded that the fuel management tools have a bias that is less than 0.3% in the delta k of depletion and an uncertainty in this bias of less than 1.2% of the delta k of depletion.

2.13

Sufficiency of available MOX experiments for criticality calculation validation of VVER burnup credit applications

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For safe application of burnup credit, a conservative safety margin should be derived, which includes the effect of nuclear cross section errors. Such margin can be derived from analysis of calculation-experiment comparison, if sufficient number of suitable critical experiments is available. The critical experiments, used in the analysis, should be similar to the investigated application. Traditionally, the parameters used in investigation of similarity, were the enrichment, fissile/moderator ratio, isotopic composition of fissile material and some spectral parameters. However, the decision on the similarity was based on the judgment of the analyst to some extent. Recently, more rigorous basis of similarity, based on sensitivity/uncertainty analysis, was developed and incorporated into SCALE program package in the form of similarity indices. The TSUNAMI module of the system evaluates more quantities, quantifying the similarity between an application and a selected experiment. These quantities are based on the sensitivity of k_{eff} to the particular cross sections and the covariance data of these cross sections.
For the actinide-only burnup credit applications, the MOX critical experiments are the possible candidates as applicable for validation of criticality calculations. However, the publicly available MOX experiments, described in the International Handbook of Evaluated Criticality Safety Benchmark Experiments, have quite different uranium/plutonium isotopic composition, than the burned fuel. For this reason, their applicability for validation is questionable.
This question has been investigated using TSUNAMI for the case of VVER-440 fuel in the compact storage pool of Paks NPP. Approximately 300 MOX and UO2 critical experiments were investigated. While high degree of coverage was found, the degree of similarity was poor. The value of c_k index was bellow 0.8 in great majority of the cases, and it was about 0.85 only in a few cases. (The perfect similarity corresponds to 1, and generally c_k higher than 0.9 is expected for validation purpose.) We can conclude that this set of critical experiments, using the presently available methods, is insufficient for burnup credit criticality validation.
Since the completion of this study, new version of SCALE, the SCALE 6 became available. This version includes new methods for investigating the influence of cross section errors (GLLSM for cross section adjustment). Study of these new methods for VVER fuel will start in the immediate future and first results are expected for the time of BUC meeting in 2009 october.

2.14

EVALUATION OF FISSION PRODUCT CRITICAL EXPERIMENTS AND ASSOCIATED BIASES FOR BURNUP CREDIT VALIDATION

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One of the challenges associated with implementation of burnup credit is the validation of criticality calculations used to support safety analyses. The purpose of the validation is to quantify the relationship between the real world and calculated results. "Conventional" bias and bias uncertainty analyses involve modeling a set of critical experiments that are similar to the criticality safety model (referred to as the "application" in this paper), using the same computer code, nuclear data, and modeling approximations as were used to model the application. Statistical analyses of the critical experiment model results are then performed to determine an overall or total bias and bias uncertainty. For burnup credit analyses, this task is made more complex by the burnup-dependent isotopic compositions of the uranium, plutonium, other actinides, and fission products in commercial spent nuclear fuel (CSNF). Validation and determination of bias and bias uncertainty require the identification of sets of critical experiments that are similar to the criticality safety models. Similarity determination has typically been based on comparisons of some integral system characteristics such as energy of average lethargy causing fission or moderation ratio (e.g., $H/^{235}U$) and, more qualitatively, on the presence and composition of materials present in the application. The SCALE sensitivity and uncertainty (S/U) analysis tools (TSUNAMI) facilitate a detailed examination and comparison of nuclear characteristics of critical experiments and criticality safety models. The TSUNAMI tools have been used to compare a typical burnup credit cask model with sensitivity information from more than 1400 critical configurations, some of which included fission product (FP) nuclides. One of the results from this comparison is that the available FP critical experiments are either not similar enough to criticality safety models or do not have FPs present at appropriate levels to use conventional methods to accurately determine overall bias and bias uncertainty for a FP burnup credit application. An evaluation of the available critical experiments that involve FPs is presented along with bounding, burnup-dependent estimates for potential FP biases generated by combining energy dependent sensitivity data for a typical burnup credit application with the nuclear data uncertainty information distributed with SCALE 6. A method has been developed for determining separate bias and bias uncertainty values for individual FPs. Sensitivity profiles generated by TSUNAMI were used as the basis for the method. The bias and bias uncertainty determination method and illustrative results are presented. A FP bias calculation method, using data adjustment techniques with reactivity sensitivity coefficients implemented in the SCALE 6 TSURFER and TSAR codes, is under development. The technique and some typical results will be presented.

2.15

Regulatory Perspective on Computer Code Validation for Burnup Credit Criticality Analyses for Spent Nuclear Fuel Transportation Packages

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In an effort to make spent fuel transportation packages more efficient, applicants for transportation package Certificates of Compliance under Part 71 of Title 10 of the U.S. Code of Federal Regulations (10 CFR Part 71), "Packaging and Transportation of Radioactive Material," have increasingly sought burnup credit in the package criticality analysis. The U.S. Nuclear Regulatory Commission (NRC) Division of Spent Fuel Storage and Transportation published Interim Staff Guidance 8 (ISG-8), "Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks," in May of 1999, with subsequent revisions in July of 1999 and September of 2002. This document provides guidance regarding acceptable approaches to burnup credit criticality analyses for intact spent PWR assemblies in transportation packages.

This paper will discuss the technical bases supporting the recommendations in ISG-8 regarding acceptable validation techniques for isotopic depletion and criticality codes necessary for burnup credit criticality analyses. Additionally, this paper will discuss NRC staff considerations for a potential future revision to ISG-8, including new data and computational techniques, which have recently become available for isotopic depletion and criticality code validation

2.16

SCALE VALIDATION EXPERIENCE USING AN EXPANDED ISOTOPIC ASSAY DATABASE FOR SPENT NUCLEAR FUEL

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Abstract - The availability of measured isotopic assay data to validate computer code predictions of spent fuel compositions applied in burnup credit criticality calculations is an essential component for bias and uncertainty determination in safety and licensing analyses. In recent years, as many countries move closer to implementing or expanding the use of burnup credit in criticality safety for licensing, there has been growing interest in acquiring additional high quality assay data. The existing open sources of assay data are viewed as potentially limiting, due to the small number of isotopes measured in many of the early experimental programs, measurements for primarily actinides with relatively few fission products, large measurement uncertainties, sometimes incomplete documentation, and the limited burnup and enrichment range of the measured samples. Oak Ridge National Laboratory (ORNL) recently initiated an extensive isotopic validation study that includes most of the data archived in the OECD/NEA electronic database, SFCOMPO, and several new datasets obtained through participation in commercial experimental programs. To date, ORNL has analyzed approximately 120 different spent fuel samples from pressurized water reactors that span a wide enrichment and burnup range, and include a broad representation of assembly designs. The validation studies, completed with the previous version of SCALE, are being revised using the recently released version of SCALE 6 and the available ENDF/B-VII cross section libraries in SCALE. The results from the ORNL validation studies are being used to support the technical basis for implementation of burnup credit, and other applications of the data including radiation source terms, radiological dose assessment, decay heat, and waste repository safety analyses. The experimental database and validation results obtained with SCALE 5.1 and preliminary results obtained using SCALE 6 are discussed in this paper.

SESSION III: APPLICATIONS AND IMPLEMENTATION

.3.1

Regulatory Perspective on Confirmatory Burnup Measurements for Burnup Credit in Spent Nuclear Fuel Transportation Packages

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In an effort to make spent fuel transportation packages more efficient, applicants for transportation package Certificates of Compliance under Part 71 of Title 10 of the U.S. Code of Federal Regulations (10 CFR Part 71), "Packaging and Transportation of Radioactive Material," have increasingly sought burnup credit in the package criticality analysis. The U.S. Nuclear Regulatory Commission (NRC) Division of Spent Fuel Storage and Transportation published Interim Staff Guidance 8 (ISG-8), "Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks," in May of 1999, with subsequent revisions in July of 1999 and September of 2002. This document provides guidance regarding acceptable approaches to burnup credit criticality analyses for intact spent PWR assemblies in transportation packages.

One of the recommendations in ISG-8 is that the user of a burnup credit spent fuel transportation package perform a measurement that confirms the reactor record for each assembly to be loaded. This also appears as a requirement in IAEA TS-R-1, "Regulations for the Safe Transport of Radioactive Material," Paragraph 674. This measurement would be difficult for the over 1000 dry cask storage systems which are already loaded in the U.S. Unloading dry spent fuel casks would increase the potential for fuel handling incidents as well as operational dose to workers. NRC is evaluating possible alternatives to the out-of-core burnup measurement recommendation in ISG-8, including additional administrative requirements for package loading, as well as a misload analysis based on the existing spent fuel inventory. This paper will discuss considerations related to the out-of-core confirmatory burnup measurement and its proposed alternatives.

[7] Managed by UT-Battelle, LLC, for the U.S. Department of Energy under contract No. DE-AC05-00OR22725.

3.2

Fuel Burnup Plant Records: Generation and Accuracy

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An accurate evaluation of the spent fuel assemblies burnup is essential to the allowance and implementation of a burnup credit program. One of the recommendations in the U.S. Nuclear Regulatory Commission (NRC) Interim Staff Guidance 8 (ISG-8) Revision 2 is an out-of-core measurement to confirm the reactor record and compliance with the assembly burnup value used for cask loading acceptance. The NRC is currently evaluating potential alternatives to the out-of-core measurement recommendation in ISG-8. This evaluation includes the determination of the reliability and accuracy of the burnup values from the reactor records and whether these values are acceptable for use in burnup credit and what additional requirements and safeguards would be needed to allow their utilization.
The fuel assembly burnup is a core-follow parameter that is treated and controlled as safety-related data and is used extensively in reactor design and in safety analysis. It is also used in regulation compliance in the on-site spent fuel management. This paper describes the instrumentation and the measurement processes, as well as the computer codes that are used to generate the reactor records in a typical pressurized water reactor (PWR) in the United States. It also reviews the parameters important to the accuracy of these records and the variability in flux map measurements resulting from the successive methodology and code changes over the past two decades.

3.3

Recommended bounding axial profiles in BUC applications from actual burnup measurement of French PWR assemblies

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The concept of taking credit for the reduction of the reactivity of nuclear spent fuel due to their burnup is referred to as “Burnup Credit” (BUC). Allowing reactivity credit for spent nuclear fuels (SNF) offers many economic incentives. The increasing ^{235}U enrichment and need for fuel transport and storage point out the interest for BUC methods. A recent and rigorous methodology[1] was developed by the CEA and AREVA-NC, carrying out the French BUC calculation route based on the code systems DARWIN[2] and CRISTAL[3]. It is accounting of:

5. 15 poisoning FPs, stable and non-gazeous, in addition to the actinides;
6. Conservative hypotheses for the depletion calculations;
7. The qualification of the spent fuel inventory obtained with DARWIN and of the reactivity worth of BUC nuclides calculated with CRISTAL;
8. A bounding axial profile of assembly BU[4,6]. The previous method using a uniform mean BU gives a non realistic cosines axial flux and is therefore not conservative for BU > 30 GWd/t [5].

As a consequence, because of the so-called “end-effect” (low irradiation of the extremities of the irradiated assembly) the use of a burnup profile becomes necessary in criticality studies. The definition of a bounding profile to be recommended for criticality studies has been studied on the basis of a representative set of axial traces obtained from gamma spectrometry measurement in La Hague facilities on spent fuel assemblies before the dissolution step[7]. More than 200 measurements were studied here, extracted from the extensive database obtained by AREVA/LH on French 17x17 PWR-UOx FAs. This important basis gives a reliable and representative range of shapes of burnup profiles. The study shows that a simplified description of the axial profile using 11 zones is sufficient, as follows from the bottom to the top of the assembly : 11 – 22 – 33 – 45 – 73 – H-98 – H-63 – H-42 – H-23 – H-11 – H, where is the fissile column height (in cm).
We recommend the use of a profile (normalized to 1) deduced from the most conservative one among the measurements of FAs of BU > 30 GWd/t in the experimental basis. It is characterized by a maximum en effect on the top of the assembly. Its conservativeness compared with the use of a mean burnup profile amounts to 860 to 1400 pcm at the BU of 30 GWd/t, depending on the cooling time from 0 to 5 years. A more asymmetric profile is also recommended for FAs with low BU < 30 GWd/t, although the end effect becomes here less sensitive to the shape of the profile.

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3.4

Inventory Prediction and BUC Calculations Related to MEU/LEU IRT Fuels of LVR-15 Research Facility

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The LVR-15 is the one of the main research facilities of NRI at Rez, Czech Republic. As one of the older pool LWR research reactors of the Russian origin situated in many countries over the world, the LVR-15 used IRT-2M HEU fuel enriched to 80 wt. % ^{235}U until recently. It was subsequently replaced by IRT-2M fuel of 36 wt. % ^{235}U (MEU) and finally by the IRT-4M fuel of 19.75 wt.% ^{235}U (LEU) being implemented at present under the Reduced Enrichment for Research and Test Reactors (RERTR) Programme. The spent fuels are stored in the inner area of NRI - in pool at the reactor and then in the interim storage. In the course of the LVR-15 operation in NRI the original reactor was upgraded to give thermal power up to 15 MW and fuel is used up to quite a high burnup of about 60% of the ^{235}U initial amount. In the framework of the International RRRFR Project (Russian Research Reactor Fuel Return) Project of US DOE/GTRI (National Nuclear Security Administration's Global Threat Reduction Initiative) the spent fuel of the highest initial enrichment and most of fuel of the middle initial enrichment were transferred to the Mayak reprocessing plant (Russian Federation) in 2007. The rest of MEU spent fuel is being stored in NRI.
To try to optimize the storage capacity and assess burnup credit of the LVR-15 storage facilities the fuels were modeled for calculations by TRITON and KENO (SCALE 5.1). Models and results of preliminary calculations are performed and presented. The new tools for the depletion and criticality safety calculations should also support the LVR-15 operational calculations currently based on 1D depletion model (WIMSD4 code modified for IRT geometry by ANL under the RERTR program) and diffusion flux calculations (NODER, four-group 3D difussion code) in connection with the requested fast and effective conversion to LEU cores.

3.5

'FUEL DEPLETION CALCULATION IN MTR-LEU NUR REACTOR'

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This report presents the results of few groups calculations for the MTR-LEU nuclear research reactor fuel elements depletion analysis until 45 000 MWD/TU, taken as the maximum fuel burn up. The WIMSD-4 cell code has been employed as calculation tool. In this study, we are interested in some actinides such as the uranium and plutonium isotopes, as well as the fission products Xe135, Sm149, Sm151, Eu155 and Gd157.
Results of calculations in five energy groups are in a good agreement with those obtained with only two energy groups which can therefore be used in all subsequent calculations. The calculation results presented in this article can be used as a microscopic data base in order to estimate the amount of radioactive sources randomly dispersed in the environment. They can also be used to monitor the fuel assemblies' inventory at the core level.

3.6

Burnup Credit in the Swedish Interim Storage Facility (CLAB)

Lennart Agrenius

In the Swedish Facility for Interim Storage of Spent Nuclear Fuel (CLAB) the criticality safety analysis is based on the assumption that the fuel is fresh. The criticality safety criterion
$k_{eff} < 0.95$ is met during normal and accident conditions if the enrichment in the fuel is less than 4.2% U235. This is valid for both PWR- and BWR-fuel. For BWR-fuel credit for integral burnable poison is required.
The Swedish nuclear power utilities have plans to increase the enrichment in the fuel above 4.2 % U235. In order to be able to store fuel with enrichments above 4.2% U235 burnup credit has to be used in CLAB. Burnup requirements for the fuel in CLAB with enrichments up to 5% U235 were developed for the limiting BWR- and PWR-fuel types taking into normal and accident conditions, uncertainties and other effects.
The study shows that burnup credit is an acceptable way to control the reactivity in CLAB for both BWR- and PWR-fuel with enrichments up to 5% U235 using a minimum set of nuclides, actinides only. If selected fission products also are credited more margin is achieved.

3.7

Lessons learnt from OECD/NEA Phase II-C through Phase II-E Benchmarks

Jens Christian Neuber (AREVA NP GmbH, Germany)

In the OECD/NEA Phase II-C Burnup Credit (BUC) Benchmark the impact of the asymmetry of realistic axial burnup profiles of spent PWR UO_2 fuel assemblies on the axial end effect was studied. It was shown that the axial end effect increases with increasing asymmetry.
In the Phase II-D BUC Benchmark the effect of Control Rod (CR) insertion during irradiation of PWR UO_2 fuel assemblies on the spent fuel composition was studied. It was shown that CR insertion results in a change of the Spent Nuclear Fuel (SNF) isotopic inventory; and it was demonstrated that, at given initial enrichment and given burnup, SNF which was exposed to CR insertion during irradiation has a higher reactivity than SNF which has not been exposed to CR insertion.
In the Phase II-D benchmark exercises it was however assumed that the fuel assemblies were exposed to control rod insertion over their full active length; and in all cases analyzed a uniform distribution of the burnup and hence a uniform distribution of the isotopic number densities were assumed.
The Phase II-E benchmark therefore combined the asymmetry effect on the end effect with the CR insertion effect on the isotopic inventory. Thus, the objective of the Phase II-E benchmark was to study the impact of changes in the spent nuclear fuel isotopic composition due to CR insertion on the reactivity and the end effect of fuel assemblies with realistic axial burnup profiles for different CR insertion depths ranging from 0 cm (no insertion) to full insertion.
Results from the three benchmark exercises will be presented, and it will be shown that the results obtained in the Phase II-E benchmark can be understood from observations made in the Phase II-C exercises.

3.8

REVIEW OF RESULTS FOR THE OECD/NEA PHASE VII BENCHMARK: STUDY OF SPENT FUEL COMPOSITIONS FOR LONG-TERM DISPOSAL

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This paper summarizes the problem specification and compares participant's results for the OECD/NEA/WPNCS Expert Group on Burn-up Credit Criticality Safety Phase VII benchmark – Study of Spent Fuel Compositions for Long-Term Disposal.

[8] Managed by UT-Battelle, LLC, for the U.S. Department of Energy under contract No. DE-AC05-00OR22725.

After spent nuclear fuel (SNF) is discharged from a reactor, the reactivity continues to vary as a function of time due to the decay of unstable isotopes. Burnup credit analyses for storage and transport consider timeframes that are extremely short (typically less than 100 years), as compared to the timeframe of interest to long-term disposal (e.g., 10,000 years after closure in the US). The Phase VII benchmark was developed to study the ability of relevant computer codes and associated nuclear data to predict spent fuel isotopic compositions and corresponding k_{eff} values in a cask configuration over the time duration relevant to SNF disposal. Expected outcomes of the benchmark exercise include improved understanding relative to potential differences in international nuclear data sets and improved understanding and/or confidence in our ability to predict k_{eff} and source terms for timeframes relevant to long-term disposal of SNF. The benchmark is divided into two sets of calculations: (1) decay calculations out to 1,000,000 years for provided PWR UO_2 discharged fuel compositions and (2) criticality (k_{eff}) calculations for a representative cask model at selected time steps. This paper will provide detailed comparisons of the numerous (>10) participant's isotopic compositions and k_{eff} values that were calculated with a diversity of computer codes and nuclear data sets.

3.9

BURNUP CREDIT APPROACH FOR THE PROPOSED UNITED STATES REPOSITORY AT YUCCA MOUNTAIN

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The United States Department of Energy, Office of Civilian Radioactive Waste Management has submitted a license application for construction authorization of a deep, geologic repository at Yucca Mountain, Nevada. The license application is currently under review by the United State Nuclear Regulatory Commission. This paper will describe the methodology and approach used to address the issue of criticality and the role of burnup credit in the proposed repository at Yucca Mountain during the postclosure disposal time period. The most significant and effective measures for prevention of criticality in the repository include: multiple, redundant barriers that act to isolate the fissionable material from water (which can act as a moderator, corrosive agent, and transporter of fissile material); inherent geometry of the waste package internals and waste forms; presence of fixed neutron absorbers in the waste package internals; and fuel burnup for commercial SNF. A probabilistic approach has been used to screen criticality from the total system performance assessment. Within the probabilistic approach, criticality is considered an event, and the total probability of a criticality event during the disposal time period is calculated and compared against the regulatory criterion. The total probability of criticality includes contributions associated with both internal (within the waste package) and external (external to the waste package) criticality for each of the initiating events that could lead to waste package breach. The occurrence of and conditions necessary for criticality in the repository have been thoroughly evaluated using a comprehensive range of parameter distributions. A simplified design basis modeling approach has been used to evaluate the probability of criticality by using numerous significant and conservative assumptions. Burnup credit is only used for evaluations of in-package configurations, and uses a combination of conservative and bounding modeling approximations to ensure conservatism. This paper will review the United States Nuclear Regulatory Commission regulatory criteria relevant to postclosure criticality, explain the role of criticality within the overall repository performance assessment, describe the strategy for preventing criticality via design features and waste form properties, and discuss the numerous considerations relevant to criticality and burnup credit for spent fuel during disposal in a geologic repository, with emphasis on the burnup credit approach and analyses.

3.10

Regulatory issues for final disposal

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The criticality safety committee of the German institute of standardization DIN ("Deutsches Institut für Normung") is working out a criticality safety standard for final disposal of radioactive waste containing fissile materials. The first complete draft version of this standard just finished give consideration to a risk-informed approach: The criticality acceptance criteria for the pre-closure period as well as for the post-closure phase are consistently derived from the fact that the probability P_{krit} of a criticality event is given by the conditional probability $P(k_{eff} \geq 1|E)$, that the k_{eff}-value of a Nuclear Fuel System under a given event E is greater than 1, and the probability of occurrence P(E) of the event E,

$$P_{krit} = P(k_{eff} \geq 1 | E) \cdot P(E) .$$

Reasons for the maximum values allowable for P_{krit} as established for the pre-closure period and the post-closure phase in the present draft version of the DIN standard will be given. The problems related to the estimation of the probabilities $P(k_{eff} \geq 1|E)$ and P(E) will be described, and examples for interfaces with probabilistic safety analysis techniques will be given. In addition, the requirements for including balancing the risks from potential criticality excursions under repository conditions in the post-closure period against the risks resulting from safety measures, that have to be taken in the pre-closure period for ensuring prevention of criticality in the post-closure phase, will be described.

3.11

Burnup Credit in the Canister for Final Disposal of Spent Nuclear Fuel

Lennart Agrenius

In the planned Swedish repository for disposal of spent nuclear fuel the fuel assemblies will be placed in disposal canisters made of cast iron and copper.
A canister consists of an insert of cast iron with a diameter of 949 mm with a 49 mm thick outer shell of copper. The outside diameter of a canister is 1050 mm. In the BWR-insert twelve storage compartments are formed with the inner measures of 160 mm x 160 mm. In the PWR-insert there are four storage compartments are formed with the inner measures of 235 mm x 235 mm.
Calculations show that the effective neutron multiplication factor exceeds 0.95 if fresh fuel is assumed. In order to meet the criticality criterion burnup credit has to be used.
Burnup requirements for the fuel were developed for the limiting fuel types taking into account uncertainties and other effects:

Material compositions
Position of fuel assemblies in the canister
Dependence on temperature
Effect of integral burnable poison
Effect of burnable poison rods
Uncertainty in declared burnup

Axial temperature distribution
Axial burnup distribution (end effect)
Control rod history
Horizontal burnup distribution

Calculational uncertainty
Uncertainty in isotopic prediction

Demolition of fuel assemblies

Manufacturing tolerances
Long term reactivity change

Change in geometry due to burnup
Defects in the canister

The study shows that burnup credit is an acceptable way to control the reactivity in the disposal canisters for BWR- and PWR-fuel using a minimum set of nuclides, actinides only. If selected fission products also are credited more margin is achieved.

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Córdoba ciudad de encuentro | Consorcio de Turismo

HORARIO DE MUSEOS Y MONUMENTOS DE CÓRDOBA- MUSEUMS AND MONUMENTS OPENING TIMES – OCTUBRE/OCTOBER
ESPAÑOL / ENGLISH

Monumento/Monuments	Lunes – Sabado / Monday to Saturday	Domingos y Festivos Sunday and holiday	Precios/Prices
Mezquita Catedral (1) 957 47 05 12	08:30-19:00 **2/10** Misa 10:00 08:30 – 10:00 12:30 – 19:00	**08:30 - 10:00 14:00 - 19:00** Misa/Mass 11:00 - 13:00 Sagrario: 10:00 - 12:00 **12/10 Misa 11:00** 08:30 – 10:00 12:30 – 19:00	Adulto/Adult **8 €** Niño/Children **4 €** 08:30-10:00 Gratis/Free excepto domingos/except Sundays. Grupos no/no groups.
Alcázar de los Reyes Cristianos (2) (C/Caballerizas Reales) 957 42 01 51 **Museo Julio Romero de Torres (32)** (Plaza del Potro) 957 49 19 09 **Baños del Alcázar Califal (11)** (Campo Santo de los Mártires)	Martes a Viernes Thusday to Friday 08:30 – 19 :30 Sábados/Saturdays 09:30 – 16:30 Lunes cerrado/Monday Closed	09:30 – 14:30 **12/10; 24/10** 09:30 – 14:30	Adulto/Adult **4€** Estudiante/student **2 €** hasta 26 años/Till 26years old **Baños del Alcázar Califal** Adulto/ Adult **2 €** Estudiante/Student **1€** Hasta 26 años/till 26 years Miércoles Gratis Wednesday free
Sinagoga (18) (c/Judíos) 957 20 29 28	09:30-14:00 15:30 -17:30 Lunes cerrado/Monday closed	09:30 – 13:30 **12/10, 24/10** 09:30 – 13:30	Gratis UE/Free EU Resto/others **0.3 €**
Casa Andalusí (17) (c/Judíos)957 29 06 42	10:00 – 19:30	10:00-19:30	Adultos/Adults **2.5 €** Estudiante/student **2 €**
Casa de Sefarad (16) (c/Judíos esquina c/Averroes) 957 42 14 04	10:00 – 17:30 Visita Guiada Opcional/Optional Tour	Domingos-Sundays 11:00 – 14:00 Festivos/Holidays 10:00 – 17:30 Guiada/Sunday Tour 12:00	Adultos/adults **4 €** Reducido **3 €**
Medina Azahara (101) 957 35 55 06 (a 7 Km de Córdoba)	10:00 – 18:30 Lunes cerrado/Monday closed	10:00 - 14:00 **12/10; 24/10** 10:00 - 14:00	Gratis UE/ EU free Resto/others **1.5 €**
Museo Bellas Artes (32) (Plaza del Potro) 957 35 55 50 **Museo Arqueológico (49)** (Plaza Jerónimo Páez) 957 35 55 17	Martes/Tuesday 14:30 – 20:30 Miércoles a Sábado/ Wednesday to Saturday 09:00 – 20:30 Lunes cerradoMonday closed	09:00 - 14:30 **12/10; 24/10** 09:00 – 14:30	Gratis UE/ EU free Resto/others **1.5 €**
Museo Torre de la Calahorra (4) (Puente Romano s/n) 957 29 39 29	10:00 – 18:00	10:00 – 18:00	Adultos/adults **4.50** Niños/children **3.00** Pensionistas/pens. **3.00** Grupos/groups **3.00**
Mercado de Artesanía de Andalucía (c/Manriquez,4) 957 10 26 17	Martes a Sábados Tuesday to Saturdays 10:00 - 14:00 17:00 - 20:00	Domingos/ Sundays 10:00 – 14:00 Festivos/ holiday 10:00 - 14:00 17:00 - 20:00	Gratis /Free
Casa Museo Arte sobre Piel.Guadamecies Omeya(26) Pza.Agrup.Cofradías,2 957 05 01 31	11:00 - 14:00 16:30 – 20:00 **24/10**11:00 - 14:00 16:30 – 20:00 Lunes cerradoMonday closed	10:30 – 14:00 **12/10** Cerrado/Closed	Adultos **3 €**

Monumentos/Monument	Lunes – Sabado / Monday to Saturday	Domingos y Festivos Sunday and holiday	Precios/Prices
Palacio de Viana (86) (Plaza de Don Gome 2) 957 49 67 41	10:00 -19:00 Sábados / Saturday 10.00 -15.00 Lunes cerrado/Monday closed	Domingo / Sunday 10.00 – 15.00 Festivos cerrado/ closed on holiday **12/10; 24/10** Cerrado/Closed	Visita completa/full visit **6€** Sólo Patios/only Patios **3€** Si el aforo de visitas guiadas al interior del Palacio–Museo se completase, solo se venderán entradas de Patios.
Molino de Martos (38) (Ronda de los Mártires) 957 75 20 08	10:00 – 15:00 Visita guiada/Tour 09:45 cada hora/every tour **24/10** 10:00 – 15:00 Lunes cerrado/Monday closed	10:00 – 15:00 **12/10** Cerrado/Closed	Adultos/Adults **2 €** Niños/Children **1,3 €** Pensionistas/Pens. **1.3 €**
Jardín Botánico (15) (avda.de Linneo s/n) 957 957 20 00 18	10:00 – 18:00 **24/10** 10:00 – 18:00 Lunes cerrado/Monday closed	10:00 – 15:00 Domingo/ Sunday **12/10** Cerrado/Closed	Adultos/adults 2 € Niños/children 1.30 € Pensionistas/pens. 1.30 €
Yacimiento Arqueológico Cercadilla (99) (Avda Vía Augusta s/n) 957 47 90 91	Miércoles-Domingo Wednesday-Sunday 10:00 – 14:00 Grupos / Phone Groups 697 95 44 45	10:00-14:00	Entrada gratuita Free entrance
Ermitas (102) (a 15 Km. de Córdoba)	10:00 – 13:30 17:30 – 19:30 Lunes cerrado/Monday closed	Mismo horario/same time Misa domingo Mass Sunday 10:00	Adultos/adults **1.5 €** Niños/children **0.7 €** Grupo/group 20 pax
Zoológico (15) (Avd. de Linneo) 957 20 08 07	10:00 –19:00 Lunes cerrado /Monday Closed	10:00 – 19:00	Adultos/Adults **4 €** Niños/Children **2€**
Ciudad de los Niños Avd. Menéndez Pidal	10.00 - 19.00 Lunes Cerrado/Monday Closed	10:00 – 19.00	Entrada gratuita Free entrance
Museo del Aceite Carbonell 957 32 04 00 (Ctra.Nac. IV, km387.8)	Necesaria cita previa Visit booked in advance, by calling Pases: 10:00,12:00,16:00	Cerrado / Closed Sábados cerrado Saturday closed	Edad mínima 8 años
Galería de la Tortura C/ Manriquez, 1 957 47 45 08	10:00 – 19:30	10:00 – 19:30	Adultos/Adults **3€** Menores 10 años/ children under 5 years old Gratis/Free

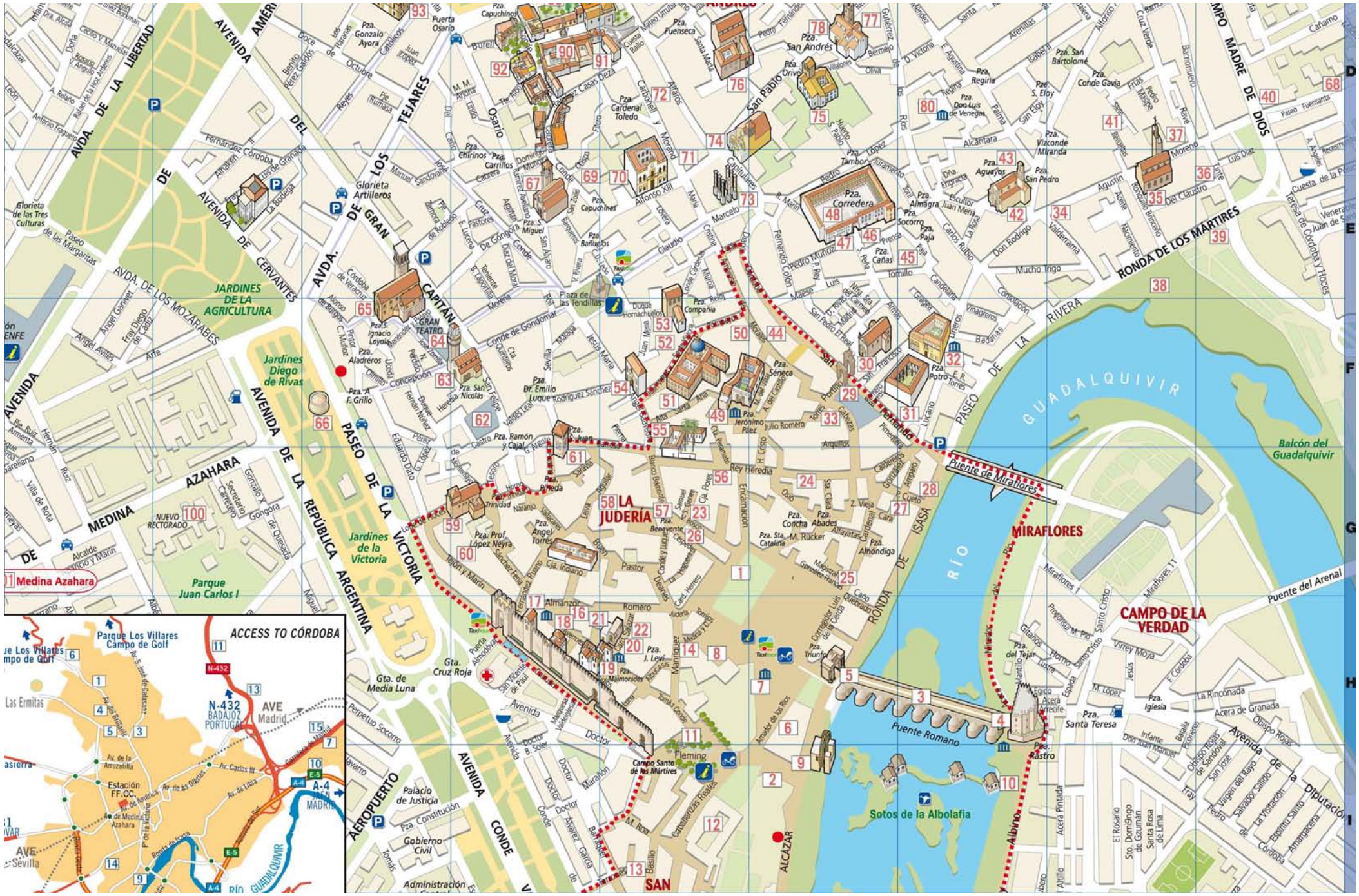
Notas: El último pase de los monumentos es media hora antes de su cierre. Los horarios y precios que se indican son los facilitados a esta Oficina de Información Turística por los monumentos, por tanto esta Oficina no se responsabiliza de los cambios que se produzcan sin notificación previa. Notes: The last visit to the monuments is half and hour before closing. The timetables and prices of entry to these monuments indicated have been provided in the different offices of each monument. Therefore this office isn't responsible for the changes that could happen at any moment.

1 Mezquita Catedral Mosque Cathedral
2 Alcázar de los Reyes Cristianos Fortress of the Christian Kings
3 Puente Romano Roman Bridge
4 Torre de la Calahorra Calahorra Tower
5 Puerta del Puente The bridge gateway
6 Seminario de San Pelagio San Pelagio Seminary
7 Palacio Episcopal y Museo Diocesano Episcopal Palace and Diocesan Museum
8 Palacio de Congresos y Exposiciones, antiguo Hospital de San Sebastián. Restos del Alcázar Omeya Palace of Congresses and Exhibitions, formerly the Hospital of San Sebastián. Remains of Omeya Fortress
9 Molino de la Albolafia Albolafia water mill
10 Molino de San Antonio San Antonio water mill
11 Baños califales Califhate Baths
12 Caballerizas reales Royal Stables
13 Iglesia de San Basilio Church of San Basilio
14 Filmoteca de Andalucía Andalusia's Film Archive
15 Jardín Botánico y Zoo Botanic Garden/Zoo
16 Casa de Sefarad Sefarad House
17 Casa Andalusí Andalusi House
18 Sinagoga Synagogue
19 Museo Taurino y Zoco Municipal Bullfighting Museum and Municipal Market
20 Capilla de San Bartolomé Chapel of San Bartolomé
21 Facultad de Filosofía y Letras, antiguo Hospital del Cardenal Salazar Arts & Philosophy School, formerly Cardenal Salazar Hospital
22 Convento de San Pedro Alcántara Convent of San Pedro Alcántara
23 Baños árabes de Santa María Santa María Arab Baths
24 Antiguo Convento de Santa Clara. Alminar de mezquita Former convent of Santa Clara. Minaret of Mosque
25 Restos de la Sala de Abluciones de Almanzor Remains of the Almanzor Ablution Hall
26 Casa Arte sobre Piel Leather Art House
27 Casa barroca en c/ Cara Baroque House at Cara Street
28 Casa barroca en c/ Amparo Baroque House at Amparo Street
29 Casa de los Marqueses del Carpio House of the Marquises of El Carpio
30 Iglesia y Claustro de San Francisco Church and Cloister of San Francisco
31 Posada del Potro Inn of El Potro
32 Museo de Bellas Artes y Museo de Julio Romero de Torres Fine Arts museum and Julio Romero de Torres museum
33 Casa de Luis de Góngora Luis de Góngora house
34 Convento de Santa Cruz Convent of Santa Cruz
35 Iglesia de Santiago Church of Santiago
36 Casa de la Encomienda de Santiago House of the Guild of Santiago
37 Palacio de los Marqueses de Benameji Palace of the Marquises of Benameji
38 Molino de Martos Martos water mill
39 Ermita de los Santos Mártires de Córdoba Hermitage of the Santos Mártires (Holy Martyrs)
40 Iglesia de Ntra. Sra. de los Remedios y San Rafael Church of Ntra. Sra. de los Remedios y San Rafael
41 Casa de las Campanas Las Campanas House
42 Parroquia de San Pedro Parish of San Pedro
43 Casa de los Aguayo. Colegio de las Francesas House of the Aguayo. Las Francesas School
44 Teatro Cómico Principal Cordoba's Main Comic Theatre
45 Convento de la Piedad La Piedad Convent
46 Mercado Sánchez Peña Sánchez Peña Market
47 Casa Doña Jacinta Doña Jacinta House
48 Plaza de la Corredera Corredera Square
49 Museo Arqueológico Archaeological Museum
50 Convento del Corpus Christi. Fundación Gala Corpus Christi Convent. Gala Foundation
51 Colegio e Iglesia de Santa Victoria School and Church of Santa Victoria
52 Iglesia de Santo Domingo. Archivo Histórico Provincial Church of Santo Domingo Provincial Archives
53 Iglesia y Colegio de la Compañía School and Church of La Compañía
54 Palacio del Marqués de la Fuensanta del Valle. Conservatorio Superior de Música Marquess of the Fuensanta del Valle Palace. High School of Music
55 Iglesia Conventual de Santa Ana Santa Ana Convent Church
56 Iglesia Conventual de la Encarnación Convent Church of La Encarnación
57 Escuela de Arte Dramático. Palacio de las Quemadas Drama School. Palace of Las Quemadas
58 Iglesia del Hospital Jesús Crucificado Church of the Hospital Jesús Crucificado
59 Iglesia Parroquial de la Trinidad Parish Church of La Trinidad
60 Casa de los Guzmanes. Archivo Histórico Municipal House of the Guzmanes. Municipal Archives
61 Iglesia de San Juan. Alminar árabe Church of San Juan. Arab Minaret
62 Casa de los Venegas Henestrosa. Gobierno militar House of the Venegas Henestrosa. Military Government
63 Parroquia de San Nicolás de la Villa Parish of San Nicolás de la Villa
64 Gran Teatro Grand Theatre
65 Iglesia de San Hipólito Church of San Hipólito
66 Mausoleos Romanos Roman Mausoleums
67 Iglesia de San Miguel Church of San Miguel
68 Santuario de Nª Sra. de la Fuensanta Santuary of Nª Sra. de la Fuensanta
69 Convento de las Capuchinas Las Capuchinas Convent
70 Círculo de la Amistad Círculo de la Amistad
71 Instituto Politécnico Polytechnic High School
72 Convento del Císter Convent of El Císter
73 Templo Romano y restos de murallas. Ayuntamiento Roman Temple and remains of walls. Town Hall
74 Iglesia Conventual de San Pablo Convent Church of San Pablo
75 Palacio de los Villalones Palace of the Villalones family
76 Convento de Santa Marta Convent of Santa Marta
77 Parroquia de San Andrés Parish of San Andrés
78 Casa de los Luna House of the Luna Family
79 Palacio de Muñices Palace of Muñices
80 Museo Regina Regina Museum
81 Iglesia de la Magdalena Church of La Magdalena
82 Convento del Carmen Convent of El Carmen
83 Iglesia Parroquial de San Lorenzo Parish Church of San Lorenzo
84 Iglesia del Juramento de San Rafael Church of Juramento de San Rafael
85 Iglesia Conventual de San Agustín Convent church of San Agustín
86 Palacio de los Marqueses de Viana Marquises of Viana Palace
87 Iglesia Parroquial de Santa Marina Parish Church of Santa Marina
88 Convento de Santa Isabel de los Ángeles Convent of Santa Isabel de los Angeles
89 Iglesia Conventual del Santo Ángel (Capuchinos). Cristo de los Faroles Convent church of Santo Angel (Capuchinos). Cristo de los Faroles
90 Hospital de San Jacinto. Iglesia de los Dolores San Jacinto Hospital. Church of Los Dolores
91 Casa Palacio del Bailío. Biblioteca Viva de al-Andalus Palace of El Bailío. Biblioteca Viva de al-Andalus Library
92 Palacio de Torres Cabrera Torres Cabrera Palace
93 Iglesia y antiguo Convento de la Merced. Diputación Provincial Church and former Palace/Convent of La Merced. Provincial Government
94 Torre de la Malmuerta Malmuerta Tower
95 Casa de Paso de la Lagunilla Way House at la Lagunilla
96 Iglesia Conventual de San José (San Cayetano) Convent Church of San José (San Cayetano)
97 Iglesia parroquial de Nuestra Señora de Gracia (PP. Trinitarios) Parish Church of Nuestra Señora de Gracia (PP. Trinitarios)
98 Restos arqueológicos Archaeological Remains
99 Zona arqueológica de Cercadillas Archaeological area of Cercadillas
100 Rectorado de la Universidad The university Rectory
101 Conjunto arqueológico de Madinat al-zahra Madinat al-zahra archaeological site
102 Las ermitas The Hermitages
103 Casa Patio C/ San Basilio, 50 Casa Patio at 50, San Basilio street

Visitable Visitable

SIMBOLOS / SYMBOLS
- Oficina de turismo / Tourist office
- Museo / Museum
- Bus Medina Azahara
- Aparcamientos / Parking
- Autobuses / Buses
- Taxi / Taxi
- TaxiTour
- Ferrocarril / Railways
- Coches de Caballo / Horse Carriage
- Policía / Police
- Servicios sanitarios / Medical Assistance
- Gasolinera / Petrol station
- Reserva / Nature reserve
- Límite Patrimonio de la Humanidad / World Heritage Limit

ACCESOS A CÓRDOBA / ACCESS TO CÓRDOBA

 AVDA. DE LA LIBERTAD
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AVDA. DE GRAN CAPITÁN
AVDA. DE LOS MOZÁRABES
AVENIDA DE MEDINA AZAHARA
AVENIDA DE LA REPÚBLICA ARGENTINA
PASEO DE LA VICTORIA
AVENIDA CONDE DE VALLELLANO
AVENIDA DEL AEROPUERTO
RONDA DE LOS MÁRTIRES
CAMPO MADRE DE DIOS
PASEO DE LA RIVERA
RONDA DE ISASA
Glorieta de las Tres Culturas
Paseo de las Margaritas
JARDINES DE LA AGRICULTURA
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Jardines de la Victoria
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Plaza de las Tendillas
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Pza. San Andrés
Pza. Capuchinos
Pza. Gonzalo Ayora
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Pza. Bañuelos
Pza. Compañía
Pza. Séneca
Pza. Jerónimo Páez
Pza. Dr. Emilio Luque
Pza. S. Ignacio Loyola
Pza. Aladreros
Pza. A F. Grillo
Pza. San Nicolás
Pza. Ramón y Cajal
Pza. S. Juan
Pza. Pineda
Pza. Trinidad
Pza. Prof López Neyra
Pza. Angel Torres
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Pza. Almagra
Pza. Socorro
Pza. Cañas
Pza. Aguayos
Pza. San Pedro
Pza. Regina
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Pza. S. Eloy
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Pza. San Bartolomé
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Pza. Potro
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Puente de Miraflores
Puente Romano
Puente del Arenal
Sotos de la Albolafia
Campo Santo de los Mártires
Fleming
Gta. de Media Luna
Gta. Cruz Roja
Puerta Almodóvar
Palacio de Justicia
Gobierno Civil
Administración Central
Medina Azahara
Taxi
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I
ACCESS TO CÓRDOBA
Parque Los Villares Campo de Golf
Las Ermitas
N-432 BADAJOZ PORTUGAL
AVE Madrid
A-4 JAEN MADRID
E-5
Estación FF.CC.
Av. de América
Av. de las Ollerías
Av. Carlos III
Av. de Libia
Av. de Medina Azahara
Ronda de Isasa
AVE Sevilla
RÍO GUADALQUIVIR