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Data Requirements and Maintenance of Records for Spent Fuel Management: A Review

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FOREWORD

One of the vital issues for long term management of spent fuel is the retention of appropriate information which must accompany the spent fuel itself for the time span required down to its endpoint. Due to safety, safeguards and security implications, as well as for operational needs, the significance of spent fuel data management will persist as long as the spent fuel has to be managed for its lifetime. Data collection and maintenance for spent fuel are required from the earliest phase of any project for spent fuel management. From a practical point of view, however, it is a challenge to define which data must be retained, for how long it must be kept and by what methods.

Spent nuclear fuels contain some nuclides of special importance in data management. While the fissile content is the key interest for recovery by reprocessing, the minor actinides are of principal concern in the case of spent fuel disposal because of their long half-life and radio-toxicity. The recent concept of total cycle management in repository performance requires accounting of all the nuclides data, while data on the spent fuel content is required for such a case as direct reuse of spent fuel by re-fabrication without separating particular radio-nuclides. It is quite obvious that adequate data must be available when necessary, in order to make an informed decision either for technical or administrative issues.

In the area of radioactive waste management, several TECDOCs have been issued by the IAEA on related subjects, i.e.

- IAEA-TECDOC-1097, Maintenance of Records for Radioactive Waste Disposal (1999).
- IAEA-TECDOC-1222, Waste Inventory Record Keeping Systems (WIRKS) for the Management and Disposal of Radioactive Waste (2001).
- IAEA-TECDOC-1398, Records for Radioactive Waste Management up to Repository Closure: Managing the Primary Level Information (PLI) Set (2004).

Whereas some of the information provided by these publications is applicable to spent fuel management, there are technical particularities pertaining to spent fuel management. This TECDOC is intended to provide additional information, by reviewing issues and identifying relevant data required for spent fuel management.

The contributions from various experts to this TECDOC work are highly appreciated. The IAEA officer responsible for this publication was J.S. Lee of the Division of Nuclear Fuel Cycle and Waste Technology.

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1. INTRODUCTION

1.1. Background

The growing inventory of spent fuel in Member States is a challenge to be resolved for a sustainable development of nuclear energy in the future. In view of this trend, keeping good track of the information on the spent fuel and its management facilities has been gaining more and more importance.

A number of countries and organizations, especially those involved in radioactive waste management, are operating or trying to establish a system for information management, as required by relevant stakeholders, which would be equally applicable to spent fuel data management. The positions of stakeholders in the nuclear area have evolved with time, as witnessed in the past decades which considered the impacts of information management on institutional arrangements.

Aside from the reinforcement of legal and regulatory systems on national and international levels, there has also been a general trend toward public concern on issues associated with nuclear safety and radioactive waste management, which has strengthened the need of better communications with the public and of a transparency on related information, demanding more information to be made public in the case of nuclear issues on spent fuel and waste management, including data on storage inventory or transportation as well as on the endpoints of spent fuel management such as reprocessing or disposal [1].

Key data on spent fuel must be available to make an informed decision and implement technical options for spent fuel management, either for storage, transportation, reprocessing or disposal of the fuel. Such data is required for the lifetime management of pertinent facilities for spent fuel management, in particular for the design, licensing, construction and operation of facilities or equipment. A systematic approach to the identification, capture, organization, maintenance, retrieval and protection of such data, as a part of a life cycle management of information on the spent fuel management facilities, gives the additional assurance that regulatory requirements are met, that only those records that need to be retained are maintained and that the continuing safety of the installations and/or equipment related to spent fuel is ensured.

1.2. Objective and scope

The main objective of this TECDOC is to address the question of what data needs to be gathered and how it should be managed. In order to answer such a question, an appropriate approach has to be identified. The first step is to identify the issues relevant to spent fuel management, including those required for safety analyses. The second step is to describe those issues in terms of data parameters suitable for use in database systems that could be operated either by the utilities or on a national basis.

As a first step in the implementation of these objectives, the various stages in the spent fuel-management routes from AR[1] to AFR storage leading to conditioning for disposal, refabrication or reprocessing are identified, with a focus on the data needed at each stage which could affect subsequent safe handling and treatment of the spent fuel.

This objective entails a subsequent question on how to manage those data for the long term to pass it to future generations. Historical experiences show that record keeping over centuries is

[1] Abbreviations are given at the end of this publication.

prone to various risks. This is an issue that some countries have begun to examine to take appropriate measures.

A variety of information generated from the activities involved in the life cycle management of spent fuel and radioactive waste can be classified in various categories of data in accordance to the technical nature, usage, etc.

The spent fuel covered in this TECDOC is assumed to be discharged from power reactors[2]. Although the scope of this TECDOC is intended to apply primarily to fuel from commercial power reactors, some of the considerations presented will also be applicable to fuel from research reactors. The specific requirements to meet safeguards, licensing and operational needs, each of which could impose additional demands for data acquisition and recording, are not addressed. Although not stated above, data recording refers not only to the non-reprocessed UOX spent fuel, but also to spent MOX fuel. Records for high level waste arising from spent fuel reprocessing, are outside the scope of this report, even though there are technical commonalities between disposal of spent UOX/MOX fuel and HLW and thus of their data management.

The accounting boundary for inventory of spent fuel begins with its discharge from the reactor and is interfaced with spent fuel management options, including reprocessing, disposal, or other future options. The identity of a spent fuel assembly can be lost, such as in reprocessing, which will be the end of assembly lifetime.

1.3. Identity of spent fuel

The boundary with which spent fuel is retained in an identifiable physical form in the back end of the fuel cycle is illustrated in Figure 1.

An approach to facilitate the identification of spent fuel is to use technical attributes which can be grouped into form and content.

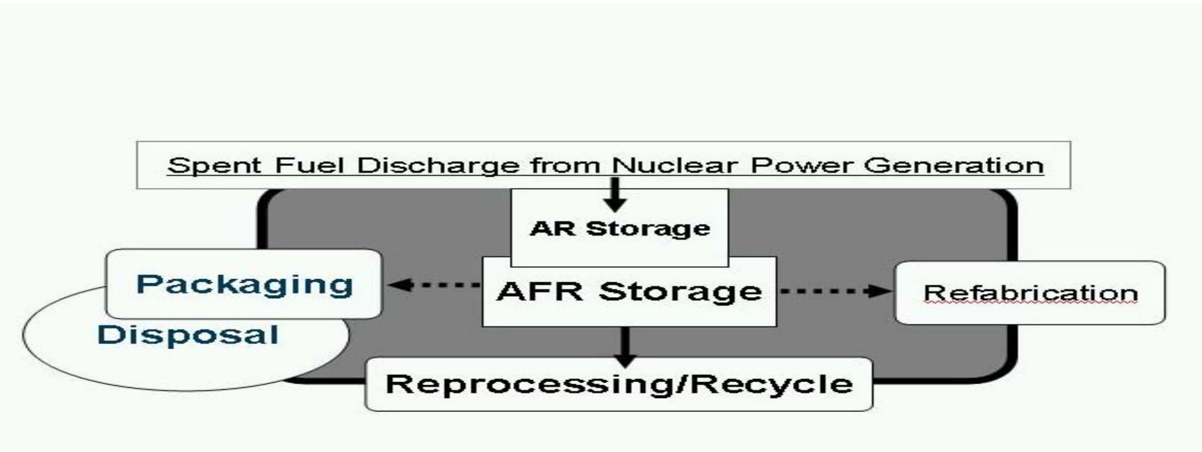

FIG. 1. Boundary of fuel cycle back end with integral form of spent fuel.

[2] However, experimental fuels that have been irradiated for research or development purposes in a power reactor and, thereafter, are part of the inventory of stored fuel at the reactor site, are included.

1.3.1. Spent fuel form

The spent fuel described in this publication refers to reactor systems employing "fuel assemblies (or bundles)" as the identifiable units. The term “assembly” is used here to mean a geometrically identifiable fuel configuration bearing (or at least capable of bearing) a durable form of identification that can, under normal circumstances, be determined by some method (e.g. optically, electronically or from pre-irradiation records) following discharge of the assembly from the reactor.

Nuclear fuel that is not constructed in the form of an assembly (such as pebble-bed reactor fuel) and therefore cannot be designated by a unique identifying number would need some special methods for proper identification of fuel forms. In the case where such fuel may have been removed from an assembly and encapsulated in other containers for storage (e.g. HTR fuel in Germany), each storage container must be assigned an identification number and its subsequent movements tracked. The same applies for individual fuel rods separated from the fuel assembly or separated from the structural parts of the fuel assemblies as long as they may be continuously tracked.

In the case of reprocessing, data recording and assembly tracking ends at the head-end of the reprocessing line where spent fuel assemblies are mechanically disassembled, i.e. when spent fuel assembly looses its identity.

For spent fuel disposal with or without fuel consolidation the available spent fuel database can be used for the disposal purposes or can be transformed to the disposal database. In the case of spent fuel disposal with fuel consolidation, the entire spent fuel assembly data must be separated into the spent fuel rod data in the fuel disposal canister and into the assembly structural data (i.e. that part not containing fuel material.)

In the case of refabrication, data recording and assembly tracking end with the start of the refabrication line — that is, when the fuel material loses its unique identifier by destruction of the assembly identy.

1.3.2. Spent fuel content

The content of spent fuel can be specified by such key parameters as chemical composition and isotopic constitution with radiological characteristics. Even after loosing the identifiable form of a spent fuel assembly, the material content can be tracked in bulk form within a process or between the adjacent stages of the nuclear fuel cycle by analyzing the material balance in terms of inventories and composition. At this time, the accounting method must be changed to a suitable one (e.g. bulk accounting) in order to maintain the balance, as the assembly identity no longer exists.

1.3.3. Simulation of spent fuel flow or cycle

It is possible to track the material flows and balances, even after the loss of assembly identity, through the entire fuel cycle by simulation techniques, along with an estimation of the isotopic composition of each stock at every stage of the fuel cycle.

The IAEA has developed a simulation system named VISTA in conjunction with the fuel cycle database NFCIS (Nuclear Fuel Cycle Information System) which is available from its web site [2].

1.4. Life time management of spent fuel information

For the life cycle consideration of information management, a classification suggested by the IAEA has 3 hierarchical levels:

- Primary Level Information (PLI),
- Intermediate Level Information (ILI),
- High Level Information (HLI).

According to this classification, the major scope of the information on spent fuel management addressed in this TECDOC belongs to the PLI (primary level information) for the radioactive waste management. The Waste Inventory Record Keeping System (WIRKS) is a subset of the PLI which comprises information generated prior to pre-closure of repository for radioactive waste disposal. In the case of spent fuel disposal, it includes all the stages from spent fuel discharge to final disposal [3].

The higher levels of information, ILI and HLI, correspond to the later phases of waste disposal, institutional control period and post-closure phase respectively [4].

1.4.1. Spent fuel data and database

In general, data to describe in detail the condition of the spent fuel at discharge from the reactor must be collected and maintained.

The technical characteristics of spent fuel are mainly determined by its history both in-reactor and following discharge (particularly during interim storage and transportation). These characteristics influence a number of subsequent activities such as siting, design, construction, operation, licensing, supervision and decommissioning of AR and AFR storage facilities and of subsequent fuel cycle back end facilities such as reprocessing plants, disposal facilities etc. Attention should also be paid to spent fuel transport, which links most stages of remaining spent fuel management stages (see Figure 1).

TABLE 1. SPENT FUEL DATA AND ASSOCIATED INFORMATION

DATA	ASSOCIATED INFORMATION
Inventory	location, number of fuel assemblies, future or planned spent fuel management route
Burnup	minimum, average and maximum burnup; quantities of fissile materials consumed
Unusual Fuel Conditions	identification of fuel historical events and of modifications to the original fuel element configuration
Basic Physical and Chemical Characteristics	fuel element geometry, size, shape, gross weight, volume and chemical composition
Heavy Metal Isotopic Mass Loading	beginning of life and end-of-life masses
Reactor Operating History	power levels, shutdowns, position of assemblies in reactor core

In practical applications, spent fuel data are input to a database for easy accessability by users. While most of the spent fuel data is generally available by a manual control off-line from the services responsible for those databases, an increasing number of on-line databases with continuous or frequent updates are becoming available. Sometimes the databases are linked to a simulation software to provide an integrated information on spent fuel management system, as is the case of the IAEA database NFCIS integrated with simulation code VISTA indicated above[3].

1.4.2. Record keeping

The spent fuel data should be retained by methods that will ensure their maintenance for the periods of preservation. The records should be available, when required, to a responsible body, such as a licensing agency, and should be maintained until that authority considers that the records are no longer required.

In the case of long term maintenance of records, such as for spent fuel disposal, selection of appropriate data to be retained by the operator will be a challenging issue, due to the various factors to be considered under the uncertainties involved in the long term. Similar questions could be raised by the generators of information on the scope and duration of record keeping. In the case of spent fuel disposal, a small fraction of the large amount of the PLI and ILI would be essential to be passed to future societies, while the rest could be considered as non-essential (or non-useful) for understanding the repository and its associated risks [5].

For more details on record keeping, the IAEA publications on the subject are referred.

1.5. Structure of the publication

This TECDOC is intended to provide some insights on a rational approach to spent fuel data management, considering the common requirements involved in spent fuel management for any Member State. In this regard, the information provided in these sections is mostly generic.

After the Section 2 on data requirements for spent fuel management, Section 3 examines technical parameters that could specify spent fuel characteristics and associated conditions, followed by Section 4 on life cycle management of spent fuel data which includes the maintenance of records and other issues.

Finally, some specific examples of the approaches already developed by a number of utilities and national organisations to characterise and track their spent-fuel data are presented in the Annex.

[3] The NFCIS is integrated with a simulation tool named VISTA which can calculate various statistics on fuel cycles including mass flow and projections. This database was also published in hardcopy and is being refurbished to make it available on the web with enhanced features as well as quality of data (www-nfcis.iaea.org). A TECDOC on VISATA is in preparation.

2. DATA NEEDS FOR SPENT FUEL MANAGEMENT

2.1. Data associated with spent fuel management

This section discusses the importance of making relevant data available for the spent fuel management in the fuel cycle back end. The sequence of activities for data management follows up the actual movement of spent fuel pattern in the fuel cycle back end; i.e. spent fuel to be turned over to the subsequent destination in the back end of the fuel cycle, together with its data on its identity and specifications, by a manifest (Fig. 2)[4].

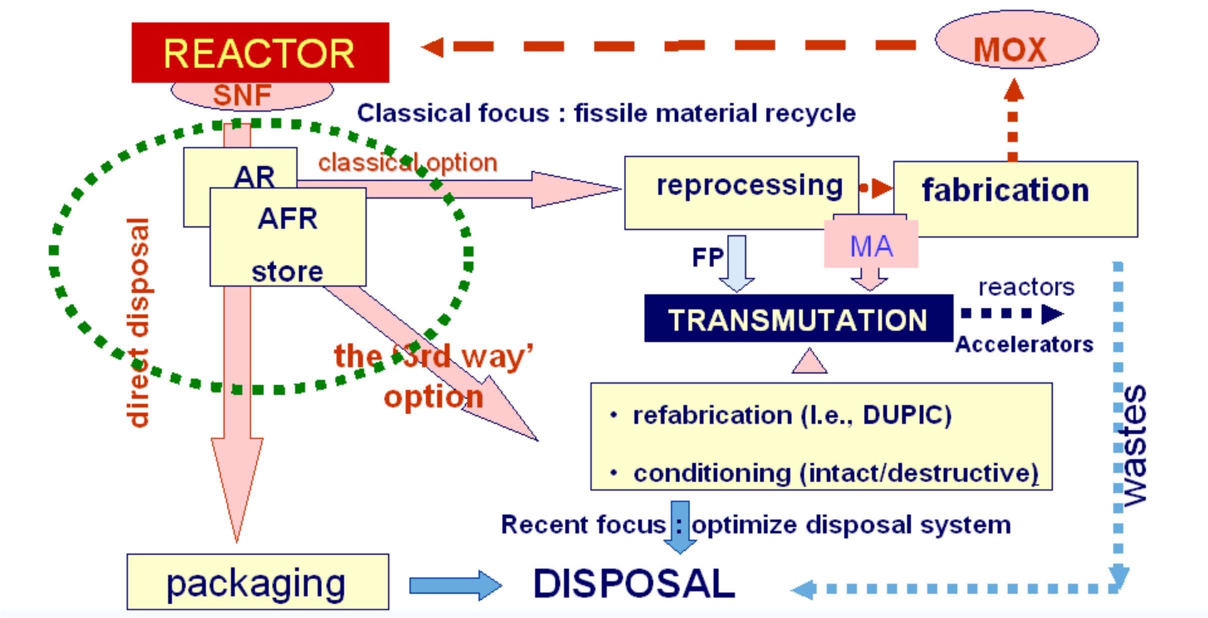

FIG. 2. Spent fuel management options for fuel cycle back end.

With respect to continuous spent fuel data maintenance, it is crucial to ensure that maintenance of data is not interrupted between a nuclear power plant closure and the beginning of the operation of a fuel cycle back end facility (e.g. deep geological disposal). In some cases this period may cover many decades.

2.1.1. Types of fuel

A clear definition of spent fuel identity is essential for handling large amount data arising from spent fuel management.

A variety of nuclear fuels have been developed for a number of reactor types. All currently used fuel types such as UO_2, MOX, CANDU fuel, etc. from power reactor types are listed in Table 2. Technical information on commercial nuclear fuel with evolving designs is available in the literature, such as Nuclear Engineering International, which provides regular updating on fuel designs [6]. More detailed specification of some of the fuel types can be found in the relevant country reports provided in the Annex.

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[4] List of the kind and quantity of items for a shipment.

TABLE 2. SUMMARY OF POWER REACTOR FUEL AND MANAGEMENT
(NU = natural uranium, SEU/LEU/HEU = low/slightly/highly enriched uranium)

TYPE	DESIGN	SPECIFICATION	MANAGEMENT
LWR	PWR BWR WWER	Cubic/hexagonal x-section, 4~5m long, 400~500 kg weight assembly, LEU	• Usually stored intact • Can be consolidated • Partly reprocessed
PHWR	CANDU	Ø 10 × 50 cm, 20 kg bundle, NU/SEU	• Handled in tray/basket • Once-through cycle
GCR	Magnox AGR	Magnox: Ø 3cm × 1.1m long slug, 24 cm diameter, NU, AGR: 1m long bundle, SEU	• Need to reprocess • Dry storage possible
OTHERS	RBMK	Ø 8cm × 10m long assembly, SEU	• Rods consolidated to a suitable size for storage • No reprocessing
	PBMR	Ø 6cm spherical form fuel element, LEU/HEU	• Canning • Possible to reprocess

2.1.2. Generation of fuel data

Among the data needs, the data described in this publication is required for licensing nuclear fuel cycle back end facilities, for implementing international obligations including safeguards, and throughout the fuel cycle for licensing and operating systems to transport the spent fuel.

2.1.2.1. Fresh fuel

In the case of spent fuel management, the originator of the physical form of fuel is the manufacturer who ships it to a reactor operator, together with the pertinent manifest with information on the fuel specifications.

Fuel fabricators are required by regulation to retain complete records of critical data generated for the length of time the fuel can be expected to be in the reactor. This pertinent fuel fabrication data is useful for purposes of assessing the quality of the fuel in reference to the chemical and isotopic analyses of fuel materials and the quality control data on the non-fuel components of the assemblies, including material specifications, acceptance testing, and production records.

2.1.2.2. Spent fuel

During its insertion in the reactor core, fuel data such as burnup, age, thermal power, etc. are used by reactor operators for a variety of purposes including fuel cycle optimisation, fuel reload planning, fuel performance warranty, and for special nuclear material reporting. Such records are administratively controlled and are tied to specific fuel assemblies by the fuel assembly identification number stamped on the upper fitting of the assembly. If the fuel is reconstituted at the reactor, the reconstitution records would provide the traceability from the reconstituted assembly identification number to the original fuel assembly data. As such information is required for special nuclear material control purposes, it must be obtained and retained.

The operator's responsibility at this stage is to store the manufacturers' data and any data concerning fuel modifications. Moreover, the power history and results of routine radionuclide content calculations are to be documented as well.

A spent fuel database including discharge dates, assembly types, burnups, initial enrichments serves as a basis for calculating the gamma, neutron and thermal source intensities and other radiological characteristics which are used for the design of spent fuel storage facilities, transport casks, reprocessing or refabrication plants or repositories.

2.1.3. Classification of spent fuel

An important technical classification in terms of physical integrity of spent fuel is needed for various operations in spent fuel management.

Although there is no international norm developed yet, an example of categorization adopted by USDOE in its standard contract with utilities for delivery to repository for disposal of spent fuel is presented in Table 3 [7].

2.1.4. Failed (damaged) fuel

Among the categories of spent fuel in the Table 3, the third category classified as "failed" fuel[5] deserves special care because of its significance in the actual operation and associated regulatory requirements.

Issues associated with dealing with failed fuel, including management of data, are discussed at both national and international levels [8].

TABLE 3. SPENT FUEL CATEGORISATION BY USDOE

FUEL CATEGORY	CLASS	REMARK
Standard Fuel	• S-1 (PWR) • S-2 (BWR)	• S-1 <(L = 14' 10", X = 9' × 9') • S-2 <(L = 14' 11", X = 6' × 6')
Non-Standard Fuel	• NS-1 (Physical dimensions • NS-2 (NFBC) • NS-3 (Short-cooled) • NS-4 (Non-LWR type) • NS-5 (Consolidated)	• Spent fuel that does not meet one or more of the General Spec. • NFBC = Non-fuel components including, but not limited to, control spiders, burnable poison rod assemblies, etc. which do not require special handling
Failed Fuel	• F-1 (Visual failure or damage) • F-2 (Radioactive Leakage) • F-3 (Encapsulated)	• Assemblies which are structurally deformed or damaged cladding • Assemblies that can not be handled with normal handling equipment for any reason • Previously encapsulated assemblies

[5] There exists currently no internationally agreed definition of glossary for this category of spent fuel (it is sometimes called 'damaged, defect, leaking, etc.)

2.2. Data requirements for spent fuel management

A variety of technical data on spent fuel would be required for various purposes in spent fuel management. The focus of this TECDOC is mainly associated with specification of spent fuel assemblies in identifiable form and characteristics that would be essential for life cycle management of concerned facilities[6].

The initial data (fuel assembly type, identification, initial enrichment, basic physical dimensions and mass, material composition, etc.) applies to both fresh and spent fuel and to both UOX and MOX fuel. Such data comes mainly from the manufacturers' database and is normally required by:

- National regulatory bodies and/or other governmental organizations, which are usually responsible for the safeguarding of nuclear material at the national level and for the licensing and control of spent fuel management facilities,
- National radioactive waste management authorities (if established) for future management of spent fuel after it is declared as radioactive waste, if such a procedure is required by a national legislation, and
- Environmental agencies involved in e.g. environmental impact assessment process.

The data on safety are required for a description of the fuel at various stages of the back end route to ensure that:

- spent fuel assemblies can be handled safely and effectively,
- decay heat output can be estimated for the design of both storage and back end facility operations and equipment, and for subsequent storage/disposal facility design,
- criticality safety criteria can be established and appropriate equipment and procedures to meet them can be developed,
- potential radiation fields can be determined for the design of storage or disposal facilities,
- adequate shielding, equipment and measures are provided to protect storage and back end facility workers and the public from radiation exposure,
- appropriate procedures can be established and equipment developed to safely and, economically process the assemblies toward their final destination, and
- licensing requirements for safe facility operation can be met.

The data that Section 3 recommends be maintained is not an exhaustive list covering all potential fuel/reactor types and situations; therefore, it may be necessary to maintain additional data records in some cases. Alternatively, some of the recommended data may not be relevant to some specific or national situations. The need for either more or less data to ensure the safe management of spent fuel in the back end route should be determined in individual applications.

2.2.1. Spent fuel storage facilities

A summary of technical information on spent fuel storage facilities is reported in various IAEA publications [9].

[6] There are an extensive list of data that would be needed to study the integrity of spent fuel for long term storage, for example, which are not covered in detail in this TECDOC (see, for example, reference 41).

TABLE 4. SUMMARY OF STORAGE SYSTEMS

TYPE	OPTION	CONTAINMENT	SHIELDING	EXAMPLES
Wet	pool	water/building	water	Common to all pools
Dry	metal cask	cask lid	metallic wall	CASTOR Series, TN Series, NAC series, MC-10
	concrete cask	canister	concrete cask/overpack	Hi-Star/Storm, CONSTOR,
	concrete module	canister	concrete module	NUHOMS, MACSTOR
	vault	canister	concrete vault	Wylfa, Paks
	drywell/tunnel	canister	drywell	not commercial

2.2.1.1. At reactor (AR) storage facilities

Spent fuel storage is usually regarded as the first stage in the back end of the nuclear fuel cycle after discharge from the reactor. An AR storage facility is integral to or associated with the reactor refuelling operation and buffer function.

All data needed for the subsequent handling of the spent fuel is derived from the operator data. The operators' spent fuel data will be used for any further calculation needed in subsequent steps of spent fuel management. These calculations may be needed for nuclear safety and radiation protection and usually cover:

- radionuclide inventory of spent fuel (currently at least safeguards relevant radionuclides are calculated in terms of mass and activity);
- burnup calculation (the credit of burnup is often used for wet type storage facilities);
- criticality calculation (loading pattern of reactor pool);
- thermal load of spent fuel.

2.2.1.2. *Away from reactor (AFR) storage facilities*

Spent fuel may be transported from the AR facility to the AFR storage facility, either at the licensed reactor site or at some distance from the reactor. The technology for AFR storage of spent fuel may be a wet or dry one.

In principle the needs for spent fuel data for wet type AFR facilities are the same as AR storage. It is important to note that, under normal conditions, the detailed information about wet storage conditions is not crucial for the purposes of spent fuel database development. When the spent fuel is to be moved from wet storage to a transport cask spent fuel data is needed to determine safe transport cask loading pattern and other conditions for safe transport of the spent fuel.

Dry storage facilities include casks, silos and vaults. The spent fuel data needed for such storage are defined in the acceptance criteria for the storage facility and in the license approval for the package unit, if needed, (e.g. for dual-purpose cask.) including not only spent fuel data itself (geometrical data, enrichment, mass of heavy metal (pre-irradiation) per assembly, burnup values), the condition of spent fuel (leakage rate, mechanically damaged, stress values, etc.), but also the number of spent fuel assemblies in a cask, and the exact spent fuel assembly identification, the moisture content of fuel/loaded canister, the canister design features (for canned fuel) such as weight, volume, handling features, etc.

Changes in spent fuel characteristics must be reflected in the acceptance criteria based on new safety analyses and may require changes in the license or approval of the dual-purpose cask.

2.2.2. *Spent fuel transportation systems*

Transportation is the vital link between all stages of spent fuel management. Therefore, an extensive set of spent fuel data is necessary to evaluate and perform safe transport of spent fuel assemblies between different nuclear installations, e.g. between the AR storage pool of nuclear power plant and the AFR dry storage facility, the reprocessing plant, etc.[7]

The spent fuel specifications related to the transport process are not much different from the information required for either AR and/or the AFR storage. Most transport casks are licensed for spent fuel with specific initial enrichment, burnup and thermal load. To meet the requirements for utilizing these licensed casks, the values of these parameters for the materials proposed for loading into the cask are added together with the total cask inventory and spent fuel identification are used as basic data to determine if transport of the spent fuel can be safely accomplished.

2.2.2.1. Procedure for data handling

An example of the sequence of activities performed for each spent fuel transport campaign is as follows:

(1) A list of spent fuel assemblies to be shipped (spent fuel assembly ID, number of spent fuel assemblies, mass of heavy metal per assembly — pre-irradiation, NFBC densities, masses and volumes, thermal output) is compiled initially by the utility and provided to both the receiver and shipper of the fuel.
(2) Comparison of the transport list and the spent fuel acceptance criteria at the receiving facility may result in special acceptance conditions.
(3) If the information provided for in the spent fuel transport list is insufficient to support either a transport safety case or assess the suitability of a given transportation cask for a shipment campaign, detailed reactor fuel data sheets are developed for each type of spent fuel assembly in accordance with the utility's quality management system. This allows to prepare new transport or storage safety cases
(4) A shipping cask is tentatively selected depending upon the physical limitations at the reactor (adequate crane, space for handling), the fuel characteristics (for the support frame), any possible limitation on the transport route and the cask availability.
(5) The packaging configuration (type of cask body, type of fuel support frame, type of fuel spacer, type of special internal needs such as a fuel canister) is then specified.
(6) The utility provides to the shipper the fuel pool positions and the confirmation of the individual burnup of each spent fuel assembly — to ensure the best configuration in terms of cask external dose rates.
(7) The cask transport documents are then prepared and include records of all prior inspections in compliance with the cask certification and the required regulatory information.

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[7] Transportation activities are often assigned to specialized services or businesses with contracts between the shipper and receiver.

(8) At the reactor site the shipper is usually responsible for the application of correct operating procedures and (i) controls the fuel identification and burnup verification, (ii) supervises the sampling and analysis of the cask cavity water, (iii) certifies the cask cavity pressure, (iv) inspects each cask orifice component's sealing face and gaskets, verifies torques specified by the cask safety analysis report and certifies the leak test results, (v) supervises the contamination surveys, dose rate levels and temperature measurements, (vi) applies the regulatory labels on the cask and transport vehicle, (vii) is responsible for proper transport vehicle turnaround inspection and maintenance.

2.2.2.2. Tracking of spent fuel

An example of an integrated software package designed to facilitate tracking and optimisation of the various stages of the life cycle of the fuel assembly is FuelWorks, developed by EPRI, which tracks each fuel assembly from its initial shipment to the site to its eventual long term storage and/or disposal stage. FuelWorks is composed of several modules that access as single main database for consistency and protection of the historical fuel assembly data, and ensures that fuel parameters are not lost or transposed.

2.2.3. *Disposal facilities*

The spent fuel data is also used to show that the spent fuel disposal packages are in compliance with the disposal facility's acceptance criteria. This requirement is articulated in the case of USNRC case as "*the license application adequately characterizes the range of parameters that describe the spent fuel*" [10].

2.2.3.1. Acceptance criteria for disposal

The acceptance criteria of the disposal facility are derived from the relevant safety analyses. This usually leads to the extension of the required data set compared to transport or storage by including data such as the thermal conductivity of spent fuel, the emissivity and moisture of components, the specific heat of spent fuel for the waste package design. Data on spent fuel to be disposed of as intact fuel assemblies (or as consolidated fuel rods) are also required for the design of the disposal facility and safety analyses for the various portions of the spent fuel management system that may have different sets of data criteria.

No set of finalised acceptance criteria for disposal of spent fuel has been developed yet. Some countries started the process of selecting sites suitable for disposal of spent fuel and may have reached a status where a preliminary set of relevant parameters is under development or being defined. An example case is the US criteria pending for the disposal package for Yucca Mountain repository for which USDOE promoted MPC currently used for AFR storage [11].

2.2.3.2. Criticality issue and burnup credit

In the case of the planned U.S. repository for spent fuel and high level waste at Yucca Mountain, it should be noted that the post-closure criticality control rationale for the repository depends strongly on burnup credit, including principal actinide and fission product burnup credit.

The transportation part of regulatory activities requires a direct measurement of burnup as a pre-condition for the use of burnup credit in spent fuel transport casks, while the disposal part of regulatory activities has not yet made a determination of any burnup measurement pre-conditions for the use of burnup credit in spent fuel disposal.

2.2.3.3. *Data quality*

The spent fuel data provided by the operator of the reactor must assure that all data needed is provided under a fully auditable quality assurance programme.

The uniformly used process of dual, independent identification and mapping of individual fuel assemblies and the subsequent comparison and confirmation of the identity and position of each assembly loaded into a cask or canister under fully approved and monitored quality assurance (QA) procedures, helps in providing full and proper assurance of the correct identity of the loaded fuel assemblies.

If these data are supplied to the national organization in charge of spent fuel management or disposal under regulatory approved QA conditions, and these data are given an independent confirmatory check for completeness and consistency, such cask or container loading data provides a suitable basis for the identity of the spent fuel for the purposes of spent fuel acceptance at the time of delivery at the repository — without the need for opening any sealed canisters for verification after transport to the repository site.

2.2.3.4. *Conditioning of spent fuel disposal package*

Spent fuel needs appropriate conditioning and packaging for safe storage or disposal. The method of conditioning and packaging is not only a function of the design concept for the repository or storage facility, but also a variety of other factors such as regulatory, economics, public acceptance, etc.

From the standpoint of spent fuel integrity, there are three kind of conditioning and packaging methods considered; intact fuel assembly, rod consolidation, destructive compaction.

The materials composition data required will depend on the disposition foreseen and on the conditioning prior to disposal. Since structural parts separated from the fuel may be disposed differently, different data may be needed.

An example of data requirements for spent fuel disposal package conditioned to POLLUX type was formulated with more detailed list of items for spent fuel and associated data, as shown in the Appendix.

Conditioning with spent fuel rod consolidation requires detailed assembly level data in order to enable safe and reliable separation of the fuel pins from the structural part (skeleton) of the spent fuel assembly. It is the operator of the conditioning facility who must define the type of data needed to perform a safe operation of spent fuel conditioning.

The conditioning of spent fuel disposal package could require destruction of spent fuel integrity, depending on the rationales to be adopted for spent fuel management that could evolve with time (i.e. safeguards, safety, economics, public acceptance, etc.). Recently, this kind of conceptual developments have already been initiated on national and/or international initiatives, which could set forth some important implications in the future spent fuel management (see Section 2.2.5 for more details).

2.2.4. *Reprocessing facilities*

Historically, reprocessing has been the classic option for spent fuel management. The PUREX technology which is illustrated in Figure 3 has been matured for decades on industrial scale in several countries. Reprocessing, therefore, can be regarded as the only industrial option currently available for spent fuel management with an end point, pending disposal in a geological repository. A summary on the recent status and trends in spent fuel reprocessing is provided in an IAEA publication [12].

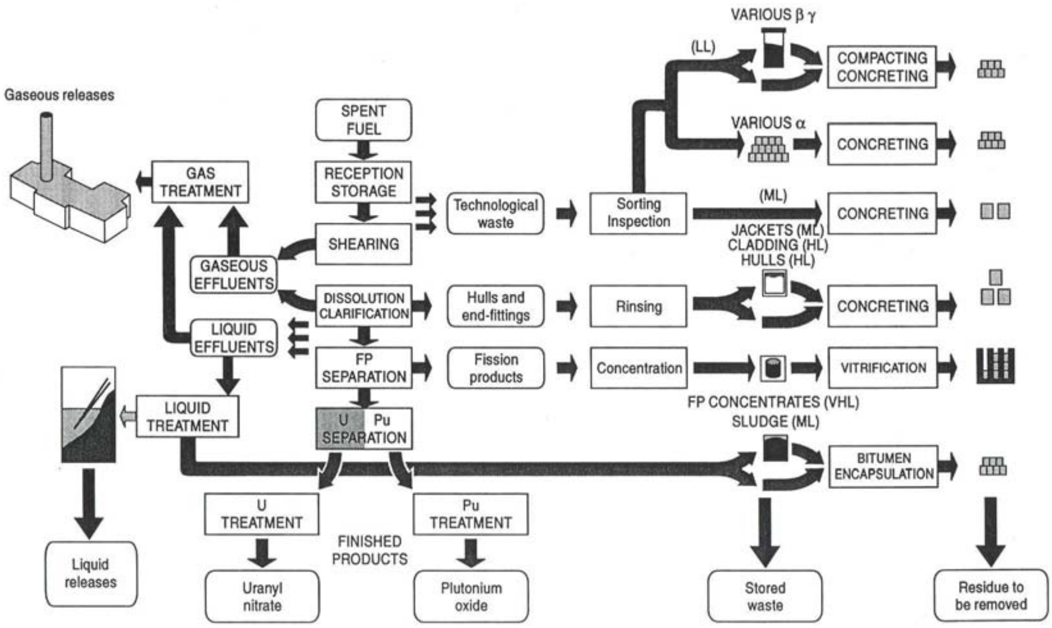

FIG. 3. Flow diagram of spent fuel reprocessing by PUREX process.

Roughly one-third of the global cumulative arisings of spent fuel has already been reprocessed. The balance of the global arisings is currently stored as inventory either in conventional pools or in the more recently developed dry storage systems.

Data needed for spent fuel delivered to a reprocessing site is now well defined as can be seen in the following industrial examples.

2.2.4.1. Industrial examples of spent fuel database

Some examples of databases for spent management in reprocessing industry are cited below [13].

- *FIS*

As an example of an integrated data management system, THORP reprocessing plant has implemented a Fuel Information Service (FIS), for the spent fuel storage areas, that replaced several independent systems and brings the Nuclear Materials Accountancy and Safeguards Records together with the Operating Records into one database. A standard "off the shelf" software package, widely used in warehousing applications and the food and drug industries, has been used as a basis of the FIS with specific enhancements and customisation developed by British Nuclear Group (BNG, former BNFL) and the software supplier. Previously, the logistics of spent fuel pool storage were managed by independent systems using a combination of computer and manual methods.

The FIS system now integrates all operating and accounting functions: data is entered into the FIS only once to avoid the problems associated with transcription errors and data inconsistencies. Information regarding spent fuel receptions, movements within the pool is entered directly by the operators, thus improving the timeliness and accuracy of records.

The FIS updates the pool maps automatically as spent fuel movements are entered and details of container inventories (the fuel assemblies being placed into containers or "bottles") are accessed by inquiries from the pool map. Historical records of the location of all fuel assemblies and containers are maintained as is a record of the contents of each location. Usually, the spent fuel is reprocessed in customer campaigns that require the fuel assemblies to be selected from stock, assessed against the safety case and technical criteria specific to the reprocessing plant, e.g. burnup, cooling time, enrichment, fuel additives, etc. and scheduled in a "feed" order to promote efficient shearing, dissolving and subsequent reprocessing operations. The FIS includes an interface to the systems implemented in the reprocessing plant each fuel assembly is identified and monitored before being sheared to verify the relevant data.

- *CONSULHA*

Other systems used for spent fuel assembly identification include an automatic spent fuel identification number reader through an image processing technique. The CONSULHA system (developed as part of the French support programme for the IAEA safeguards) that processes the input flow of spent fuel assemblies to the pool, the output flow of fuel assemblies from the transportation casks, the internal movement of the spent fuel within the dry cask unloading cell, and miscellaneous other movements: the log supplies data such as date of movement, time of start of movement, time of end of movement, presence of gamma radiation counting, presence of neutron counting, sequence number, etc.

- *BUD*

Another system installed at La Hague is the BUD (burnup device) that counts and identifies spent fuel assemblies transferred from the pool to the entrance of the reprocessing plant, checks the burnup of the fuel without interfering with plant operations, transfers the video and nuclear data via an optical fiber network, and processes the data to enable a straightforward review by Euratom inspectors.

2.2.4.2. Data management on high level waste from reprocessing

Data on high level waste from the reprocessing of spent fuel may form another database for disposal purposes. High level waste data may be documented/archived within the data management systems used for other wastes, (for example the AVK-system in Germany deals with high level waste in a special module beside the modules for operational and decommissioning wastes from power reactors [14].

However, data management for high level waste is not covered in any detail in this publication.

2.2.5. Advanced and/or special treatment technologies

Even though it may be too early to speculate about the perspectives of innovative nuclear systems anticipated for the future, the Group participating in the technical meeting did agree that such contingencies of spent fuel management should be covered as a future possibility in view of the growing interest attracted to the global energy and environmental issues. It was noted that IAEA should keep abreast of the new development in innovative nuclear systems, to determine if the recommended spent fuel tracking procedures are adequate.

2.2.5.1. Refabrication

Even though there has not yet been any industrial refabrication of spent fuel from a power reactor for reuse in another reactor, research activities have been conducted on lab scale, a representative case being the DUPIC project intended for reuse of spent fuel from PWR in CANDU reactors without separation of sensitive materials like plutonium. The basic rationale of the DUPIC fuel cycle is that the typical remnant fissile contents of spent PWR fuel (approximately double those of natural uranium) can be reused with doubling burnup in a CANDU reactor, which is designed to be fuelled with natural uranium. Another example is the AIROX concept which had been tested on laboratory scale in the mid-sixties at Atomics International in the USA, with a view to reuse spent LWR fuel in LWRs by adding enriched uranium to the depleted spent fuel. From a technological point of view, the DUPIC (or the AIROX, for that matter) fuel cycle concept bears some interesting features that are anticipated from innovative fuel cycle options. All the fuel fabrication processes are remotely conducted in shielded hot cell facility [15].

The concept of the DUPIC fuel cycle is illustrated in Figure 4. The identity of spent PWR fuel is destroyed by the DUPIC process and a new “fresh” fuel is regenerated for reuse in CANDU reactors (of CANFLEX type).

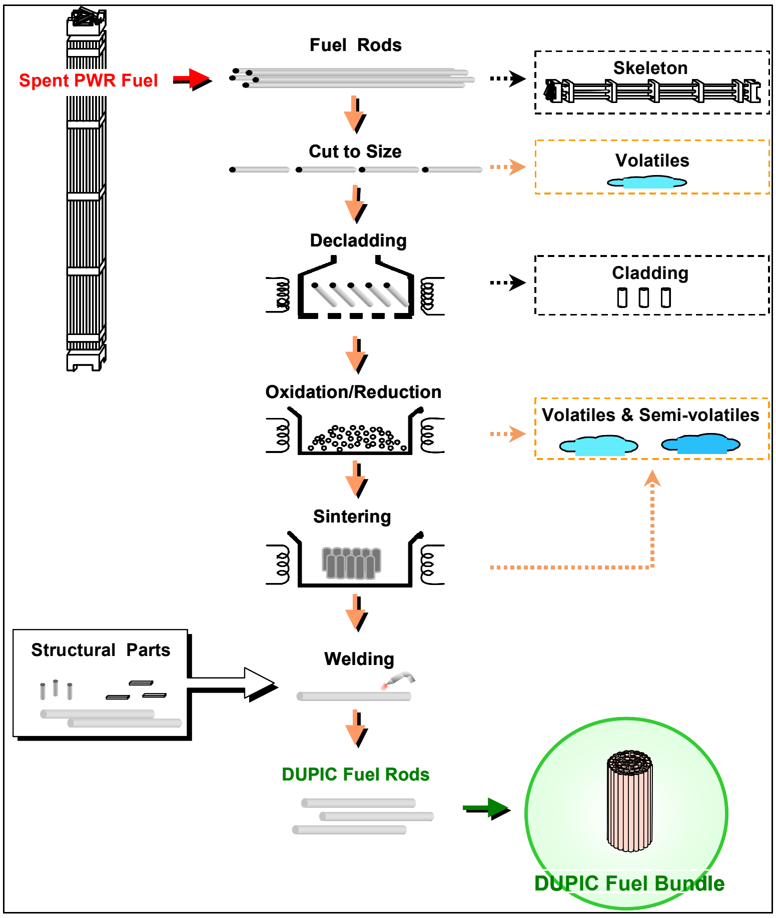

FIG. 4. Schematic diagram of DUPIC fuel fabrication process.

The DUPIC fuel cycle is an extreme case representing the complexity of spent fuel data requirement due to the complexity of fuel contents to be processed. With a view to resolve the disparity in spent PWR fuel burnup to come up with a homogeneous compositions in CANDU fuel to be refabricated, a study on algorithm for optimal combination of different burnups was conducted [16].

2.2.5.2. PBMR (pebble-bed modular reactor)

The technology for pebble-bed modular reactor (PBMR) is attracting industrial attention as one of the major reactor models for innovative nuclear systems to be deployed in the future. Because of the technical features of PBMR fuel in a bulk of hard spheres in pebble form, the management of spent PBMR fuel and its data management call for special considerations [17].

In the particular case of PBMR, tracking information of its fuel in pebble form is considered to be a challenge. The pebble form fuel is neither amenable to handle nor identifiable as a unique item in accounting for safeguards and this issue is being looked at by IAEA.

2.2.5.3. Partitioning and transmutation

As the first step in the transmutation process is partitioning, a process similar to reprocessing, the spent fuel data requirements for both activities should be similar. However, the tracking requirements for a transmutation reactor are much greater than for a power reactor. In a power reactor, the main concern is what happens to the uranium, plutonium, and thorium isotopes in the fresh fuel. In a transmutation reactor, one has to track the disposition of all the radioactive isotopes that are transmuted.

With the development of particle accelerator technologies in 1980s and 1990s there is an increasing interest in determining if these technologies can fulfill some basic nuclear missions. One of them is the transmutation of long lived nuclides in spent fuel and high level waste streams to short or medium lived radionuclides or stable isotopes (projects like, MICANET – EU, ATW-USA, Omega-Japan, …) [18]. The reduction of radiotoxicity by transmutation of minor actinides can be shown as Figure 5.

The objectives of transmutation and partitioning technologies may be very different, but it is possible to distinguish several approaches discussed in different countries [19].

From the point of view of spent fuel data management systems, the future transmutation and partitioning technologies are treated similarly to the spent fuel conditioning or reprocessing technologies — the operator of the transmutation and partitioning facility must define the type of data needed to perform a safe and efficient spent fuel transmutation.

Additional data may be required for the disposal of generated radioactive waste; for which data would be generated by the operator of the transmutation and partitioning facility. Additional data might further be needed, depending on the number of independent installations involved in the transmutation and partitioning. However to be consistent with the statements in previous sections of this publication, the spent fuel data management is followed only up to the steps of transmutation and partitioning process, since this technology usually requires mechanical and chemical destruction of spent fuel assemblies (see Figure 6 for a case of electrorefining process for actinide partitioning[8]).

[8] For recent trends on this subject, see Global Nuclear Energy Partnership website(www.gnep.energy.gov).

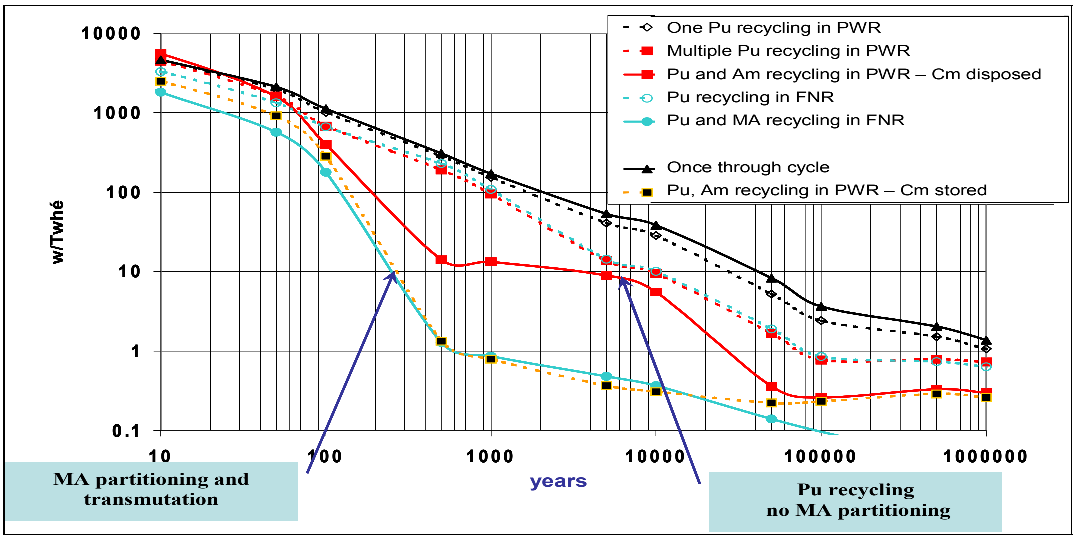

FIG. 5. Reduction of radiotoxicity by transmutation of minor actinides.

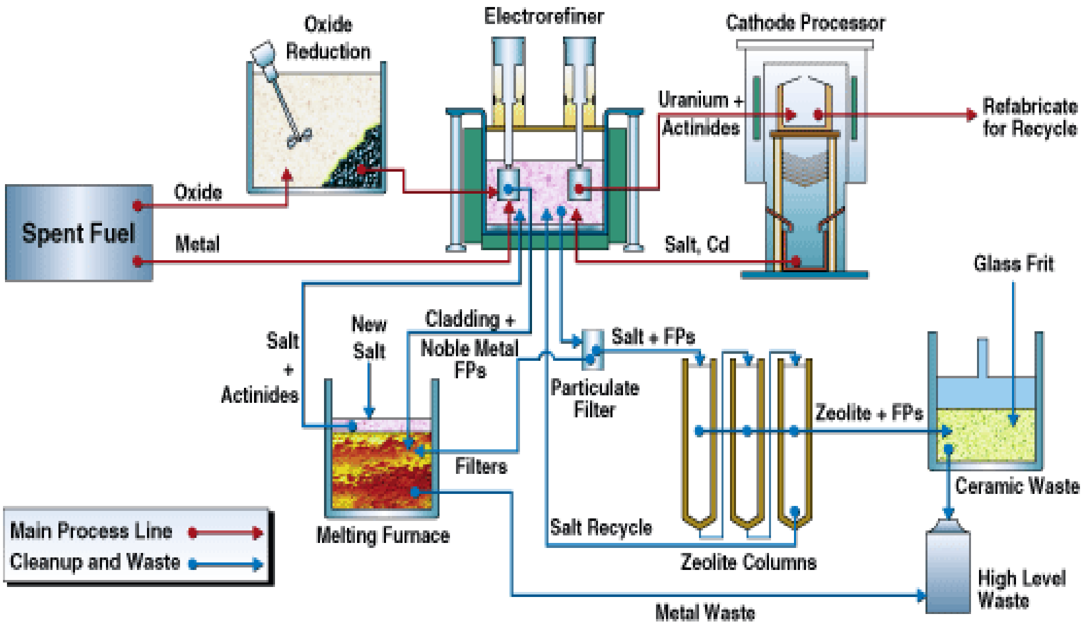

FIG. 6. Illustrative flow diagram of actinide partitioning system by electrorefining process.

2.3. International obligations

There are international agreements or treaties on some nuclear issues regarding spent fuel management. The obligation for spent fuel data management is embedded in several international documents prepared by the IAEA. The most important of these publications are:

- Safeguards agreements
- Several safety standards
- Joint Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management

2.3.1. IAEA Safeguards Agreements

The IAEA is authorized by Article III.A.5 of its Statute to apply safeguards. There are three types of safeguards agreement.

2.3.1.1. Comprehensive safeguards agreements

Virtually all comprehensive safeguards agreements with the IAEA have been concluded by non-nuclear-weapon States pursuant to the Treaty on the Non-Proliferation of Nuclear Weapons (NPT). Each of these agreements, concluded along the lines of INFCIRC/153 (Corrected)[9], requires a State to accept IAEA safeguards on all source or special fissionable material in all peaceful nuclear activities within the territory of the State, under its jurisdiction, or carried out under its control anywhere. It requires that the State establish and maintain a system to account for and control all nuclear material subject to safeguards.

Other bilateral or multilateral arrangements require that comparable provisions be contained in comprehensive safeguards agreements concluded pursuant thereto. These include: (a) the Treaty for the Prohibition of Nuclear Weapons in Latin America and the Caribbean (Tlatelolco Treaty); (b) the South Pacific Nuclear Free Zone Treaty (Rarotonga Treaty); (c) “the Argentine-Brazilian Declaration on Common Nuclear Policy”; (d) the Treaty on the Southeast Asia Nuclear Weapon Free Zone (Bangkok Treaty); and (e) the African Nuclear Weapon Free Zone Treaty (Pelindaba Treaty).

2.3.1.2. INFCIRC/66-type safeguards agreements

In some States, the IAEA implements safeguards under agreements that are not comprehensive, but rather are item specific. These safeguards agreements, based on the guidelines contained in INFCIRC/66/Rev.2[10], specify the nuclear material, non-nuclear material (e.g. heavy water, zirconium tubes), facilities and equipment to be safeguarded.

Under such agreements, the IAEA is required to ensure that the nuclear material and other specified items are not used in such a way as to further any military purpose.

2.3.1.3. Voluntary offer agreements

The five nuclear-weapon States[11] have offered some or all civilian nuclear material and/or facilities, from which the IAEA may select material or facilities for the application of safeguards. These voluntary offer safeguards agreements generally follow the format of INFCIRC/153-type agreements, but they vary in scope.

2.3.1.4. Protocols additional to safeguards agreements

Beside the implementation of those strengthening measures, which were within the legal authority provided by safeguards agreements, and with the aim to reinforce the effectiveness and improve the efficiency of the safeguards system, the IAEA was requested to use the

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[9] The Structure and Content of Agreements between the IAEA and States Required in Connection with the Treaty on the Non-Proliferation of Nuclear Weapons *(INFCIRC/153 (Corrected)), 1972*.

[10] The IAEA’s Safeguards System (1965, as Provisionally Extended in 1966 and 1968), *INFCIRC/66/Rev.2, 1968*

[11] Article IX.3 of the NPT defines a nuclear weapon State as one, which manufactured and exploded a nuclear weapon or other nuclear explosive device prior to 1 January 1967. There are five such States: China, France, the Russian Federation (the Soviet Union when the NPT entered into force), the United Kingdom and the United States of America.

Model Additional Protocol, published as INFCIRC/540 (Corrected)[12] as the standard for individual additional protocols to be concluded with States that have comprehensive safeguards agreements. The IAEA was also requested to negotiate additional protocols or other legally binding agreements with States that have other types of safeguards agreements (i.e. voluntary offer and INFCIRC/66-type) and are prepared to accept measures provided for in the Model Additional Protocol.

Under the implementation of comprehensive safeguards agreements *with* additional protocols, the overall objective with respect to a State is to provide credible assurance of both the non-diversion of nuclear material from declared activities and of the absence of undeclared nuclear material and activities in the State as a whole.

2.3.1.5. Safeguards implementation

Under all types of safeguards agreement, the State provides information ("declarations") to the IAEA, and the IAEA verifies and evaluates that information, with the aim to draw safeguards conclusions for that State.

For a State with a comprehensive safeguards agreement only, its declaration consists primarily of nuclear material accounting reports and facility design information, and the verification activities by the IAEA focus primarily on verifying these declarations. The IAEA then evaluates the results of its verification activities and all other available information about the State's nuclear and nuclear-related activities in order to draw a conclusion about the non-diversion of declared nuclear material.

For a State with a comprehensive safeguards agreement and an additional protocol, the implementation of safeguards involves, in addition to the foregoing nuclear material verification activities, the provision by the State of a much broader range of information about its nuclear and nuclear-related activities, and the performance by the IAEA of activities under complementary access, as necessary, to assure the absence of undeclared nuclear material and activities, or to resolve any questions or inconsistencies related to the information provided by the State.

2.3.1.6. Nuclear material verification

Nuclear material verification activities are implemented under safeguards agreements and are based on the principle of nuclear material accountancy, complemented by containment and surveillance. In that regard, the facility operator/State authority is required to maintain for each facility nuclear material accounting records on inventory and inventory changes of all nuclear material. The facility operator declares to the IAEA through the State authority the nuclear material accounting data, and safeguards-relevant design information. These declarations constitute the basis for the IAEA verification activities. The scope of the verification activities at a given facility is ruled by the safeguards agreement and the subsidiary arrangements concluded with the State.

The verification activities that the IAEA inspectors may perform during inspections include:

- Examination of facility accounting and operating records and comparison of these records with accounting reports submitted by the State;

[12] Model Additional Protocol to the Agreement (s) between State (s) and the International Atomic Energy IAEA for the Application of Safeguards (*INFCIRC/540 (Corrected)*), 1997.

- Application of containment and surveillance measures;
- Verification of inventories of nuclear material and, under certain types of agreements, of non-nuclear material and equipment, and of inventory changes at a facility;
- Verification of nuclear material flows, including transfers between facilities and, in certain cases, transfers within facilities (e.g. material flows into and out of the process area).

2.3.1.7. Safeguards criteria

Safeguards criteria have been established for all facility-types under safeguards, including storage facilities[13]. These criteria specify the scope, the normal frequency and the extent of the verification activities required to achieve the inspection goal at each facility. They are also used for planning the implementation of verification activities and for evaluating the results thereafter.

2.3.1.8. List of inventory items

The list of inventory items (LII) is the facility operator's declaration[14] regarding the safeguarded nuclear material, which is provided to the IAEA inspector(s) in advance of an IAEA's physical inventory verification (PIV). The LII lists the measured values or derived estimates of each item of nuclear material, (of other material, and of equipment subject to safeguards) physically present at the facility at the declared closing date of the material balance period. The grouping of items (stratification) with similar physical and chemical characteristics and with the same measurement errors is a requirement (with respect to IAEA'S PIV activities); grouping of items according to locations is usually optional, but desirable to speed up verification activities.

The operator of a facility subject to safeguards is required to maintain accounting and operating records that permit the IAEA to determine the book inventory and to verify the physical inventory of nuclear material.

The IAEA recognizes two types of inventory verification:

(i) The PIV, which coincides with closing a material balance period and the PIT by the operator; and
(ii) The interim inventory verification (IIV), which does not coincide with closing a material balance period and during which part or the entire inventory may be verified.

The detailed procedures for PIT are specific to each facility type; therefore a careful completion of the relevant information by the State is needed.

(a) The main elements that the design information should comprise are as follows:

  - General procedures,
  - PIT frequency,
  - Nuclear material distribution,

[13] An internal document of the IAEA Safeguards Department, Vienna.

[14] The LII normally reflects the results of operator's Physical Inventory Taking (PIT); these results vary in nature, they may be the results of actual item measurements carried out during PIT or they may be simply the confirmation of the presence of an item along with the transcription of item data from a tag or a computer file.
In some cases, such as quasi-simultaneous PIT/PIV, the LII prepared by the operator may mainly reflect operator's confirmation of the presence of items and to a lesser extent actual measurement results; even then the LII is considered the operator's commitment prior to PIV; the IAEA should consider this list LII.

- Measurement methods for item and bulk material,
- Accuracy of measurement methods (random and systematic measurement errors),
- Accessibility of material,
- Use of factors, nominal values, calculated values,
- Use of measured values.

(b) The following data elements are mandatory for an LII:

- Cutoff time and date for PIT,
- Material Balance Area (under INFCIRC/153), or accounting area (under INFCIRC/66),
- Location (key measurement point, area identification),
- Stratum identification,
- Item identification no.,
- Batch identification no.,
- Element code,
- Element weight,
- Isotope weight.

(c) The following data elements may be required for an LII, as appropriate:

- Irradiation status (fresh, irradiated),
- Inventory tag number,
- Seal number,
- Item description (drum, tray, rod, assembly, ...),
- Material description (MOX, UO_2 sintered., alloy, ...),
- Material description code, MDC,
- Gross weight,
- Tare weight,
- Net weight (weight of chemical compound),
- Element concentration factor (indicate whether: nominal-calculated-measured),
- Isotope enrichment factor (indicate whether: nominal-calculated-measured),
- Volume,
- Density,
- Cooling time of irradiated fuel,
- Burnup of irradiated fuel (Measurement basis, method or instrument used, accessibility...).

2.3.1.9. Management of spent fuel data for safeguards

In view of the foregoing, it is important when a facility (AFR storage, reprocessing facility or disposal site) receives or is going to receive spent fuel, the operator requests from the shipping facility complete identification of each spent fuel item, as well as information regarding measurement or calculation results, and containment/surveillance methods used and any conditions that might have adversely impacted their integrity.

Example of data needed for safeguards purposes on spent low-enriched uranium (LEU) fuel received at a storage facility is as follows:

- Number of spent fuel assemblies,
- Mass of heavy metal per assembly, pre- and post-irradiation,

- Type of spent fuel assembly,
- Canister identification number,
- Tamper indicating device (canister/cask),
- Initial enrichment (% with tolerance),
- Fuel burnup (average, maximum),
- Last date of irradiation,
- Pre-irradiation information (^{235}U, total U),
- Post-irradiation information (total U, Pu, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{238}U, ^{235}U).

2.3.2. IAEA safety standards

The establishment, and promotion, of advisory international standards and guides is a key part of the IAEA's statutory mandate. Standards are issued as series publications and cover nuclear safety, radiation protection, radioactive waste management, the transport of radioactive materials, the safety of nuclear fuel cycle facilities and quality assurance. These standards also contain different requirements on the data collection and maintenance.

At the level of general safety standards one of the key publications is the IAEA Safety Standards Series No. GS-G-1.4 [20]. which clearly defines the operator's responsibility for the safety of the facility including the keeping of records of all activities that are considered to be safety related. Regulations or licence conditions should establish the types of records that should be kept and the periods for which they should be retained. In specifying the retention period, account should be taken of the possible future need to refer to these records and of the difficulties of regenerating the information.

2.3.2.1. Records on relevant data

In general there are several categories of records at the operator of nuclear facility:

- records of site evaluation and construction,
- records made during commissioning,
- commissioning reports,
- operational records,
- records of modifications to the facility,
- records and evaluation of events, and
- decommissioning and licence termination records.

The data relevant to spent fuel data management are usually covered in the last five categories of records managed by the operator of nuclear facility. Mainly the operational documents to be retained by the operator contain an extensive set of data that should be examined by the regulatory body and should include:

- output and performance records of the facility,
- operating log books,
- inventories of fissile and radioactive materials,
- records of periodic calibration of equipment,
- records of periodic testing of equipment and systems,
- records of in-service inspections,
- records of preventive maintenance and repairs,
- records of personnel training,

- records of personnel radiation monitoring,
- records of radiation monitoring and contamination records for the facility,
- records of radioactive waste management including spent fuel management,
- records of effluent discharges and of the environmental monitoring programme, and
- records of fault conditions.

All modifications relevant to safety and their evaluations (e.g. change of the type of nuclear fuel used) should be recorded for possible re-examination.

The regulatory body should periodically examine the complete set of modifications made to the facility in order to evaluate the effectiveness of the operator's control process and to ensure that all modifications relevant to safety have been submitted for approval in accordance with applicable regulations. Also the event evaluation process and its results should be recorded for all events above an established threshold of significance. Recorded events should be periodically reviewed by the operator to identify trends and possible deterioration of safety levels.

The general and specific requirements on the record keeping concerning matters important to safety are covered in the Section 6 of the IAEA Safety Standards Series No. NS-G-2.7 [21]. It also contains standards for keeping these records in order to maintain up to date information and historical information on important aspects of the radiation protection and the spent fuel and radioactive waste management.

2.3.2.2. Maintenance of records

Records should be prepared and stored in such a manner that they are readily retrievable and can be understood later. They should be classified as requiring retention for the long term (such as records of personnel doses), the medium term (such as records of shipments of radioactive waste) or the short term (such as records of survey results for controlled areas). The regulatory body should specify the minimum periods of time for their retention.

As-built drawings should be kept for all facilities associated with spent fuel and radioactive waste. Records, including all relevant details, should be kept on spent fuel, radioactive waste, on packages and on the contents of spent fuel and radioactive waste stores. At any time during storage it should be possible to determine from the records the type, activity and characteristics of the spent fuel and radioactive waste stored in each specified location. Computer assisted programmes should be used for the continuous updating of the radionuclide inventory, with account taken of radioactive decay.

All shipments of radioactive material and dispatch of treated or untreated spent fuel and radioactive waste for processing or disposal should be documented and recorded, including the type and quantity of such spent fuel and/or radioactive waste, the type of packaging and the destination. These records should be retained even after the spent fuel and/or radioactive waste or the sources have been disposed of.

Reports on any investigations into abnormal conditions or deficiencies, such as unplanned releases or spills, in the programme for spent fuel and radioactive waste management should be kept. In particular, records of contamination levels in structures and components of the plant should be maintained in order to facilitate decommissioning.

2.3.2.3. Quality assurance (QA)

Under the IAEA programme for safety standards for nuclear power plants document quality assurance and documentation requirement on identification, location and movement of fuel assemblies are identified [22].

The design of all equipment for fuel handling and storage should incorporate features that are necessary for verification of the records on:

- the number and identification of fuel assemblies and other core components;
- the location of each fuel assembly or core component.

Identification features should be made so durable that they will remain effective during the handling and operation procedures.

Recommendations on how to meet the safety requirements for the predisposal management of high level radioactive waste, which includes high level waste in liquid form and solidified form from the reprocessing of spent fuel and also spent fuel itself are summarised in [23]. Section 6 provides recommendations on record keeping and reporting.

A quality assurance programme for the predisposal management of spent fuel is required to be established and implemented by the operator of the facility. As a part of quality assurance programme, quality records should be established and maintained for each conditioned package. These records should be reviewed against the specifications to determine the acceptability of the waste package. A record of the results of the review should be made and retained for a specified period of time as approved by the regulatory body. A system for documentation that includes the development of such records should be established.

2.3.3. The "Joint Convention"

The Joint Convention on the Safety of Radioactive Waste Management and the Safety of Spent Fuel Management ("Joint Convention", for short) entered into force on 18 June 2001. The Joint Convention has, as its main objective, "to achieve and maintain a high level of safety world-wide in spent fuel and radioactive waste management" [24].

Each Contracting Party should provide a list of spent fuel management facilities and inventory of spent fuel that is subject to this Convention.

In particular:

- Article 9 "Operation of Facilities" in paragraph (vi) clearly states that "Each Contracting Party shall take the appropriate steps to ensure that programmes to collect and analyse relevant operating experience are established and that the results are acted upon, where appropriate."
- Article 17 states furthermore that "Each contracting party shall take the appropriate steps to ensure that after closure of a disposal facility, records of the location, design and inventory of that facility are preserved".
- Article 19 also specifies that the legislative and regulatory framework shall provide for "a system of appropriate institutional control, regulatory inspection and documentation and reporting".

These articles provide an official justification for the collection of information concerning conditions of storage and transport of spent fuel.

Each Contracting Party shall submit a National Report to the Review Meeting held every 3 years. The first Review Meeting was held in 2003 and the second is to be held in 2006.

3. SPECIFICATION ON SPENT FUEL DATA

In Section 2, the data need for a spent fuel management is discussed. This section identifies parameters that are necessary to provide a technical description of the data in order to define the categories to which various spent fuel data would belong.

3.1. Data parameters

A method to describe spent fuel by data parameters is to use the different phases of life cycle through which spent fuel form and content are altered.

These data parameters generally fall into following groups relating to:

- general data,
- pre-irradiation data,
- irradiation data (in core fuel management data),
- post-irradiation description, and
- data on conditions of long term storage and transport.

Within these groups, some data will apply to specific assemblies (e.g. final discharge burnup) and some will apply to groups of assemblies (e.g. general fuel assembly descriptions).

In many cases, these parameters identify other sets of data, which further describe the assembly and/or its condition. The responsibility of different institutions is clearly identified for each subgroup of data (fuel designer, fuel custodian, fuel manufacturer, reactor operator, etc.).

General data comprise an initial set of parameters related to the design and ownership of the fresh fuel. These data can be identical for large number of individual fuel assemblies, for which the current ownership of the fuel assembly should be specified. Typically, the owner is the operator of the reactor in which the fuel was irradiated. In cases where the fuel was irradiated in more than one reactor, the owner is typically the operator of the reactor in which the fuel was last irradiated. In addition, the identity of the reactor(s) in which the assembly was irradiated should be maintained for tracking purposes.

3.2. Reliability of spent fuel data

In defining an approach to the verification of utility-supplied spent fuel data for reliability purposes, it must be recognized that generally the most accurate data on fuel assembly characteristics is the operational fuel data developed by the nuclear plant operators using accurate measurement of total core thermal output, and using in-core measurement systems and validated computer programs to allocate total core energy output to individual assemblies, thereby determining assembly burnups at all times.

These operational assembly fuel data result from the regulatory requirements for a complete quality assurance based level knowledge of reactor core conditions at all times, including assembly burnups and radiological content at the time of final discharge. No currently known post-discharge assembly measurement system can match the accuracy of the burnup and other radiological data developed from the extensive core-follow measurement and calculation systems being used by nuclear plant operators.

From this, it can be concluded that if these data are supplied under full QA conditions, and these data are given an independent confirmatory check for completeness and consistency with other available data, such data provide the soundest basis for operational planning and for material control and accounting (MC&A) safeguards requirements for any back end facility that will accept the spent fuel.

On this same basis, it can also concluded that there is no additional benefit to be gained by performing confirmatory measurements on individual fuel assemblies after delivery to the back end facility (long term storage, reprocessing, refabrication or disposal facility), provided that all post-delivery handling of fuel assemblies or other containers is done under approved and monitored QA procedures.

The extent to which these observations apply to all national spent fuel management programmes depends primarily on how closely the utility operational core follow-up programmes and spent fuel cask loading practices have the ability to deliver safety level QA data to the regulatory body in charge of spent fuel management or to the national disposal agency.

3.3. Specification on spent fuel by data parameters

The data parameters summarized in Table 5 are suggested to be collected and retained on an individual assembly and they build a structure of reference spent fuel database developed for the illustrative purposes only.

3.3.1. Pre-irradiation data

These data belong to fresh fuel assemblies manufactured by fuel suppliers, before irradiation in the reactor.

3.3.1.1. Assembly identification and characteristics

General to a spent fuel management database is the unique identification of each fuel assembly. Ideally, identification is accomplished through a unique assembly fabrication serial number, which is permanently attached to, or imprinted on the fuel assembly. This serial number should be unique, not only at the reactor, but within the entire system, whether the system spans spent fuel discharges from a reactor site, a utility, or a country.

TABLE 5. STRUCTURE OF REFERENCE SPENT FUEL DATABASE

GENERAL DATA

FUEL DESIGNER
- Reactor type
- Design serial No. [-]
- Fuel type [-]
- Drawing

FUEL CUSTODIAN
- Owner of fuel [-]
- NPP and reactor unit [-]

PRE-IRRADIATION DATA

MANUFACTURER
- Fabrication serial number [-]
- Physical dimensions
  - Length [mm]
  - Effective diameter [mm]
  - Weight total [kg]
  - Weight HM [kg]
- Physical Form [-]
- Chemical form [-]
- Initial isotopic composition
  - Uranium Isotopes [kg]
  - (for MOX fuel also Pu, Th, Am and Np isotopes)
  - Solubility
- Hardware
  - Cladding - Composition [-]; Weight [kg]
  - Spacers - Composition [-]; Weight [kg]
  - Grids - Composition [-]; Weight [kg]
  - Hanger rods - Composition [-]; Weight [kg]
  - Springs - Composition [-]; Weight [kg]
  - Sleeves - Composition [-]; Weight [kg]
  - Coatings - Composition [-]; Weight [kg]
  - Burnable absorbers - Composition [-]; Weight [kg]
  - Flux suppressors - Composition [-]; Weight [kg]
  - Pin inserts - Composition [-]; Weight [kg]
- Date of fabrication [dd/mm/yyyy]
- Interim storage conditions
  - Temperature [°C]
  - Humidity [%]

IRRADIATION DATA

REACTOR OPERATOR
- Date of receipt from manufacturer [dd/mm/yyyy]
- Total burnup [MWd/tU]
- Data of first load in core
  - Date of first loading [dd/mm/yyyy]
  - Position in core [-]
  - Burnup [MWd/tU]
  - Peak rod burnup [MWd/tU]
  - Date of unload [dd/mm/yyyy]
- Data of n-th load in core
  - Date of n-th loading [dd/mm/yyyy]
  - Position in core [-]
  - Burnup [MWd/tU]
  - Peak rod burnup [MWd/tU]
  - Date of unload [dd/mm/yyyy]
- Isotopic composition
  - Date of calculation [dd/mm/yyyy]
  - Activation products [Bq/FA]
  - Fission products [Bq/FA]
  - Actinides [Bq/FA]
  - Total activity [Bq/FA] incl. α, β/γ, and n [Bq/FA]
- Thermal output [W/FA]
  - Date of calculation [dd/mm/yyyy]
- Special considerations
  - Defective fuel [Y/N]
    - Date of defect occurring [dd/mm/yyyy]
    - Short description of defects [-]
  - Canned fuel [Y/N]
    - Date of canning [dd/mm/yyyy]
    - Reason for canning [-]
  - Other considerations [e. g. removal of fuel rods, consolidation of FA, alteration of FA, etc. -]

POST-IRRADIATION DATA

AR STORAGE OPERATOR
- Storage facility ID [-]
- Position of FA in store [-]
- Date of loading [dd/mm/yyyy]
- Storage environment
  - Type of medium [-]
  - pH
  - Conductivity [-]
  - Selected ion concentration [mol/l]
  - Temperature [°C]
- Storage of Canned FAs
  - Position in Store [-]
- Transient or abnormal events
  - Date [dd/mm/yyyy]
  - Description (modification, overpackaging, consolidation, etc.)

LONG-TERM STORAGE DATA & DATA CESSATION

AFR STORAGE OPERATOR (example for dry cask storage facility)
- Date to storage [dd/mm/yyyy]
- Storage environment
  - Type of medium [-]
  - Pressure [MPa]
  - Temperature [°C]
- Transient or abnormal events
  - Date [dd/mm/yyyy]
  - Description (modification, overpackaging, consolidation, etc.)

TRANSPORTATION

MANUFACTURER
- Fuel physical characteristics
- Fuel composition
  - Uranium Isotopes [kg]
  - (for MOX fuel also Pu, Th, Am and Np isotopes)
  - Thermal output [W]
- Date of transport cask loading (fresh fuel) [dd/mm/yyyy]
- Cask ID [-]
- Mode of operation [-]
- Peak temperature during transport [°C]
- Date of transport cask (fresh fuel) unloading [dd/mm/yyyy]

AR STORAGE OPERATOR (example for transport from AR to AFR dry cask storage facility)
- Cask manufacturer [-]
- Physical characteristics
  - Dimensions [mm]
  - Total weight [kg]
- Inventory
  - Date of Calculation [dd/mm/yyyy]
  - Total No. of FAs in Cask [-]
  - No. of free loaded FAs in Cask [-]
  - No. of canned FAs in Cask [-]
  - Total Activity [Bq/Cask]
    - Activation Products [Bq/Cask]
    - Fission Products [Bq/Cask]
    - Actinides [Bq/Cask]
    - α [Bq/Cask]
    - β/γ [Bq/Cask]
    - n [Bq/Cask]
- Total Thermal Output [W/cask]
- Date of cask loading [dd/mm/yyyy]
- Cask ID [-]
- Date and mode of transport [dd/mm/yyyy, -]
- Peak temperature during transport [°C]
- Radiological Properties
  - Date [dd/mm/yyyy]
  - Surface γ-dose rate [Sv/h]
  - Surface n-dose rate [Sv/h]
  - γ- dose rate at x-meter distance [Sv/h]
  - n- dose rate at x-meter distance [Sv/h]
  - Surface α contamination [Bq/cm^2]
  - Surface β/γ contamination [Bq/cm^2]

AFR STORAGE OPERATOR (use of identical cask also for transport from AFR storage facility to the next backend facility)
- Date and mode of transport [dd/mm/yyyy, -]
- Peak temperature during transport [°C]
- Radiological Properties
  - Date [dd/mm/yyyy]
  - Surface γ-dose rate [Sv/h]
  - Surface n-dose rate [Sv/h]
  - γ- dose rate at x-meter distance [Sv/h]
  - n- dose rate at x-meter distance [Sv/h]
  - Surface α contamination [Bq/cm^2]
  - Surface β/γ contamination [Bq/cm^2]

REACTOR OPERATOR
- Date of transfer of fresh fuel from manufacturer [dd/mm/yyyy]
- Interim storage conditions (fresh fuel)
  - Temperature [°C]
  - Humidity [%]

OPERATOR OF SUBSEQUENT BACKEND FACILITY (geological repository, reprocessing plant, etc.)
- Date of fuel arrival [dd/mm/yyyy]
- Date of data cessation [dd/mm/yyyy]
- Identification of data transfer procedure (HLW database, MOX fuel database, etc)

Potential obstacles to assembly identifications can arise in the following cases:

- If the assembly has been damaged, either in-reactor or during discharge, its identification marker may have been obscured or even obliterated. In some cases, part of the assembly may have been dislodged. Such assemblies (depending on power-station practice) may have to be separately encapsulated and/or re-identified with a durable marker.
- Where fuel assemblies are stored in close proximity to, or overlaying each other, the ability to readily identify or gain access to specific assemblies may be reduced.
- Fuel pins that have been dislodged or deliberately removed from an assembly may not bear an identification relating them to their assembly of origin.
- The transfer of fuel from one storage medium to another may include an encapsulation step. Thus, the ability to gain access to or observe a specific assembly may be diminished following such transfers.

Sufficient physical description data should be available to adequately describe the fuel assemblies. Such descriptions will be needed for the design of fuel storage and transportation facilities, and fuel handling and processing components at a back end facility.

3.3.1.2. Assembly type

Each assembly serial number must have associated with it physical description data, but that physical description need not be specific to a single assembly. The physical description may vary according to the specific types of fuel being described, but will typically include parameters such as:

- fuel material (UO_2, UC, U/metal alloy, U metal, mixed oxide (MOX), etc.) and density,
- burnable absorbers details, if present,
- fuel assembly details, including dimensions, materials, and characteristics,
- fuel hardware[15] materials and quantities,
- typical fuel and reactor-coolant temperatures under normal operating conditions, and
- detailed fuel assembly diagrams/technical drawings and location(s) of such drawings.

Typically these data come from the fuel fabricator. These data may either be generic to production-scale quantities of assemblies or specific to a single assembly (e.g. where only one assembly has this description).

3.3.1.3. Initial weight of heavy metal and initial isotopic composition

The initial weight of heavy metal (total U, total Pu, total Th, — that is, the total weight of the actinides present) of each individual fuel assembly should be specified, as well as the initial isotopic composition of the fuel. For example, in the USA, all current fuel is manufactured from previously unirradiated uranium. In this case, only the total U content and the ^{235}U enrichment need to be specified to provide the initial weight of heavy metal and the initial isotopic composition.

Where mixed-oxide (MOX) fuels are being used, as in several counties in Europe, both the total U and the total Pu must be specified. In addition, the isotopic composition of all the uranium and

[15] Such as cladding, spacers, grids, hanger rods, springs, graphite sleeves, coatings, burnable absorbers, flux suppressors, fuel-pin inserts, etc.: all components of an assembly other than the fuel.

plutonium isotopes ($^{232\text{-}236}$U inclusive, ^{238}U, ^{236}Pu, $^{238\text{-}242}$Pu inclusive, and ^{244}Pu) must be specified.

For cases where the fuel assembly has been refbricated from other, possibly irradiated, assemblies, e.g. DUPIC assemblies, the isotopic composition may include fission products and other heavy-metal isotopes, in addition to the U and Pu isotopes. For such cases, the maintained records should show that the original fuel assemblies no longer exist as such, and that the new assembly has been assigned a new, unique identification number (see Subsection 2.2.4).

3.3.2. Irradiation data

Irradiation of fresh fuel during reactor operation brings forth major changes in fuel content (and also possibly in form) and therewith modifications in technical characteristics.

3.3.2.1. Historical records

Characterization or classification of spent nuclear fuel is sometimes based on the operational records together with the fuel fabricator's description. When a certain calculation such as cladding integrity evaluation during the dry storage is needed, historical data is as critical to the calculation as is the calculation methodology. Along with the fuel fabrication data, therefore, historical information and records of in-core management must be collected and preserved for the entire duration of spent fuel management.

The information to be retained is as follows:

- date of receipt from the manufacturer,
- location in the fresh fuel storage,
- date of first load in the core,
- the number of the first cycle in which the assembly was irradiated,
- position in the first loading core during the cycle operation,
- date of first unload,
- date of next cycle loading in the core,
- the number of next cycle in core,
- position in the core during the next cycle operation,
- date of next unload (date of discharge if discharged after twice burnup),
- date of third load in the core,
- the number of third loading cycle,
- position in the third loading core during the cycle operation,
- date of third unload (date of discharge if discharged after thrice burnup),
- entry date to the spent fuel pool.

If fuel rod or assembly damage took place and the damage was repaired during the overhaul period, repair date and details of the repair must be included in the historical records.

3.3.2.2. Power history and cumulative burnup

Cumulative burnup is, in general, the most important single calculated data element in the spent fuel management because this data element is used to estimate the inventory of each fission product generated inside the fuel rod.

Required information on the power history is as follows:

- average power of fuel assembly in each operating cycle,
- peak rod burnup in the assembly in each operating cycle,
- axial power distribution of the assembly in each operating cycle.

Power history of the nuclear fuels or fuel assembly, however, becomes critically important depending on the back end fuel cycle options. For example, total amount of iodine released from the pellet to the pellet/clad gap, is critically dependent of power history. The total iodine release, in turn, affects the integrity of spent fuel via stress corrosion cracking during dry storage. If fuels are consolidated or reconstituted, such traceable information is also required for spent fuel control purpose.

3.3.2.3. Modification of the fuel during irradiation

If fuel rod failure or assembly damage took place during the cycle operation, the following information is desired to supplement the in-core management records:

- visual inspection,
- sipping test,
- ultrasonic test, and
- summary of failure or damage.

If a damaged or failed fuel assembly is repaired, details of the repair must also be included in that assembly's records.

3.3.3. Post-irradiation description

After final discharge from the reactor, additional data are required to provide a complete description of the fuel assembly.

3.3.3.1. Final discharge burnup

The average discharge burnup of each fuel assembly should be recorded either as a part of irradiation or post-irradiation data category. In certain cases, the burnup distribution within the assembly may also be pertinent. For example, such data would be needed if dismantling of the assembly is planned for purposes of consolidation with fuel pins from other assemblies or where the assemblies may be refabricated into new assemblies for further irradiation.

The variation of burnup within an assembly could also impact criticality analyses and shielding requirements. Although discharge burnup is a derived quantity, because of its extreme importance in calculating the radiological properties of spent fuel and because discharge burnup is often directly available from the utilities, it is included here as a base, or non-derived data element.

The following data, in conjunction with the pre-irradiation data on initial weight of heavy metal and initial isotopic composition, should be sufficient to determine the average discharge burnup of the fuel:

- the date the assembly was installed in the reactor,
- the date the assembly was discharged from the reactor,

- the reactor power history during the residence time of the fuel assembly (i.e. power level as a function of time),
- the fuel channel(s) and/or reactor position(s) in which the assembly was irradiated and times or dates of residence in each location, and
- the positions of reactivity-adjustment-devices during the in-reactor residence time of the fuel assembly.[16]

3.3.3.2. Inventory of radionuclides in spent fuel

The inventory of radionuclides in spent fuel may be calculated from the fuel assembly's pre irradiation physical description, the initial weight of heavy metal, the initial isotopic composition and the in-core fuel management data, including the final discharge burnup. Retaining an assembly's detailed power history for more than five years after discharge is not essential, because after five years the radiological characteristics are directly related to the final discharge burnup, but are relatively independent of the details of the power history.

Computer codes are available to calculate the isotopic content of the fuel assembly at discharge and, if necessary, radioactive isotopes produced by activation in non-fuel components of the assembly. These codes typically focus on the calculation of the isotopic abundances of three sets of data: actinides and their daughters, activation products, and fission products. The actinides and their daughters will consist of the following:

- isotopes of uranium,
- isotopes of higher actinides produced by neutron activation (Np, Pu, Am, Cm, Bk, and Cf), and
- isotopes of actinide daughters (Pa, Th, Ac, Ra, Rn, Fr, Po, At, Bi, Pb, Tl, and He4, from alpha decay).

Both fission-product and activation-product isotopic inventories can be similarly calculated.

3.3.3.3. Radiological characteristics

From the isotopic inventories detail radiological characteristics can be obtained, including activities, photon spectra, and neutron spectra. Additionally, the variation of these characteristics with time can be calculated.

3.3.3.4. Decay heat

The decay heat can be also calculated on the basis of the isotopic inventories.

3.3.3.5. Non-fuel components

If discharged fuel assembly contains non-fuel components, those components must be included in the description of that assembly. Non- fuel components are not associated with a particular fuel type. These include, but are not limited to, control spiders, burnable poison rod assemblies, control rod elements, thimble plugs, fission chambers, primary and secondary neutron sources and boiling water reactor channels.

[16] For reactor types in which power adjustment or trimming within the reactor core may lead to significant power and burnup variations in some fuel assemblies. For example, such records are kept for some CANDU reactors.

3.3.3.6. Results and consequences of spent fuel tests

After final discharge from the reactor, the ‘leak tightness test’, the ‘visual inspection’ and the ‘geometry inspection’ can be performed. As a result of these tests some fuel assemblies can be identified as defective or failed assemblies: such identified assemblies must be handled according to special procedures or require special consideration. Such assemblies may not be able to fit into a spent fuel rack, may not be able to be lifted normally, or may have cladding defects (leakage) greater than established limits.

3.3.3.7. Special considerations

Several special conditions have been identified which, when present in an individual assembly, will require special considerations. These include fuel assemblies that

- are known to contain defective fuel pins,
- have been enclosed in a special containers (“canned”),
- have had fuel pins removed,
- that have been consolidated, or
- that have been altered in some way.

3.3.3.8. Data collection form: US example RW-859

Assemblies meeting any of these conditions should be identified, and they should have details regarding the nature and extent of the condition recorded with other assembly data. For example, if an assembly is known to contain defective fuel pins, the nature and extent of the defect or damage and the location of the defective pins should be recorded, if known. For “canned” assemblies, a cross-reference to a unique container identifier should be provided. For assemblies from which fuel pins have been removed, the initial weight of heavy metal should be modified to reflect the removed pins, and the location, mass, and burnup of the removed pins should be specified (preferably by the new unique identifier for the container where they are stored). Containers into which fuel pins from consolidated fuel assemblies are placed should be identified, along with the mass and burnup of the fuel contained.

A survey form in use by USDOE/EIA for collection of data on every fuel assembly irradiated in commercial nuclear reactors operating in the US is the Nuclear Fuel Data Survey Form RW-859 which was formulated with a view to application to the Commercial Radioactive Waste Management Programme in the US.

The data files of the RW-859 Form include the following categories of information:

- Facility data,
- Reactor and operation cycle data,
- Data on permanently discharged fuel,
- Pool storage,
- Reinserted fuel and shipment of fuel,
- Canisters and on-fuel components,
- Dry storage,
- Projected assembly discharge,
- Comments.

Further details on the RW-859 Form is attached in the country report of the USA, attached in the Annex, together with a paper which provides background information [25].

3.3.4. Data specification for spent fuel management facilities

3.3.4.1. Storage

The conditions under which spent fuel is stored should be recorded because these conditions can have considerable influence on the subsequent safe management of fuel assemblies. For example, for advanced gas reactor fuel, the initial storage medium can have a considerable influence on the fuel assembly’s subsequent corrosion durability.

In this report it is assumed that a storage facility is composed of one or more storage units. Each storage unit can be wet (pool) or dry (cask) storage. Each storage unit may have a unique position identification system where any stored assembly can be positioned. The data that should be recorded for each facility include:

- storage facility identifier,
- kind of facility (reactor, away from reactor storage, etc.),
- storage unit inventory (storage unit identifier, maximum capacity of storage unit, currently unused (available) capacity), and
- storage unit special documentation inventory.

For each storage unit the data recorded should include:

- unit identifier,
- kind of storage unit (wet or dry),
- position inventory (position identifier, spent fuel assembly identifier (void if not used), initial storage data, final storage data.

For each spent fuel assembly the data recorded should include:

- spent fuel assembly identifier,
- kind of spent fuel assembly.

Generic information common to specific kind of storage unit or specific kind of fuel assembly should also be recorded. This information should include:

- licensed content,
- storage medium (air, water, inert gas, other),
- general physical and chemical characteristics, where applicable (e.g. for water storage, temperature, pH, Cl — content, and other parameters relevant to the maintenance of good storage pool chemistry control such as chemical agents added to reduce biological growth within, and maintain clarity of the pool water),
- temperature of the assemblies in storage and variation, if any, and
- transient or abnormal events, and their date, duration and consequences, if any.

Since an individual fuel assembly may have several storage locations over time, the dates and duration of storage in each facility should be maintained. The data parameters describing the transport of fuel assemblies between storage facilities are described in the following section.

It may be prudent to track the precise location of individual assemblies; however, in many circumstances it may be sufficient only to know that a specific fuel assembly is in a particular storage facility.

As an example, the CaskLoader software developed by EPRI assists utilities in the task of loading spent fuel assemblies into dry casks for storage (see Subsection 2.2.2.2). The software enables commercial cask loads for radiation and heat loading, within regulatory limits and helps select the appropriate spent fuel assembly for each cask by evaluating and calculating (i) neutron and gamma sources for different types of fuel as a function of initial enrichment, Burnup and cooling time, (ii) dose contribution factors for each location on the cask, and (iii) decay heat algorithms.

A key goal is to enable utilities to avoid using up all their "oldest, coldest" fuel in the first casks loaded. This goal can be met by distributing the coldest fuel among more casks and mixing it with hotter fuel and the fuel will be arranged in the cask to maximize self-shielding and thereby minimize external dose, while not compromising the cask thermal limits.

Experience shows that such software packages provide an easy interface for plant personnel responsible for loading casks, performs calculations needed for task optimisation or as required by the regulators, and significantly reduces documentation errors. It also allows to create necessary reports and records to demonstrate compliance with regulatory and specific utility plant requirements, while allowing an easy update for new reporting and control requirements that may emerge in the future.

3.3.4.2. Transport

Each back end route may involve a number of transport stages, including possible transfers of fuel assemblies between storage facilities. Because transport may affect the condition of the fuel, certain data should be retained for each transport. These include

- start and end dates of transport,
- method of transport (e.g. road, rail),
- type of transport cask used (e.g. wet or dry transfer, water, air, or inert-gas medium),
- nominal and worst case temperatures during transfer, and
- unusual conditions or incidents during transport that could affect fuelassembly condition.

During transportation, fuel assemblies may be subjected to vibrational and/or impact loading which, if sufficiently severe, can cause damage. Similarly, a loss of cooling may result in overheating. Any such irregular occurrences should be recorded and the records maintained.

A reference model for database could be developed for transportation of spent fuel, which is stored at AR storage facility after the irradiation in reactor core and then unloaded into a dual-purpose cask and transported and stored at dry type cask AFR store. The same cask is also used after a long term storage period for the transport into a subsequent fuel cycle back end facility as geological repository or reprocessing facility.

3.3.4.3. Reprocessing

In the case of spent fuel reprocessing it must be noted, that due to the destruction of the physical form of the assembly the spent fuel data tracking is considered as finished. However data on MOX fuel and high level waste generated as a by-product of spent fuel reprocessing have to be covered by other database systems and have to be included in the original spent fuel database.

3.3.4.4. Disposal

The importance of each radionuclide must be determined with respect to the safety assessment of the disposal systems for spent fuel or high level waste generated from the reprocessing of spent fuel. These radionuclides have to be included in the spent fuel database in the section on isotopic composition of the spent fuel.

As a part of several ongoing deep geological disposal projects, screening calculations were performed to show the contribution of each radionuclide to the total dose. The importance of radionuclides evaluated by screening calculations depends on:

- the design of geological repository including host material type (clay, salt, granite, tuff),
- the scenarios considered, and
- the physical and chemical properties of near field, far field and biosphere.

The SPA[17] project has identified following safety-relevant radionuclides [26]:

- 10 fission and activation products: (^{14}C, ^{36}Cl, ^{90}Sr, ^{93}Mo, ^{94}Nb, ^{99}Tc, ^{107}Pd, ^{126}Sn, ^{129}I, ^{135}Cs), and
- 13 actinides ($^{236}U \rightarrow {}^{232}Th$; $^{237}Np \rightarrow {}^{233}U \rightarrow {}^{229}Th$; $^{242}Pu \rightarrow {}^{238}U \rightarrow {}^{234}U \rightarrow {}^{230}Th \rightarrow {}^{226}Ra$; $^{239}Pu \rightarrow {}^{235}U \rightarrow {}^{231}Pa$).

The radionuclides evaluated as safetyrelevant in the screening analysis for the Japanese deep geological repository containing high level waste in form of vitrified matrixes from the spent fuel reprocessing contain [27]:

- 8 activation/fission products (^{151}Sm, ^{135}Cs, ^{126}Sn, ^{107}Pd, ^{94}Nb, ^{99}Tc, ^{93}Zr, ^{79}Se),
- 23 actinides ($^{240}Pu \rightarrow {}^{236}U \rightarrow {}^{232}Th$; $^{245}Cm \rightarrow {}^{241}Pu \rightarrow {}^{241}Am \rightarrow {}^{237}Np \rightarrow {}^{233}U \rightarrow {}^{229}Th$; $^{246}Cm \rightarrow {}^{242m}$ $Am \rightarrow {}^{238}Pu \rightarrow {}^{242}Pu \rightarrow {}^{238}U \rightarrow {}^{234}U \rightarrow {}^{230Th} \rightarrow {}^{226}Ra \rightarrow {}^{210}Pb$; $^{243}Am \rightarrow {}^{239}Pu \rightarrow {}^{235}U \rightarrow {}^{231}Pa \rightarrow {}^{227}Ac$).

Similar recommendations with respect to the identification of radiologically important radionuclides have been developed as a part of Yucca Mountain Project [28]. For the TSPA-SR project, the following radionuclides from different waste forms (high level waste, boiling water reactor spent fuel, pressurized water reactor spent fuel) are identified for further examination in the performance assessment of Yucca Mountain repository:

- for a direct release from a disruptive event scenario: ^{90}Sr, ^{137}Cs, ^{227}Ac, ^{229Th}, ^{231}Pa, ^{232}U, ^{233}U, ^{234}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Am, ^{243}Am. These are the isotopes that contribute most to the dose when the release is not mitigated by either solubility or transport.
- nominal release calculations should include: ^{14}C, ^{99}Tc, ^{129}I, ^{227}Ac, ^{229Th}, ^{232}U, ^{233}U, ^{234}U, ^{236}U, ^{238}U, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Am, ^{243}Am.
- numan intrusion calculations should include ^{14}C, ^{99}Tc, ^{129}I, ^{227}Ac, ^{229Th}, ^{232}U, ^{233}U, ^{234}U, ^{236}U, ^{238}U, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Am, ^{243}Am.

[17] Spent fuel disposal performance assessment (SPA) project of European Union (ref. 26)

- in addition, ^{235}U must be tracked through the transport calculation if an accurate release of ^{231}Pa and ^{227}Ac is to be calculated. ^{231}Pa and ^{227}Ac are important contributors to dose. ^{232}Th and ^{228}Ra must be tracked through the transport calculation for the groundwater protection scenario.

3.3.4.5. Advanced/special treatment of spent fuel

The issues associated with data management for advanced and special treatment technologies for spent fuel management have not been well established yet, mainly because of the immature development of technologies and industrialization.

If these technologies are established as in the longer term future, the methodologies being established for conventional options for spent fuel management, as reviewed above, could be applied in the development of required systems.

4. LIFE CYCLE MANAGEMENT OF SPENT FUEL DATA

4.1. Responsibility of data management

The institutional framework for spent fuel data arrangement will mirror that of spent fuel management on national level.

Depending on the regulations, the responsibility for the spent fuel management may be shifted from the operator of the reactor to another party, e.g. to a different private organization or a regulatory or a governmental organisation. In many countries, the operator of the reactor remains responsible for the spent fuel until disposal.

4.1.1. Origin of data on fresh fuel

The fuel manufacturer supplements the design identification protocol with a numbering system to uniquely identify a unit from the total number of assemblies of the same design. Specification compliance details are recorded, together with notifications of concessions.

4.1.2. Reactor operator

4.1.2.1. Pre-irradiation data

The continuity of interim storage conditions for the fresh fuel is the responsibility of the reactor operator until the fuel is irradiated.

4.1.2.2. Irradiation data

The reactor operator is required to record the irradiation history of the fuel assembly to determine the post-irradiation characteristics of the assembly. Information provided by the reactor operator consists mainly on loading details and special considerations.

4.1.2.3. Post-irradiation data

The reactor operator is required to record the post-irradiation history of the fuel assembly. This is because the storage environment can have a considerable influence on the selection and suitability of future management options. The continuity of storage conditions is the responsibility of the reactor operator until the fuel is delivered for AFR storage or for final treatment.

The continuity of storage conditions after transportation is the responsibility of the AFR store operator until the fuel is delivered final treatment.

4.1.3. Operator of subsequent back end facility

The operator undertaking next step in nuclear fuel cycle back end is responsible to record the activity that results in the cessation or transfer of the record.

4.2. Development and maintenance of spent fuel database

The common goal of databases is to provide a uniform and unified source of data on a common basis and to an adequate level of detail. Such reactor specific databases can be combined to track and estimate nuclear material inventories.

The structure for spent fuel tracking system is largely determined to a great extent by a national organizational framework. Nevertheless there are some general issues, reviewing of which can hardly be avoided during spent fuel database development

4.2.1. Database for spent fuel inventory

The most basic data for spent fuel management are the inventories of spent fuel in term of quantity, location, characteristics, etc., often used for analysis or planning. The specific need for spent fuel inventory data varies depending on the ultimate purpose:

- International Level — compilation on a gross tonnage basis mainly for statistical purposes and global trend analysis both for use by IAEA and at the request of Member States;
- National Level — compilation for industry and regulatory purposes on either a gross tonnage or individual assembly basis to assist in planning and public awareness;
- Operator Level — the origination and maintenance of detailed data on individual assemblies by the utility for operational needs or to meet regulatory requirements.

Most of the spent fuel databases have been developed and used at operator or national level. The need for data compilation and reliability is usually proportional to the amount of spent fuel to be management. For planning purpose, accurate projection of spent fuel arising in the future would be also important. A methodology for the projection of spent fuel arising in the future was discussed in the literature [29].

For the global compilation of spent fuel inventory data, at a minimum on a national level, the following information would be useful for statistical status and trend analyses. Where available, the information can be collected at the facility or site level.

The spent fuel inventory should be collected individually for:

- AR and AFR storage facilities;
- Pool (wet) storage and dry storage systems (vault, metal/concrete cask).

Such distinction of storage type and method could be useful in compiling of trends in spent fuel management.

The accounting boundary for inventory of spent fuel begins with the discharge from the reactor and is interfaced with spent fuel management options, including reprocessing, disposal, or other future options such as direct re-fabrication. The identity of a spent fuel assembly can be lost, such as in reprocessing, which will be the end of assembly lifetime. At this time, the accounting method must be changed to a suitable one (e.g. bulk accounting) in order to maintain the balance as the assembly identity no longer exists.

As most of the off-site shipments were or are still delivered to reprocessing facilities (either planned or operating), reliable data on the quantity of material reprocessed as well as the quantity of material in buffer storage at reprocessing facilities is important to enhance the global data on spent fuel inventories and management.

A survey of international databases on spent fuel with a focus on IAEA activities was presented with a recommendation to enhance the data collection and compilation [30]. Need for management of spent fuel inventory data in a context of recent initiative for multi-national cooperation for spent fuel management was also discussed in the literature [31].

4.2.2. Development of spent fuel database

(1) Initial activities for database development

At the first stage of the database development the interactions between regulatory, governmental institutions and nuclear facility operators, which are involved in the spent fuel management activity, should be considered and the organization, which will be responsible for database development activity, should be established. This organization has to coordinate development activity of the spent fuel database and establish requirements to the spent fuel database. At the development stage, the following issues should be solved:

- Who will develop the spent fuel database?
- Which information should be collected/calculated and stored in the spent fuel database?
- Which software/codes should be applied?

(2) The development of database structure, including forms for inputting/outputting data, reports, the information retrieval system. The development of procedures of collecting information and handling it over.

These procedures should guarantee the completeness and correctness of stored data and should be approved by the regulatory body or a governmental institution. The possibility of mistaken data input, caused by human factor, should be analysed and the actions for its minimization should be taken. The software has to provide the control of data completeness and correctness during input of data:

- Who will responsible for collecting and storing information?
- The access level to the data, which is stored in the database, should be determined and the information security measures should be developed.
- Who will responsible for the maintenance of spent fuel tracking system (including software and equipment). The spent fuel database service guide should be elaborated.

4.2.3. Types of data

Data needed on the physical and radiological characteristics of the spent fuel and associated wastes, such as non-fuel bearing components includes radiological and thermal characteristics of the materials. Physical information (dimensions, weight, configuration, identification, etc.)

is needed to allow for efficient planning for transportation and receiving the shipment at facilities in terms of the equipment required. Unique information related to the spent fuel, such as test history and known defects, should also be documented and retained by in a record system.

The specific information that needs to be provided to the spent fuel management organization by utilities, once the specific items to be delivered has been identified, is summarized below:

4.2.3.1. Identification of fuel assemblies

(1) A unique identification number for the item delivered

For spent fuel, this will be the manufacturer's assigned identification number, while canisters will have a separate identification number. This identification number must be unique at least within the inventory of each individual reactor. In this way, this identification number can be combined with the utility's reactor identification to form a completely unique number for purposes of item control.

(2) Date of final discharge

This is the date that the spent fuel was last irradiated, which is necessary to perform decay calculations.

(3) The spent fuel and its associated waste, as an "other" category to accommodate deliveries of non fuel assembly hardware and components

A more complete description of this type of waste will be obtained through other characteristics discussed below.

4.2.3.2. Fuel forms

(1) Physical description of the spent fuel

This will include the drawings that describe the item to be delivered (e.g. type of assembly, control rods, etc). From the drawing, the active length of the fuel assembly, the cladding material, the cross-section of the assembly, the overall assembly length, and the overall assembly weight must be able to be determined.

(2) Physical description of canisters

This will include the drawings that describe the detailed dimensions of the canisters, the weights, the materials of construction, the method of drainage, the lifting feature, and the types of material that are intended to be contained therein. The weight of the canister and contents will be provided for each loaded canister along with an identification of the relevant type of canister.

4.2.3.3. Fuel content

(1) Final discharge burnup

The final discharge burnup of the spent fuel must be reported to allow for the calculation of isotopic contents.

(2) Isotopic composition of spent fuel

(a) Initial enrichment and uranium mass. The weight percent initial ^{235}U enrichment and the total initial uranium mass must be reported to allow for the calculation of isotopic content as well as for criticality calculations.
(b) Post irradiation isotopic content. Specifically, ^{232}Th (for thorium based fuel) and ^{233}U (for thorium based fuel), ^{235}U, ^{238}Pu, ^{239}Pu, ^{240}Pu and ^{241}Pu content (in weight percent concentrations) at the time that the spent fuel is permanently discharged from the reactor (final cycle end date) must be calculated and submitted, and each must be reported separately.
(c) Post irradiation uranium and plutonium content. Total fissile uranium and plutonium content in grams must be calculated and submitted for each assembly as of the date of discharge. ^{238}Pu should be reported in tenths of a gram. This information must be collected to allow the spent fuel management organization to submit required reports to regulatory authorities and IAEA after it becomes responsible for such reporting.

(3) Irradiation history in the reactor

This includes a history of the power levels experienced during fuel residency in the reactor, shutdowns, change of position of the fuel in the core, data on insertion/usage of control rods in and adjacent to the assemblies, and the use of burnable poison absorbers — which, along with the initial enrichment, allow the accurate calculation of burnup and isotopic composition of the spent fuel.

4.2.3.4. Other types of data

(1) Description of any unusual incidents associated with the use or handling of the spent fuel and associated waste or any changes, whether intentional or accidental, to the fissile material content.

This should include a discussion of anything that might affect the way that the waste is handled, packaged, transported, stored, or disposed of. Any removal of rods within the assembly must be described.

(2) The results of any measurements, tests, inspections, or observations associated with spent fuel.

This should not include the results of any testing that may be determined necessary for acceptance at another facility for either long term storage, reprocessing or disposal (such as burnup measurements), but should instead include any measurements, tests, inspections or observations that have been conducted by the utility independent of these requirements.

(3) Fabricators special nuclear material transfer documents.

A copy of the documents that were provided by the fabricator with the fresh fuel when it was transferred to the utility that showed the weight and isotopic content of the contained uranium and weight of UO_2, including any changes subsequently made by the utility.

4.2.4. *Verification of data*

Validation of data received from utilities should be accomplished by the following:

(1) First, confirming that all necessary information has been submitted to the spent fuel management organization for each item of spent fuel and non fuel bearing component delivered by the utilities.
(2) Then comparing the information submitted to a set of bounding parameters. If a data item is identified that falls outside of the bounding parameters, the utility should be asked to confirm that the value as reported is correct. If the utility reports that the value is correct, the value would be retained.
(3) No information would be accepted that is not collected under a qualified QA programme, approved by the spent fuel or waste management organization. This QA programme should require extensive confirmation of utility calculations and records to the point that it should be unnecessary for the spent fuel management organization to perform any additional confirmation exercises.

Physical verification of the unique identity of each item of spent fuel and associated non-fuel bearing component should be made by the utility immediately prior to emplacement of the material into a cask and/or canister. For example, this verification might be accomplished by two individuals independently reading the number of the assembly, recording this number on paper, and then comparing the two records.

The verified information would be recorded on a form developed by the spent fuel management organization to establish a continuity of knowledge record for the material. If sealed canisters are used to store spent fuel, this information should be recorded at the time the canister is sealed. Completing the form at the time of canister sealing will maximize the probability that all required information is recorded in an acceptable level of detail. Placing nuclear material in a sealed canister will also require the recording of the canister's unique identification number.

Verification of the data must be expanded to include the collection of information necessary to establish continuity of knowledge regarding the spent fuel in order to identify the quantity of radioactive material per item, locate the material by item identification, provide a recorded number and identify the storage or disposal location.

The final component of verification should consist of verifying that the spent fuel to be accepted at another facility is properly loaded, packaged, marked, and ready for transportation, and that appropriate documents have been executed for the transfer of title, if any. This verification should occur immediately before title is transferred. This component of verification should include confirmation that the identity of the spent fuel is as expected, whether non-canistered or canistered spent fuel is being delivered.

As in the case of the verification process described for individual assembly identification, verification of either the identity of the non-canistered spent fuel placed in a transport cask or of a canister to be delivered should be accomplished by the utility using a method approved by the regulatory authorities for item identification.

4.2.5. Auditable by regulators/fuel users

Procedures for collecting, transmitting and incorporation into each of the data sets of the spent fuel database need to be established and include provisions for verifying the acceptability of each data. To be acceptable, each data must be legible, authentic, accurate and complete. It will be beneficial to audit regularly the design, construction and operation of spent fuel database to assess it properly manages the data important for spent fuel management. The auditing process is intended to ensure that proper data are flagged for inclusion or consideration at the time of transportation for AFR storage, reprocessing, re-fabrication and disposal and also at the time of any conditioning.

Each data contained in the spent fuel database is essential to the safe operation of the nuclear facilities such as storage, reprocessing, re-fabrication and repository including transportation. Regulatory body or governmental institution should approve the list of data, which will be stored in the database from the point of information completeness.

4.2.6. Enhancement of data reliability

4.2.6.1. Upgradeable systems

Improvement of digital information technology is fast evolving and might result in the incompatibility of inputting data form, reading pickup technology, recording media and software. Durable spent fuel database needs the persistent migration and retrievability of recorded data and also standard forms and languages of inputting and outputting data.

4.2.6.2. Capability to include improvements in measurement technology and commercial capability

Improvement of accuracy on measurement and calculation should be accepted to the spent fuel database, which requires contextual information framework to show how to have got the data sets. Standardization is needed to manage the different style of data in the spent fuel database.

4.2.6.3. Completeness of records

The records do not exist in isolation, but exist in a multi-layered contextual framework defined by various entities including the individuals, the organisations, the regulatory framework, the bureaucratic governmental structure, the international community and the scientific basis for the industry. To understand the records and their content in the spent fuel database it is not sufficient to attempt to interpret the records in isolation, they must be interpreted in context. Access to contextual information is necessary for the realistic interpretation of records.

Recommendations concerning information, which should be stored, are in the Chapters 2 and 3 of this publication and a sample database structure is shown on Fig. 4.

As mentioned in the previous sub-chapters, audit by regulators and fuel users, approval by regulators and quality assurance are needed to be established for completeness of records such as inputting error correction, data maintenance and quality assurance of the spent fuel database system.

4.2.6.4. Frequency of review

Review should be done at the approved frequency to keep the quality at the predetermined level. For example frequency of review should be regularly or at the time of records migration and alteration of measurement and calculation methodology.

4.2.6.5. QA of the data acquisition and maintenance

The responsible organization will have a QA programme for a nuclear facility as an integral part of its management system. The QA programme should be modified at various stages (e.g. at the design, operation and decommissioning of a facility) at a time consistent with the schedule for accomplishing stage related activities.

A spent fuel database is an important part of the overall QA programme for each facility. This system ensures that data are specified, prepared, authenticated and maintained, as required by applicable codes, standards and specifications. Controls need to be established to identify the personnel authorized to make modifications to data, and the conditions under which modification may be made. Methods of controlling access to data need to be established and documented to prevent loss, destruction or unauthorized alteration of data. Controls might include identification of organizational responsibility for authorizing and controlling access to data.

The QA programme also provides some form of routine review of the quality and completeness of the data, based on the information required. The spent fuel database needs to incorporate the archive requirements into instructions, procedures and plans. Those data identified for the highest importance are to be maintained at the highest requirement level. In the event that an international archiving body is established, that body might produce specific requirements to be incorporated in the spent fuel database.

4.3. Lifetime management of spent fuel data

Member States now have diverse policy on spent fuel management such as direct disposal, reprocessing and re-fabrication through storage. The period of storage is now considered to be several decades, but might be extended up to or more than hundred years, for which period the spent fuel database also should be maintained.

During such long periods, it is unlikely that the ownership, operation, maintenance and management system would be the single responsible organization and/or covered by one continuous management system. Experience in other industries has shown that it is at times of change in institutional control that the spent fuel database is at most risk of failure.

4.3.1. Record keeping

Records must be tangible if they are to be useful over time. As is still generally the case, many records relating to the design of the spent fuel database exist on paper. Many contemporary records will also exist in a digital form and be housed on variety of media including magnetic drives in servers, compact discs and other media [32].

However, significant change affecting the spent fuel database is inevitable not just in the organisations directly responsible for long term management of spent fuels, but in all the ancillary organisations and entities. Change is also an inevitable product of science and technology. Digital information system has proper weakness such as possible sudden total loss of recording media and frequent change of the reading pickup technology and computer software and systems.

Furthermore, records exist within systems and networks, including contextual frameworks, administrative, scientific and cultural environments that are constantly changing. The issue of continuous and pervasive change is perhaps the most fundamental challenge facing the industry. They have to be robust and to be capable of evolving to meet the needs of a continuously changing environment.

4.3.2. Backup of data

Data controls for periodically ensuring durability of the information contained in the spent fuel database need to be established based on the data form(s) (e.g. microfilm, paper, digital form, etc.). The expected life for each data form needs to be established and controlled to ensure that data are migrated to the same or another form prior to the end of their expected life. Controls to ensure and verify the legibility and integrity of migrated information also need to be established. Appropriate remedial actions must be taken to restore deteriorated data.

For long term retrievability, procedures must be established to ensure that the tools necessary for reading the data (for example, microfilm readers, computer software and systems) continue to be available. Any loss of information during migration of data must be documented. The document may determine or estimate the extent and contents of the lost data.

4.3.3. Search for record keeping methods for long period of time

To transfer the information to future generations related to safety of radioactive waste disposal facilities, historical and current issues relating to long term record preservation were comprehensively reviewed and discussed in several countries [33], [34], [35] and IAEA [36] and clarified the requirements to strengthen the robustness and redundancy of long term record preservation system.

Recently an extremely long lasting robust system has been developed in Japan. That is laser-engraving information onto silicon carbide plates, which is one promising technology for recording data on robust media. Silicon carbide plate is the most durable artificial material invented in terms of strength, thermal and corrosion resistance and wear due to abrasion. It would be possible to preserve records without the need for sophisticated preservation or environmental controls and without the need for human intervention to initiate a duplication programme for over 1,000 years.

For records to be useful they have to be accessible, readable and understandable. Open networks [37] are the most suitable challenge to transfer the information to the future generations in the contextual framework based on the international standards by the International Organization for Standardization (ISO) and the International Council on Archives (ICA). They consist of main and local computers, being supported by paper documents and digital archives and well matured as the robust information transfer system.

4.4. Maintenance of records

As mentioned in Section 1, the concept of hierarchical record management is a key factor for long term management of data over different phases, like disposal of spent fuel.

Depending on the legal regulations in Member States, the responsibility for collecting, inputting and outputting, changing and storing data of the records in the spent fuel database could be imposed on nuclear facility operator, regulatory or governmental organisations.

The organization, which is responsible for records maintenance, should provide the reliable information storage by means of:

- The fulfilment of procedures for information handling.

The elaborated procedures, which provide the completeness and correctness of stored data, should be revised periodically. Furthermore completeness and correctness of data should be controlled by means of software.

- The availability of skilled staffs.

After changing procedures or software, the staffs have to be trained.

- Creation of an archive database.

The archive of spent fuel database should be kept in a different place away from the location of the main spent fuel database. The archive should be updated with established periodicity, which depends on the records change frequency of spent fuel database.

Regulatory or governmental institutions establish the information keeping term as well as the data list. Any changes in the keeping terms, data list or procedures instructions should be approved by authorities.

Responsibility for the records maintenance can not be shifted from one side to another one without authorization given by regulatory body or governmental institutions.

4.4.1. Records maintenance responsibility transfer

Spent fuel information may also have to be considered in the context of nuclear knowledge preservation. In addition to current inventories, historical and projected data is important for some purposes such as consistency analysis.

The records maintenance responsibility may be shifted in the following cases:

- Shift of responsibility for spent fuel management,
- Spent fuel shipment between nuclear facility sites,
- Reorganization or ownership transfer.

The information/database transfer should be performed according to established procedures, which prevent the data loss. These transfer procedures should be appproved by regulatory or governmental institutions. Before transfer responsibility for data keeping, the following main issues should be agreed:

- Content of transmitted information,
- Information format and information-carrying medium,
- Responsibility for collecting information concerning of the spent fuel shipment conditions.

If the content of transmitted data is more than that stored in the spent fuel database, the redundant information issue must be resolved. The redundant information can be archived or removed. Any information can not be eliminated without authorization given by regulatory or governmental institution.

4.4.2. Elimination of unnecessary data

The spent fuel records should be maintained until the authority considers that the records are no longer required. Mostly the spent fuel data can become unnecessary when fuel is reprocessed, re-fabricated or conditioned for disposal. In order to prevent the premature or mistaken spent fuel data removal, the elimination procedures should be established and be proved by regulatory or governmental institutions.

In order to reduce the risk of deleting information by mistake, a redundant set of records may have to be archived for a required period before deleting data permanenetly.

4.4.3. Inspection of spent fuel database

Representatives of regulatory or governmental institutions should periodically inspect the spent fuel database system for the following issues:

- Availability of documentation, which should meet the regulatory/governmental institutions requirements,
- Effectiveness and observance of the established procedures,
- Guarding information reliability,
- Staff training level.

The organization responsible for spent fuel database records maintenance must take appropriate measures for full compliance with regulatory or national or international requirements.

4.4.4. Maintenance of spent fuel database

Maintenance of the spent fuel database, including equipments maintenance, should be provided during all period of spent fuel database utilization.

Taking into account the extended period of spent fuel database utilization, the spent fuel database will need to be upgraded or modified because of operator complaints or due to changes in regulatory requirements. After any modification or upgrade of the spent fuel database, the procedure adequacy should be analyzed and appropriate changes be made, as necessary.

For data backtracking purposes it is highly recommended to store all data sources, in both electronic and hardcopy versions if possible. The responsibility for maintenance of this data archive has to be clearly identified and a list of all changes of data, codes and versions has to be available to all spent fuel database users on a regular basis. The same is true also for the technical support on a day-to-day-basis.

4.4.5. Cessation of data tracking (termination of record keeping on reprocessing or refabrication)

As mentioned above spent fuel management policy could be pursued with different paths: to dispose of as waste, reprocess or re-fabricate. If the spent fuel management policy shifts from one to another, the spent fuel database system would also have to be transferred accordingly and divided into some groups with the contextual information framework.

Cessation of data tracking determines the duration of the life cycle of the spent fuel database. It is related to the next step of nuclear fuel cycle, where the spent fuel assemblycould either be reprocessed, thus loosing all uniquely identifiable properties, or is declared as radioactive waste for disposal in which case selected data are included into the radioactive waste database. In the latter case spent fuel data tracking continues after its disposal.

4.4.5.1. Cessation at reprocessing/refabrication

Cessation of spent fuel tracking may occur at the head end of a reprocessing line, or when the fuel assembly is refabricated and assigned a separate and unique identification number (see 2.2.4). In any of these cases, the reason and date for the cessation of tracking should be recorded.

4.4.5.2. Cessation at disposal

The requirement to continue tracking spent fuel assemblies and their characteristics after they are placed into waste disposal containers may be necessary for safety or safeguards purposes. This is especially important since repository design may encompass a period of potential spent fuel retrieval and/or for either transport to further storage or to another back end facility

4.5. Other issues associated with spent fuel data management

The past evolution in nuclear industry and subsequent institutional arrangements for spent fuel management in the Member States has, nevertheless, affected the positions of the stakeholders with respect to information management and thereby the demand on concerned information to the responsible organizations.

4.5.1. Trends toward transparency

There is in general a global trend towards greater transparency of information with the general public which may require more information to be made public on spent fuel management, including data on storage inventory or transportation. With the increase in the commercialization of the nuclear industry, the trend is away from national government operation of nuclear activities. This results in the spread of information on spent fuel as it is not concentrated in a vertical manner at the government level, but is instead held by various organizations in the private sector in a more horizontal manner.

4.5.2. Policy factors

The management of information on spent fuel is also affected by national policy on spent fuel management. Some countries are pursuing reprocessing of spent fuel which is regarded as resource, in contrast to others adopting once-through (direct disposal) policy declaring spent fuel as waste. In the former case, information on spent fuel is in general managed separately from radioactive waste either by pertinent national organization or by the commercial entity involved in the reprocessing business, while in the latter case spent fuel data are managed in the frame of radioactive waste information management.

Recently, the issue of security in spent fuel management has also begun to be considered in the information management associated with spent fuel or radioactive waste. Such consideration brings a debate on the level of information on the spent fuel management to be released in the public domain.

4.5.3. Responsibilities for data management

The spent fuel database recognizes the regulatory body, producer of spent fuels and operator of nuclear facilities as responsible for maintaining the documentation and data consistent with the legal requirements and their own needs. The regulatory body may choose to take the responsibility for the long term retention of the data. Nevertheless, it would be prudent if the Member State would identify a responsible body tasked with the definition of the objectives, goals and minimum contents of the database. This body would identify the organization(s) who is responsible for defining, developing, operating and maintaining the spent fuel database. There are three kinds of responsibilities to be identified for different tasks. These responsibilities have to be clearly distinguished concerning:

- regulatory process (licensing authority),
- implementation of the data collection,
- management of spent fuel database.

Special attention should be paid to the accuracy of data included the spent fuel database as it can significantly affect the usability of the database for other purposes e.g. as a source of inventory data for safety case of deep geologic repository. The national regulatory bodies and other national authorities involved in spent fuel management should therefore identify the requirements on data accuracy and its recording.

5. CONCLUSIONS AND RECOMMENDATIONS

Data for spent fuel are produced and collected in all phases of the spent fuel management activities. The quantity of information will depend on the management options used (long term storage, reprocessing or refabrication, disposal) and the rules and regulations regarding documentation procedures and quality assurance requirements in each Member State.

Key data on spent fuel must be available to make an informed decision in the planning and implementation of technical options for spent fuel management, including storage, transportation, reprocessing or disposal. They are required for the lifetime management of facilities including design, licensing, construction and operation and decommissioning of facilities or equipment.

Generic issues in spent fuel data management are to:

- identify what data level sets will be needed and the level of detail of each data,
- identify and document procedures for extracting the data needed for spent fuel management options that may be implemented in the future,
- transfer such data to record media with high longevity,
- establish the interfaces between the different data level needs.

The prospect for availablity of information on fuel cycles in general and on spent fuel management in particular will depend on the necessity of such data. There is on the one hand a global trend towards greater transparency of information on nuclear activities with regard to the general public, requiring more information open to the public on spent fuel management, including data on storage inventory or transportation. On the other hand, there is a trend to globalization and privatization of the nuclear industry, driving away from national government operation of nuclear activities. Should any of the initiatives for mult-national or regional approach bear fruit, management of the spent fuel data at international level would become an essential requirement [38].

Appendix I

EXAMPLE OF DATA REQUIREMENTS FOR DISPOSAL

Disposal-Relevant Properties and Characteristics of Spent LWR Fuel Elements Conditioned in Disposal Containers *(POLLUX Container and Fuel Rod Canister)*

(Status 02 March1999)

No.	Disposal-Relevant Properties Characteristics/Requirements	Dimension	Documenttation	Control Procedure
1.	**Total Activity** – β/γ Activity – α Activity – Neutron emission rate	 Bq/pack. Bq/pack. s^{-1}/pack.	 D/S D/S D/S	Determination of the activity by means of burnup, decay and activation calculations
2.	**Activities of relevant radionuclides** – Activation products (Co-60, Cl-36 etc.) – Volatile radionuclides (H-3, Kr-85, C-14, I-129, Rn-222 {as Ra-226} etc.) – Fission products (Cs-137, Sr-90, etc.) – Actinides (Np-237, Am-241, Cm-244, U and Pu isotopes etc.) *A compilation of all safety relevant radionuclides can be found at the end of this table.*	 Bq/pack. Bq/pack. Bq/pack. Bq/pack.	 D/S D/S D/S D/S	Burnup, decay and activation calculation
4.	**Thermal characteristics** – Thermal output	 kW/pack.	 D/S	Calculation by means of activity inventories
5.	**Dose rate at the waste package** – γ (surface) – γ (in 1 m distance) – n (surface) – n (in 1 m distance)	 Gy/h Gy/h Gy/h Gy/h	 D/S D/S D/S D/S	Measurement at the POLLUX disposal container and the shipping container respectively, calculation for fuel rod canister
6.	**Surface contamination** – α contamination – β/γ contamination	 Bq/cm^2 Bq/cm^2	 D/S D/S	Measurement at the POLLUX disposal container and the shipping container respectively, technical description for fuel rod canister as verification

No.	Disposal-Relevant Properties Characteristics/Requirements	Dimension	Documenttation	Control Procedure
7.	**Description of the waste product** – Fuel element type, nature and origin (e.g. PWR, BWR, MOX; initial enrichment; Pu/U percentage, Pu and U vector (reprocessed uranium) for MOX fuel elements; NPP delivering the fuel element; lifetime; date of unloading) – Burnup – Mean linear power rate of the rod – Length and diameter of the fuel rods – Fraction of defect rods, material and service data of substitute blind rods (dummies) – Mass of heavy metal (U, Pu) in the • waste package • fuel rod – Mass of the cladding material – Type of cladding material – Mass of spacer grids, end pieces and other structural pieces – Type of the pieces mentioned above – Free volume in the fuel rod plenum – Gaseous constituents – Residual moisture – Remaining volume – Filling gas (e.g. He, air)	– MWd/t W/m mm – kg/pack. kg/fuel rod kg/pack. – kg/pack. – l/pack. – – l/pack. –	D/S D/S D/S D/S D/S D/S D/S D/S D/S S S S D/S D/S S S	Description of the fuel element, material, process, waste package; Identification and tracing of individual fuel elements and fuel rods – Documentation for defect rods – Documentation for defect rods
8.	**Hydrolytic resistance and release of radionuclides** – Waste product • Corrosion behaviour in relevant solutions, long term corrosion rate • Instantaneous release of relevant nuclides • Hydrogen generation • Behaviour of radionuclides being released from the fuel matrix, cladding material and structural pieces – Disposal container (incl. welding seams) • Corrosion behaviour in relevant solutions, long term corrosion rate • Hydrogen generation	$FIAP_i$/a $FIAP_i$ m³/a – µm/a m³/a	S S S S S S	Results of relevant R&D work

No.	Disposal-Relevant Properties Characteristics/Requirements	Dimension	Documenttation	Control Procedure
9.	**Description and quality of the disposal container** – Material specification – Packaging (inner container, outer container, seal) – Empty weight – Dimension • Height • Diameter • Wall thickness – Leak tightness of welding seams – Corrosion protection (base material, welding seam) – Mechanical and thermal stability of the POLLUX container	 name – kg mm mm mm – – –	 D/S D/S D/S D/S D/S D/S D/S D/S D/S	Documentation and accompanying inspection, type testing – Qualification of the welding process, welding parameters, accompanying inspections during fabrication – Type testing as disposal container
10.	**Mass of Disposal package**	kg	D/S	Weighing or calculation respectively
11.	**Labelling of the disposal package** – unique, distinctive, permanently	 –	 D/S	 – visual inspection, for fuel rod canisters combined with inspection of documentation

FIAP$_i$ **F**raction of **I**nventory of Radionuclide **i** Released into **A**queous **P**hase,
FR **F**uel **R**od
D Supporting documentation of the waste package (Gebindebegleitdokumentation),
S Specification

Safety relevant radionuclides

The following compilation contains the safety relevant radionuclides for the safe implacement in the repository according to the current status of experiences in the field of radioactive waste management in Germany. The compilation is based on safety relevant criteria like “Specified normal emplacement operation” (shielding, retention of volatile radionuclides), “Incidents during the operational phase”, “Heat impact to the host rock”, “Criticality safety” and Long term safety”. This list can be amended, if necessary, when the acceptance requirements for heat-generating wastes are definitely defined.

β/γ emitters: H-3, C-14, Cl-36, Ca-41, Mn-54, Fe-55, Co-58, Co-60, Ni-59, Ni-63, Se-79, Kr-85, Sr-90/Y-90, Mo-93, Zr-93, Zr-95/Nb-95, Nb-94, Tc-99, Ru-103, Ru-106/Rh-106, Pd-107, Ag-110m, Sb-124, Sn-126, Sb-125, I-129, Cs-134, Cs-135, Cs-137, Ce-144/Pr-144, Pm-147, Sm-147, Sm-151, Eu-152, Eu-154, Eu-155, Ho-166m, Ac-227, Pu-241, Pu-243, Am-242m.

α emitters: Ra-226, Th-232, Pa-231, U-232, U-233, U-234, U-235, U-236, U-238, Np-237, Pu-238, Pu-239, Pu-240, Pu-242, Am-241, Am-243, Cm-242, Cm-243, Cm-244, Cm-245, Cm-247, Cf-249, Cf-251, Cf-252.

REFERENCES

[1] UNITED STATES DEPARTMENY OF ENERGY, Central Internet Database (http://cid.em.doe.gov).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, VISTA — A Fuel Cycle Simulation System, (http://www.iaea.org/OurWork/ST/NE/NEFW/nfcms_b4_03_VISTA.html).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Records for Radioactive Waste Management Up to Repository Closure: Managing the Primary Level Information (PLI) Set, IAEA-TECDOC-1398, IAEA, Vienna (2004).

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Waste Inventory Record Keeping Systems (WIRKS) for the Management and Disposal of Radioactive Waste, IAEA-TECDOC-1222, IAEA, Vienna (2001).

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Maintenance of Records for Radioactive Waste Disposal, IAEA-TECDOC-1097, IAEA, Vienna (1999).

[6] NUCLEAR ENGINEERING INTERNATIONAL, Fuel Review: Design Data, Nuclear Engineering International, 2005.

[7] JOHNSON ASSOCIATES, E.R., Classification of Spent Nuclear Fuel, Oak Ridge National Laboratory, ORNL-sub-89-41/1, Washington, DC (1990).

[8] AMERICAN NATIONAL STANDARD INSTITUTE, Characterizing Damaged Spent Nuclear Fuel for the Purpose of Storage and Transport, ANSI N14.33-2005) Washington, DC (2005).

[9] INTERNATIONAL ATOMIC ENERGY AGENCY, Survey of Wet and Dry Spent Fuel Storage, IAEA-TECDOC-1100, IAEA, Vienna (1999).

[10] UNITED STATES NUCLEAR REGULATORY COMMISSION, Yucca Mountain Review Plan, NUREG-1804, Final Rev. 2.

[11] HIRUO, E., "Repository target dates unknown as DOE shifts to canister system", Nuclear Fuel, 7 Nov. 2005.

[12] INTERNATIONAL ATOMIC ENERGY AGENCY, Status and Trends in Spent Fuel Reprocessing, IAEA-TECDOC-1347, IAEA, Vienna (2005).

[13] SAVEROT, P., "Spent Fuel Burnup Measurements Systems in France". Proceedings of the 11th INMM Spent Fuel Management Seminar (1994).

[14] SCHLESINGER, H-J., "AVK-The German NPP Waste Flow Tracking System", ICEM 2001 (8th International Conference on Radioactive Waste Management and Environmental Remediation, Sept 30th–4th Oct 2001, Brugge, Belgium).

[15] KO, W-I., et al., "Economic Analysis on DUPIC Fuel Cycle", Nuclear Technology, Vol. 134 (May 2001), pp. 110–186.

[16] NA, M-K., "A Spent Fuel Combination Method for Minimizing Composition Variability of DUPIC Fuel", Nuclear Science and Engineering, vol. 142 (2002), pp. 315–326.

[17] INTERNATIONAL ATOMIC ENERGY AGENCY, High Temperature Gas Cooled Reactor Technology Development, IAEA-TECDOC-988, IAEA, Vienna (1997).

[18] NUCLEAR ENERGY AGENCY, "Accelerator-driven Systems and Fast Reactors in Advanced Nuclear Fuel Cycles — A Comparative Summary", OECD/NEA, Paris 2002.

[19] INTERNATIONAL ATOMIC ENERGY AGENCY, Implications of Partitioning and Transmutation in Radioactive Waste Management, Technical Reports Series No. 435, IAEA, Vienna (2004).

[20] INTERNATIONAL ATOMIC ENERGY AGENCY, Documentation for Use in Regulating Nuclear Facilities, IAEA Safety Standards Series No. GS-G-1.4, IAEA, Vienna (2002).
[21] INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation Protection and Radioactive Waste Management in the Operation of Nuclear Power Plants Safety Guide, IAEA Safety Standards Series No. NS-G-2.7, IAEA, Vienna (2002).
[22] INTERNATIONAL ATOMIC ENERGY AGENCY, The Management System for Facilities and Activities, IAEA Safety Standards No. GS-R-3, IAEA, Vienna (2006).
[23] INTERNATIONAL ATOMIC ENERGY AGENCY, Predisposal Management of High Level Radioactive Waste Safety Guide, IAEA Safety Standards Series No. WS-G-2.6, IAEA, Vienna (2003).
[24] INTERNATIONAL ATOMIC ENERGY AGENCY, Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management, INFCIRC/546, IAEA, Vienna (2001).
[25] PAYTON, M-L, GIBBARD, K., "RW-859 — A Key Link Between Government and Utilities", Proceedings of the 3rd International Conference on HLRWM (12–16 April 1992, Las Vegas).
[26] EUROPEAN UNION, Spent Fuel Disposal Performance Assessment (SPA Project), EUR 19132 EN, 2000.
[27] JAPAN NUCLEAR FUEL CYCLE INSTITUTE, Project to Establish the Scientific and Technical Basis for HLW Disposal in Japan, JNC, 2000.
[28] USDOE, Status of Radionuclide Screening for Total Systems Performance Assessment — Site Recommendation (TSPA-SR). Input Transmittal R&E-PA-99217.Td. Las Vegas, Nevada: CRWMS M&O 1999d. ACC: MOL.19991117.0104.
[29] MCLEOD, B., POLLOG, T., "Methodology for the projection of spent nuclear fuel discharge", International Conference on High Level Radioactive Waste Management (IHLRWM) 2003, 30 March–2 April, 2003, Las Vegas.
[30] LEE, J.S., "International database on spent fuel inventories and management", paper presented at the Technical Meeting on Fissile Materials Management Strategies for Sustainable Nuclear Energy, 12–15 Sept. 2005, IAEA, Vienna.
[31] SMITH, M., "Status of Nuclear Materials in Power Reactor Programs", paper presented at the Multilateral Technical and Organizational Approaches to the Nuclear Fuel Cycle Aimed at Strengthening the Non-proliferation Regime (13–15 July, 2005, Moscow), IAEA, Vienna (2005).
[32] WISE, M., et al., "For the Record…", Nuclear Engineering International, Nov. 2005.
[33] ENG, T., NORBERG, E., TORBACKE, J., JENSEN, M., Information, Conservation and Retrieval, SKB Technical Report 96-18, SKB, Stockholm (1996).
[34] TRAUTH, K.M., HORA, S.C., Expert Judgement on Markers to Deter Inadvertent Human Intrusion into the Waste Isolation Pilot Plant, SAND92-1382, Sandia National Lab. (1993).
[35] OHUCHI, J., et al., "Conceptual System of Robust Record Preservation on Geological Disposal", International Conference on High Level Radioactive Waste Management (IHLRWM) 2003, 30 March–2 April, 2003, Las Vegas, American Nuclear Society.
[36] INTERNATIONAL ATOMIC ENERGY AGENCY, Maintenance of Records for Radioactive Waste Disposal, IAEA-TECDOC-1097, IAEA, Vienna (1999).

[37] McCARTHY G., IAEA, Hungarian Radioactive Waste Disposal, Contextual Information Framework — Demonstration Web site (http://barney.asap.unimelb.edu.au/iaea/IAEAhome.htm).

[38] INTERNATIONAL ATOMIC ENERGY AGENCY, Technical, Economic and Institutional Aspects of Regional Spent Fuel Storage Facilities, IAEA-TECDOC-1482, IAEA, Vienna (2005).

Annex

COUNTRY REPORTS

As indicated in Section 1.5, “Structure of the publication”, this annex provides country reports as actual examples in the practice of spent fuel data management in some Member States, thanks to those contributors who were willing to submit the country reports.

In addition to the ‘generic’ information provided in the main text, the country reports are expected to show some specific examples on the institutional arrangements and methodologies used by those Member States for spent fuel data management, which might be informative to some others for improving existing systems or for future implementation of a new plan.

DATA MANAGEMENT FOR SPENT FUEL FROM POWER REACTORS IN ARGENTINA

A.M. BEVILACQUA[1], O.E. NOVARA[2], M.A. AUDERO[3]

[1] Nuclear Materials Division and Instituto Balseiro, Comisión Nacional de Energía Atómica (CNEA) and Universidad Nacional de Cuyo, San Carlos de Bariloche, Argentina
[2] Nuclear Fuels Department, Centro Atómico Constituyentes, CNEA, San Martín, Argentina
[3] Radioactive Waste Management National Program, CNEA, Buenos Aires, Argentina

Abstract

Updated data for the spent fuel management from the operating power reactors — as of December 31st, 2004 — as well as research reactors — as of March 1st, 2003 — in Argentina are presented. Data for the power reactor spent fuel are received from the nuclear power plant operator (Nucleoeléctrica Argentina S.A.) twice a year for the cumulative spent fuel arising up to June 30th and December 31st. Data for the research reactor spent fuel are collected once a year from CNEA operators. At the time being, such data are not managed in a database management system but some of them are handled with a spreadsheet program in order to get total, average, lower and higher values. These values are being used to built the input of codes for calculating the composition, activity and thermal power for the spent fuel as a whole as well as the mass, activity and thermal power for spent fuel elements or nuclides.

1. ORGANISATIONAL FRAMEWORK FOR THE SPENT FUEL MANAGEMENT

In Argentina the spent fuel (SF) and radioactive waste (RW) activities are carried out according to the National Act Nr. 25018 ("Radioactive Waste Management Regime") passed by the Parliament in 1998. This Act compels the Comisión Nacional de Energía Atómica (CNEA) as the governmental organisation responsible for the strategic planning for the management of the spent fuel and the radioactive waste generated in the country. Furthermore, Argentina is a Contracting Party of the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management entered into force on June 18th, 2001.

The SF generators have to pay for their management and are responsible for its conditioning and storage until it is transferred to the CNEA, in accordance with the acceptance conditions established by that organisation. SF generators as well as CNEA activities related to radiological and nuclear safety are regulated and controlled by the Autoridad Regulatoria Nuclear (ARN).

Particularly in the case of the spent fuel from power reactors, the owner and operator Nucleoeléctrica Argentina S.A. is responsible for its management, including the storage, during the operation of the nuclear power plants (NPPs). After the operating life of the NPPs the spent fuel will be transferred to the CNEA, that it is also responsible for the NPPs decommissioning.

The Radioactive Waste Management National Program (RWMNP) of the CNEA is responsible, as stated by the National Act Nr. 25018, for requesting, collecting and manage the data related to the spent fuel.

2. POWER REACTORS

Argentina has two power reactors in operation: one 357 MWe pressurised vessel heavy water reactor (HWR) designed by KWU at the Atucha-1 NPP in operation since 1974 and a 648 MWe pressurised tubes HWR (CANDU) designed by AECL at the Embalse NPP in operation since 1983.

2.1. Atucha-1 NPP

The fuel assembly (FA) has an active length of 5.3 m and a circular cross-section of 0,10 m diameter either with 36 fuel rods plus one structural rod, for natural uranium (U_{nat}) as well as slightly enriched uranium (SEU) FAs, or with 37 fuel rods, for SEU FAs. Values presented below correspond to the update performed on December 31st, 2004 for the spent fuel generated from November 16th, 1974 to December 31st, 2004.

For almost 27 years, from November 16th, 1974 to July 31st, 2001 this NPP discharged FAs that were loaded with an average of 152.82 kg of U_{nat}. A total amount of 8055 U_{nat} spent fuel assemblies, corresponding to 1231.00 metric tonnes (ton) of initial U_{nat}, were generated with an average burnup of 5957 MWd/tU. The total amount of plutonium contained in these FAs is 4124.34 kg and the fission products (FP) activity amounts 2.23×10^6 TBq.

In January 1995 Atucha-1 NPP started a programme to gradually convert its core to 0.85% ^{235}U SEU that was completed on July 31st, 2001. Therefore, since October 30th, 1995 Atucha-1 NPP discharges FAs that are loaded with an average of 155.19 kg of SEU. A total amount of 912 SEU spent fuel, corresponding to 141.53 ton of initial SEU, were generated with an average burnup of 10389 MWd/tU. The total amount of plutonium contained in these FAs is 627.54 kg and the FP activity amounts 7.49×10^6 TBq.

The core conversion to SEU produces an important saving in FAs consumption and spent fuel generation: from approximately 395 FA/full power years (fpy), equivalent to 60.36 ton of initial U_{nat}, to approximately 210 FA/fpy, equivalent to 32.59 ton of initial SEU.

The spent fuel in interim storage in the pools, made of concrete with stainless steel lining, amounts a total of 8967 FAs corresponding to 1372.53 ton of initial uranium both natural and slightly enriched. The total amount of plutonium contained in these FAs is 4751.87 kg and the FP activity amounts 9.72×10^6 TBq.

Data are received from Atucha-1 NPP as an ASCII format file as shown in Appendix 1.

The preliminary management strategy considered is to transfer the SF to dry interim storage after the final shutdown of the NPP. Nevertheless, it is foreseen the need of operating the wet interim storage installation during at least 10 years after the final shut down, to allow for thermal cooling and radioactive decay of the SF from the last core.

Currently Atucha-1 has re-racked the spent fuel in order to enlarge the capacity of the wet storage pools. However, it is foreseen the need of additional storage capacity to comply with the remaining 11 full power years of life of the reactor. Therefore, it is necessary to anticipate the preliminary planning in order to operate a dry interim storage facility before the final shutdown of the NPP. In this framework the design of a modular system at the reactor site for dry interim storage, composed by an arrangement of reinforced concrete structures into which welded metallic canisters containing 37 spent fuel assemblies each are stored in horizontal position, is being developed.

2.2. Embalse NPP

The CANDU fuel assembly has a length of 0.5 m, a circular cross-section of 0,10 m diameter with 37 fuel rods. Values presented below correspond to the update performed on December 31st, 2004 for the spent fuel generated from September 1st, 1983 to December 31st, 2004.

For slightly more than 21 years, from September 1st, 1983 to December 31st, 2004 this NPP discharged FAs that were loaded with an average of 18.85 kg U_{nat}. A total amount of 98117 U_{nat} spent fuel assemblies, corresponding to 1849.87 ton of initial U_{nat}, were generated with an average burnup of 7357 MWd/tU. The FAs consumption and spent fuel generation is about 4800 FA/fpy, equivalent to 90.48 ton of initial U_{nat}.

Spent fuel is initially in wet interim storage in a pool and after an average cooling time of 7.2 years (minimum 3.7 years, maximum 12.3 years, standard deviation 0.7 years) the FAs are transferred to dry interim storage in silos at the reactor site.

The spent fuel in wet interim storage in the pool, made of concrete with epoxy liner, amounts a total of 40877 FAs corresponding to 772.72 ton of initial U_{nat}. The total amount of plutonium contained in these FAs is 2782.88 kg and the FP activity amounts 1.73×10^7 TBq.

The spent fuel transferred to dry interim storage, 106 silos made of concrete containing 9 sealed stainless steel baskets with 60 FAs each, amounts a total of 57240 FAs corresponding to 1077.15 ton of initial U_{nat}. The total amount of plutonium contained in these FAs is 3904.75 kg and the FP activity amounts 2.59×10^6 TBq.

For the 98117 FAs, corresponding to 1849.87 ton of initial U_{nat}, interim stored both in wet and dry, in the pool and silos respectively, the total amount of plutonium is 6687.64 kg and the FP activity amounts 1.99×10^7 TBq.

From the point of view of the data management the smaller FA has an impact on the number of items to be managed that increase for about one order of magnitude.

Data are received from this NPP as separated ASCII format files for spent fuel stored in the pool and stored in silos as shown in Appendix 1.

The dry interim storage, of modular type, has already 120 silos with a total capacity of 64,800 FAs and, when needed, new silos will be added to the existing ones. It is planned that 6 to 8 years after the final shutdown of the NPP the total spent fuel arising will be in dry interim storage.

2.3. Power reactor spent fuel interim storage

The following table shows, in tonnes of initial uranium, the spent fuel stored until December 31st, 2004 as well as the current capacity and expected arising at the end of life of the NPPs.

NPP	Currently stored			Current capacity			Expected at the end of life
	Wet	Dry	Total	Wet	Dry	Total	
Atucha-1 357 MWe	1372.53	0	1372.53	1690	0	1690	1779
Embalse 648 MWe	772.72	1077.15	1849.87	900	1221	2121	2719

3. CONCLUSIONS

Standardisation of the database management systems and calculation codes like ORIGEN (2.1, –S or –ARP) are considered essential for inter-comparison purposes. Experimental determinations of burnup, activity and thermal power to validate the calculation codes are welcomed.

Harmonisation of non-numerical data like the state of preservation of the spent fuel along the various management activities are considered, among others, relevant issues to be discussed. Integrated packages that combine database management systems with calculation codes are considered a powerful tool for the cost effective generation of relevant information for spent fuel managers from the raw spent fuel data requested from the operators.

As this information have to be readable in the long term the selected data management system should properly address this feature.

Appendix I

NPP DATA ARGENTINA

Atucha 1 – NPP Example of the ASCII format file for U_{nat} spent fuel assemblies

POS. PILE	NOMBRE	BURN/UP	ULT.POS. REACTOR	FECHA DE EXTRACCION	DIAS DE ESTADIA (CAL.)	(PLENA POT.)	PU (Gr)	DPP EN EL MOM. DE EXTRACCION	COMENTARIOS	URANIO (Kg)	U235inic. (Gr)
A — 1	RH570	5644	N02	04/06/1981	243	228.81	495	2032.17		152.46400	
B — 1	RH536	5661	J02	07/06/1981	224	209.48	496	2035.13		152.46400	

Atucha 1 – NPP Example of the ASCII format file for ULE spent fuel assemblies

POS. PILE	NOMBRE	BURN/UP		ULT.POS. REACTOR	FECHA DE EXTRACCION	DIAS DE ESTADIA (CAL.)	(PLENA POT.)	PU (Gr)	DPP EN EL MOM. DE EXTRACCION	COMENTARIOS	URANIO (Kg)	U235inic. (Gr)
E -60	C9388	10598	A	C22	13/12/2001	672	332.55	699	6998.70	URANIO ENRIQUECIDO	154.45379	1312.860
E -59	C9267	10367	A	G32	14/12/2001	820	458.25	691	6999.94	URANIO ENRIQUECIDO	154.45731	1312.890

CNE – NPP Example of the ASCII format file for U_{nat} spent fuel assemblies in wet interim storage

Ident EC	Ubicación	Fecha Desc	DPP Desc.	Peso U	Quemado	Pu
A0994W	P	01/05/1984	163,10	18,811	4170,8	47,64
A0997W	P	01/05/1984	163,10	18,855	4389,0	49,40

CNE – NPP Example of the ASCII format file for U_{nat} spent fuel assemblies in dry interim storage

Ident EC	Ubic.	Silo	Ubic Silo	Fecha Descarga	DPP Desc.	Fecha Carga a Silo	Peso U [kg]	Quemado [MWD/tU]	Pu [g]
A0538W	S	1	1	05/03/1984	115.07	22/03/1993	18.885	3768.7	44.46
A0544W	S	1	1	05/03/1984	113.76	22/03/1993	18.885	4156.5	47.72

SPENT FUEL DATA COLLECTION AND RECORDING IN CANADA

J.L. CROSTHWAITE
Atomic Energy of Canada Ltd,
Chalk River, Ontario, Canada

Abstract

This paper gives a summary of status in spent fuel data management at Ontario Power Generation, which is the major producer of nuclear power in Canada, with a focus on the method of data collection and recording for spent CANDU fuel bundles which are stored in AR pools and AFR dry storage systems.

1. BACKGROUND

Ontario Power Generation (OPG) is the electricity-supply company for the province of Ontario. The company operates CANDU-PHW reactors at its Pickering, Bruce and Darlington sites.

All fuel assemblies used in every current CANDU-PHW reactor have the same overall dimensions and are called fuel "bundles".

TABLE 1. PICKERING FUEL BUNDLE DATA

Fissionable Material	Sintered Pellets of Natural UO_2		UO_2 Pellets		
Structural Material	Zircaloy-4		Outside diameter, mm	14	
Bundle Assembly			Length, mm	23 Stack Length,	
Pitch circle diameter			mm	480	
— outer pins (16), mm		85	Average density, Mg/m^3	0.6***	
— intermediate pins (8), mm		54	No. of pellets per fuel pin	21	
— inner pins (4), mm		24	End Plate		
Maximum diameter, mm**		104.2	Diameter, mm	88	
Length, mm*	497.1		Thickness, mm	1.6	
Fuel Pin Cladding			Material Volumes per Bundle		
Average inside diameter, mm		14	UO_2, cm^3		2100
Average cladding thickness, mm		0.4	Total Zircaloy, cm^3		300
Length, mm	486		Material Weights per Bundle		
			UO_2, kg		22.5
			U, kg (initially)	19.9	
			Total Zircaloy, kg		2.0
			Total bundle weight, kg	24.6	
			Average bundle burnup, MW·d/t U 8125		

* Nominal values reported in the table.
** Recommended dimensions for irradiated fuel bundles.
*** More recently, the average UO_2 pellet density used in CANDU fuel has been approaching 10.8 Mg/ m^3.

The inner surfaces of the cladding of all 28 pins in the bundle are coated with a thin layer of CANLUB™, a commercial graphite suspension, which decreases cladding susceptibility to stress-corrosion cracking during in-reactor power ramps.

2. SPENT FUEL CHARACTERISTICS

Each bundle is about 0.5 m long and 0.1 m in diameter. Fuel bundles used in all eight Pickering reactors are comprised of 28 fuel pins of ≈15 mm diameter. Fuel bundles used in all eight Bruce reactors and in all four Darlington reactors are comprised of 37 pins of ≈13 mm diameter. The typical average burnup is ≈over 8000 MW d/t U. Depending on design, each bundle contains about 21–23 kg UO_2 of natural enrichment.

In normal operation, not all fuel bundles in a channel are discharged at once. Rather, depending on the location of the fuel channel in the core, and other operational factors, either two, four, or eight bundles are replaced in a single channel refueling operation. After a further period of reactor operation, another two, four or eight bundles would be replaced, and so on. Normally, each bundle is irradiated in one channel only. The frequency of refueling of any particular channel depends on a number of factors, including position of the channel in the core, reactor power-level history since the previous refueling, etc.

3. OPG PRACTICE IN SPENT FUEL DATA MANAGEMENT

OPG maintains up-to-date databases on the locations and movements of all unirradiated fuel, in-reactor fuel and irradiated fuel discharged from reactors, at each of its reactor sites. These databases are maintained in particular to satisfy the requirements for fuel accounting under the terms of agreement between the Government of Canada and the International Atomic Energy Agency for the application of safeguards pursuant to the Treaty on the Non-Proliferation of Nuclear Weapons.

Fuel bundles are tracked by individual identity, according to a serial number stamped on both end plates by the manufacturer. Once the serial number is known, then the model number of the bundle, specific features that distinguish it from all other models, and the bundle manufacturer can be determined. From this, the manufacturer's records reveal other information such as date of manufacture, shipment date, etc.

Tracking of the irradiation history of a bundle begins with the date of its insertion into a channel in the reactor. The specific core location of a bundle, during any period in its irradiation, is designated by the channel coordinates in the core and the bundle position number in the channel (from 1 to 13). From the recorded operational parameters for the reactor, the reactor refueling history, and the reactor physics characteristics of the core, the detailed power history of a specific bundle can be determined.

Following discharge, and using the appropriate computer codes, the bundle's power history and time since discharge can be used to determine bundle burnup, actinide content, decay heat, and fission-and activation-product inventories.

Tracking of the movement and location of individual irradiated bundles is routinely entered into the irradiated-fuel database. In the water bays currently used for storage, bundles are stored either in 32 bundle capacity baskets, 24 bundle trays or 96 bundle modules (depending on the reactor station. Each basket, tray or module bears a unique identification number. The irradiated fuel data management system used by OPG makes it possible to locate the individual storage receptacle, and the position of individual fuel bundles within it.

The dry-storage facility involves the transfer of 96-bundle modules into concrete containers, each capable of holding four modules. Therefore, tracking of fuel bundles from wet into dry storage forms a component of the Ontario Hydro fuel database.

Typical data that can be determined on a representative, rather than individual bundle basis include

- discharge burnup distribution,
- maximum linear power distribution,
- annual spent fuel arisings,
- decay heat, and
- fission product and actinide content.

Typical data that can be determined and/or derived on an individual bundle basis include

- design and fabrication data,
- irradiation history,
- burnup and power rating,
- fission product and actinide content, and
- decay heat output.

Where detailed examinations may have been conducted on specific bundles in, for example, a shielded facility such as a hot cell, data such as the following are sometimes included in the database:

- post-irradiation dimensional data;
- fission product, activation product, and actinide isotope data, as determined by laboratory analysis;
- gamma scan data;
- unusual physical features of a particular fuel assembly; and
- other data as appropriate.

A record of the chemical control of its spent-fuel storage pools is maintained by Ontario Hydro, including, for example, pH, Cl^- and conductivity levels, see Table 2.

TABLE 2. EXAMPLE OF SPENT-FUEL STORAGE-POOL SPECIFICATIONS: PICKERING NUCLEAR GENERATING STATION

Parameter	Specified Range
pH	5.9 to 9.0
Chloride Content	<1.0 mg/kg
Conductivity	<1.0 mS/m

WASTE INVENTORY RECORD KEEPING SYSTEMS IN THE CZECH REPUBLIC

P. LIETAVA
State Office for Nuclear Safety, Praha, Czech Republic

V. PŘÍMAN
ČEZ, a. s., Praha, Czech Republic

M. KESELICA
ČEZ, a. s., NPP Dukovany, Czech Republic

Abstract

In the Czech Republic, the minimum requirements on waste data, which have to be included into so called "standard document for radioactive waste", are defined by the Atomic Act and its implementing regulations. These documents have to accompany radioactive waste for any physical consignment to a licensee. All information on radioactive waste including used sealed sources are collected by a state organisation — Radioactive Waste Repository Authority (SURAO), which is, among others, responsible for keeping records of radioactive waste received for disposal and of their originators. Currently the existing SURAO database system has been reviewed and a new one will be developed soon. For definition of the database architecture the national regulatory body — State Office for Nuclear Safety (SUJB) is also involved. SUJB is responsible for the contributions to the IAEA NEWMDB database and for the preparation of National Report for the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management. Another database, which is under development, is the spent fuel management record system. This database is intended to be used as a data source for all operations with spent fuel after its long term storage. Because the national policy expects that the spent fuel will be disposed directly into the deep geological repository, this database can be considered as a second waste inventory database. Even if currently the spent fuel is not declared to be radioactive waste according to the definition in the Atomic Act, it is subject to the same requirements that apply to radioactive waste and at the furthest after taking over by SURAO it will be classified as radioactive waste.

1. INTRODUCTION

In the Czech Republic, Waste Inventory Record-Keeping Systems (WIRKS) can be divided into two main categories — the radioactive waste tracking system and the spent fuel management record system. This distribution of record keeping systems is based on two facts:

The Atomic Act, which is the main legislative document regulating among others "*conditions for safe management of radioactive waste*", clearly declares, that the spent fuel is not radioactive waste "*until a generator or the (State) Office (for Nuclear Safety) declares spent or irradiated fuel to be radioactive waste*." However the same Act requires, that the management of spent fuel, "*apart from the requirements arising out of other provisions of this Act, is subject to the same requirements as apply to radioactive waste. An owner of spent or irradiated fuel shall manage it in such a way as not to encumber the potential for subsequent conditioning*;"

A State organisation called the Radioactive Waste Repository Authority (SURAO) has been established to perform a set of activities related to the disposal of radioactive waste including "*keeping records of radioactive waste receipts and their generators*". Because spent fuel is currently not included into the classification scheme for radioactive waste and it is owned by the nuclear power plant operator (CEZ, a. s.), the spent fuel database cannot be managed by SURAO.

The information from both WIRKS is submitted to the national regulatory body — State Office for Nuclear Safety (SUJB) on a regular basis. The SUJB uses the data as a source of information for:

- the preparation of information provided on an annual basis to municipalities and Regional Authorities concerning radioactive waste management within their territory of administration;
- the preparation of the National Report under the Joint Convention on Safety in Spent Fuel Management and Safety in Radioactive Waste Management (the Joint Convention) every three years;
- the annual submissions to the IAEA's Net Enabled Waste Management Database (NEWMDB); and
- the annual submissions to the IAEA Integrated Nuclear Fuel Cycle Information System (iNFCIS).

2. LEGISLATIVE ENVIRONMENT

The Chamber of Deputies of the Czech Parliament approved the new Atomic Act, No. 18/1997 Coll., in January 1997. The Act entrusted execution of state administration and supervision of the utilization of nuclear energy and radiation practices to SUJB and defined its competence. The Atomic Act newly defines conditions for the peaceful utilization of nuclear energy and ionizing radiation, including activities requiring a licence from SUJB. An extensive list of obligations of the licensees also includes their obligations concerning WIRKS.

According to the Atomic Act, the licensee has an extensive set of obligations, one of them is to "*keep and archive records of ionising radiation sources, facilities, materials, activities, quantities and parameters and other facts impacting on nuclear safety, radiation protection, physical protection and emergency preparedness, and submit the recorded information to the Office in the manner set out in an implementing regulation*". The implementing regulation defining the minimum set of data is the Decree No. 307/2002 Coll., on radiation protection. In its Chapter III, "Radioactive Waste Management", in Articles 53 and 54, requirements on waste record systems are provided.

Quantities and specific activities of radionuclides in radioactive waste have to be recorded during its collection, sorting, processing, treatment, storage, transportation and disposal. These records shall serve to control the licensee's radioactive waste flow. A standard document for radioactive waste has to accompany radioactive waste for any physical consignment. The standard document shall be issued for all types of radioactive waste and for each radioactive waste package that is an independent manipulation unit, such as a cask.

The standard document for radioactive waste shall contain the following information:

(a) indication of physical and chemical form and properties of radioactive waste, or a code characterising radioactive waste, and for solid waste, its category;
(b) description of the package type and an external label or marking, identifying the package (identification number);
(c) mass activity of those radionuclides whose content shall be limited by the acceptance criteria and of those radionuclides whose quantity is greater than 1 percent of the total activity;
(d) dose equivalent rate on the package surface;
(e) data on surface contamination of the package with radionuclides;

(f) magnitude of the leachability coefficient of radioactive waste treated for disposal if the coefficient is limited by the acceptance criteria of the repository;
(g) total weight of radioactive waste;
(h) weight of the radioactive waste package if it is to be stored prior to disposal and when it is being disposed of;
(i) the packaging filling date or period;
(j) the issue date of the standard document;
(k) business company and identification number (if it has been assigned) of a person who is consigning radioactive waste as well as the first name, surname, job position and signature of an authorised representative of this person; and
(l) business company and identification number (if it has been assigned) of a person who is receiving radioactive waste, and the first name, surname, job position and signature of an authorised representative of this person.

An integral part of the standard document for radioactive waste that will be disposed or stored is a written statement by the generator of the treated radioactive waste stating that the waste was treated in accordance with approved limits and conditions for treatment thereof and that it complies with the acceptance criteria for given repository or storage facility. The standard document for radioactive waste shall be permanently archived by the SURAO. In addition, other licensees authorised for radioactive waste management or the generators shall archive the same information for a minimum of 10 years from consignment or disposal of the given radioactive waste.

Other licensee obligations cover:

- preparation and submittal to the legal person authorised to dispose of radioactive waste, data on short term and long term production of radioactive waste and spent nuclear fuel together with other background information to determine the amount and method of transfer of payments to the nuclear account;
- record keeping of radioactive waste by type of waste in such a manner that all characteristics affecting its safe management are apparent; and
- handing over to the SUJB and to the European Commission data required by the Act and by the EC legislation; the scope of data, the form and the manner of the handover shall be established in an implementing regulation.

3. RADIOACTIVE WASTE DATABASE SYSTEMS

The information on radioactive waste is submitted from all waste generators to the SURAO, where it is further processed. SURAO also submits requested radioactive waste data to the regulatory body, SUJB, for further use (e.g. preparation of the National Report under the Joint Convention, submission of waste data to NEWMDB database, preparation of annual waste management reports for the local authorities, ...). Details on material and data flow are provided in Figures 1 and 2.

In the year 2000, the SURAO launched a project on the development and implementation of a basic information system called ZISS for registration of radioactive waste generators, record-keeping of accepted and disposed waste, monitoring of repository sites and some other supporting functions. The information system was designed as on line system allowing waste generators to directly communicate with SURAO by the Internet.

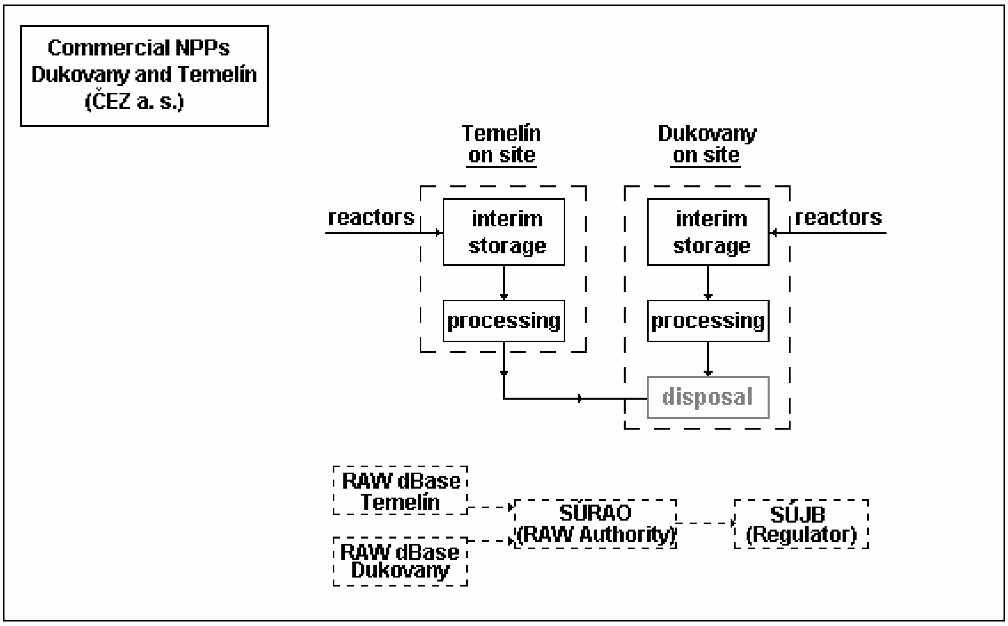


FIG. 1. Material flow and data tracking for operational waste (grey marked facility is operated by SÚRAO).

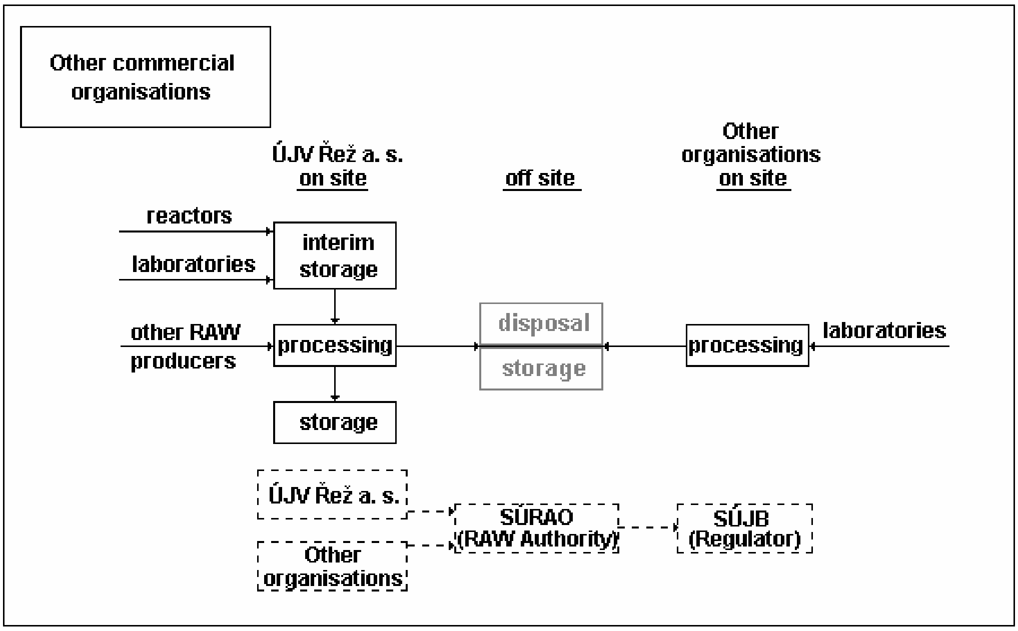


FIG. 2. Material flow and data tracking for institutional waste (grey marked facilities are operated by SÚRAO).

After the waste generator registers himself with help of registration form, he can download or update all of his provided data. After confirmation of data acceptance by the SURAO administrator, the generator cannot make any corrections to his data. In addition to waste generators, there are three more groups of ZISS users:

- Guest — access for the general public, without username or password to subsections documents (includes waste acceptance criteria for operated repositories and a template for cargo form), radioactive waste generator registration (includes new registration form, selection of registration forms, and waste generators catalogue) and register (lists of standardized units, expressions, …);

- Manager — access to all defined data, without any possibility for editing; and
- Administrator — access to defined sections for maintenance of the data; the system administrator has access to all data, including the SQL server.

The structure of the ZISS database, covering not only data on waste generators and waste packages, is provided in Table 1.

Under the PHARE project, a new Waste Tracking System (WTS) is currently under development in the Czech Republic. The need for the development of new waste database is based on several weak points of the ZISS system, which can be summarised according to the following groups [4]:

- functional deficiencies/shortcomings — sorting of lists can be done only in alphabetic order and reports with the nuclide mass number in the leading column cannot be produced, repeated queries without modifying the database do not necessarily produce identical results, …;
- structural deficiencies/shortcomings — database structure seems to be unnecessarily complicated; and
- other deficiencies/shortcomings — poor performance regarding the response time.

In general the new WTS shall fulfil the following tasks:

- radioactive material accountancy (including waste tracking);
- reporting commitments; and
- publication of selected information.

Additional requirements on WTS have been defined by SUJB and SURAO:

- graphical representation of important results accessible for internal use and for direct access by the general public;
- use of offline copies of the database;
- enhance the possibility to track charges and to model exactly the complete life cycle of the waste. Special functions are needed for the tracking of movements of charges in and between waste repositories;
- provide information on:
  - sealed sources placed in a cask (radionuclide inventory, activity of each radionuclide, No. of sealed sources in one cask, …);
  - total amount of radioactive waste (in kg or m3) and inventory (in Bq) of radionuclides listed in operational limits and conditions stored and/or disposed in repositories adjusted for decay half-life;
- location of each cask in repository;
- allow to generate the output queries for the IAEA NEWMDB database, containing information on:
  - used storage and disposal capacity of each operated repository in % of its current capacity and planned capacity of the repository;
  - the category of disposed waste for disposal areas of each repository using the IAEA categorisation scheme (exempt waste, LILW-SL, LILW-LL, HLW), processed or unprocessed waste, its volume (in m3) and origin according to the waste origin (RO — reactor operations, FF/FE — fuel fabrication/ enrichment, RP — reprocessing, NA — nuclear application, DF — defence, DC/RE — decommissioning/remediation, ND — not determined);

TABLE. 1. STRUCTURE OF ZISS DATABASE [3]

SECTION	SUBSECTION
Radioactive Waste Generators Registration	Catalogue of waste generators List of contacts List of registration forms New contact New registration form Waste generators registration overview
Radioactive Waste Acceptance and Record Keeping	Selection of waste package passports New waste package passport List of shipments New shipment Import of data from Dukovany repository List of disposal (storage) records Selection of accepted waste Information on disposed of waste
Monitoring	Export from subsystem Monitoring Import of measurements and results List of measurements and results Methods New measurement New sample Sampling places Samples Monitoring plan and overview Monitored items
Doses	Export of personal doses Export of entries to the controlled zone Places of doses New personal dose New collective entry to the controlled zone New entry to the controlled zone Personal doses Entries to the controlled zone
Entry of Persons to Facilities	New collective entry to the facility New entry to the facility Selection of entries
Equipment Maintenance	New equipment List of activities (selection of activities planned, performed, etc.) List of equipment (selection of equipment by place, type, etc.)
Wallpaper	Publishing of a new document List of published documents
Documents	Document templates
Tools	Registers Export of system log System users administration Change of password

— disposed sealed sources divided into two main groups according their half-life (≤30 y, >30 y).

According to the time schedule of the whole project the WTS system shall be fully developed in the middle of 2005.

4. SPENT FUEL MANAGEMENT RECORD SYSTEM

One of the vital issues in long term management of spent fuel is the retention of appropriate information. This information must accompany the spent fuel during its whole lifetime; i.e. from its unloading from reactor core till its disposal in repository or reprocessing. Due to safety and security considerations, spent fuel must be managed for very long periods of time. The significance of spent fuel data management will persist for its lifetime.

4.1. Current situation

The spent fuel information flow in the Czech Republic is quite straight forward. The company ČEZ, a. s., which is the only operator of power reactors in the Czech Republic, annually submits the requested spent fuel data to the regulator (SUJB). The operator performs spent fuel data management in compliance with national regulations and international recommendations. The spent fuel data are provided to SUJB also for the safeguard purposes.

The operator of NPP Dukovany manages three databases — spent fuel database, manufacturer database containing the detailed information about the spent fuel assembly construction and used materials and operational database with detailed information about the fuel assembly operation in reactor core. Similar data is also available at the operator of NPP Temelín.

The structure of spent fuel database managed by the staff of NPP Dukovany covers also the data needs for dry Interim Spent Fuel Storage Facility (ISFSF Dukovany), which is placed at the same site as the NPP. The capacity of ISFSF Dukovany is 60 CASTOR 440/84-type casks each containing 84 VVER-440 type fuel assemblies. For each fuel assembly discharged from reactor core following information are stored in the SF database:

- Serial No. of spent fuel assembly;
- Manufacturing date;
- No. of spent fuel assembly drawing (defines the construction type of spent fuel assembly);
- Initial enrichment,
- U weight (total),
- ^{235}U weight in fresh fuel assembly,
- UO_2 weight in fresh fuel assembly (total),
- Date of fuel delivery to NPP Dukovany,
- Movement of fuel assembly in NPP,
- History of fuel assembly usage in reactor core (date of loading, BU at the end of campaign, position in reactor core, effective days, date of unloading),
- Information about the fuel assembly control,
- Final BU and the position of spent fuel assembly in spent fuel pond,
- Isotopic composition of U and Pu in spent fuel,
- Date of CASTOR cask loading,
- No. of CASTOR cask and spent fuel assembly position,
- Date of CASTOR cask transport from reactor unit to ISFSF Dukovany, and
- CASTOR cask position in storage hall of ISFSF Dukovany.

4.2. Planned activities

Based on ongoing IAEA activities, the SUJB has initiated preliminary discussion with the spent fuel owner, the CEZ, a. s. power company, about the development of a comprehensive spent fuel database based on a draft IAEA publication [5]. The publication has two main objectives — identify the issues that must be addressed to implement safe management of spent fuel and to identify the data needed to address those issues and to determine how that data should be maintained.

Discharged fuel data are needed to establish and maintain a simple, reliable and accessible database. Those data must permit discharge burnup to be determined, together with tracking of the history of the fuel from discharge through subsequent stages of handling, transfer and storage. The starting point for fuel tracking is its delivery to the NPP and is transferred into another database system, e.g. disposal database or MOX fuel database, only when the fuel is reprocessed, re-fabricated or conditioned for disposal.

The spent fuel data parameters should be recorded and secured by methods that will ensure their maintenance and preservation for extended periods of 100 years or more. The records should be available to the responsible regulatory authority and should be maintained until that authority considers that the records are no longer required.

The spent fuel data parameters generally fall into discrete groups:

- general data,
- pre-irradiation data,
- irradiation data (in core fuel management data),
- post-irradiation description,
- data on conditions of storage and transport,
- cessation of data tracking, and
- information concerning fuel cycle back end options.

In addition to these basic parameters, particular derived data, such as isotope abundance in the discharged fuel and the radiological characteristics of fuel following substantial storage periods may be required. Based on a structure of reference spent fuel database developed for illustrative purposes only in [5], a preliminary structure of a future spent fuel database has been proposed (see Fig. 3).

Depending on available financial resources, the database could be available for the operator in about 5 to 10 years. However, despite this limitation, the operator of NPP Dukovany has already launched a project together with the company Škoda JS a. s. leading to the elaboration of methodology for total activity calculation of spent fuel loaded and stored in dry CASTOR 440/84 casks. Except the total spent fuel assembly activity, the activities of following 19 radionuclides are calculated:

> H-3, Kr-85, Sr-90, Ru-106, I-129, Cs-134, Cs-137, Ce-144, U-235, U-236, U-238, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Am-241, Cm-242 and Cm-244.

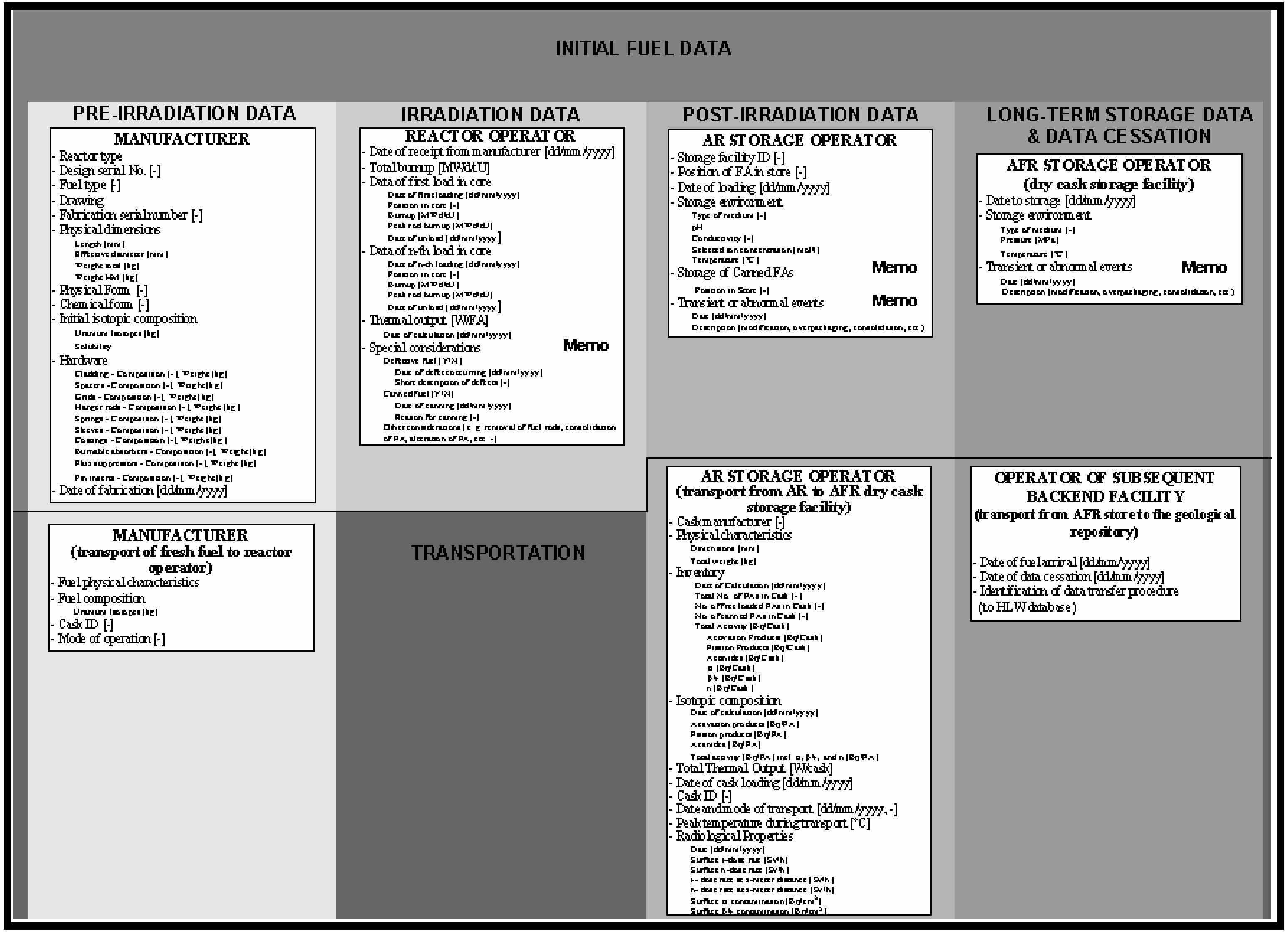


FIG. 3. Proposed structure of spent fuel database.

For VVER-440 fuel with initial enrichment 3,82 % wt. ^{235}U, with burnup 10 to 60 MWd/kgU and with cooling period 5 to 10 y, characteristic volumetric activity (activity per 1 cm^3 of UO_2) curves for each of 19 selected radionuclides and cooling times were calculated. These curves will be used for the determination of total and radionuclide specific activities of spent fuel loaded into CASTOR 440/84 casks. It is expected, that this, or a similar calculation scheme, will be included into the proposed spent fuel database.

Initial fuel data of a spent fuel management database generally fall into groups (vertical structure) relating to:

- pre-irradiation data,
- irradiation data (in core fuel management data),
- post-irradiation description, and
- data on conditions of long term storage and transport.

Additional information concerning transportation and fuel cycle back end options can affect some basic parameters, particular derived data, such as isotope abundances in the discharged fuel and the radiological characteristics of fuel following transport and storage periods etc. Cessation of data tracking determines the length of the life cycle of the spent fuel database. It is related to the next step of nuclear fuel cycle, where the spent fuel is declared as radioactive waste (according to the national policy the spent fuel will be directly disposed in deep geologic repository) and selected data are included into the radioactive waste database. The spent fuel data tracking continues after its disposal.

The structure of the proposed spent fuel database in Fig. 3 was developed for spent fuel that is stored at an AR (at reactor) storage facility after the irradiation in reactor core and then unloaded into a dual-purpose cask and transported and stored at dry type cask AFR store. The same cask is also used after a long term storage period for the transport into a subsequent fuel cycle back end facility as geological repository or reprocessing facility.

5. CONCLUSION

In the Czech Republic, several projects related to the development of radioactive waste and spent fuel databases are under way at the present time. These database systems are being developed to be operated by the waste disposal authority SURAO and by the power company ČEZ, a. s. in close cooperation with national regulatory body — SUJB. Recently (1 May 2004), the power plant operator has, under the Euratom treaty, the obligation to submit the information on fissile material to the Euratom Safeguards Office.

SUJB as the organisation responsible for the data transfer to the IAEA and other international organisations not only defines additional requirements on functional properties of database systems, but also provides feedback for their development. The final goal of the SUJB is to participate at the development of effective and reliable radioactive waste and spent fuel data management systems, which will ensure that these data will pass to the future generation contributing to the safe management of radioactive waste and spent fuel. These database systems have to be harmonised with the structure of the IAEA databases to provide reliable information flow not only towards national, but also international bodies involved in radioactive waste and spent fuel management.

REFERENCES

[1] Act No. 18/1997 Coll. of 24 January 1997, on Peaceful Utilisation of Nuclear Energy and Ionising Radiation (the Atomic Act) and on Amendments and Alterations to Some Acts.

[2] Regulation No. 307/2002 Coll. of the State Office for Nuclear Safety of 13 June 2002, on Radiation Protection.

[3] KUCERKA, M., MOCHANOVA, D., Radioactive Waste Generators Registration and Waste Record Keeping System at the Radioactive Waste Repository Authority, Czech Republic, WM'01 Conference, Tucson, AZ, USA, February 25–March 1, 2001.

[4] Project Development of a Waste Tracking System, Task 1, Review of the Existing Database System, Europaid/113987/D/SV/CZ, April 2004.

[5] Data Requirements and Maintenance of Records for Spent Fuel Management, TecDoc Draft, IAEA, Vienna, 2004.

[6] HEP, J., Methodology of VVER-440 Spent Fuel Activity Calculation by the CASTOR Cask Loading, Part 1, Škoda JS a. s., December 2003 (in Czech).

FUEL DATA COLLECTION AND RECORDS MANAGEMENT IN FRANCE

P. SAVEROT JAI

Abstract

This paper describes a Nuclear Fuel database and data collection method implemented at the EDF, France. The system has been developed to achieve control and accountancy of the fuel assemblies, to optimize consistency of data, for automatic transport of reports to safeguard authorities and to serve as source of nuclear material data for technical purposes.

Fuel Data Collection and Records Management

A comprehensive fuel database has been implemented at EDF to provide a single, reliable, nuclear material control and accounting system for all reactors. The objectives of the database are as follows:

(1) the control and accountancy of the fuel assemblies at the reactor,
(2) the optimization of the consistency of the data by minimizing transcription errors,
(3) the automatic transfer of reports to national and international safeguards authorities,
(4) the servicing of other EDF users for nuclear material data for technical purposes.

This Nuclear Fuel Management system meets the needs of all reactor sites both from the operational and the regulatory bodies viewpoints. It enables the fuel managers to perform the following functions:

(i) the physical management (location and history) of the fuel assemblies and fuel rods from the time of their delivery to the reactor site to the time of the shipment of the spent fuel outside of the reactor,
(ii) the physical management (location and history) of rod control clusters and other non-fuel hardware items used for the physical management of the fuel assemblies and the rods,
(iii) the accountancy of the nuclear material,
(iv) the fulfillment of statutory requirements,
(v) the servicing of other EDF users for technical or commercial purposes.

According to the French regulations, the operator must know at any time the quantity and location of the nuclear material on his plant site. The Nuclear Fuel Management system is designed to provide the operator with such data long with a complete history of any item through its residence on the site. The Nuclear Fuel Management System provides the fuel manager with a complete file of all the characteristics of the fuel assemblies, which is updated at each movement or when changes occur. The system performs also the accountancy of the nuclear material and calculates book inventories.

The spent fuel acceptance criteria for shipment to La Hague, subsequent storage and reprocessing can be subdivided into those applicable to sound assemblies, questionable spent fuel assemblies, and damaged fuel assemblies. These are discussed below.

General acceptance criteria for sound fuel are as follows:

(1) *Geometry and mechanical integrity*: spent fuel assemblies must not be deformed or bent, i.e. they should not exhibit any interference for loading/unloading the transport cask. Each fuel assembly should be individually handled to check that it is structurally sound.
(2) *Non-leaking fuel*: spent fuel must be declared as “non-leaker” by the reactor operator. After reactor unloading, the operator must reconfirm the integrity of the fuel and provide pool water inspection characteristics with supporting information concerning the last storage emplacement of the fuel.
(3) *Crud deposit*: if P > 1000 g per fuel assembly, the fuel is accepted with a cleaning at the reactor. If P < 100 g per fuel assembly, the fuel is accepted without cleaning. P: Reactor estimated value.
(4) *Cooling time*: 1 year minimum at the time of shipment and in compliance with the shipping cask CoC.
(5) *Gas checking in the loaded cask*: the beta-gamma activity (excluding tritium) of a gas sample from a loaded cask shall not exceed $0.37 MBq/m^3$.
(6) *Final visual check that each spent fuel assembly is structurally sound and as documented*: this verification is performed at the reactor to monitor the fuel during cask loading. In case of any doubt during this inspection, electronic images are taken and transmitted along with dimensional data to La Hague for a final decision.

The demonstration of fuel integrity is very strict and consists of the following procedures, either with reactor primary coolant activity or sipping, as explained below:

(1) *With reactor primary coolant activity during the last irradiation cycle and corresponding shut down*: The I^{131} concentration gives an indication of the soundness of the fuel. There are no “leakers” if the water primary coolant activity is smaller than 10^{-3} Ci/m^3 and the additional release accumulation during shutdown is below 0.1 Ci or comprised between 10^{-3} and 10^{-2} Ci/m^3. Xe^{133} is also used with similar criteria.
(2) *With sipping demonstration*: When the reactor primary coolant activity doesn't meet the above criteria, at least one damaged fuel is present in the core and a sipping test must be done for each fuel assembly. The sipping box technique (which can be either wet or dry) is used for PWR reactors:

    (i) the wet sipping technique detects Cs^{134} — Cs^{137} and/or I^{131} in a liquid sample. the dry sipping technique detects Xe^{133} and/or Kr^{85} in a gas sample:

        (a) The acceptance criteria for both “in core sipping” and the “wet sipping methods” are as follows:

        - $\frac{A_1}{A_o} < 2$ for sound fuel
        - $2 < \frac{A_1}{A_o} < 5$ for suspect fuel
        - $\frac{A_1}{A_o} > 5$ for leaking fuel

    where A_0 is the background value and A_1 the fission product count for each fuel assembly with elimination of high leaker values.

(b) The acceptance criteria for the dry sipping technique are:

$$\frac{A_1}{A_o} < 2 \text{ for sound fuel}$$

$$\frac{A_1}{A_o} > 2 \text{ for leaking fuel}$$

where

$A_1 < \overline{A_1} + 3\,\sigma$ for sound fuel, and $A_1 > \overline{A_1} + 3\,\sigma$ for leaking fuel where σ is the standard deviation of the sample.

with $\overline{A_1} = \frac{1}{M}\sum_1^M A_1$

Spent fuel that meet the criteria for "sound fuel", as mentioned above, are generally accepted. Spent fuel that does not meet one or more of the above criteria, and which is not heavily damaged, is considered as "questionable". Such fuel may eventually be accepted and transported without canning, provided that further detailed examinations are conducted, e.g. fuel inspections and sipping tests coupled with additional controls on each loaded cask. Additional criteria that have to be met are as follows:

(1) Spent fuel must be physically sound so that there shall not be any dissemination of radioactive materials during handling or transport.
(2) With a geometrically and mechanically sound spent fuel assembly that exhibits signs of corrosion, spent fuel will be accepted only after a satisfactory radioactivity release rate measured by a specific sipping test. The individual assembly is immersed overnight in demineralized water to remove surface contamination prior to the beginning of the test; water is then circulated through the sipping test tank; samples are taken regularly over a 100 hour period and analyzed for Cs^{137}, Cs^{134}, Co^{60}, Co^{58}, gross alpha, and gross beta. A global indicative soundness threshold of 10KBq/hr gross beta is used for acceptance of that spent fuel for shipment.

Damaged fuel is generally fuel that has undergone significant corrosion and/or loss of mechanical or geometrical integrity. Damaged fuel assemblies release fission products through their cracks, which contaminate the cask, the unloading facilities, the pool storage facilities as well as all the equipment used for fuel transfer. Both contamination and irradiation risks for operators could shut down the receiving facilities. Such spent fuel is canned for transportation and further storage to La Hague.

Acceptance Criteria Specific to Transport

The implementation of burnup credit for transport (approved since 1987) is subject to several conservative conditions as follows:

- The allowed burnup credit must be reached on the least irradiated 50 cm of the fuel's active length instead of an average value over the total active length,
- The reactor operator must guarantee, on the basis of its fuel management and in-core measurement records, that the minimum burnup on the least irradiated 50 cm of the fuel's active length exceeds the allowed burnup credit,

- The irradiation status of each fuel assembly must be checked by an independent quantitative measurement of the actual fuel burnup (the previous qualitative go/no-go physical measurement is no longer accepted). The acceptance of each fuel assembly is now subject to the comparison of the physical measurements with the minimum criteria and a cross-check with utility-supplied data.

Two types of physical measurement are required prior to cask loading:

(1) For limited burnup credit (below 5600 MWD/MTU for transport), for storage of 215 mm square fuel assemblies (3.5% to 3.75% U^{235}) and for storage of 230 mm square fuel assemblies (3.3% to 3.55% U^{235}), only the irradiation status of the fuel assemblies is verified by "on-line" measurements, with a gamma detector immersed in the pool, during the fuel transfer from its pool storage position to the cask compartments. It is considered that a fuel element is irradiated if the dose rate measured at a minimum distance of 0.75 m is higher than 100 mSv/hour.

(2) For burnup credit exceeding 5600 MWD/MTU for transport, for storage of 215 mm square fuel assemblies (3.75% to 4% U^{235}), for 230 mm square fuel assemblies (3.55% to 4% U^{235}), a direct and quantitative measurement of the actual fuel burnup is required. Either the "Python" or the "Fork" detector is used for passive and active fuel burnup and reactivity measurements. The input data concerning the selected assembly (cooling time, identification number, etc.) is entered into the computer and the burnup profile is recorded along the active part of the fuel. The average value over the least irradiated 50 cm is calculated and compared with the acceptance criteria before final positioning of the fuel assembly into either its preloading position in the pool storage rack or into the cask.

ACQUISITION OF DATA RELATED TO SPENT FUEL MANAGEMENT IN GERMANY

C. GÖTZ
Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit (BMU), Bonn, Germany

W. MESTER
Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Köln, Germany

W. WEBER
Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Garching, Germany

Abstract

This paper describes data collection and records keeping related to Spent fuel Management in Germany. The paper provides a review of policy and management of Spent Fuel in German nuclear power plants. It also provides data on stored fuel in power plants and data requirements for spent fuel management. Methods to acquire spent fuel data by annual inquiries are also described. Trends on spent fuel accumulation and data on essential spent fuel parameters are shown on graphs and tables. There is no centralized spent fuel data management system but the regulatory bodies dispose data required for safety assessment and licensing.

1. INTRODUCTION

Annually about 10,000 t of spent nuclear fuel are unloaded from nuclear reactors worldwide. By 2004 a total amount of 270.000 t has been accumulated. Though this is a relatively small quantity compared with other industrial hazardous waste, it requires special measures because of its very high radioactivity. So it is obvious that adequate provisions have to be taken for the safe management of the fuel not only with regard to the radiological risks but also for non-proliferation reasons.

Roughly one third of the accumulated spent fuel has been reprocessed, the rest will be stored until repositories for their disposal are available. For the planning of storage and disposal facilities it is necessary to have information about certain key data like mass of heavy metal, total activity, composition, container materials etc. The level of accuracy of these data has to increase in line with the planning stage.

2. MANAGEMENT OF SPENT NUCLEAR FUEL

2.1. Nuclear power plants in Germany

In Germany 17 nuclear power plants (11 PWR, 6 BWR) were in operation at the end of May 2005. Twelve nuclear power plants including the six reactors of the former GDR have already been shut down and are in different stages of decommissioning (Table 1).

2.2. Policy on the management of spent fuel

Germany’s policy on the management of spent fuel has undergone a number of changes. Until 1994, the Atomic Energy Act included the requirement of reusing the fissile material in the spent fuel assemblies. This requirement changed with the amendment of the Act in 1994, and the operators of nuclear power stations then had the option of either reuse by means of reprocessing, or else direct disposal.

TABLE 1. NUCLEAR POWER PLANTS IN GERMANY

In operation (17 plants):

Brokdorf	PWR	1440 MWe
Unterweser	PWR	1410 MWe
Grohnde	PWR	1430 MWe
Emsland	PWR	1400 MWe
Biblis A	PWR	1225 MWe
Biblis B	PWR	1300 MWe
Philippsburg 2	PWR	1458 MWe
Neckarwestheim 1	PWR	840 MWe
Neckarwestheim 2	PWR	1395 MWe
Isar 2	PWR	1475 MWe
Grafenrheinfeld	PWR	1345 MWe
Brunsbüttel	BWR	806 MWe
Krümmel	BWR	1316 MWe
Philippsburg 1	BWR	926 MWe
Gundremmingen B	BWR	1344 MWe
Gundremmingen C	BWR	1344 MWe
Isar 1	BWR	912 MWe

Shut-down (12 plants):

Mülheim-Kärlich	PWR	1302 MWe
Stade	PWR	772 MWe
Obrigheim	PWR	357 MWe
Rheinsberg	PWR	70 MWe
Greifswald 1 to 5	PWR	5x440 MWe
Lingen	BWR	252 MWe
Gundremmingen A	BWR	250 MWe
Würgassen	BWR	670 MWe

After the elections of 1998 the new government formed by a coalition of social democrats and greens decided the phase-out of nuclear energy. The main elements of this new policy were negotiated with the utilities and legally fixed in the revision of the Atomic Energy Act of 2002.

As of 1 July 2005, delivery for the purposes of reprocessing will be prohibited in accordance with an amendment to the Atomic Energy Act (AtG) to this effect of 22 April 2002, and only direct disposal of the spent fuel assemblies from nuclear power plants then existing in Germany will be possible. By 30 June 2005, most of the spent fuel assemblies had been shipped to said facilities by the nuclear power plant operators; the last spent fuel assemblies from the Stade nuclear power plant were dispatched for reprocessing before the end-of-July-2005 deadline.

The spent nuclear fuel delivered to France and the United Kingdom until 30 June 2005 will be reprocessed. For those spent fuel assemblies proof of reuse must be kept of the recycled plutonium separated during reprocessing. This is designed to ensure that throughout the remaining residual operating lives of the nuclear power plants, all separated plutonium is processed in the fabrication of MOX fuel assemblies and thus re-used. For the reprocessed uranium a proof of disposition has to be provided by the utilities according to the Atomic Energy Act. A small part of the reprocessed uranium has been or will be re-enriched and reused in nuclear power reactors. The re-enrichment can be performed either by feeding the uranium into an enrichment plant or by blending it with surplus highly enriched uranium.

During the last years fuel elements containing enriched reprocessed uranium (ERU) have been introduced in five nuclear power plants.

All fuel assemblies remaining in Germany, and those which will continue to be generated, will be stored in storage facilities until their final transportation into a repository. In line with statutory requirements this is done in storage facilities that have been or are yet to be constructed at the sites of the nuclear power plants, which will be reserved solely for spent fuel arising at that particular site. The spent fuel is stored dry in containers licensed for transport and storage. Spent fuel assemblies from decommissioned power reactors of Soviet design in Greifswald and Rheinsberg are likewise stored dry in containers at a central storage facility in Greifswald.

Only for exceptional cases, if storage at the site of the nuclear power plants is not possible on technical grounds, are two central storage facilities operational and on stand-by at Ahaus and Gorleben.

Usually, the spent fuel assemblies from research reactors will be returned to their country of origin. If that is not possible, these too will be intermediately stored until their final transportation to the repository.

The Federal Government is aiming to dispose of all kinds of radioactive waste including spent fuel assemblies in geological disposal. Until the year 2000, the Gorleben salt dome was explored with regard to its suitability as a repository, especially for heat-generating waste. Since then, a moratorium has been in place on the exploration, running for between 3 and 10 years, depending on the clarification of safety-related and conceptual issues regarding final storage. This clarification process has not yet been concluded. The moratorium on Gorleben does not mean that the salt dome has been abandoned as a possible repository site, also for the disposal of spent fuel assemblies [1].

The Konrad repository is designed for waste with negligible heat generation. The plan approval procedure was completed and a plan approval decision adopted on 22 May 2002. However, the decision is not yet legally valid since objections have been filed.

2.3. Storage of spent fuel

For the storage of spent fuel there are different facilities and procedures available:

The cooling ponds in the reactor buildings,

- the dry storage facilities at the reactor sites, including temporary storage facilities (so-called *Interimslager*) which serve as transitional solution until the on-site storage facilities are complete,
- the storage facilities at Greifswald (wet: ZAB, dry: ZLN) for spent fuel from the nuclear power plants at Rheinsberg and Greifswald, and the dry storage facility at Jülich for spent fuel from the high-temperature reactor AVR, and
- the central dry storage facilities at Gorleben and Ahaus.

A list of storage facilities for spent fuel assemblies, their storage capacities and the inventories at 31 December 2004 are shown in Table 2.

TABLE 2. STORAGE FACILITIES FOR SPENT FUEL ASSEMBLIES (AS AT 31 DECEMBER 2004) [1]

Site	Storage capacity	Storage capacity	Status		Emplaced
	(Number of storage positions)	(tHM)	Applied for	Licensed	(tHM) as at 12/04
Fuel pools in reactor buildings					
Nuclear power plants total	19776 positions[1]	approx. 6119 tHM[1]		X	3358
Onsite storage facilities					
Biblis	135 container positions	1400 tHM		X	
Brokdorf	100 container positions	1000 tHM		X	
Brunsbüttel	80 container positions	450 tHM		X	
Grafenrheinfeld	88 container positions	800 tHM		X	
Grohnde	100 container positions	1000 tHM		X	
Gundremmingen	192 container positions	1850 tHM		X	
Isar	152 container positions	1500 tHM		X	
Krümmel	80 container positions	775 tHM		X	
Lingen/Emsland	120 container positions	1250 tHM		X	153
Neckarwestheim	151 container positions	1600 tHM		X	
Obrigheim[2]	980 positions	286 tHM		X	44
Philippsburg	152 container positions	1600 tHM		X	
Unterweser	80 container positions	800 tHM		X	
Temporary storage facilities					
Biblis	28 container positions	300 tHM		X	234
Brunsbüttel	18 container positions	140 tHM	X		
Krümmel	12 container positions	120 tHM		X	9
Neckarwestheim	24 container positions	250 tHM		X	149
Philippsburg	24 container positions	250 tHM		X	99
Centralised storage facilities					
Gorleben	420 container positions[3]	3800 tHM		X	38
Ahaus	420 container positions	3960 tHM		X	58[4]
Local storage facilities outside the reactor sites					
ZAB Greifswald	4680 positions	560 tHM		X	150
ZLN Greifswald	80 container positions	585 tHM		X	407
Jülich	158 containers	0.225 t nuclear fuel[5]		X	0,075[5]

[1] Part of the storage capacity has to be kept free for unloaded cores.
[2] The storage facility at Obrigheim is a wet storage facility outside of the reactor building that was commissioned in 1999.
[3] Including the positions for HAW canisters.
[4] Total amount from power reactors; an additional approx. 6 tHM from the THTR.
[5] Excluding thorium.

3. DATA REQUIREMENTS FOR SPENT FUEL MANAGEMENT

In order to take the appropriate safety measures, specific data are required for the planning of storage and disposal facilities for spent nuclear fuel. The type of data needed to perform a safety assessment depends on the type of facility. In Germany only the storage of spent fuel is currently being practiced. At all NPP sites with operating reactors dry storage facilities are under construction or already in operation. As for the disposal of spent fuel elements R&D work is still going on.

3.1. Storage of spent fuel elements in containers

An applicant for a license has to prove that he has taken all necessary measures to guarantee the safety of such a facility according to the existing safety regulations. Important requirements are laid down in the RSK Recommendation "Safety Guidelines for the dry Storage of Spent Fuel Elements in Containers". These guidelines address safety requirements with regard to the containment of radioactive substances, criticality safety, heat removal, shielding and radiation protection.

In order to fulfil these requirements certain conditions have to be met by the spent fuel and the containers which are linked to the compliance with specific limits. Within this context the operators of the nuclear power plants have to provide information on the spent fuel that is going to be stored. These data are used by Gesellschaft für Nuklear-Service (GNS) mbH to prepare the documentation for the loading of the storage containers in order to meet the "Technical Acceptance Requirements" that are part of the licenses for storage facilities together with the corresponding regulations for their application. The texts of the licenses are published at the BfS Website (http://www.bfs.de/transport/gv/dezentrale_zl/standort).

Typical data being required to analyse the safety of an storage facility and the associated storage containers within the licensing process are:

- Type of fuel (PWR fuel, BWR fuel, UO_2 fuel, MOX fuel);
- Dimensions and technical specifications of the fuel element;
- Weight of the fuel element;
- Burnup of the fuel;
- Number of defect fuel rods;
- Initial enrichment;
- Content and distribution of fissile material;
- Total activity;
- Dose rates;
- Heat generation;
- Residual moisture;
- Most of these data are provided by the "Technical Acceptance Requirements".

3.2. Disposal of spent fuel in POLLUX containers or fuel rod canisters

For the specification of requirements on radioactive waste for disposal it is necessary to identify the disposal-relevant characteristics and parameters of these waste. For non heat-generating waste, HLW glass blocks and compacted hulls and structural parts this has been done in Germany in the middle of the nineties. In 1999 an expert working group also issued a statement on disposal-relevant characteristics and parameters of conditioned LWR fuel elements in disposal packages, e.g. of the POLLUX type. This document contains a list of 12 groups of disposal-relevant characteristics and parameters which should sufficiently characterize the waste package. The list includes the following parameter groups:

- Total activity,
- Activities of relevant radionuclides,
- Criticality safety,
- Thermal characteristics,
- Dose rates,
- Surface contamination,
- Description of the waste product,
- Hydrolytic resistance and release of radionuclides,
- Description and quality of the disposal container,
- Mass of the disposal package,
- Mechanical characteristics,
- Labelling of the disposal package.

4. METHODS TO ACQUIRE SPENT FUEL DATA

4.1. Data provided by fuel manufacturer and reactor operator

Relevant data of spent fuel may be provided by the manufacturer of the fuel (e. g. distribution of fissile material, initial enrichment) and/or by the reactor operator (type of fuel, burnup, radioactive inventory, dose rates, heat generation). These data are used by Gesellschaft für Nuklear-Service (GNS) mbH to prepare the documentation for the loading of the storage containers in order to meet the technical acceptance requirements at the storage facility.

4.2. Annual inquiry of spent fuel quantities

4.2.1. Legal framework and procedure

From each nuclear power reactor in Germany an amount of 10 to 30 t HM are unloaded annually according to its respective power. This results in a total amount of about 400 t HM every year.

According to the Atomic Energy Act (AtG) the utilities are committed to make provisions for the safe management of the residues from reactor operations. Section 9a AtG requires that radioactive residues are to be re-used without any harm or disposed of as radioactive waste. In this context utilities have to prove:

- adequate provisions for the safe management of spent fuel and reprocessing waste;
- the feasibility of the recycling of separated plutonium in power reactors;
- the safe disposition of reprocessed uranium in storage facilities.

In order to fulfil these requirements the Federal Ministry of the Environment, Nature Conservation and Reactor Safety (BMU) performs an annual inquiry asking for information on the amounts of spent fuel, the provisions taken for the safe management of this fuel and for the management of uranium and plutonium separated by reprocessing as well as the resulting waste. The questionnaire has been developed by a working group including representatives of the utilities.

The utilities fill out the questionnaire and send it to the responsible state ("Länder") authorities. The authorities check whether the legal requirements of Section 9a AtG are fulfilled and send a copy to BMU in order to allow a comprehensive overview of all German nuclear power plants. Under this viewpoint BMU also checks the information and forwards the forms to GRS for further evaluations.

GRS evaluates the information with regard to consistency, plausibility of the data and feasibility of the indicated provisions. The findings and aggregated data are included in a report which is prepared every year for the respective inquiry.

4.2.2. Types of data registered within the inquiry

The contents of the questionnaire and the types of the requested data have only slightly changed until 2001. However, in 2002 the forms were completely revised due to the new requirements in the revised Atomic Energy Act. Until 2001 the following date were inquired:

- General data (reactor type, electrical power),
- Number of spent fuel elements in the reactor cores (uranium fuel, MOX fuel, others),
- Number of spent fuel elements in storage (type, age, storage capacity),
- Fuel element data (mass, planned burnup, decay time),
- Number of fuel elements to be unloaded in the next years,
- Transport to reprocessing plants or storage facilities performed in the past and planned in the future.

In the new questionnaire since 2002 any general and fuel element data and most information on past actions (e.g. transports) have been abandoned. On the other hand some information has been added, in particular:

- Information on plutonium recycling,
- Information on disposition of reprocessed uranium,
- No radiological data are collected within these questionnaires.

4.2.3. Results of the 2004 inquiry

The main results of the 2004 inquiry are the following:

(1) About 11,380 t HM of spent nuclear fuel have been unloaded from German power reactors since the beginning of nuclear power production in the sixties.
(2) 3,510 t HM of spent nuclear fuel were stored in wet storage facilities by the end of 2004. That means that there remained free storage capacity of 1,190 t HM.

(3) 1,220 t HM of spent nuclear fuel were stored in dry storage facilities including the AFR storage facilities at Gorleben and Ahaus as well as the storage facilities at the reactors. Compared to the planned capacities there was a free dry storage capacity of 13,830 t HM at the reactor sites.

(4) 6,210 t HM have been shipped to the reprocessing plants at La Hague (5,360 t HM) and Sellafield (850 t HM).

The spent fuel arisings accumulated by 2025 is estimated at 17,200 t HM. 6,620 t HM of this quantity will be shipped to reprocessing plants and a small amount of 50 t HM for storage or re-use to Sweden and Hungary. The largest part, however, will be disposed of (10,530 t HM).

4.2.4. Trends

The cumulated amounts of spent nuclear fuel discharged from German NPPs for different years are shown in Figure 1.

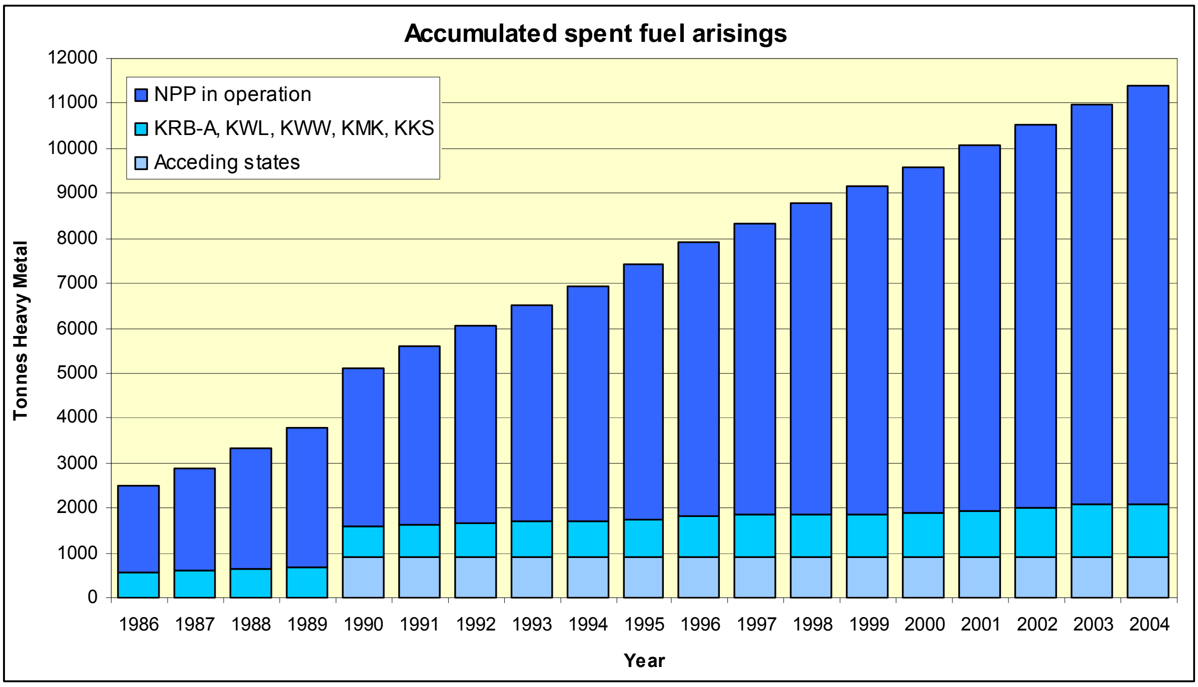


FIG.1. Cumulated spent fuel arisings in Germany.

The burnup of the spent fuel has been increased continuously during the past 20 years. This trend can be seen in Figure 2. The left part shows the targets which were declared by the operators to be reached in the future while the right part of the figure shows the same trend for the real burnup values being expected for the next discharge.

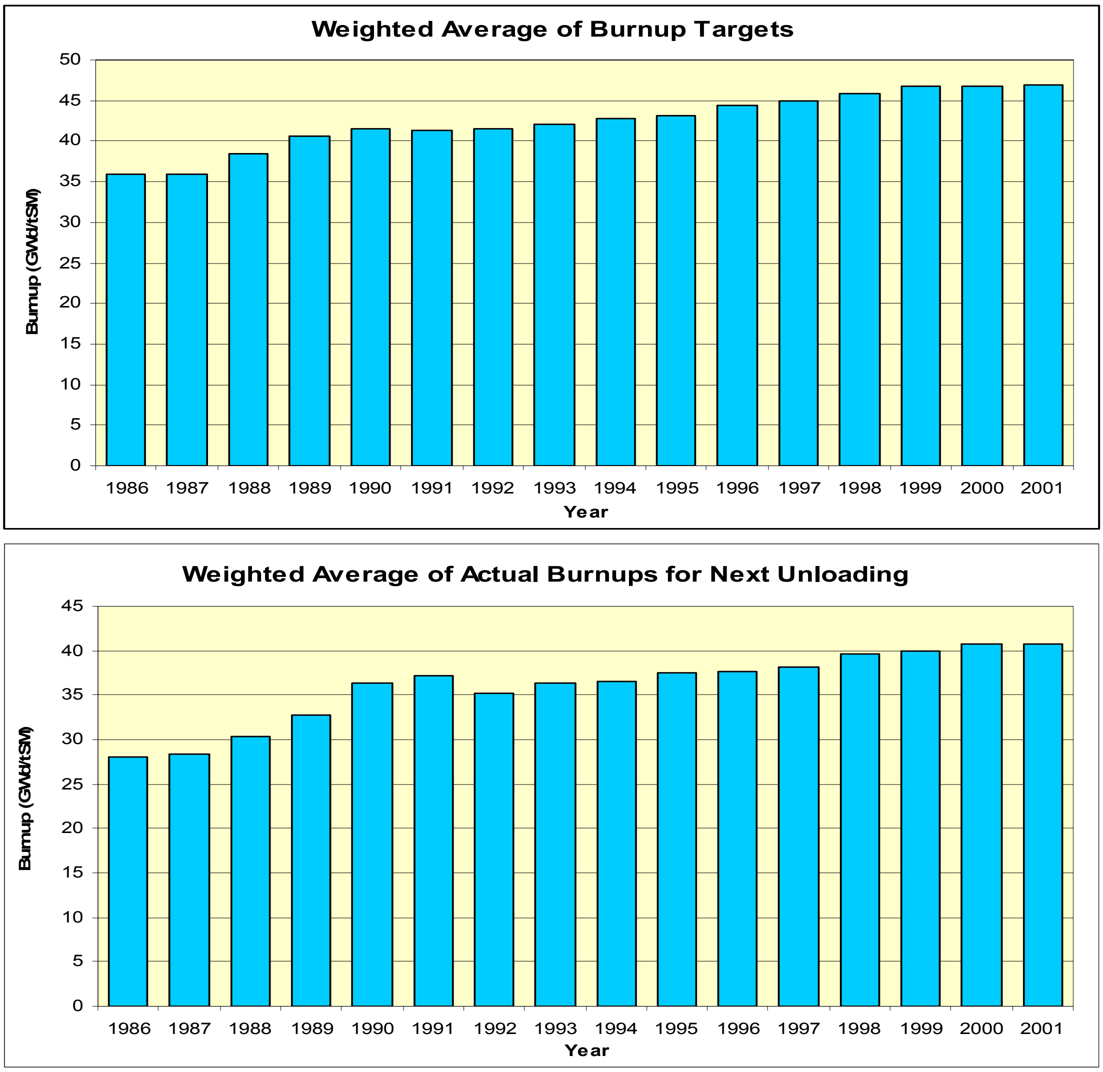


FIG. 2. Trends on the expected burnup of spent fuel elements (weighted averages; left: long term expectations; right: short term expectations).

4.3. Calculation of radioactive constituents of spent fuel

4.3.1. Overall activity

Estimates of radiological parameters like radionuclide contents etc. can be achieved by calculations. Examples of such calculations are summarized in the following tables [2]. Table 2 contains data on the content of fission products, actinides and some major radionuclides for different burnups, each with a decay time of 5 years. Table 3 gives the corresponding values for a decay time of 50 years.

TABLE 2. INVENTORY OF FISSION PRODUCTS, ACTINIDES AND SOME MAJOR RADIONUCLIDES IN SPENT FUEL FOR DIFFERENT BURNUPS, EACH WITH A DECAY TIME OF 5 YEARS (IN BQ/THM) [2]

Initial enrichment →	3.2%	3.6%	3.9%	4.3%	4.6%	4.6%
Burnup [GWd/tHM] →	35	40	45	50	55	60
Fission products. total	1.7×10^{16}	2.0×10^{16}	2.2×10^{16}	2.5×10^{16}	2.7×10^{16}	3.0×10^{16}
Actinides	4.3×10^{15}	4.8×10^{15}	5.4×10^{15}	5.8×10^{15}	6.3×10^{15}	6.8×10^{15}
Sr-90	2.4×10^{15}	2.8×10^{15}	3.1×10^{15}	3.5×10^{15}	3.8×10^{15}	4.0×10^{15}
I-129	1.2×10^{9}	1.4×10^{9}	1.5×10^{9}	1.7×10^{9}	1.9×10^{9}	2.0×10^{9}
Cs-137	3.6×10^{15}	4.1×10^{15}	4.6×10^{15}	5.1×10^{15}	5.6×10^{15}	6.1×10^{15}
Pu (alpha)	1.4×10^{14}	1.7×10^{14}	2.1×10^{14}	2.5×10^{14}	2.9×10^{14}	3.3×10^{14}
Americium	4.3×10^{13}	4.8×10^{13}	5.2×10^{13}	5.6×10^{13}	6.0×10^{13}	6.3×10^{13}
Np-237	1.2×10^{10}	1.5×10^{10}	1.8×10^{10}	2.0×10^{10}	2.3×10^{10}	2.5×10^{10}
Curium	9.1×10^{13}	1.3×10^{14}	1.8×10^{14}	2.3×10^{14}	3.0×10^{14}	4.3×10^{14}

TABLE 3. INVENTORY OF FISSION PRODUCTS. ACTINIDES AND SOME MAJOR RADIONUCLIDES IN SPENT FUEL FOR DIFFERENT BURNUPS. EACH WITH A DECAY TIME OF 50 YEARS (IN BQ/THM) [2]

Initial enrichment →	3.2%	3.6%	3.9%	4.3%	4.6%	4.6%
Burnup [GWd/tHM] →	35	40	45	50	55	60
Fission products, total	4.2×10^{15}	4.8×10^{15}	5.3×10^{15}	5.9×10^{15}	6.5×10^{15}	7.0×10^{15}
Actinides	7.7×10^{14}	8.7×10^{14}	9.7×10^{14}	1.1×10^{15}	1.2×10^{15}	1.3×10^{15}
Sr-90	8.3×10^{14}	9.5×10^{14}	1.1×10^{15}	1.2×10^{15}	1.3×10^{15}	1.4×10^{15}
I-129	1.2×10^{9}	1.4×10^{9}	1.5×10^{9}	1.7×10^{9}	1.9×10^{9}	2.0×10^{9}
Cs-137	1.3×10^{15}	1.5×10^{15}	1.6×10^{15}	1.8×10^{15}	2.0×10^{15}	2.2×10^{15}
Pu (alpha)	1.1×10^{14}	1.3×10^{14}	1.6×10^{14}	1.9×10^{14}	2.2×10^{14}	2.5×10^{14}
Americium	1.6×10^{14}	1.7×10^{14}	1.9×10^{14}	2.0×10^{14}	2.2×10^{14}	2.3×10^{14}
Np-237	1.4×10^{10}	1.5×10^{10}	2.0×10^{10}	2.3×10^{10}	2.5×10^{10}	2.8×10^{10}
Curium	1.6×10^{13}	2.3×10^{13}	3.3×10^{13}	4.2×10^{13}	5.4×10^{13}	7.7×10^{13}

The dependence of emission rates and heat production from burnup and decay time can be seen from Tables 4 and 5.

TABLE 4. EMISSION RATES AND HEAT PRODUCTION OF SPENT FUEL AS A FUNCTION OF BURNUP (DECAY TIME: 5 YEARS) [2]

Initial enrichment →	3.2%	3.6%	3.9%	4.3%	4.6%	4.6%
Burnup [GWd/tHM] →	35	40	45	50	55	60
Neutrons/sec	3.6×10^8	5.2×10^8	7.2×10^8	9.2×10^8	1.2×10^9	1.7×10^9
Photons/sec	7.8×10^{15}	9.3×10^{15}	1.1×10^{16}	1.2×10^{16}	1.4×10^{16}	1.6×10^{16}
Heat production [kW]	1.82	2.16	2.52	2.88	3.27	3.73

TABLE 5. EMISSION RATES AND HEAT PRODUCTION OF SPENT FUEL AS A FUNCTION OF BURNUP (DECAY TIME: 50 YEARS) [2]

Initial enrichment →	3.2%	3.6%	3.9%	4.3%	4.6%	4.6%
Burnup [GWd/tHM] →	35	40	45	50	55	60
Neutrons/sec	8.2×10^7	1.1×10^8	1.5×10^8	1.9×10^8	2.5×10^8	3.5×10^8
Photons/sec	1.5×10^{15}	1.7×10^{15}	1.9×10^{15}	2.1×10^{15}	2.3×10^{15}	2.5×10^{16}
Heat production [kW]	0,57	0,66	0,75	0,84	0,94	1,04

The activity of the total number of spent fuel assemblies (reference date: 31 December 2004) stored on-site at the reactors and in the container storage facilities can be estimated based on the following assumptions [1]

In an initial approximation, only uranium dioxide fuel is considered. The fuel assemblies are divided into different categories on the basis of age: for those fuel assemblies unloaded prior to 1998, the assumed mean burnup is 40 GWd/tHM, whilst for those unloaded between 1999 and 2004, the mean burn-up is assumed as 45 GWd/tHM.

Based on these assumptions, the radioactive inventories may be estimated as follows:

(1) Inventory of spent fuel stored in NPP cooling ponds 1.4×10^{20} Bq (corresponding to 3402 t HM);
(2) Spent fuel assemblies in containers, interim and temporary storage facilities 2.6×10^{19} Bq (corresponding to 1336 t HM);
(3) Thus, the total activity of all spent fuel assemblies currently in storage as per the reference date is approximately 1.6×10^{20} Bq.

The activity of Sr-90/Y-90 can be estimated at 2.8×10^{19} Bq, whilst the activity of Cs-137/Ba-137m can be estimated at 3.9×10^{19} Bq.

4.3.2. Uranium and plutonium content

The factors that influence the content of uranium and plutonium in the spent fuel include:

- the type of reactor (LWR, Candu, Magnox, AGR),
- the type of fuel (uranium oxide, U/Pu mixed oxide),
- the initial enrichment,
- the burnup,
- the decay time after unloading.

The following tables present some figures of the amounts of fissile materials in spent fuel for different conditions. Tables 6 and 7 give examples for the uranium and plutonium contents in uranium or MOX fuel respectively. The influence of the target burnup on the plutonium content in spent uranium fuel is shown in Table 8.

TABLE 6. URANIUM AND PLUTONIUM QUANTITIES IN SPENT URANIUM FUEL WITH 3.6% INITIAL ENRICHMENT, 40 GWD/THM BURNUP AND 7 YEARS DECAY TIME [3]

Uranium		
Isotope	g/tHM	Wt%
U-234	159	0.02
U-235	7,737	0.8
U-236	4,549	0.5
U-238	934,200	98.7
Total	946,700	100

Plutonium		
Isotope	g/tHM	Wt%
Pu-236	0.0004	<0.001
Pu-238	212	2.0
Pu-239	5,773	55.0
Pu-240	2,715	25.9
Pu-241	1,094	10.4
Pu-242	709	6.8
Total	10,500	100

TABLE 7. URANIUM AND PLUTONIUM QUANTITIES IN SPENT MOX FUEL WITH 4.4% PU_{FISS} INITIAL ENRICHMENT, 40 GWD/THM BURNUP AND 7 YEARS DECAY TIME /GRS 96/

Uranium		
Isotope	g/tHM	Wt%
U-234	147	0.02
U-235	3,477	0.4
U-236	688	0.08
U-238	901,500	99.5
Total	905,800	100

Plutonium		
Isotope	g/tHM	Wt%
Pu-236	0.0003	<0.001
Pu-238	1,537	3.2
Pu-239	19,100	39.8
Pu-240	15,310	31.9
Pu-241	6,278	13.1
Pu-242	5,808	1.2
Total	48,030	100

TABLE 8. AMOUNT OF PLUTONIUM IN GRAMS PER GWD OF PRODUCED ENERGY IN URANIUM FUEL ELEMENTS FOR DIFFERENT TARGET BURNUPS AND CORRESPONDING INITIAL ENRICHMENTS [3]

Target burnup (GWd/tHM)	Initial enrichment (wt%)	g Pu/GWd					
		Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Pu total
30	2.8	4.2	172	79	29	17	305
40	3.4	5.4	142	69	28	19	263
50	4.1	6.5	123	62	26	20	237
60	4.8	7.5	111	57	24	20	220

5. REGISTRATION SYSTEMS

5.1. Spent nuclear fuel

As already mentioned before, the spent fuel inventories of the German nuclear power plants are assessed by annual inquiries. This leads to a continuously updated data set with regard to the number of fuel elements and the corresponding masses of heavy metal. For the evaluation of these data GRS has developed a computer code called LUISE in 1995. This code has to be adapted to the new contents of the questionnaire.

With the exception of this annual inquiry (cf. also Section 4.1) there is no centralized data management system for spent fuel data covering all nuclear power plants in Germany. Of course, however, each NPP operator has also separate and much more detailed information on his own spent fuel elements but this information is not registered centrally and only available on demand for specific analyses. One important application where those data are needed is the licensing and operation of onsite storage facilities for spent fuel elements.

Of course, BfS as the regulatory body for the licensing of all spent fuel storage facilities disposes of all data and information needed for the safety assessments. Besides that there exists a Coordinating Group for Information on the Handling of Fuel Containers (KOBAF) which is composed of Federal States' representatives and gathers information being relevant for the storage and handling of the containers. The Federal States which are responsible for the supervision of the storage facilities in their respective territory have access to all information.

5.2. Radioactive waste

Other registration systems exist for radioactive waste.

Similar to the annual spent fuel inquiries the Federal Office for Radiation Protection (BfS) performs nationwide annual inventories for the radioactive waste except for spent fuel. The information covers all waste producers in Germany but is limited to waste volumes and does not include radioactive inventories.

The utilities use a registration system called AVK tracking radioactive waste from the operation and decommissioning of nuclear facilities during all management steps, beginning at the production of the waste and ending at its disposal. The system is used and operated by GNS.

Of course several installations like research centers or storage facilities use their own tracking systems. One example is the KADABRA system used by the radioactive waste management department in the research center Karlsruhe.

GRS is currently developing a database for the radioactive waste in the so-called “State Collecting Facilities” where radwaste from hospitals, industry and research is collected and stored under the responsibility of the Federal States.

REFERENCES

[1] Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management: Report under the Joint Convention by the Government of the Federal Republic of Germany for the Second Review Meeting in May 2006, September 2005.

[2] CLEMENTE, M., et al., Sicherheitsaspekte bei der Erhöhung des Abbrandes der Brennelemente, GRS-A-1763, February 1991

[3] GEWEHR, K., et al., Entsorgungskonzepte für Plutonium, GRS-A-2343, March 1996.

SPENT FUEL DATA MANAGEMENT IN THE REPUBLIC OF KOREA

JE-KEUN CHON
Korea Institute of Nuclear Safety,
Daejeon, Republic of Korea

Abstract

The Republic of Korea has implemented a database which integrates information on radioactive waste management on national level. This database, named WACID, is designed to enhance efficiency and safety of managing radioactive waste and spent fuel in Korea, by facilitating any stakeholders in need of associated information.

1. INTRODUCTION

The Korean government has maintained a consistent national policy for stable energy supply by fostering nuclear power industries under the insufficient energy resources in the country. Nuclear power reached approximately 40 % of total domestic electricity generation.

Since the commencement of the first commercial operation of Kori Unit 1 in April 1978, 20 units of NPPs are commercially operating in October 2005. Four units out of the 20 operating NPPs are Pressurized Heavy Water Reactors (PHWRs) at Wolsong. The 16 units located in Kori, Younggwang, and Ulchin are Pressurized Water Reactors (PWRs). The spent fuels generated from these NPPs are stored in spent fuel storage pools at the reactors or an on-site dry storage facility. The low and intermediate-level radioactive wastes (LILW) generated from the NPPs are stored at the on-site radioactive waste storage facilities.

Only one research reactor is now in operation: the HANARO reactor at the Korea Atomic Energy Research Institute (KAERI) located in Daejeon. Its operations commenced in 1995 and it has thermal power of 30 MW. The two research reactors, KRR-1 & 2, located at the former KAERI site in Seoul, were shut down and the reactors and the auxiliary facilities had been decommissioned since 1997.

2. NATIONAL POLICY FOR SPENT FUEL MANAGEMENT

The 249th meeting of Atomic Energy Commission (AEC) held in September, 1998, decided on the “National Radioactive Waste Management Policy” aiming to construct and operate a LILW disposal facility by 2008 and a centralized spent fuel interim-storage facility by 2016, but the site selection had not been successful yet. Therefore, the 253rd meeting of AEC held on December 17, 2004, decided that the construction and operation of LILW disposal facility will be accomplished by 2008, and but the national policy for spent fuel management including construction of the centralized spent fuel interim-storage facility will be decided in the view of the domestic and international technology development.

In this regards, the national policy for spent fuel management including the construction of the interim storage facility for spent fuel shall be timely decided considering the saturation of spent fuel storage capacity from 2016 through national consensus by public consultation among stakeholders.

3. LEGISLATIVE FRAMEWORK

National laws related to the safety of spent fuel management are the Atomic Energy Act (AEA), the Electricity Business Act (EBA), the Environmental Impact Assessment Act and etc. All the provisions on nuclear safety regulation and radiation protection are entrusted to the Atomic Energy Act. The Atomic Energy Act was enacted as the main law concerning safety regulations for spent fuel and radioactive waste.

The laws concerning nuclear regulation, as shown in Figure 3-1, consist of 4 stages: the Atomic Energy Act, the Enforcement Decree of the same Act, the Enforcement Regulations of the same Act (including regulations concerning technical standards of nuclear facilities, etc., and regulations concerning technical standards of radiation safety management), and the Notices of the MOST.

4. SPENT FUEL MANAGEMENT PRACTICES

Spent fuels generated from nuclear power plants are stored in the spent fuel storage facility in each unit. The storage capacity for spent fuel had to be expanded in connection with the delayed schedule for construction of the Away-From-Reactor (AFR) Storage according to the 249th and the 253rd of Atomic Energy Commission. For PWR reactor, Kori unit 3 and Ulchin units 1 and 2 had already expanded their storage capacity by adopting high-density storage racks, Kori unit 4 and Yonggwang units 1, 3 and 4 will be expanded their storage capacity by adopting high-density storage racks. Kori units 1 and 2 encountered the shortage of the spent fuel storage capacity, so spent fuel which was in excess of the SFP storage capacity has been transferred to the Spent Fuel Storage Pool (SFSP) of Kori units 3 and 4. For PHWR at Wolsung site, an on-site Dry Storage Facility (DSF) has already been constructed and additional DSF will be constructed to solve the shortage capacity of the SFSP of Wolsong units 1, 2, 3 and 4.

5. SPENT FUEL INVENTORY

As of December 2004, spent fuel inventories for PWRs and PHWRs are 3397 MTU and 3889 MTU, respectively. The inventories, initial enrichment of fuel and types of spent fuel in storage are as given in Table 1.

TABLE 1. INVENTORY OF SPENT FUELS STORED IN NPPS (AS OF DECEMBER 2004)

NPP	TYPE	STORED AMOUNT (MTU)	INITIAL ENRICHMENT (W/O)	FUEL TYPE
Kori Site	Wet storage	1415	3.4～4.2	PWR
Yonggwang Site	Wet storage	1140	3.8～4.4	PWR
Ulchin Site	Wet storage	842	3.8～4.4	PWR
Wolsong Site	Wet storage	2570	natural uranium	CANDU
	Dry storage	1319		

6. SPENT FUEL INFORMATION DATABASE SYSTEM

6.1. Waste comprehensive information database system (WACID)

As the generation and accumulation of radioactive waste continues to increase with the domestic use of nuclear energy, the necessity arises for establishing a national level comprehensive database system which applies to the state-of-the-art information technology in order to manage information related to the safety management of various and massive radioactive waste sources in the systematic manner.

Coping with the urgent national demand, a WACID (WAste Comprehensive Information Database system) was developed by KINS from July 2002 to May 2004 as the 2-year project. The WACID system was tested to confirm the integrity and then, its practical operation was started from January of 2005.

The system collects data on the radioactive waste information from domestic nuclear installations through the internet quarterly and verifies the data integrity. After verification of data, the system operator produces a variety of report to offer the information to the public. By using this system, individual waste generators can report the waste inventory more efficiently.

The WACID DB system has employed 8-sub modules (solid radioactive waste, liquid effluent, gaseous effluent, spent fuel etc.) so as to maximize data sharing, to minimize data redundancy, to enhance the effectiveness of system in operation, and to avoid unexpected complication of system itself, due to involvement of a variety of data characteristics from numerous waste generators. Figure 6.1. shows the WACID system structure.

The WACID system will play an important role to direct the radioactive waste policy by Government, to promote R&D activities and to upgrade the domestic level of safe management of radioactive waste. In addition, the developed system will do much for achieving 5 principles (i.e. independence, openness, clarity, efficiency, reliability) of nuclear safety regulation, by providing essential information to the general public.

6.2. DREAMS-RWM database system

The KHNP has developed the integrated Radioactive Waste Management system (DREAMS-RWM) which allows real time management of radioactive waste generation, treatment and disposal. And this system allows interface with national radioactive waste information system (WACID).

The DREAMS-RWM system development project began on April, 2002. Through system design and configuration by February 2002, system operation started on May, 2002. This system was comprised of gaseous and liquid radioactive waste release managing programme, solid radioactive waste managing programme, clearance control programme and statistical management of radioactive waste.

This system will contribute to allow integrated management of regulatory routine gaseous and liquid radioactive waste release management empowered by an aligned information system and so, improved commodity of rapid data transaction, creating reports, data inquiry of numerous statuses and preparing a database platform to interface with the government information system.

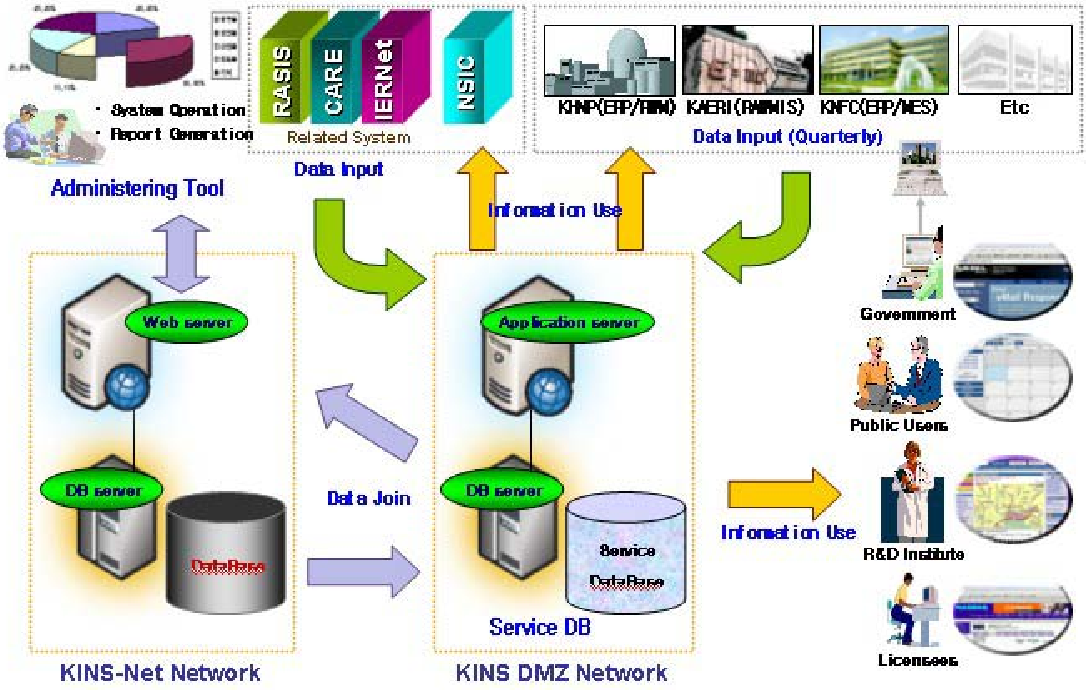


FIG. 6-1. WACID system structure.

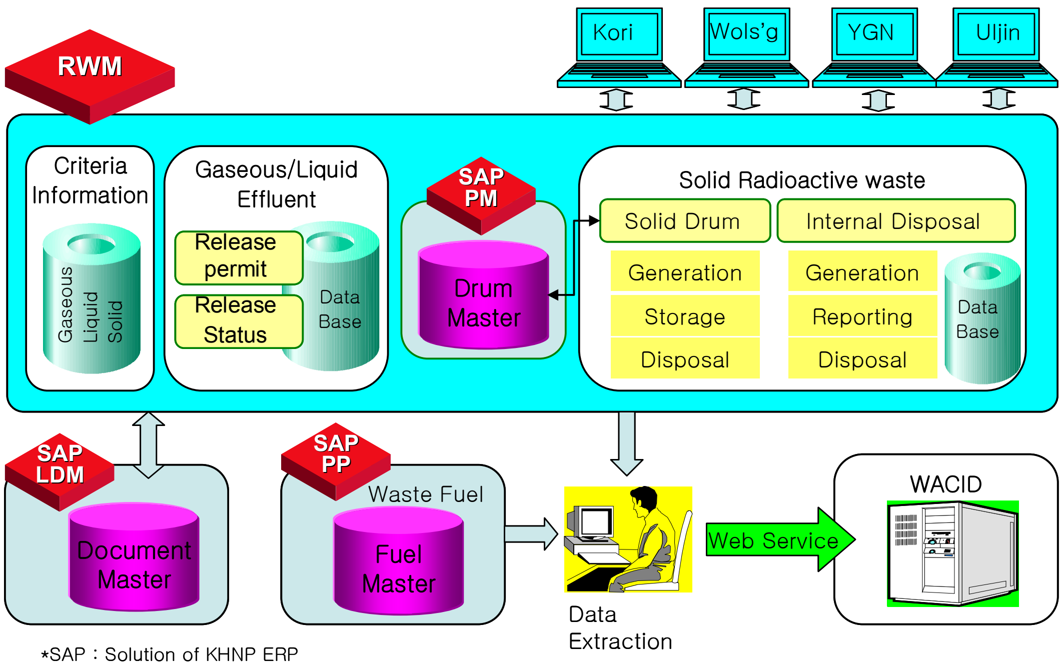


FIG. 6-2. Structure of DREAMS-RWM system.

To enable life cycle tracking of solid radioactive waste management; generation, temporary storage and relocation for permanent disposal, the KHNP standardized the drum serial numbers and putting in all related waste information to the DREAMS-RWM system. This improved the efficiency of information search. The database of clearance wastes which are discarded as industrial wastes through the regulatory approval has been integrated in DREAMS-RWM system as well.

The KHNP has made it possible for the DREAMS-RWM system end users to search, display and report information, using statistics provided by real time computer system. Through this prompt transaction and reference of information, end users can access the major reports such as generating situation of solid radioactive waste in each site, monthly/quarterly reports, and reports about monthly/yearly control situation.

The DREAMS-RWM system has made it enable to provide effective radioactive waste relating information from its computer system to national WACID system.

Annex A

LIST OF SPENT FUEL STORAGE FACILITIES FOR NPPS (AS OF DEC 2004)

(Unit: MTU)

Facility	Storage Type	Storage Capacity		
		Volume stored	Extension Method	Total Capacity
Kori # 1	Wet	116	Trans shipment	209
Kori # 2	Wet	254	Trans shipment	360
Kori # 3	Wet	633	Addition Re-racking	953
Kori # 4	Wet	412	Addition	485
YGN # 1	Wet	375.3	Addition	485
YGN # 2	Wet	331.8	Addition	485
YGN #3	Wet	193.5	-	283
YGN #4	Wet	172.4	-	283
YGN #5	Wet	47	-	292
YGN #6	Wet	20.3		292
Ulchin #1	Wet	319	Re-racking	470
Ulchin #2	Wet	297	Re-racking	370
Ulchin #3	Wet	127	-	283
Ulchin #4	Wet	99	-	283
Ulchin #5	Wet	0		292
Ulchin #6	Wet	0		292
Wolsong #1	Wet	704.4	-	827
	Dry	1316.7	Canister	2041
Wolsong #2	Wet	709.6	-	734
Wolsong #3	Wet	638.1	-	734
Wolsong #4	Wet	520.1	-	734

※ Including emergency cores reserves and additional dry storage canister extension plan for yrs 04–06 at Wolsung site

SPENT FUEL DATA SYSTEM AT IGNALIA NPP

G. KRIVOSEIN
Ignalina NPP, Lithuania

Abstract

An integrated database for on-line tracking of spent fuel has been implemented at Ignalina Nuclear Power Plant. This paper gives an introduction to the spent fuel database system which provides important assistance to the management of spent fuel at the Ignalina NPP.

1. INTRODUCTION

The Ignalina nuclear power plant (INPP) is located in the north-east of Lithuania, closer to the borders with Belarus and Latvia. There are 2 units at INPP, each of which is equipped with RBMK-1500 reactor. The RBMK-1500 is a graphite moderated, channel-type, boiling water reactor. Its design thermal power is 4800 MW. However, for safety reasons, these reactors are currently running at reduced power of maximum 4200 MW. The RBMK-1500 reactor is the most advanced version of RBMK design and the RBMK-1500 fuel assembly has advanced version too.

The fuel assembly contains two fuel bundles. Each bundle has 18 fuel rods arranged within two concentric rings in a central carried rod. The lower bundle of the fuel assembly is provided with an end grid and ten spacing grids. The top bundle has additionally 18 specifically design spacers, which act as turbulence enhances to improve the heat transfer characteristics. The hoop has 12 inclined grooves from which a steam and water mixture gets additional turbulence. This detail is one of distinctive features of RBMK-1500 fuel assembly. The schematic representation of the principal features of the RBMK-1500 fuel assembly is shown in Fig. 1. The main parameters of the fuel assembly are presented in Table 1.

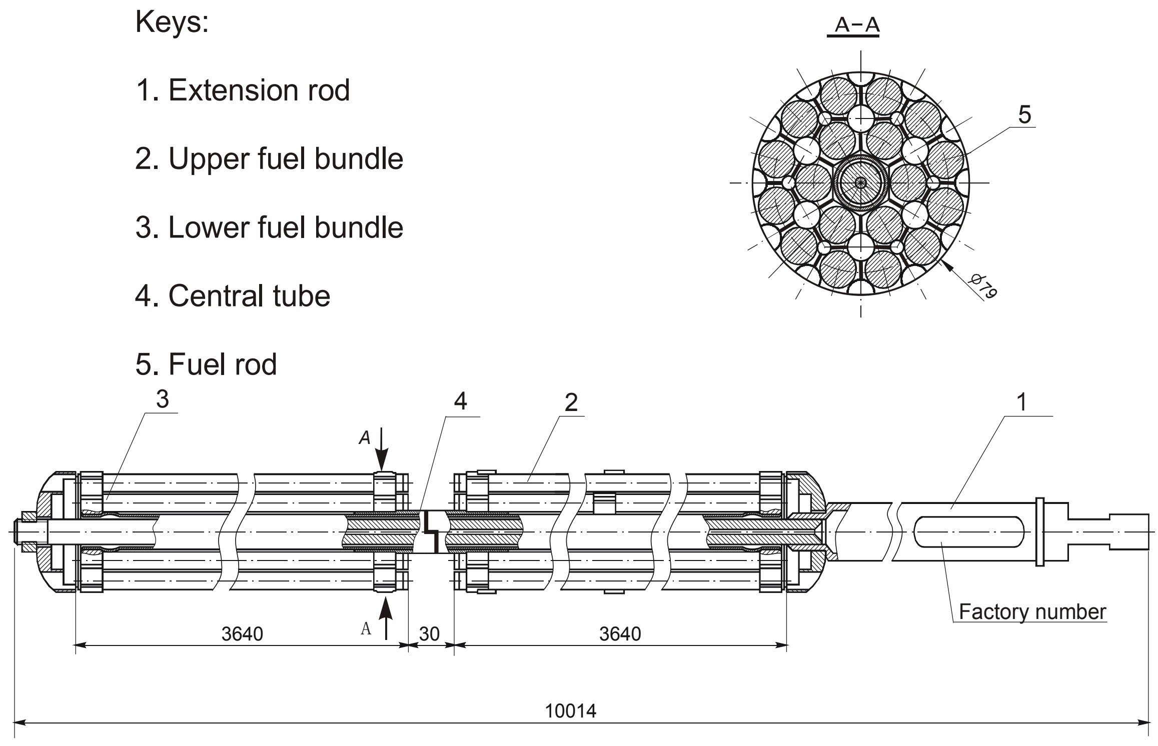


FIG.1. RBMK-1500 fuel assembly.

TABLE 1. RBMK-1500 FUEL ASSEMBLY PARAMETERS

5.1.1.1. FUEL PELLET	
Fuel	Uranium dioxide
Fuel enrichment, % of U^{235}	2, 2.4, 2.6
Fuel pellet diameter, mm	11.5
Fuel pellet length, mm	15
Pellet central orifice diameter, mm	2
FUEL ELEMENT	
Cladding material	Zr+1%Nb
Outside diameter, mm	13.6
Length, m	3.64
Cladding thickness, mm	0.825
Pellet/clad gap, mm	0.22-0.38
Helium pressure in the cladding, MPa	0.5
Maximum linear heat generation rate, W/cm	485
5.1.1.2. FUEL ASSEMBLY	
Number of bundles	2
Number of fuel rods per bundle	18
Diameter (in the core), mm	79
Mass of uranium within fuel pellet, kg	111.2
Mass of uranium within edge fuel pellet, kg	1.016
Maximum permissible power of fuel channel, MW	4.25

Currently the FAs with the fuel of 2.0% enrichment by U^{235}; 2.4% enrichment by U^{235} and burnable poison -0.41% of Erbium; 2.6% enrichment by U^{235} and 0.5% of Erbium are used.

2. SPENT FUEL HANDLING

The unloaded spent FAs are stored for a year in the deep pools of the Cooling Pools Area and then moved to the Hot Cell to be cut up into two bundles. The bundles are placed in the transport baskets which are positioned within the shallow pools. The FA cutting enables to provide a compact way to store the spent FAs. After five or more year's storage in the Cooling Pools Area the transport baskets are to be loaded into the CASTOR-RBMK (or CONSTOR-RBMK) casks to be shipped to the on-site dry storage facility.

Transport baskets with spent FAs subject to loading to shielding casks and transportation to Dry Spent Fuel Storage Facility (DSFSF) shall meet the following requirements:

- fuel bundles placed in basket have been cooled for, minimum, 5 years after withdrawal from reactor;
- calculated value of residual power distribution of spent fuel does not exceed 6 kW;
- activity of isotope Cs^{137} in shielding cask water samples does not exceed 5×10^{-6} Ci/l.

Figure 2 represents the rout scheme of FA handling within the Unit.

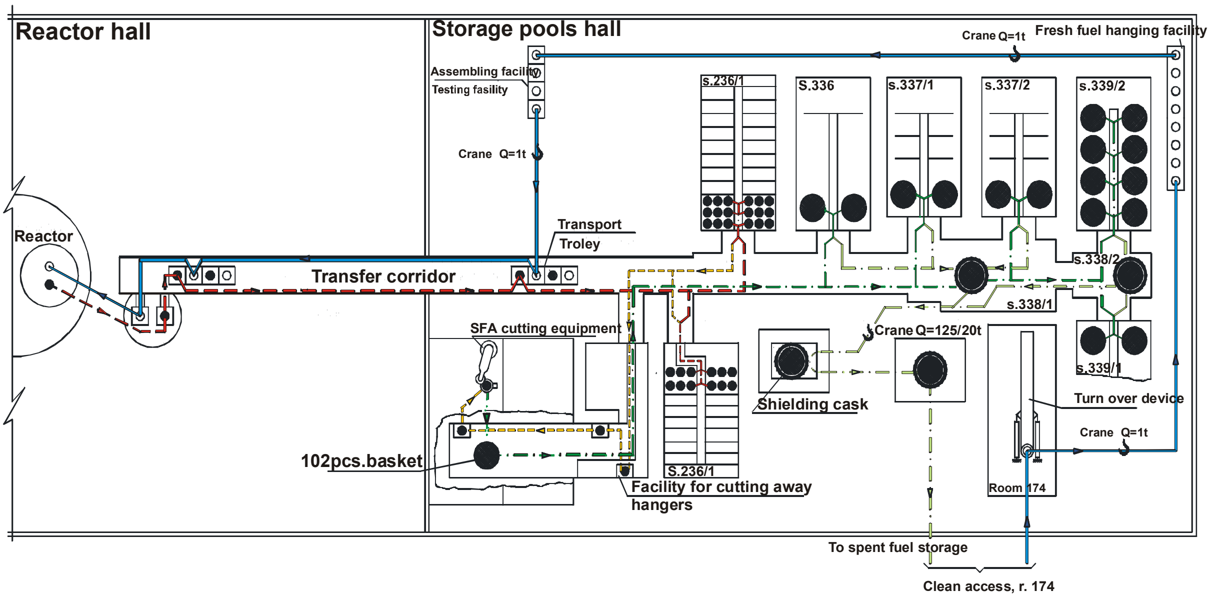


FIG.2. Scheme of the fuel assembly handling.

After a transport basket is loaded into the shielding cask and the cask is closed by leak-tight lid, complete drainage and vacuum drying of internal cavities of the shielding cask is performed.

The control of tightness of spent nuclear fuel is made after the transport basket with spent nuclear fuel is loaded into the shielding cask and during drainage of the shielding cask. The control of tightness is carried out by measurement of activity of radionuclide Cs^{137} in shielding cask water samples by means of gamma-spectrometry.

The cask is filled with inert gas (He) which provides anti-corrosive protection, improves a passive heat-removal and gives the possibility for leak-tightness monitoring of the cask sealing system. The shielding lid of the cask is closed upon confirming the fuel tightness; the cask is closed by the leak-tight lid. Shielding lid of cask CASTOR RBMK is closed by bolts and cask CONSTOR RBMK is welded up.

By means of bridge crane the shielding cask is delivered to room 174 to be placed on the vehicle TP1-3 and be transported by railway to long term storage (till 50 years) to the dry spent fuel storage facility located on Site at 1 km distance from INPP Units.

DSFSF platform (facility 192) is intended for storage of 72 shielding casks, loaded with SNF (20 positions — for shielding casks CASTOR RBMK and 52 positions — for shielding casks CONSTOR RBMK). Currently 20 containers CASTOR RBMK and 29 containers CONSTOR RBMK are stored in DSFSF.

For delivery of the shielding cask from the supplier to SDFSF platform and delivery of shielding cask loaded with SNF from INPP to SDFSF platform there is a railway adjacent to INPP main railway.

TABLE 2. CASK CHARACTERISTICS

DESCRIPTION	VALUE	
	CASTOR	CONSTOR
OD, mm	2120	2340
ID, mm	1500	1482
External height, mm	4385	4746
Internal height, mm	3770	3810
Wall thickness: — bottom, mm — side wall, mm — leak-tight lid, mm — shielding lid, mm — sealing lid, mm — secondary lid, mm — shielding cap, mm	 330 310 210 70 — — 150	 430 429 250 — 40 40 150
Materials: enclosure, lid, cap,	 Malleable iron Carbon steel concrete	 Steel/concrete Carbon steel concrete
Weight, empty casks, kg filled cask, kg	65000 78000	70910 84430

For transportation process purposes a traveling bridge crane GK-100 is installed to be used for cask handling within SDFSF.

All containers with SNF transported for storage are subject to complete entrance control: temperature inspection, check of external surface radioactive contamination. The temperature condition of casks is much lower than the established design value.

Special support pedestals are arranged for storage of shielding casks at facility 192, six under each cask.

Ignalina NPP stores 15300 irradiated fuel assemblies now: in pools-12300, in the dry storage – 3000. Taking into account a plan to close the first unit in 2004 and the second in 2009 the total amount of accumulated spent fuel will be about 21700 fuel assemblies or 2400 ton of Uranium.

3. REGULATORY BODY REQUIREMENTS FOR ACCOUNTING OF NUCLEAR AND RADIOACTIVE SUBSTANCES

In June, 1999 the State Nuclear Power Safety Inspectorate (VATESI) issued the General Requirements For Dry Type Storage Of Spent Nuclear Fuel, VD-B-03-99. The publication contains the requirements for accounting of nuclear and radioactive substances.

"...4.2.1. A licensee shall manage the accounting documentation of nuclear materials in accordance with Regulations Of Accounting And Control Of Nuclear materials In Nuclear And Non-Nuclear Objects, Clause 7.1. of Safety Regulations In Storing And Transportation Of Nuclear Fuel In Nuclear Energy Objects, and Agreement Between the Government of the Republic of Lithuania and the International Atomic Energy Agency for the Application of Safeguards in Connection with the Treaty on the Non-proliferation of Nuclear Weapons. An applicant shall present a program (instruction) of account and control of nuclear and radioactive substances for co-ordination of VATESI.

4.2.2.. A licensee shall keep the accounting documentation of nuclear and radioactive substances as long as the spent nuclear fuel is stored in a storage facility and five years more after disposal of the fuel or its removal from a storage facility. Present accounting documents shall be dubbed. The second set of documents shall be kept in a separate place rather remote from the place of the first copy in order to avoid destruction of both sets during an accident

4.2.3. Information about nuclear materials stored in a storage facility shall be accumulated in state nuclear substance information system, located at VATESI, according to the agreement with IAEA concerning guarantee application.

4.2.4. Besides the information accumulated in a storage facility in accordance with [15] requirements, an operating organisation shall make a database of the spent nuclear fuel loaded to a container. The database shall contain the following:

4.2.4.1. date and time of fuel loading into the reactor and removal from the reactor, indicating the number of technological channel;

4.2.4.2. chroniology of alteration of reactor's power (loading);

4.2.4.3. date and time of fuel placing into maintenance waterpool and its removal from the pool, by indicating a place in a pool;

4.2.4.4. results of analysis of water chemical composition in a case and of measuring and analyses performed during of container draining and vacuuming;

4.2.4.5. the scheme of assemblers' layout in a container;

4.2.4.6. alteration of chemical composition of the fuel;

4.2.4.7. dissipation of residual heat;

4.2.4.8. experiment results of ionising radiation and measuring;

4.2.4.9. periodical measuring of container state;

4.2.4.10. other significant information...."

4. FUEL ACCOUNTING SYSTEM

The Ignalina NPP in the light of applicable nuclear material accounting and control system is considered as a Nuclear Material Balance Area (Fig. 1) in which the physical inventory and the quantity of nuclear materials are determined in accordance with specified procedures. The Dry Spent Fuel Storage Facility is a separate Nuclear Material Balance Area.

In conformity with the FAs handling specified and in order to establish nuclear material accounting and control, the Nuclear Material Balance Area is divided into the separate locations called the Key Measurement Points showing the currently locations of nuclear materials: KMP-A — Fresh Fuel Storehouse;

- KMP-B,F — Fresh Fuel Storage Locations at the Units 1,2 respectively;
- KMP-G — Reactor Cores at the Units 1,2 respectively;
- KMP-D,H — Uncut FAs Cooling Pools at the Units 1,2 respectively;
- KMP-E,I — Hot Cells and the Transport Baskets Storage Pools at the Units 1,2 respectively.

All information being available in the data base relevant to the FAs is shown in two screens titled "Fuel Assembly Record Card" (Fig. 3, 4).

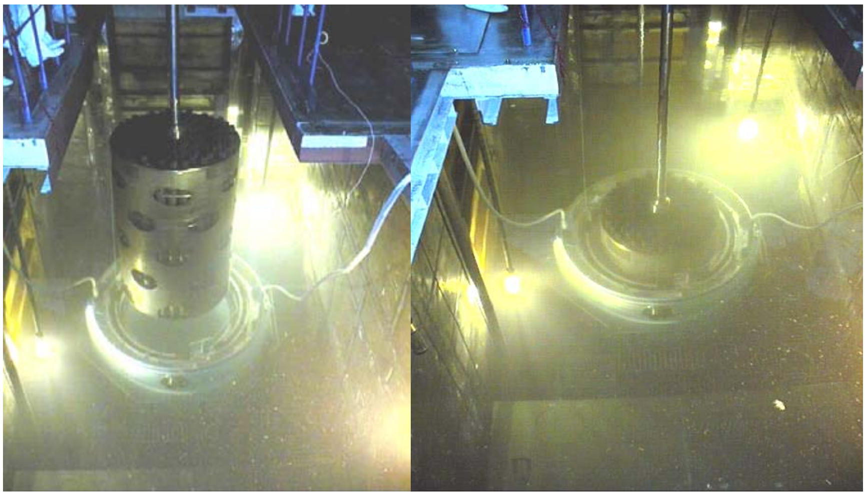


FIG.3. The loading of the basket into the cask.

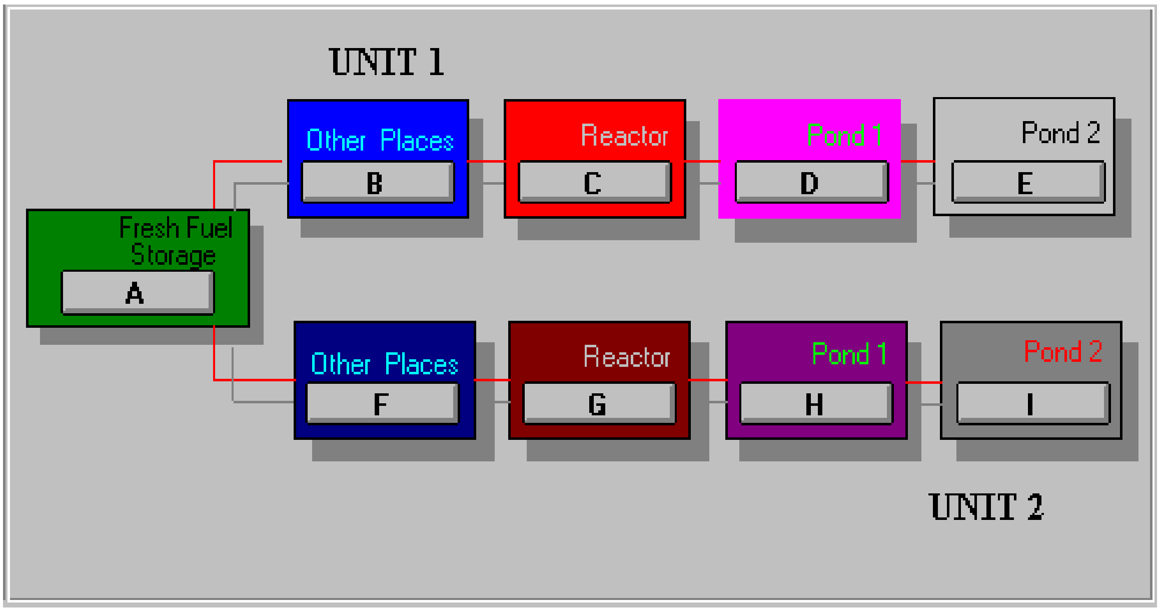


FIG.4. The Nuclear Material Balance Area block chart.

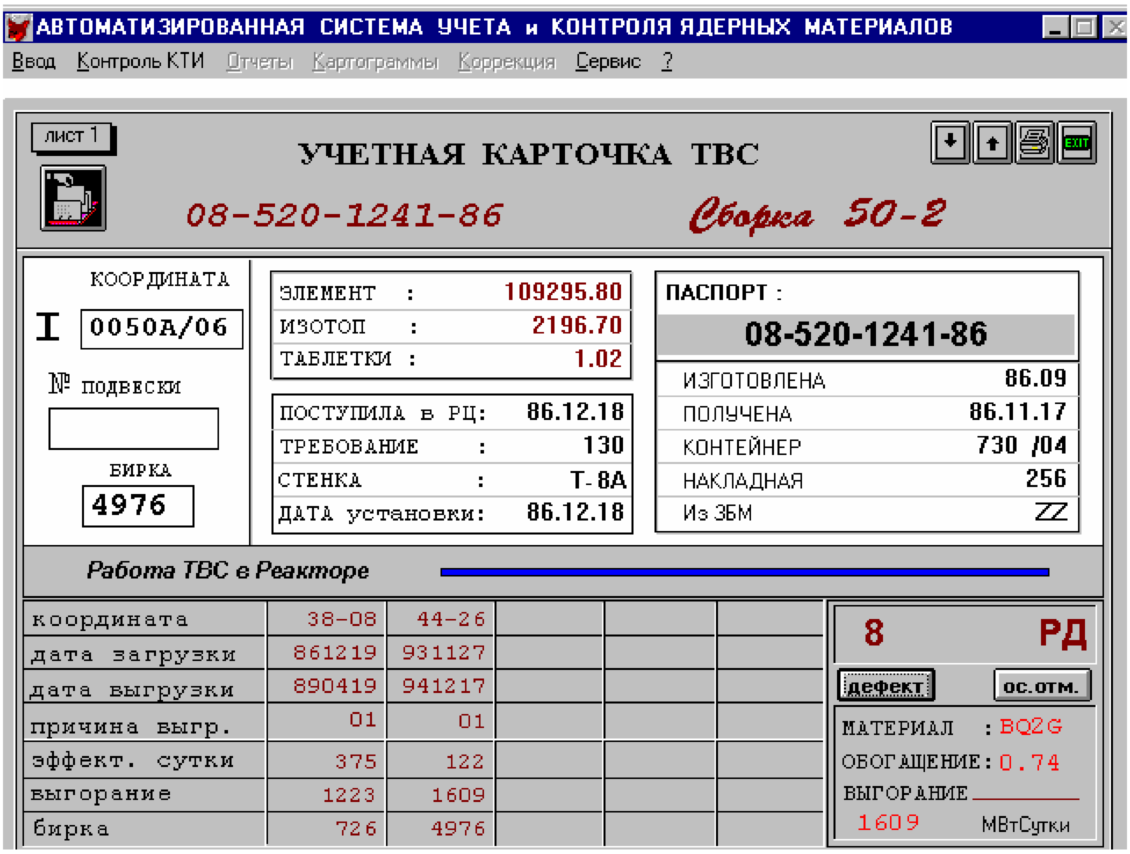


FIG.5. Fuel Assembly Record Card, sheet 1.

TABLE 3. BASKET'S PASSPORT

Cask number 016	Basket number 009
Date of loading	**22.03.2000**
Average burnup, MW. D/kg	**15.55**
Decay heat, kW	**2.74**
Isotope	**Activity, MBq**
^{85}Kr	5.10×10^5
^{90}Y	6.40×10^6
^{90}Sr	6.40×10^6
^{106}Rh	6.18×10^4
^{106}Ru	6.18×10^4
^{134}Cs	2.61×10^5
^{137}Cs	8.49×10^6
^{137m}Ba	8.49×10^6
^{144}Ce	2.43×10^4
^{144}Pr	2.43×10^4
^{147}Pm	1.70×10^6
^{154}Eu	1.18×10^5
^{238}Pu	5.78×10^4
^{239}Pu	3.28×10^4
^{240}Pu	6.35×10^4
^{241}Pu	5.46×10^6
^{241}Am	1.22×10^5
^{244}Cm	1.76×10^4
Total activity of isotops in BASKET, MBq	$\mathbf{3.83 \times 10^7}$

The following data are specified in the fuel assembly record card: FA type; manufacturer's number of FA; enrichment; mass of U-235; mass of U-238; mass of Pu-239; mass of Pu-241; mass of isotopic composition of Plutonium; date of FA shipment from the manufacturer's site; container number; date of shipment from the fresh fuel storage to the reactor building; date of loading into the reactor; number of fuel channel; date of unloading; a reason for unloading; burnup;movments in pools (coordinates); date of the cutting in the hot cell; the number of transport basket; coordinates of bundles in the basket; decay heat as of the date of the basket loading completion.

Following the commissioning in 1992 of the computerized system on nuclear material accounting and control allowed to substantially decrease time required for collecting and processing information relevant to FAs, revealing and eliminating mistakes made in accounting records, enhancing reliability of the nuclear material accounting and control procedures.

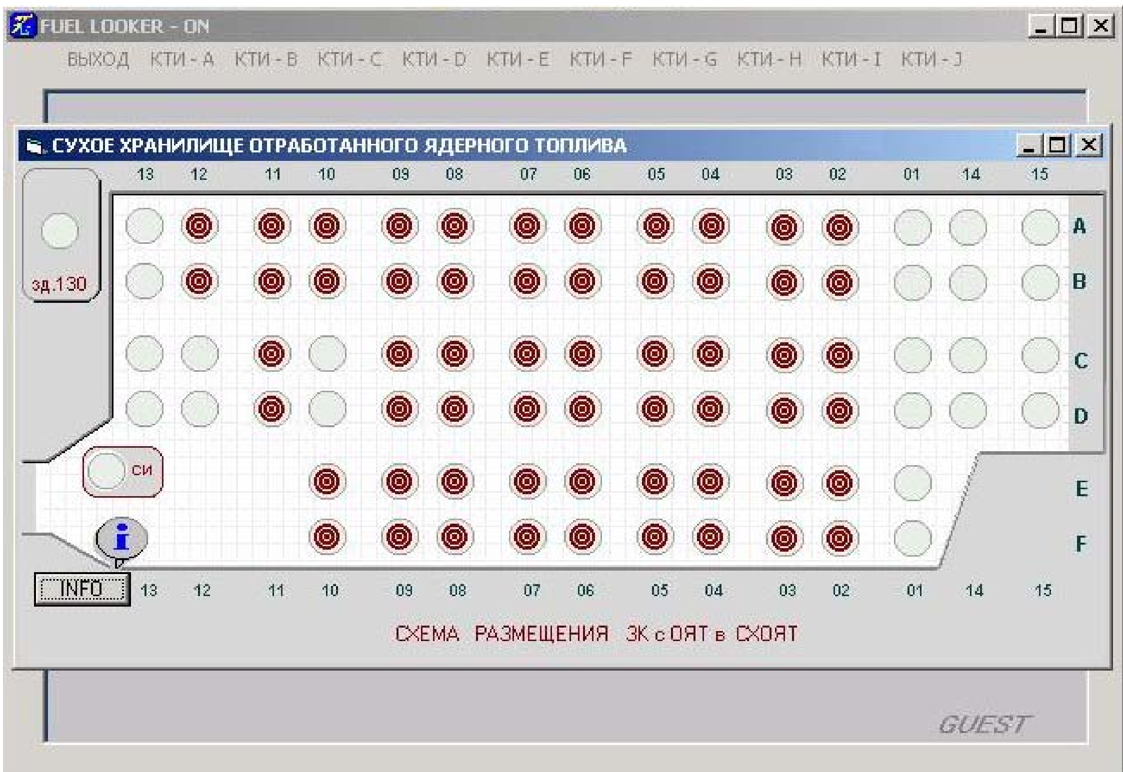
5. RADIOISOTOPE CONTENT IN CASKS WITH SPENT FUEL

In order to meet the State Nuclear Power Safety Inspectorate requirements a special computer code has been developed. The code allows to calculate the radioisotopes content in casks with spent fuel. For this purpose the computer codes system SCALE 4.3 has been implemented. Depleted calculations have been performed by the SAS2H code. Macroscopic cross sections library has been created for RBMK-1500 geometry and type of fuel. The list of isotopes has been endorsed by the Regulatory Body.

6. DRY SPENT FUEL STORAGE DATABASE

The Dry Spent Fuel Storage is a separate Nuclear Material Balance Area (MBA). Ignalina NPP carries out the nuclear fuel accounting in this MBA and submits inventory- changes reports and material balance reports to the Regulatory Body. For each fuel assembly the same data are available in the database. Each cask with spent fuel has its own file of documents which includes:

- Input checklist of the container;
- Checklist of cask loading;
  - Report on fuel leak tightness test;
- Report on vacuum dryout;
  - Basket's passport;
  - Report on radiation measurements;
- Report on cask leak tightness test;
- Report on temperature measurements;
- Report on welding;
- Cask manufacturer documents;



7. CONCLUSION

The management of data on spent nuclear fuel at Ignalina NPP is performed in accordance with the Regulatory Body requirements.

MANAGEMENT OF DATA ON SPENT FUEL STREAMS OF NUCLEAR POWER PLANTS OF THE RUSSIAN FEDERATION

M.F. VOROTYNTSEV, A.I. VOROPAEV, G.M. PSHAKIN,
SSC RF — Institute for Physics and Power Engineering, Obninsk, Russian Federation

A.V. NOVIKOV, V.A.LUPPOV,
ROSENERGOATOM Concern, Moscow, Russian Federation

Abstract

This paper describes the institutional arrangement in Russian Federation for management of information on spent fuel. The institutional arrangement for spent fuel information on several levels includes the federal level, the utility level, and at reactor level. The considerations on creation of «ideal» database on spent nuclear fuel of power reactors in Russian Federation is also discussed.

1. BACKGROUND

The goal and strategy of SNF stream management for the period until 2025–2030 were defined in the Concept of the Ministry of the Russian Federation for Atomic Energy (hereafter referred to as Concept). The Concept outlines the basic trends of scientific and technological policy in this area and the main propositions on the development and implementation of all stages of SNF handling. The Concept is based on postponed decision on reprocessing implying the following solutions:

- construction of centralized dry storage facility on the MCC site for the long term storage of SNF of VVER-1000 reactors followed by reprocessing at the RT-2 plant under construction;
- long term storage of SNF of RBMK-1000 reactors on the reactor sites and in the dry storage facility under construction on the MCC site and making decision on SNF reprocessing or disposal.

The main features of the new system of SNF handling/management are as follows:

- creation of the long term dry storage facilities on MCC site for SNF of VVER-1000 reactors (9000 ton capacity) and that of RBMK-1000 reactors (24000 ton capacity);
- change from wet storage to the long term dry storage in centralized storage facility at MCC, interim (10–15 years) dry storage being arranged on the reactor sites for SNF of channel type reactors;
- transportation of all SNF of VVER-1000, RBMK-1000, EGP-6 and AMB reactors to MCC site and its further long term storage;
- modification of RT-1 plant for the purpose of extension and development of existing technological capabilities of SNF reprocessing as applicable to SNF of VVER-1000 (PWR, BWR) reactors with nuclides partitioning, and continuation of reprocessing of SNF of VVER-440, BN-600 and other reactors;
- reprocessing of SNF of VVER-1000, PWR, BWR and, possibly, RBMK reactors using new technology of chemical reprocessing at the RT-2 plant under construction with partitioning and further conditioning of RW for the purpose of environmentally safe disposal of cured RW in the appropriate geological strata;
- production of MOX fuel at the RT-2 plant;
- creation of infrastructure for SNF storage and transportation meeting up-to-date international requirements.

2. INSTITUTIONAL BACKGROUND OF SPENT FUEL MANAGEMENT DATA FROM REACTORS IN RUSSIA

Nuclear power plants in Russia are included in the nuclear complex of Russia, its management on the federal level being implemented by the Ministry of the Russian Federation for Atomic Energy (The Minatom of Russia).

The whole scope of responsibility for operation and assurance of nuclear and radiation safety of the reactor on all stages (design, construction, operation and preservation) is placed on the utility, i.e. Rosenergoatom Concern. Each one out of ten reactors is now a branch of Rosenergoatom Company.

Federal control of all activities in the area of peaceful use of atomic energy is performed by Federal Supervisory Body on Nuclear and Radiation Safety (Gosatomnadzor of Russia).

SNF refers to special nuclear materials (NM), i.e. those containing fissionable substances or those capable of breeding fissionable substances. By now, NM safety system meeting international standards has been created in the RF, maintained and improved. This includes Federal System of Control and Accounting of Nuclear Materials (NM FSC&A), assuring NM control and accounting at the three levels, namely:

- in material balance areas (MBA) of plants,
- in the organizations implementing NM handling, and
- in the regulatory bodies on the use of atomic energy at the branch and federal levels.

Minatom of Russia designated FSUE CNII Atominform, Moscow as organization responsible for the development and maintenance of the database on federal level of NMFSC&A.

Within the framework of NMFSC&A requirements, Rosenergoatom Concern maintains databases on fresh and spent nuclear fuel for all reactors in Russia.

3. FEDERAL SYSTEM OF CONTROL AND ACCOUNTING OF RADIOACTIVE WASTE AND REACTOR SPENT FUEL

In 1977, Federal System of Control and Accounting of Radioactive Materials (RM) and radioactive waste (RW) was established in Russia by the Government's Decree. The RF Minatom has developed and approved «Regulations on Federal Control and Accounting of Radioactive Materials and Radioactive Waste». In order to assure control and accounting of RM and RW at the federal level,

The Minatom of Russia has established Central Information and Analytical Center (CIAC) FSC&A of RM and RW on the basis of VNIIHT, Moscow, and CIAC Branch on RM Control and Accounting on the basis of Nuclide MKC, St. Petersburg.

SNF to be disposed refers to RW category. Decision on reprocessing or disposal of SNF of uranium-graphite reactors and on the methods of SNF disposal will be made in 2007.

In this view, CIAC activity concerning reactor SNF is currently limited to collection, maintenance and transfer to the higher level authorities data by No. 2-TP form (radioactivity). According to this form, reactor should transfer every year the following information to CIAC:

- SA type;
- number of this type SSAs;
- total mass of this type SNF;
- total α and β activity of this type SNF.

Columns indicating total α and β activity are not filled now, since activity evaluation codes have not yet been adopted at the reactor.

4. DATA ON REACTOR SNF MAINTAINED AT THE DIFFERENT LEVELS IN RUSSIA

4.1. Federal level

At the federal level, CNIIAtominform maintains DB within the framework of NM FSC&A. In this DB, data are maintained at the level of reported batches of accounting units, but not at the level of each accounting unit. The reported batch of accounting units is formed in the organizations using and storing NM [10]. All SSAs with oxide SNF form common accounting batch on the reactor site, irrespective of the type of the reactor where these were irradiated, reached burnup, etc.

For this SSA batch, the following data should be recorded:

- mass of uranium (g);
- uranium mass isotope composition (U-233, U-235, U-238) (g);
- mass of Pu (g);
- plutonium mass isotope composition (Pu-238, Pu-239,Pu-240, Pu-241, Pu-242) (g).

These records are refreshed in this database:

- every year, according to the form of «Master List of NM Inventory in Organization»;
- every quarter, according to the form of «Master Report on Change of NM Inventory in Organization».

4.2. Utility level — Rosenergoatom Company

Since 2001, Rosenergoatom Company has been maintaining DB on nuclear materials in fresh and spent SA, the range of data referring to SNF being much larger as compared to that required by the federal level documents of NM FSC&A.

First of all, DB is maintained at the level of each SA/SSA.

Maintained data are as follows:

- reactor and power unit;
- manufacturer's number of SA;
- SA type (including information on the fuel and type of the reactor);
- initial uranium enrichment;
- dates of the first charging to the reactor, unloading from the reactor to the near-reactor storage pool and back, final unloading from the reactor, on-site transportation, and shipment for reprocessing or to centralized storage;
- data on SSA location on each handling stage;

- final burnup (MW-days/kg);
- data on failures (failed/non-failed SSA);
- mass isotope composition of uranium before and after irradiation (U-235, U-236 and U-238);
- mass isotope composition of plutonium before and after irradiation (Pu-238, Pu-239, Pu-240, Pu-241 and Pu-242).

The Company assures preparation and sending to the Federal Information System of «Master Lists of Available NM in Organization» (every year as of 01.01) and «Master Reports on Changes of Available NM in Organization» (every quarter).

At the end of 2002, software of «Archive of SA types» DB was determined within the framework of Information System of Control and Accounting of Nuclear Materials of Rosenergoatom Concern. Data input is currently performed to this DB. About 30 parameters are indicated in this DB tables:

- SA type identifier;
- type of reactor, in which SA is used;
- fuel type and density;
- fuel enrichment;
- standard dimensions of SA, fuel elements and pellets;
- structural materials and their characteristics;
- fuel mass in the fuel elements and SA;
- mass of fuel subassembly;
- gas filler, its pressure and other parameters.

Besides, complete texts of Specifications for each SA type are added to this DB (total number of pages of the document is ~ 40 plus drawings).

There are the following sources of information:

- TVEL public company — for SA types currently manufactured;
- reactor — for SA types that are not manufactured now.

About 120 SA types have been used in power reactors.

4.3. Reactor level

At the reactor, SNF data control is performed at the level of each SA/SSA, the main pursued objectives being as follows:

- maintenance of set of input data characterizing required SA from the date of SA delivery to the site to the first charge to the reactor,
- maintenance of data obtained as a result of evaluation of SSA characteristics and determining conditions of safe handling and storage of SSAs on site,
- maintenance of data required for sending SSA for reprocessing or to the long term centralized storage,
- maintenance of data required for NM FSC&A.

The following two main documents are SA input data sources:

- Specifications for SA type, and
- Certificate for fresh SA.

In Specifications, there are requirements for manufacture and operation of this type SA. There is a description of SA materials and rated characteristics, permissible values of uncertainties of the main geometrical, physical and operational SA parameters, instructions on operation transportation and storage, etc.

SA transportation from the manufacturer to reactor site is carried out with its certificate.

The following data are specified in fresh SA certificate:

- SA type;
- manufacturer's number of SA;
- enrichment (rated value);
- mass of uranium dioxide (and plutonium dioxide — for MOX fuel);
- mass of U-235;
- mass of plutonium (for MOX fuel);
- mass isotopic composition of uranium (U-235, U-236, U-238);
- mass isotopic composition of plutonium (Pu-238, Pu-239,Pu-240, Pu-241, Pu-242);
- rated value of mass fraction of burnable poison (erbium and gadolinium);
- date of SA shipment from the manufacturer's site;
- container number;
- location in the container.

Just after completion of procedures of SA receipt/transfer at the reactor, special form (accounting card) is filled for SA. This form includes the following sections:

- SA certificate data (type, rated enrichment, mass isotopic composition, etc.);
- data on irradiation history (date of the first charge to the reactor, number of reactor cell, date of completion of the first irradiation cycle, date of the second charge to the reactor, and so on to the final unloading from the reactor);
- date of SSA transportation from near-reactor storage pool to the common on-site storage and its location in the pool storage;
- date of SSA shipment from reactor site for reprocessing or the long term storage;
- average fuel burnup as of the date of final unloading;
- data on failures;
- mass isotopic composition of uranium and plutonium as of the date of final unloading;
- decay heat as of the date of SSA installing into SP.

Besides, at the reactor, data on the history of reactor operation on power are maintained, as well as core load maps, data on location of absorber rods, technological parameters of reactors and SSA storage facilities.

Methods of control and maintenance of the totality of data are different in different reactor sites. On some reactors, technological databases/fuel archives have been created and maintained. Data required for NM FSC&A are maintained in electronic form in all types of reactor. Some technological information required for SNF control under conditions of its long term storage and postponed decision on management methods, for instance, for uranium-graphite reactors, is stored in the paper archives in the form of various logbooks and other documents.

Valid regulatory documents on operation of nuclear power reactors and SNF handling do not imply evaluation of radiological characteristics, such as content, activity and decay heat of fission products, minor actinides and products of activation of structural materials.

4.4. Considerations on creation of «ideal» database on spent nuclear fuel of power reactors

It is indicated in the Concept that it covers all kinds of activity concerning handling SNF of power, propulsion and research nuclear reactors. In particular, this document implies «creation and running of database on SNF and fissile materials». However, activity of this kind is not specified within the framework of the Concept. Neither there is planned date of creation of such DB.

Future DB on SNF in Russia should obviously be a totality of several DBs:

- reactor SNF DB;
- DB of SNF of research reactors;
- DB of SNF of propulsion reactors;
- DB of SNF of nuclear facilities of space application.

Spent fuel database should be developed within the framework of Rosenergoatom Company, since this Concern and its branches maintain all SNF data required for safe reactor operation and safe SNF handling. There should be two levels of reactor SNF DB:

- reactor level;
- Rosenergoatom Concern level.

Apparently, DB of reactor level should not include section on «Detailed environmental characteristics of SNF», since this task is not function of reactor. These characteristics should be evaluated at the Rosenergoatom Company level.

Besides, SSA data are tracked at the reactor level up to the moment of SSA shipment for reprocessing or the long term storage. Tracking of SSA history after this point of time could be implemented in the Concern level DB. There are some organizational problems related to that plants for SNF reprocessing and long term storage are not subordinated to Rosenergoatom Concern. Therefore, a system of control of reactor SNF data capable of tracking SSA life cycle outside reactor sites of Rosenergoatom Concern should be developed and established.

Some problems are expected in connection with SNF delivery to Russia from abroad. Tracking of this SNF data would require change of forms and contents of covering documents.

SPENT FUEL MANAGEMENT IN UKRAINE AND SPENT FUEL DATA TRACKING

L. METELEVA
Nuclear Fuel, Radiation Protection and Radioactive Waste Division,
Nuclear Energy Department,
Ministry of Fuel and Energy of Ukraine,
Kyiv, Ukraine

Abstract

Ukraine has eleven WWER-1000 and two WWER-440 operating reactors at four nuclear plants. These reactors generated almost 45% of Ukraine's electricity. The last of the three RBMK-1000 reactors of Chornobyl NPP was shut down on December 15, 2000. Two WWER-1000 units (one at Khmelnytskyy NPP and another at Rivne NPP) are under construction. According to the Spent Fuel Management Program of Ukraine [1], which was approved in 2000, the state policy in the spent fuel management field is "wait and see". In order to implement this state policy the following problems should be solved: Construction of interim spent fuel storage facilities; Provision of spent fuel transportation from the reactor site to the interim storage facility; Provision of scientific and technical support of the spent fuel management

1. INTRODUCTION

In the former Soviet Union the spent fuel from Ukrainian WWER reactors was stored in the reactor site pools or transported to Russia. The spent fuel of the RBMK-1000 reactors was stored in the reactor site pools and in the wet storage facility at the Chornobyl NPP site.

Although we haven't made a final decision concerning the necessity of Ukrainian spent fuel reprocessing we have been transporting spent fuel from WWER reactors to Russian reprocessing plants. This activity depends on reactor pools filling. We are going to suspend spent fuel shipment as soon as the interim spent fuel storage facilities construction project will have been implemented, namely:

- the dry spent fuel storage facility construction at Zaporizhzhya NPP site,
- the of new modular type spent fuel storage facility (horizontal concrete modules "NUHOMS" designed by Pacific Nuclear, USA and Framatom ATEA) construction at Chornobyl NPP site,
- the construction of centralized spent nuclear fuel facility which will able to contain spent fuel from Khmelnytskyy, South Ukraine and Rivne NPPs.

2. SPENT FUEL ARISING DATA

At present the total amount of spent fuel which has arisen from Ukrainian reactors operation is **6 299** tHM:

- **521** tHM from WWER-440,
- **3 331** tHM from WWER-1000,
- **2 447** tHM from RBMK-1000.

Concerning WWER-440 reactors spent fuel:

- **115** tHM is stored in the reactor pools,
- **406** tHM was shipped to Russian reprocessing plant RT-1 ("MAYAK").

As for WWER-1000 reactors spent fuel:

- **1 458** tHM is stored in the reactor pools,
- **57** tHM is stored in the dry spent fuel storage facility at Zaporizhzhya NPP site,
- **1 814** tHM was shipped and is being stored at Russian reprocessing plant RT-2.

The total amount of spent fuel which were arising during RBMK-1000 unites operation is

- **2 447** tHM., of which:
- **273** tHM is stored at reactor pools,
- **2 174** tHM is stored in the wet spent storage facility at the Chornobyl NPP site.

3. NUCLEAR FUEL DATA TRACKING

There are two utilities in Ukraine. One of them "Energoatom" operates all Ukrainian power plants with WWER reactors, namely:

- Khmelnytskyy NPP with one WWER-1000,
- South Ukraine NPP with three WWER-1000,
- Zaporizhzhya NPP with six WWER-1000 and a dry spent fuel storage facility,
- Rivne NPP with one WWER-1000 and two WWER-440 units.

Every year the “Energoatom" obtains an interim permit for each unit operation. To get the permit the Utility has to develop the Technical Specification which should be approved by the Regulatory Body. In addition to neutronic calculation this specification contains some data concerning the previous loading. Among them: the scheme (or cartogram) of fuel assemblies disposition in the core, the time when the FAs were loaded and unloaded from the reactor core, the coolant properties, the burnup, the leakage test results, the fuel failure data and so on. In addition to this "Energoatom" collects the data concerning nuclear fuel and spent fuel movement, namely:

- the nuclear fuel reserve,
- the spent fuel shipment to the reprocessing plant,
- the spent fuel transportation to the interim spent fuel storage facility,
- the change of reactor pools capacity,
- the transportation of spent fuel from one reactor pool to another.

All this data is brought to the Ministry of Fuel and Energy of Ukraine as often as any change happens.

For data storage and automatic data processing two databases were created. One of them at “Energoatom” and another one at the Ministry. Both databases have some advantages and disadvantages.

4. THE TRACKING OF SPENT FUEL DATA IN INTERIM SPENT FUEL STORAGE FACILITY OF ZAPORIZHZHYA NPP

In 2001 the dry spent fuel storage facility on the basis of VSC-24 container designed by the American company Sierra Nuclear was put into trial operation at Zaporizhzhya NPP site. The full design capacity of this spent fuel storage facility is 380 containers. Each container can be loaded with 24 SFAs[18].

The trial operation provided for the first three containers loading and their storage during one-year period. These three containers, which were loaded in August 2001, contain only 22 SFAs per unit. After successful trial operation, the Regulatory Body gave the permit for loading a container with 24 SFAs. The term of the trial operational license was extended due to the change of loading configuration. So the next three containers, which were loaded in January 2003, contain 24 SFAs per cask.

The Trial Operation Result Report, which proves safety operation of the storage facility within operational and project limits, was given to the Regulatory Body in the end of March. Thus there is only one issue which impede the license receiving. According to the Ukrainian legislation the "Energoatom" company has to produce the proofs of its financial responsibility in case of accident. The corresponding document is on approval of the Ukraine Cabinet of Ministers now. If the document is approved, the license for Zaporizhzhya spent fuel facility operation will be issued.

But even on receiving the license, "Energoatom" has to obtain the permit of the Ukrainian Regulatory Body for each container loading [3]. For that the Utility has to develop the Technical Specification. After radiation and nuclear safety expertise this specification is submitted to the Regulatory Body for approval. The Technical Specification contains the following information:

- The SFA type, the serial number of the SFA and the number of cask cell, in which this assembly will be stored,
- The initial fuel enrichment,
- The scheme of SFAs (along with discharged absorber rods and control rods) disposition in the cask,
- The time when the FA was loaded and unloaded from the reactor core,
- The average burnup,
- The leakage test results,
- The SFA storage time in the reactor pool,
- The residual heat.

The Technical Specification, which is approved by Regulatory Body, is an argument for container loading beginning.

During loading activity the Container Passport, which contains the information about container placement in the storage facility, is filled in [2]. To the Container Passport the Spent Fuel Passport is attached. The Spent Fuel Passport contains information about:

- The serial number of SFA and the number of cask cell, in which this assembly will be stored,
- The SFA drawing number,
- The SFA weight,
- The UO_2 weight,
- The fabrication date,
- The isotopic mass for U^{235},

[18] SFA = spent fuel assembly

- The time when the FA was loaded into the reactor core,
- The time when the SFA was unloaded from the reactor core,
- The SFA storage time in the reactor pool,
- The burnup,
- The estimated mass of Uranium isotopes (including U^{235}, U^{238}),
- The estimated mass of Plutonium isotopes,
- The residual heat.

The Technical Specification, Container Passport and Spent Fuel Passport are kept during storage lifetime. The Ukrainian utility "Energoatom" is responsible for safe keeping of these documents. The data concerning storage facility operation are given to the Regulatory body and to the Ministry regularly.

5. SPENT FUEL STORAGE FACILITY AT CHORNOBYL NPP

The second Ukrainian utility is Chornobyl NPP. This Utility operates Chornobyl NPP, which was shout down, the operating wet spent fuel storage facility and dry spent fuel facility which is under construction now.

The old wet spent storage facility which is being operated at Chornobyl NPP site, will be shut down in the year 2016. Instead of this one the new modular type (horizontal concrete modules “NUHOMS” designed by Pacific Nuclear, USA and Framatom ATEA) spent fuel storage facility will be constructed at Chornobyl NPP site. The NUHOMS modules are being built in two parallel lines of 116 modules. Each module will contain 1 canister with 196 spent fuel bundle cartridges or in other words 98 RBMK SFAs which will be divided into halves before their loading into the canister. The SFAs will be divided into two hot sells which are being installed in the main storage building.

The capacity of the new storage is planned for 21356 SFAs and 2000 discharged absorber rods for 100 years.

The spent fuel data tracking in Chornobyl NPP resembles on the similar activity at “Energoatom” company.

6. CENTRALIZED INTERIM SPENT FUEL STORAGE FACILITY PROJECT

Due to spent fuel storage facilities construction on Zaporizhzhya NPP and Chornobyl NPP sites the problem of spent fuel management at these two plants will be solved. As for spent fuel management at South Ukraine, Rivne and Khmelnytskyy NPPs, this issue will be discussed.

On the basis of the preliminary assessment performed by Kyiv “Enegoproject” Institute the following conclusions were suggested:

- the preferable spent nuclear strategy is centralized storage facility,
- the preferable technology for this facility is modular or container storage,
- the preferable site location is the area near Chornobyl NPP site.

Thus in the year 2002 the Ministry of Fuel and Energy of Ukraine and the state utility “Energoatom” have begun the work concerning centralized interim spent fuel storage facility construction.

According to Ukrainian legislation the technical and economic project assessment have to be performed and three sites for facility location have to be suggested. After that, the documents, approved by Ministry, have to pass through the state and juridical expertise and to be submitted to the Cabinet of Ministers. The Ukrainian Cabinet of Ministers or the Ukrainian Parliament makes the final decision concerning the storage facility construction possibility.

Now we are only in the first stage. Kyiv "Enegoproject" Institute is performing the technical and economic project assessment. The investigation of three possible sites is completed and we have to choose the storage technology. For this the international tender will be held. Now the tender documents are almost developed and we are going to select the storage technology by the end of October. In this case it's possible that the decision concerning storage facility construction possibility will have been made at the beginning of the next year.

The centralized interim spent fuel storage facility is the first nuclear facility in Ukraine which is planned and will be constructed according to the Ukrainian nuclear legislation. So we have an opportunity to check its efficiency.

Moreover it's possible that a new utility, which will be responsible for spent fuel storage, will be created. In this case we would face the situation when one utility would buy and use nuclear fuel and than hand it over to another utility, which would store it. So we will have to review our data tracking system and may introduce some changes into it.

REFERENCES

[1] Program "Spent Fuel Management in Ukraine" Kyiv, 1999.
[2] Branch standard "Spent fuel storage facility. Container for storage of the spent fuel from WWER-1000. The loading requirements, the transportation requirements, operating requirements." Kyiv, 2002.
[3] Branch standard "Dry spent fuel storage facility for the spent fuel from WWER-1000. permit obtaining procedure. Requirements to the neutronic calculation" Kyiv, 2002.

UNITED KINGDOM COUNTRY REPORT

A.J. SCULLION
UKAEA

Abstract

This paper gives a summary on spent fuel characteristics and management status in the UK with an introduction on institutional arrangement.

1. SOURCES OF SPENT FUEL

1.1. The primary sources of spent fuel in the United Kingdom (UK) relate to the operation of commercial power stations, listed in Table 1.

1.2. The Advanced Gas-cooled Reactor (AGR) stations were designed with limited spent fuel storage capacities and their continued operation is dependant on dispatching spent fuel to an alternative management facility for reprocessing or storage pending final disposal. British Energy (BE) has contracts with British Nuclear Fuels Limited (BNFL) for the management of AGR spent fuel, and some contracts allow for the reprocessing or storage of spent fuel at BNFL's discretion.

1.3. The Pressurised Water Reactor (PWR) at Sizewell B has substantial capacity for the storage of irradiated fuel and no decision needs to be taken at present on whether to reprocess or directly dispose of this fuel in the long term.

1.4. All of BNFL's spent fuel arises from the Magnox programme and ultimately needs to be treated in some way to prevent corrosion. At present the preferred route is to reprocess the fuel.

1.5. The United Kingdom Atomic Energy Authority (UKAEA) has been involved in the research and development (R&D) of civil nuclear power for nearly fifty years. The R&D included reactor systems described as zero energy facilities, Material Test Reactors (MTR), Steam Generating Heavy Water Reactors (SGHWR) and Fast Breeder Reactors (FBR).

1.6. Educational and medical institutes produce additional smaller quantities of irradiated nuclear materials.

2. OPTIONS FOR SPENT FUEL

2.1. Currently, the decision to reprocess or hold spent fuel in long term storage pending direct disposal is a matter for the commercial judgement of its owners.

2.2. In September 2001 the UK Government initiated a consultation process aimed to develop, and implement, a UK nuclear waste management programme which inspires public support and confidence.

2.3. Plutonium and spent fuel are not currently classed as waste. However, if at some point it were decided that there was no further use for some or all of these materials, consideration would be required of how to handle them as part of a waste management strategy.

TABLE 1. UK NUCLEAR POWER GENERATING STATIONS

Operator	Site	Type
British Energy	Dungeness B	1254MW AGR
	Hartlepool	1332MW AGR
	Hinkley Point B	1230MW AGR
	Heysham 1	1320MW AGR
	Heysham 2	1344MW AGR
	Hunterston B	1320MW AGR
	Sizewell B	1258MW PWR
	Torness	1364MW AGR
British Nuclear Fuels Limited	Hunterston	500MW Magnox
	Oldbury	434MW Magnox
	Berkley	276MW Magnox
	Chapelcross	168MW Magnox
	Trawsfynydd	470MW Magnox
	Bradwell	246MW Magnox
	Dungeness A	450MW Magnox
	Wylfa	980MW Magnox
	Calder Hall	168MW Magnox
	Hinkley Point A	470MW Magnox
	Sizewell A	420MW Magnox

2.4. Reprocessing facilities in the UK are sited at Sellafield (operated by BNFL):

(1) The design capacity of the Magnox Fuel Reprocessing Plant, located at Sellafield is 1500 tHM/y.
(2) A second facility at Sellafield, the Thermal Oxide Reprocessing Plant (THORP), has a designed capacity of 850 tHM/y in the form of AGR, BWR and PWR type fuels.
(3) The Material Test Reactor Reprocessing Plant at Dounreay was closed in 1997, after almost 40 years of operation, during which time over 12,000 fuel elements were reprocessed.
(4) The Fast Reactor Fuel Reprocessing Plant at Dounreay successfully reprocessed approximately 10 tHM of DFR fuel (niobium clad uranium/molybdenum alloy fuel) before undergoing refurbishment to reprocess PFR mixed oxide fuel. Approximately 30 tHM of unirradiated and irradiated core and breeder fuel was reprocessed up to 1996, when the main dissolver developed an irreparable failure. In July 2001 the UK Government announced that no further reprocessing of fast reactor fuel would be undertaken at Dounreay.

2.5. The Nuclear Industry Radioactive Waste Executive (Nirex) was set up in the early 1980s by the nuclear industry, with the agreement of the Government, to examine safety, environmental and economic aspects of deep geological disposal of radioactive waste. Nirex deals with intermediate level waste (ILW) which accounts for the majority of radioactive waste currently in storage and with some low level waste.

2.6. Nirex give advice to the producers of radioactive waste on how waste should be packaged so that it would be suitable for eventual deep disposal, should that be the preferred option. They also set standards for radioactive waste containers and their contents for the guidance of waste producers.

2.7. Advice from the Department of Trade and Industry, in discussion with representatives from the Department of the Environment, Transport and the Regions, has indicated that:

(5) Used fuel which is destined for reprocessing or for which the owner has not yet made any decision on its future should be classed as Spent Fuel and not as waste.

(6) Used fuel which the owner has decided not to reprocess, and for which no further use is foreseen, should be regarded as radioactive waste.

(7) The categorisation of used fuel which, according to the above, has been defined as a waste should be determined according to its radioactivity level and heat generation using the UK Government definitions of Radioactive Waste given in the White Paper Cm. 2919.

2.8. Nirex has prepared a Waste Package Specifications and Guidance Documentation (WPSGD), to assist waste producers to plan effectively for the packaging of radioactive wastes. The intention is to maximise the likelihood that wastes prepared in accordance with the guidance will meet the repository Waste Acceptance Criteria when they are published. The WPSGD does not address the issues of the disposal of fuel residues directly.

3. MANAGEMENT OF SPENT FUEL

3.1. In the United Kingdom Magnox, AGR and LWR spent fuel is traditionally kept in pond storage at the reactor site until transported for reprocessing. Both the Magnox Fuel Reprocessing Plant and the Thermal Oxide Reprocessing Plant (THORP) have pond front-end receipt and storage facilities. The one exception is the Wylfa power station, which is equipped with a dry spent fuel storage facility.

3.2. BNFL have found a quantity of long-stored fuel that has suffered varying degrees of deterioration due to fuel/cladding/water reaction. In such cases the fuel is conditioned by partial oxidation before treatment.

3.3. The majority of spent fuel under the direct management of the UKAEA is located on the Dounreay Site. This inventory includes spent fuel from the Material Test Reactor (MTR), the Dounreay Fast Reactor (DFR) and the Prototype Fast Reactor (PFR) programmes, together smaller quantities of legacy fuels associated from former commercial reprocessing (see Table 2). As much of the Dounreay fuel was experimental, there is a considerable variation in fuel design and burnup. In way of example, Table 3 provides data for PFR fuel. All the spent fuel is held in interim storage, in a number of facilities on the Dounreay Site, pending treatment, disposal or storage elsewhere.

TABLE 2. DOUNREAY IRRADIATED FUEL

Miscellaneous types	Highly enriched uranium metal and alloy	Elements
		Medical target residues
		Other
	Thorium	Metal
		Oxide
Oxide with plutonium	Core fuel	Assemblies
		Experimental clusters
		Pins (intact & cut)
	Breeder	Radial assemblies
		Mixer clusters
	Miscellaneous	With HEU
		Pins
		Containers
		PIE remnants
Depleted Uranium metal		In reactor
		Dry stored
Carbide	Natural/ Depleted uranium	Experimental clusters
		Radial assemblies
		Mixer clusters
	Enriched uranium	Containers

TABLE 3. VARIATIONS IN PFR FUEL DESIGN

Assembly type	core, breeder, cluster
Fuel type	oxide, carbide
Fuel form	pellet, vibro-compacted, gel
Number of pins	325, 271, 265, 169, 127
Pin design	grid held, end held, wire wrapped
Pin cladding	M316, PE16, HL548, FV548
Grid material	PE16, 321S12, 9C1Y, FV448, 12M2P, 7F1Y, M316, 4M10P
Wrapper material	PE16, EN58B, FV448, FV607, FV548, 12R72, HL548
Irradiation	Core fuel — 0.38 to 19.57% burnup Breeder fuel — 0.07 to 2.75% burnup

3.4. Similar to the commercial power stations, spent fuel discharged from the Dounreay reactors was held in pond storage. Unlike the commercial power stations, much of the Dounreay fast reactor spent fuel was over-canned (air environment).

3.5. Processes have been identified that could be applied to each nuclear fuel item to permit its removal from the Dounreay Site either as a useful product (for unirradiated fuel), in an intermediate form suitable for subsequent treatment by a third party or as a conditioned waste ready for disposal. Preferred processes have to be of mature technology and consistent with the UKAEA mission to decommission its nuclear facilities and restore the environment of its sites in a way which is:

(1) Safe and secure,
(2) Environmentally responsible,
(3) Value for money,
(4) Publicly acceptable.

3.6. If not reprocessed, the spent fuel will be stored or (after interim storage) conditioned for disposal. An assessment of information requirements by UKAEA for spent fuel handling, transportation, reprocessing and storage/disposal is given in Table 4. In the absence of a reprocessing solution the UKAEA have initiated a study to consider long term storage options.

3.7. The UKAEA at Dounreay has identified a need for a facility to package irradiated nuclear materials to render them suitable for transportation, subsequent treatment or long term storage. The preliminary design principles for such a facility are:

(1) Have the capability of receiving irradiated material, potentially in a variety of physical forms.
(2) Have the capability to safely handle a variety of fuel types.
(3) Have the capability to provide safe storage and be designed to minimise deterioration of the fuel condition and monitor instances of primary containment rupture.
(4) Give consideration to appropriate standards of packaging and inspection and be compatible with all possible final treatment options
(5) Allow for the safe and hygienic handling of materials and permit effective and efficient nuclear material control.
(6) Be designed to allow effective and efficient maintenance and decommissioning.
(7) Be designed to minimise cross contamination of uranium with plutonium.
(8) Be designed and built to appropriate standards.
(9) Be designed to minimise environmental discharges and to minimise the radiation and chemical hazard to workers.
(10) Be sited so as to minimise the number of nuclear material movements required outside of the engineered containment and in a location commensurate with the Dounreay Site Restoration Plan (DSRP).

3.8. The facility will allow irradiated fuel to be received from its current location, in shielded containers, and then be characterised as necessary and packaged to a form suitable for receipt by reprocessing facilities, or for long term storage. The facility will also be equipped to permit routine checks of the material in store and be equipped to allow some remedial repackaging. The repackaging stage will include equipment to provide quality assured information on the fuel condition to meet the requirements for reprocessing or long term storage. The facility may include a storage area.

TABLE 4. PROPOSED INFORMATION REQUIREMENTS

Field	Data	Identification of item & handling	Transportation	Safety case — treatment	Safety case — disposal
Identifier	Name/Barcode	3	3	3	3
Description	Fuel type	3		3	3
	Design reference	3		3	3
	Drawings	3	3		
Composition	Plutonium isotopic composition Pu238, Pu239, Pu240, Pu241, Pu242, total plutonium	3	3	3	3
	Uranium isotopic composition U233, U235, U238, total uranium	3	3	3	3
	Total thorium	3	3	3	3
	Americium isotopic composition Am241, total americium	3	3	3	3
	Neptunium	3	3	3	3
	Date of calculation	3	3	3	3
Weight	Gross	3	3		3
	Net	3	3		3
Physical form & dimensions	State — Solid, Liquid, Gas	3	3	3	3
	Form — Powder, pellets, residue, pin, assembly, etc.	3	3	3	3
	Containment type				
	Length Diameter	3	3	3	3
	Sub-items		3	3	3
Chemical form	Oxide, Metal, Carbide, Nitride, Alloy, Nitrate, Other	3	3	3	3
Cladding	Material		3	3	3
	Weight	3	3	3	3

Field	Data	Identification of item & handling	Transportation	Safety case — treatment	Safety case — disposal
Irradiation history	Irradiated/Unirradiated	3	3	3	3
	Reactor		3	3	3
	Burnup		3	3	3
	Date of discharge		3	3	3
	Fission products	3	3	3	3
	Cooling time	3	3	3	3
	Date of calculation		3	3	3
Additives/Major impurities	Aluminium, Gadolinium, Molybdenum, Silicon, Sodium, etc	3	3	3	3
	Moderator	3	3	3	3
Location	Site	3	3		
	Building	3	3		
Treatment selection	Primary option		3	3	3
	Secondary option		3	3	3
Other	Condition — oxidised, hydrided, etc.	3	3	3	3
	Special handling requirements – inert atmosphere	3	3	3	3
	Solubility	3	3	3	3

3.9. As shown schematically in Figure 1, the facility will contain a number of inter-linked units. The reference option for fuel storage is for a cask-based approach. However, because of the diverse nature of the Dounreay fuels the option of vault storage will remain a possibility for the time being. The unit operations that may be incorporated are:

(1) Fuel receipt and examination,
(2) Treatment to meet storage requirements,
(3) Assay, over-packaging and testing,
(4) Cask loading and sealing,
(5) Dispatch to Cask store,
(6) Routine return of Casks for QA checks,
(7) Unloading and repack capability.

3.10. Details of the facility will be determined through the application of engineering design principles that meet the user specification.

3.11. The routine checking of casks and their contents will require equipment to be available until the fuel is transferred off-Site. The cask store may therefore exist on site for the long term.

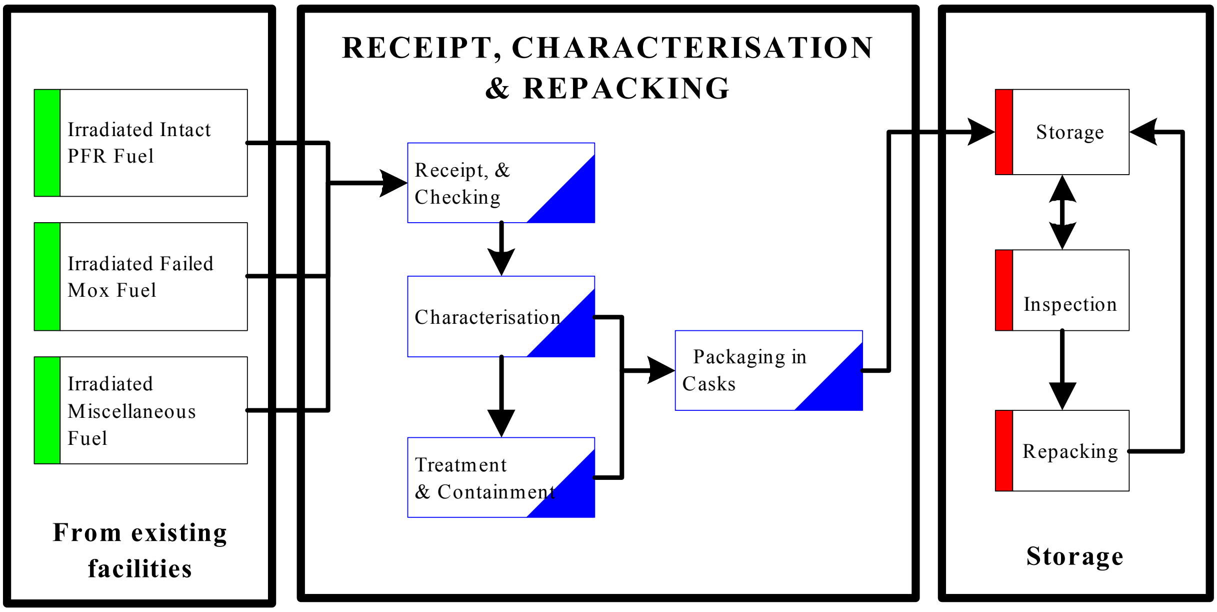


FIG.1. Schematic diagram of material flow.

GLOSSARY

AGR	Advanced Gas-cooled Reactor
BE	British Energy
BNFL	British Nuclear Fuels Limited
BPEO	Best Practical Environmental Option
BPM	Best Practical Means
Bq	Becquerel
DFR	Dounreay Fast Reactor
DMTR	Dounreay Material Test Reactor
DSRP	Dounreay Site Restoration Plan
DTI	Department of Trade & Industry
FBR	Fast Breeder Reactor
G	Giga
HEU	Highly Enriched Uranium
HLW	High Level Waste
HM	Heavy Metal
ILW	Intermediate Level Waste
LLW	Low Level Waste
MTR	Material Test Reactor
Nirex	Nuclear Industry Radioactive waste Executive
PIE	Post Irradiation Examination
PFR	Prototype Fast Reactor
SGHWR	Steam Generating Heavy Water Reactor
t	Tonne
THORP	Thermal Oxide Reprocessing Plant
UK	United Kingdom
UKAEA	United Kingdom Atomic Energy Authority
y	Year

UNITED STATES OF AMERICA COUNTRY REPORT

J. FINUCANE, D. NIKKODEM
USDOE/EIA,
Washington, DC,
USA

Abstract

The USDOE's Energy Information Administration (EIA) the responsibility for collecting data on spent fuel discharges for the purposes of planning, fee calculation, and input to the design of Federal Waste Management System which includes as a major scope the permanent disposal of spent nuclear fuel. These data are collected annually on Form RW-859 Nuclear Fuel Data. The data collected by the EIA include spent fuel discharge data, cycle data, quantities and types of spent fuel discharges, spent fuel storage locations and inventories, and a record of transportation of spent fuel. This paper gives an overview of the spent fuel data collection form RW-859 which is intended to provide necessary information to the U.S. Department of Energy (DOE) and its Office of Civilian Radioactive Waste Management (OCRWM).

BACKGROUND

The operation of commercial power reactors in the United States is the responsibility of the individual reactor owners and is regulated by the U.S. Nuclear Regulatory Commission. The U.S. Congress has given responsibility for the design, licensing, and operation of a Federal Waste Management System (FWMS) for the permanent disposal of spent nuclear fuel to the U.S. Department of Energy (DOE) and its Office of Civilian Radioactive Waste Management (OCRWM). The geological repository will be licensed and regulated by the U.S. Nuclear Regulatory Commission.

The Department of Energy collects fees from utilities, based upon the net electricity generated by nuclear power plants, to pay for the costs of the Federal Waste Management System. OCRWM has given to DOE's Energy Information Administration (EIA) the responsibility for collecting data on spent fuel discharges for the purposes of planning, fee calculation, and input to FWMS design. These data are collected annually on Form RW-859 Nuclear Fuel Data. The data collected by the EIA include spent fuel discharge data, cycle data, quantities and types of spent fuel discharges, spent fuel storage locations and inventories, and a record of transportation of spent fuel.

In addition to the data collected by the EIA on Form RW-859, the OCRWM has sponsored the development of the Characteristics Database System (CDB) at the Oak Ridge National Laboratory. The CDB combines detailed physical descriptions of the fuel assemblies used in the United States, the quantitative information supplied by the EIA Form RW-859 (see ATTACHMENT at the end of this report), and the radiological characteristics of spent fuel (as calculated using the computer code ORIGEN2).

The Characteristics Database System

The Characteristics Database System (CDB) consists of six PC-based modules, which describe the characteristics of spent fuel discharges in the United States. The six PC modules that make up the CDB are:

- LWR Fuel Assemblies Database
- LWR Radiological Database
- LWR NFA-Hardware Database
- LWR Quantities Database
- LWR Serial Numbers Database
- High-level Waste Database

A brief description of each of these modules is given below.

The **LWR Fuel Assemblies Database** (FADB) contains physical descriptions of more than 120 specific versions of fuel assemblies used in the United States (these are often referred to as "assembly types"). The physical descriptions include fuel assembly dimensions, materials, and characteristics and descriptions of fuel assembly hardware. The FADB also contains summary radiological descriptions of individual fuel rods and radiological descriptions of the activation products in disassembly hardware (top and bottom nozzles, grid spacers, instrument tubes, etc).

The **LWR Quantities Database** (QTYDB) contains information on the quantities and characteristics of spent fuel that has been discharged from commercial power reactors in the United States. Final discharge burnup, final discharge date, fuel assembly type, initial weight of heavy metal, initial enrichment, and current storage location are among the data parameters which are included in this database. Reports are also capable of distinguishing both defective and consolidated fuel assemblies.

The **LWR Radiological Database** (RADDB) contains information on the radiological characteristics of spent fuel based on user-specified reactor type, final discharge burnup, initial enrichment, and decay time since discharge. Radiological characteristics that are available include gram quantities, activities, heat outputs, total neutron generation rates, and photon spectra. Gram quantities, activities, and heat outputs are available as totals and on an isotope-by-isotope basis.

The **LWR Serial Numbers Database** (SNDB) combines the quantitative data in the LWR Quantities Database with summary radiological information from the LWR Radiological Database to enable the user to estimate the radiological characteristics of an individual assembly, based on the serial number of the assembly. The SNDB considers the reactor type, initial enrichment, initial weight of heavy metal, final discharge burnup, final discharge date, and calculates estimates of total activity, heat output, and neutron and photon generation.

The **LWR NFA Hardware Database** (NFADB) describes the physical characteristics of the non-fuel assembly (NFA) hardware associated with LWRs in the United States. These NFA hardware components include control-rod assemblies, burnable absorber assemblies, neutron sources, fuel channels, incore instrumentation, and other hardware..

The **High-level Waste Database** (HLWDB) describes the interim and planned immobilized wastes forms of high-level wastes resulting from defense reprocessing activities at the Savannah River Plant, the Idaho National Engineering Laboratory, and the Hanford Reservation, and from commercial reprocessing at West Valley, NY.

A unique feature of the system used in the United States is the methodology use to provide physical descriptions of the fuel assembly types. Because many varieties of reactors and spent fuels already exist, the need for a classification system to collect spent fuel physical descriptions in a systematic and organized manner is a management strategy.

The DOE Approach to the Verification of Utility-Delivered spent fuel is as follows:_DOE needs specific data on each delivered spent fuel assembly, in order to:

(a) Plan and execute handling, transport, storage, and disposal in compliance with license requirements, and

(b) For each Special Nuclear Material (SNM) waste item, demonstrate and document complete item control and the continuity of knowledge from item origin (typically fuel assembly fabrication) to disposal, in compliance with NRC Material Control and Accountability (MC&A) requirements, and, if applicable, in compliance with the U.S. agreement to implement IAEA safeguards.

DOE plans no at-reactor inspections relative to waste acceptance. Casks certified by NRC under 10 CFR 71 will be loaded by reactor operators under their individual 10 CFR 50 licenses. They will classify spent fuel in accordance with the standard contract Appendix E, prepare all other paperwork required for MC&A and the standard contract under their 10 CFR 50 QA programme — and DOE plans to accept that.

Should cask loading require burnup verification for burnup credit, the reactor operator would make necessary measurements. In short, DOE does not plan to interject itself into reactor operations at all. "Inspections" will consist of confirmation that all paperwork is in order at the time of title transfer, and this would probably be done using contractors acting as agents of the government.

ATTACHMENT: Nuclear Fuel Data Survey Form RW-859

The Nuclear Fuel Data Survey is the primary tool used by the Department of Energy (DOE) Office of Civilian Radioactive Waste Management (OCRWM) for the collection, from owners and generators of commercial spent nuclear fuel, of data necessary to carry out the requirements of the Nuclear Waste Policy Act of 1982, (P.L. 97-425) (NWPA).

This form is a mandatory data collection and is a major data link between the Federal Government and the utilities. The form is the primary vehicle through which the utilities communicate information regarding their specific fuel assemblies, storage capacities, site-related data and near-term fuel loading plans to the Federal Government. It provides the basis for several OCRWM programme activities in planning and implementing the safe and efficient handling and disposal of spent nuclear fuel and in supporting the accuracy of certain fees paid into the Nuclear Waste Fund.

The data (collected approximately every three years) are a "snapshot in time" and reflect the inventories of the spent nuclear fuel as of December 31 of the previous year.

1. USES OF DATA FROM THE NUCLEAR FUEL DATA FORM

The data provided by this form are used in a variety of ways by OCRWM. The data will be used for such purposes as the design of the repository and the Monitored Retrievable storage (MRS) facilities, equipment, and emplacement strategies. The transportation programme requires the data as input to the design of casks, and to make the most efficient use of available casks. The data are also used to allocate annual waste acceptance capacity to individual utilities as noted in the *Annual Capacity Report* and the *Acceptance Priority Ranking*. It will also be used extensively in implementing the waste acceptance procedures described in the Standard Disposal Contract. The data are also used, in part, for fee verification purposes. Both the Legislative branch and elements of the Executive branch of the government regularly request information on the amount of spent fuel in each state or district and this data is easily retrieved using the EIA's Nuclear Data Information System. The Oak Ridge National Laboratory (ORNL) also produced the report, *Integrated Data Base for* 1991: *U.S. Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics.*

Regulatory agencies, intervenors, businesses and private citizens also frequently make requests for the data. The data appear in a variety of publications.

The data collected on this form also constitute one of the inputs to the spent fuel Characteristics Data Base (COB) maintained by the ORNL. The discharge dates, assembly types, burnups, and initial enrichments are used by ORNL to calculate the gamma, neutron, and thermal source intensities. These radiological characteristics, along with reported fuel quantities and dimensions, are then used by Federal Waste Management System designers for shielding design, thermal design, and sizing of facilities and equipment.

Trends based on historical spent nuclear fuel data provided by the respondents are used by the DOE to estimate future discharges from U.S. commercial nuclear reactors and the characteristics of those discharges. The need for additional spent fuel storage capacity is based on these estimated cumulative discharges, and on the estimated maximum storage capacity of both at-reactor and away-from-reactor storage facilities.

2. FUTURE CHANGES

The Form is an evolutionary document, changing as circumstances, understanding and requirements change. It is typically changed every three years with a focus on changing data requirements and minimizing reporting burden of respondents.

3. DESCRIPTION OF THE NUCLEAR FUEL DATA

The 'The Nuclear Fuel Data' survey collects data on every fuel assembly irradiated in commercial nuclear reactors operating in the United States. The data that is collected is organized into tables which are listed below. In the material that follows, table names begin with the letters tbl and they are printed in *italics* and underlined.

Survey Form Cover Page

tblRespondent Respondents reported reactor name and survey years (for example: 1999 through 2002) and indicated if the survey submittal was a resubmission. The data elements in this table are:

RE_REACTORID — The reactor ID number. The reactor name appears in *tblFacility;*

RE_BEGINRPTDT — The beginning of the reporting period, (for example: January 1, 1999);

RE_ENDRPTDT — The end of the reporting period, (for example: December 31, 2002);

RE_RESUBMISSION — A 1 indicates a resubmission; a 0 indicates an original survey submission.

SECTION ON FACILITY DATA

tblFacility Respondents reported utility and reactor identification and contact information. The data elements in this table are:

FA_REACTORID — The reactor ID number;

FA_REACNAME — The reactor name;

FA_UTILNAME — The utility or operating company name;

FA_LICXDATE — Reactor operating license expiration date for operating reactors;

FA_POLICXDATE — possession-only license expiration date for permanently shutdown reactors;

FA_CONTADDR — The utility or operating company street address;

FA_CONTCITY — The utility or operating company city;

FA_CONTSTATE — The utility or operating company state;

FA_CONTZIP — The utility or operating company zip code.

Section on Reactor Cycle Data

tblCycles Respondents reported cycle startup and cycle shutdown dates for all discharge cycles since the previous Form RW-859 survey. Respondents were provided with all previously reported dates and asked to verify that these dates were correct. The file *tblCycles* contains all historical cycle startup and cycle shutdown dates for each reactor. The data elements in this table are:

CY_REACTORID — The reactor ID number. Reactor name appears in *tblFacility;*

CY_CYCLENUM — The cycle number;

CY_CYUPDATE — The cycle start date;

CY_CYDNDATE — The cycle shutdown (subcriticality) date;

Even though respondents were only required to report cycle dates for the latest period, they were supplied with and asked to verify all previous cycle dates. Any discrepancies from previously reported dates were verified with the respondents.

Section on Permanently Discharged Fuel

tblFuelFresh Respondents reported assembly characteristics (burnup, initial loading weight, enrichment, discharge (subcriticality) date, etc.) for all discharged assemblies. Respondents were also asked to identify any fuel assembly classified as failed or nonstandard. Respondents also indicated the storage pool or dry storage facility where each assembly was stored. All assembly data submitted were compared to previously submitted data. Changes to previously submitted data were verified. The data elements in this table are:

FF_REACTORID — The reactor ID number. Reactor name appears in *tblFacility;*

FF_CYCLENUM – This field is from previous surveys. [Data for the current survey are entered in CY_CYCLENUM and copied here];

FF_ASSMID — The individual ID for each assembly;

FF_INITIALU — The initial loading weight for each assembly (in kilograms of uranium);

FF_INITENR — The initial enrichment for each assembly (in weight percent);

FF_MAXBURN — The final burnup for each assembly (in megawatt days thermal per metric ton of uranium);

FF_ASSMTYPE — The assembly type for each assembly. [A table with all assembly types is periodically updated and this table is provided to respondents];

FF_ASSMCODE — The status code for each assembly. A table of various 'statuses' is provided to respondents with the survey form. A blank in this field represents standard fuel, while a code represents failed or nonstandard fuel;

FF_POOLID – The pool storage ID or dry storage ID number where the assembly is stored;

*CY_CYCLENUM – The cycle number. Cycle numbers and dates were compared to cycle numbers and dates reported in the section on cycle data (*tblCycles)*;

CY_CYDNDATE – Cycle shutdown (subcriticality) date;

Section on Pool Storage

tblStoreWet Respondents reported licensed and current installed pool capacity. All changes to capacities from those reported on the previous surveys were identified and verified. Any differences between licensed and current installed capacity were also verified. Comments were entered in the comment section as needed to clarify differences. The data elements in this table are:

SW_REACTORID — The reactor ID number;

SW_POOLID — The storage pool ID where the discharged assemblies are stored;

SW_CAPNBWR — Licensed storage capacity in number of BWR assemblies;

SW_CAPNPWR — Licensed storage capacity in number of PWR assemblies;

SW_CAPIBWR — Installed storage capacity in number of BWR assemblies;

SW_CAPIPWR — Installed storage capacity in number of PWR assemblies.

tblStoreWet_Inventory Respondents indicate the number of assemblies stored in each storage pool. The data elements in this table are:

SI_REACTORID — The reactor ID number;

SI_POOLID — The storage pool ID where the discharged assemblies are stored.

SI_NUMASSM – The number of assemblies stored in each pool.

Section on Reinserted Fuel

tblStoreWet_Reinsert Information in this table is incomplete because many respondents reported only reinserted assemblies for the last cycle. The table is used for internal quality assurance purposes only. The data elements in this table are:

SR_REACTOR ID — The reactor ID number;

SR_ASSMID — Individual assembly ID number.

SR_NoReinserted – Number of reinserted assemblies.

Section on Shipments of Fuel

tblStoreWet_FuelShip Information in this table is incomplete because many respondents reported only shipments and transfers for the last cycle. The table is used for internal quality assurance purposes only. The data elements in this table are:

SF_REACTORID — The reactor ID number;

SF_ASSMID — Individual assembly ID number;

SF_OLDPOOLID — The pool from where the assembly was shipped;

SF_NEWPOOLID — The pool or dry storage site to which the assembly was shipped.

Sections on Canisters and Nonfuel Components

Respondents reported canister identification information, the contents of canisters, and noncanistered nonfuel components in these sections. Though most respondents reported canister identification correctly, some respondents did not report complete information for the contents of these canisters. Therefore, data in these sections may be incomplete.

tblCanister This table contains information on canister description, size, general contents, nonfuel components in canisters, and canister closure. The data elements in this table are:

CA_REACID — The reactor ID number;

CA_CANID — The individual canister identifiers;

CA_CANinPool — This column should be disregarded. Only reactor sites with canisters are included in the file;

CA_CANShape — Canister cross sectional shape, (either rectangular or cylindrical);

CA_CANLength — The canister length (in inches);

CA_CANCylinDiameter — The canister diameter (in inches);

CA_CANWidth — The canister width (in inches);

CA_CANDepth — The canister depth (in inches);

CA_CANWeight — The canister's estimated weight (in pounds) when loaded;

CA_CANOther — A description of the canister's features;

CA_DateClose — The date the canister was closed in MMDDYYYY format;

CA_CANSealed — Indicates whether or not a canister is sealed. A 1 indicates the canister is sealed, a 0 indicates it is not sealed;

CA_CANSealedMethod — A description of the canister sealing method;

CA_Bolted – A checkmark indicates whether or not a canister is bolted;

CA_Welded – A checkmark indicates whether or not a canister is welded.

The following five fields correspond to the canister's contents. A 1 indicates the canister contains the item shown, a 0 indicates the canister does not contain the item shown.

CA_IntactFailed — An intact failed fuel assembly;

CA_IntactReconstituted — A reconstituted assembly;

CA_IntactFuelRods — Intact fuel rods;

CA_FuelDebris — Fuel rods or pieces;

CA_SpentFuelDisassembly — Spent fuel disassembly hardware.

The remaining 11 fields indicate the number of various spent fuel disassembly hardware items or nonfuel components or the volume in cubic inches of these items:

CA_PWRControlRods — The number or volume of PWR control rods;

CA_PWRControlRodsSpiders — The number or volume of PWR control rod spiders;

CA_ControlCruciIntact — The number or volume of intact control cruciforms;

CA_ControlCruciBlades — The number or volume of control cruciform blades;

CA_ControlCruciBases — The number or volume of control cruciform bases;

CA_BurnablePoisonAssm — The number or volume of burnable poison assemblies;

CA_NeutronSources — The number or volume of neutron sources;

CA_ThimblePlugs — The number or volume of thimble plugs;

CA_INCoreInstrumention — The number or volume of incore instrumentation elements;

CA_BWRFuelChannels — The number or volume of BWR fuel channels;

CA_Other — The number or volume of other items and a listing of these items.

tblCanister_SpentFuel This table contains specific canister contents information for canisters containing fuel assemblies, consolidated assemblies, fuel rods, fuel pieces, or fuel pellets. The data elements in this table are:

CS_REACID — The reactor ID number;

CS_CANID — The individual canister identifiers;

CS_ASSMID — The ID of the assembly from which the fuel rods or pieces were removed;

CS_NumberFuelRod — The number of fuel rods from the assembly;

CS_ INITIALU — The initial loading weight of the fuel rods;

CS_MAXBURN — The maximum burnup for the assembly from which the fuel rods were removed.

tblCanister_NonFuel This table contains general information on uncanistered fuel and nonfuel components. The data elements in this table are:

CN_REACID — The reactor ID number;

CN_PoolID — The storage pool where the uncanistered fuel rods and uncanistered nonfuel components are stored;

CN_CANSpecial — An indication of whether or not the pool contains noncanistered fuel or nonfuel components requiring special handling. A 1 indicates yes and a 0 indicates no.

The following fields contain an indication of whether or not the specific items listed are in the pool. A 1 indicates the presence of the item shown, a 0 indicates the item is not present

CN_FailedFuel — A noncanistered failed fuel assembly;

CN_Consolidated — A noncanistered reconstituted assembly;

CN_IntactFuelRods — Noncanistered intact fuel rods;

CN_FuelDebris — Noncanistered fuel rods or pieces;

CN_SeparatedAssm — Noncanistered spent fuel disassembly hardware or nonfuel components.

The following fields contain the number or volume of each of the specific items listed:

CN_NoFailedFuel — The number of noncanistered failed fuel assemblies;

CN_NoConsolidatedAssm — The number of noncanistered reconstituted assemblies;

CN_NoIntactFuelRods — The number of noncanistered intact fuel rods;

CN_NoFuelDebris — The number of noncanistered fuel rods or pieces;

CN_NoFuelDebrisVol — The volume in cubic inches of noncanistered fuel rods or pieces;

CN_NoFuelDebrisMass — The mass in initial kilograms of uranium of noncanistered fuel rods or pieces;

CN_NoSeparatedAssm — The number of noncanistered spent fuel disassembly hardware items or nonfuel components;

OptUnCanYes – A checkmark indicates there are nonfuel components;

OptUnCanNo – A checkmark indicates there are no nonfuel components.

The remaining 11 fields indicate the number of various spent fuel disassembly hardware items or nonfuel components or the volume in cubic inches of these items:

CN_PWRControlRods — The number or volume of PWR control rods;

CN_PWRControlRodsSpiders — The number or volume of PWR control rod spiders;

CN_ControlCruciIntact — The number or volume of intact control cruciforms;

CN_ControlCruciBlades — The number or volume of control cruciform blades;

CN_ControlCruciBases — The number or volume of control cruciform bases;

CN_BurnablePoisonAssm — The number or volume of burnable poison assemblies;

CN_NeutronSources — The number or volume of neutron sources;

CN_ThimblePlugs — The number or volume of thimble plugs;

CN_INCoreInstrumention — The number or volume of incore instrumentation elements;

CN_BWRFuelChannels — The number or volume of BWR fuel channels;

CN_Other — The number or volume of other items and a listing of these items.

tblCanister_NonFuel_Consolidated This table contains specific information on uncanistered fuel rods and uncanistered consolidated or reconstituted assemblies. The data elements in this table are:

CNC_REACID — The reactor ID number;

CNC_PoolID — The storage pool where the uncanistered fuel rods and uncanistered consolidated or reconstituted assemblies are stored;

CNC_ASSMID — The ID of the assembly from which the fuel rods or pieces were removed;

CNC_NumberFuelRod — The number of fuel rods from the assembly;

CNC_INITIALU — The initial loading weight (in kilograms of uranium) of the fuel rods;

CNC_MAXBURN — The maximum burnup (in megawatt days thermal per metric ton of uranium) for the assembly from which the fuel rods were removed;

CNC_CurrentLocation — The location in the storage pool where the items are stored.

tblCanister_NonFuel_UnCanistered_Failed_Assm This table contains descriptions of uncanistered failed assemblies. The data elements in this table are:

CNUF_REACID — The reactor ID number;

CNUF_PoolID — The storage pool where the uncanistered failed assemblies are stored

CNUF_ASSMID — The ID of the uncanistered failed assembly;

CNUF_Description — A description as to why the assembly is classified as failed.

Section on Dry Storage

tblStoreDry Respondents indicated whether or not they had implemented dry storage at their site, and if implemented, the dry storage site ID. Respondents also entered the number of multi-element dry storage modules in service. The data elements in this table are:

SD_REACTORID — The discharging reactor ID;

SD_DRYSTORID — The dry storage facility ID;

SD_NUMMODCASK — The number of dry storage modules at a dry storage facility;

SD_NonfuelComponent — Indicates whether a respondent's reported nonfuel components in dry storage. A "1" indicates "yes;"

OptImpYes — A checkmark indicates that a respondent has implemented dry storage;

OptImpNo — A checkmark indicates that a respondent has not implemented dry storage.

TblStoreDry_Assembly Respondents now report only the number of assemblies stored in each dry storage module, not the individual assembly ID's. For each dry storage module in service, respondents also reported the vendor and model number. The data elements in the table are:

DA_REACTORID — The discharging reactor ID;

DA_DRYSTORAGEID — The dry storage facility ID;

DA_MULTIELEMENTCASKID — The ID of each individual dry storage cask;

DA_VENDOR — The manufacturer of the dry storage modules;

DA_MODELID — The dry storage module type;

DA_DRYOPDATE — The date the dry storage module was loaded in MMYYY format;

DA_NUMASSMSTORED — The number of assemblies stored in each dry storage module.

tblStoreDry_Nonfuel Respondents report nonfuel components in dry storage. The data elements in this table are:

DN_REACTORID — The discharging reactor ID;

DN_DRYSTORAGEID — The dry storage facility ID;

DN_MULTIELEMENTCASKID — The ID of each individual dry storage cask;

DN_VENDOR — The manufacturer of the dry storage modules;

DN_MODELID — The dry storage module type;

DN_DRYOPDATE — The date the dry storage module was loaded in MMYYY format;

DN_Content — A listing of the nonfuel components stored in the module.

Section on Projected Assembly Discharges

tblFuelProj Respondents reported projected discharges on a batch or group basis for the next five cycles (n+1 through n+5).

Respondents from operating reactors who did not project five future cycle discharges were contacted to obtain the required information. Reactors with no projected cycles are already shut down. The data elements in this table are:

FP_REACTORID — The reactor ID number;

FP_CYCLENUM — The cycle number. Each reactor shutdown for refueling is assigned a sequential cycle number starting with 01. Respondents reported both projected cycle numbers and projected cycle shutdown dates.

FP_GROUPID — The batch or group ID assigned by the respondent;

FP_NUMASSEM — The number of assemblies in each batch;

FPF_AVGWT — The average initial loading weight for each assembly (in kilograms of uranium);

FP_AVGENR — The average initial enrichment for each assembly (in weight percent);

FP_AVGBURN — The average final burnup for each assembly (in gigawatt days thermal per metric ton of uranium);

CY_CYDNDATE — The projected cycle shutdown (subcriticality) date.

Final Section: Comments

tblComments Comments by respondents with reference to the section to which they pertain are included in this table. To clarify information reported on other parts of the survey form, contractor staff have added additional comments. Pertinent information obtained from respondents during error correction phone calls has also been added. The data elements in this table are:

REACID — The reactor ID;

SECTION — The survey form section to which the comment pertains;

COMMENT — The comment as submitted by the respondent or as entered by survey personnel.

ABBREVIATIONS

AIROX	Atomic International Reduction and Oxidation process
AR	At Reactor
AFR	Away-from-Reactor
AGR	Advanced Gas Cooled Reactors
AVK	Abfallfluss Verfolgungs- und Produkt-Kontollssystem (Waste Flow Tracking and Quality Assurance System)
BNG	British Nuclear Group
BWR	Boiling Water Reactors
CANDU	Canadian Deuterium and Uranium Reactor
DUPIC	Direct Use of spent PWR fuel in CANDU Reactors
EPRI	Electric Power Research Institute (located in California, USA)
GCR	Gas Cooled Reactors
GNS	Gesellschaft für Nukleare Service
LWR	Light Water Reactors
MACSTOR	Concrete cask developed by AECL (Canada)
MOX	Mixed Oxide Fuel
MPC	Multi Purpose Canister
MVDS	Modular Vault Dry Store system
NFBC	Non-Fuel Bearing Component
NUHOMS	Nutech Horizontal Modular Storage System
PUREX	Plutonium and Uranium Reduction and Extraction
PWR	Pressurized Water Reactors
RBMK	Water cooled, graphite-moderated pressurized tube reactors
THTR/AVR	High temperature reactor developed in Germany
WWER	Wodo Wodyanoi Energetichecki Reactor (Russian type of PWR)

CONTRIBUTORS TO DRAFTING AND REVIEW

Bevilacqua, A.M.	Instituto Balseiro, Argentina
Chun, K-S, .	Korea Atomic Energy Research Institute, Republic of Korea
Clarke, R.	European Union
Csullog, G.	International Atomic Energy Agency
Finucane, J., Nikkodem, D.	USDOE/Energy Information Administration, United States of America
Graf, R.	NS Gesellschaft fuer Nuklear-Service mbH, Germany
Hioki, K.	International Atomic Energy Agency
Kahrman, A.	International Atomic Energy Agency
Kim, Y-S.	Han Yang University, Republic of Korea
Kins, J-K. Cheon	Korea Institute of Nuclear Safety, Republic of Korea
Ko, W-I.	Korea Atomic Energy Research Institute, Republic of Korea
Krivošein, G.	Ignalina NPP, Lithuania
Lietava, P.	State Office for Nuclear Safety (SUJB), Czech Republic
Mester, W.	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Germany
Ohlsson, M.	OKG AB, Sweden
Ohuchi, J.	RWMC (Radioactive Waste Management Funding and Research Center), Japan
Park, H-T.	KHNP (Korea Hydro and Nuclear Power Corp), Republic of Korea
Saverot, P.	JAI Corporation, United States of America
Scullion, A.J.	UKAEA (United Kingdom Atomic Energy Authority), United Kingdom
Sokolova, I.	Center for Public Information on Nuclear Energy (CNIIAtomInform), Russian Federation
Strydom, F.	NAC International, United States of America
Vorotyntsev, M.F.	Institute of Physics & Power Eng., Russian Federation
Warnecke, E.	International Atomic Energy Agency
Weh, R.	NS Gesellschaft fuer Nuklear-Service mbH, Germany

Technical Meeting
Vienna, Austria: 16–19 June 2003

Consultants Meetings
Vienna, Austria: 24–26 September 2001, 16–18 April 2002, 26–28 April 2004