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Radioactivity Buildup During Fuel Irradiation in Light Water Reactors (LWR)

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Abstract: Power production in LWRs is economic and very efficient. A by-product of the fission process, which generates the thermal power, is radioactivity associated with fission and activation nuclides. This article describes the ways in which radionuclides are produced and then dissipated in the systems of a LWR. Radionuclides in and on the surface of components create radiation fields which can cause harm to workers. The article describes methods which help to minimize the buildup of these radiation fields. The difficulties in determining the radionuclide inventories for structural materials in LWRs are discussed and appropriate code packages are discussed. The basic principles adopted in the context of treatment of radioactive waste are summarized, including an overview of typical conditioning strategies. Radiation protection issues are briefly described and methods used to protect workers are discussed.

Keywords: Fission in uranium and MOX fuel · Neutron activation · Radioactivity buildup in LWRs · Radiation protection · Radioactive waste treatment

Introduction

In LWRs, power production starts with the fission of U-235 or Pu-239 in the fuel elements. The nuclear energy that is created by the fission process is transformed into thermal energy, mainly by moderation of the fission products in the fuel elements. This thermal energy is used to heat the cooling water in the reactor pressure vessel (RPV) and potential energy thus accumulates in the water. In a pressurized water reactor (PWR), the cooling water (temperature about 320 °C, pressure about 150 bar) is pumped through a steam generator. Due to the high pressure, the water remains liquid in the primary circuit. In the steam generator, the potential energy of the primary circuit is transformed into kinetic energy; in the secondary circuit, steam is formed which drives the turbine. Connected to the turbine is a generator in which the kinetic energy of the turbine is converted to electric energy. In a boiling water reactor (BWR), only one circuit exists. The water in the RPV has a temperature of about 280 °C and a pressure of about 73 bar. Under these conditions, the water in the RPV turns to steam, which directly impinges on the turbine [1][2]. The water circuits of a BWR are shown in [3].

For radioactivity buildup, the following aspects, which are described in detail later, are of concern:

- *Fission of U-235 and Pu-239 in the Fuel Elements*: the fission process creates fission products and two to three free neutrons. The greatest activity produced in a LWR is associated with the fission products and remains in the fuel elements. The neutrons which are necessary to trigger the chain reaction, namely the fission of other uranium and plutonium nuclei, escape from the fuel elements and produce activation products in the cooling water and in structural materials in and around the RPV.
- *Erosion and Corrosion of Structural Materials*: erosion and corrosion of structural materials of the RPV and of systems connected to it occur. The erosion and corrosion products are either already activated or can be activated in the cooling water in the RPV. These activation products are then distributed with the cooling water to various systems. In a BWR, these activation products are also fed in a small amount as carry over to the turbine circuit. This means that systems and components are also contaminated outside the RPV, *i.e.* polluted with activation products.
- *Fuel Element Cladding Failure*: the fuel (uranium and plutonium oxide) is contained in tubes (cladding). From time to time cladding failure occurs and fuel, fission, and activation products escape into the cooling water and contribute to the contamination of systems and components.
- *Radioactive Waste*: radioactive waste is produced during operation of a nuclear power plant. Parts of components which are activated have to be replaced, tools used for maintenance and repair work may be contaminated, cleaning material used in the nuclear power plant has to be treated as radioactive waste, ion-exchange resins used in the reactor water cleaning system accumulate radioactive substances, *etc*.

The activation of materials and the contamination of systems and components create special circumstances when maintenance and repair work has to be performed. A high dose rate (see section on Radiation Protection) can prevail, which forces workers to use shielding against the radiation or robotic tools to protect themselves and to fulfil the requirements of the radiation protection law and ordinance [4][5]. Airborne

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contamination produced when working on contaminated surfaces of components can create a risk of inhaling radioactive particles.

To reduce the difficulties arising due to activated and contaminated systems and components, the operator of a nuclear power plant uses materials and techniques that minimize activation and contamination.

Radiation Protection

Considerable effort is invested in determining the activity production and reducing activation and contamination of materials and systems. Why is this effort needed? Ionizing radiation can be harmful. The link between the amount of radiation absorbed and the associated health risk is expressed as effective dose, measured in Sieverts (Sv). For radiation protection purposes, the milli-Sievert (mSv) is used. Health effects caused by radiation can be acute, *i.e.* specific health effects correlated to well-defined doses. Below scientifically based and internationally accepted threshold dose values, acute health effects cannot be observed. Doses below these threshold values create a risk of developing certain illnesses, for instance cancer (stochastic effects). Since everybody is exposed to natural radiation – which in Switzerland amounts to about 4 mSv per year [6] – and since the exact molecular-biological mechanisms by which radiation at low doses causes harm are not known, the International Commission on Radiological Protection (ICRP) has adopted a linear dose–risk relationship without a threshold, *i.e.* very low doses can also be harmful [7].

To avoid acute health effects, dose limits for radiation workers have been established [5]. To minimize stochastic health effects due to anthropogenically produced radiation, a dose limit has been set for the population. For radiation workers, the stochastic effects should also be low. The regulations therefore state that, for essential tasks that have to be performed in radiation fields, radiation protection measures have to be adopted which are optimized and ensure that the doses to workers remain well below the specified limits. What does optimization mean for the work in a nuclear power plant?

- *Choice of Materials*: the materials used for components exposed to neutrons should contain only a very small amount of nuclides which can be significantly activated and can produce high radiation levels. One such nuclide is Co-60, which is produced mainly by activation of Co-59. Co-60 emits highly energetic gamma-rays. In the 1970s, a cobalt alloy (stellite) was often used in nuclear power plants for control blade rollers and bearings. The stellite was eroded and corroded, the cobalt was activated in the neutron field, and Co-60 was transported in the water circuit, causing high radiation fields on systems and components. Therefore, maintenance and repair work on these systems and components was difficult and gave rise to high doses. In the meantime, most of the stellite has been replaced by other alloys without cobalt.
- *Design of the Reactor Core, the Pressure Vessel and Surrounding Structures*: the design should take into account two aspects. One is the construction and layout of components in the plant in such a way that maintenance and repair work can be performed without generating high doses to workers. The means of achieving this goal are the geometric position of a component in a room, the ease of maintenance, and the possibility to provide shielding (lead covers) around the component. The second is the construction of the structures surrounding the reactor pressure vessel in such a way that, in the decommissioning phase, only a few components have to be treated as radioactive waste.
- *Design of the Fuel Cladding*: fuel cladding failures should be avoided. Fuel and fission products which are washed out in the case of a severe cladding failure cause high radiation and contamination levels in systems and rooms. Moreover, gaseous fission products, especially I-131, can contaminate breathing air and make strict protection measures necessary for workers.
- *Decontamination of Systems*: some systems become heavily contaminated during operation of the plant. This means that a high dose rate exists in the vicinity of these systems and that the inner surface of the circuit is contaminated with radioactive materials. Some plants have developed a strategy of bundling maintenance work, *i.e.* not performing it every year, and carrying out a chemical decontamination before the maintenance work starts. By doing so, the dose rate can be reduced by factors between 2 and 100 [3].
- *Water Chemistry*: in BWRs, but also in PWRs, chemical additions to the water are used to avoid the deposition of activated nuclides, for example Co-60, at inner surfaces of the circuit. For more details see [3].
- *Training of Personnel*: for work to be performed in high radiation fields, it is essential that workers are trained accordingly and are familiar with the working environment. For difficult geometric working conditions and for complex working procedures, mock-ups of the components are provided on which the workers can train under realistic conditions.
- *Robotics*: in high radiation fields, some work can be performed by automatic tools or by robotic devices. For most in-service inspections (inspections of welded seams), robotic devices are used nowadays.

To be able to plan all these optimization measures effectively, a good knowledge of the activity buildup in the plant is necessary.

Fission and Activation Products in the Fuel Matrix

Two different types of fuel, UO_2 and MOX (mixed oxide), are commonly used in commercial LWRs. Natural uranium consists mainly of atoms of the isotope U-238 mixed with a minor component of U-235, *i.e.* 0.7 wt%. To enable use as fuel in a LWR, the uranium has to be enriched in the fissile isotope U-235. A common content was around 3.5 wt%, with the trend nowadays going up towards 5 wt% U-235. MOX fuel is a mixture of U- and Pu-oxides, with Pu-239 and Pu-241 being the major fissile isotopes. In addition, a minor amount of U-235 will be present.

The formation of fission- and activation products in the fuel during irradiation in a reactor is dependant on the local power (fission rate) over time resulting in a burn-up of the fissile content. For the two largest Swiss reactors Leibstadt (BWR) and Gösgen (PWR), the burn-up in a fuel assembly today could be 55 and 60 MWd/kg HM (HM stands for heavy metals, here U and Pu), respectively. In the following, some numbers are presented, based on a real fuel pin geometry and power history from a Swiss LWR. The predictions were made with the internationally known and continuously updated fuel behaviour code TRANSURANUS [8].

The burn-up distribution along the fuel pin, commonly with a fuel active length of over 3 m, after irradiation in a reactor is fairly flat with a pronounced drop at both ends. The radial power profile is inhomogeneous from the start. The fast neutrons are thermalized (slowed down) in the moderator (coolant) outside the pin and, when re-entering in the pin, are used up at the fuel pin periphery by thermal fissioning of the U-235 isotope and activating the U-238 isotope. Thus the fuel shields itself from fissioning towards the fuel pin centre. The result is an increased number of fission reactions leading to a significantly increased power density and burn-up at the pin periphery (rim), which becomes more and more accentuated with increasing irradiation time (Fig. 1). Already after an irradiation period of several hundred days,

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this increase is noticeable. For high burn-up levels, the maximum value can be twice or even three times as high as the pin average for both UO_2 and MOX fuel pins.

The fission/consumption of the U-235 isotope results in different fission products, but also the U-238 is slightly depleted due to the mentioned neutron capture or activation, leading to the buildup of different Pu isotopes (and Pu-total) (Fig. 2). Noticeable is that the fissile isotope Pu-239, as soon as it is formed, also participates in thermal fissioning, and, after reaching a certain content, its concentration stagnates and even slowly decreases with higher burn-up. When irradiating MOX fuel in which the main fissionable isotopes are Pu-239 and Pu-241 from the very beginning, these isotopes are again consumed from periphery to centre and Pu-239 is consumed in function of time or burn-up from the very beginning as shown in Fig. 3.

The activity built up in fuel during irradiation is very high. The content and activity can be predicted with computer codes, *e.g.* the isotope generation and depletion code ORIGEN [10]. To make predictions it needs, among others, the appropriate neutron cross-section, *i.e.* the energy-dependent reaction probability for a neutron to interact with a given nucleus. The cross-sections depend on the properties of the target nuclei and the relative speed between the neutron and the target nucleus and are distinguished according to the type of interaction of the neutron with the nucleus, *i.e.* fission, capture, elastic or inelastic scattering, *etc*. While fission cross sections and reactivities for fresh fuel have been intensely measured, tabulated and modelled, remaining reactivities at high burn-up show some uncertainties due to the complex interaction of all built up fission products and actinide nuclides. To reduce such uncertainties, the zero-power research reactor PROTEUS at PSI has been used lately to measure remaining reactivities of fuel irradiated in Swiss power reactors to high burn-up levels, *i.e.* ≥100 MWd/kg HM for UO_2 and 60 MWd/kg HM for MOX, respectively.

The prediction of radionuclide activity levels of fuel after different irradiation and decay periods (beyond the end of irradiation) were made with the ORIGEN code and some results are given in Table 1 and Table 2 for UO_2 and MOX fuel, respectively. Both cases are based on typical and realistic fuel compositions. The burn-up of 60 MWd/kg HM is a high value, but is representative for what could be reached today as a fuel pin average. The tables include a selection of nuclides with the highest activities. A comparison between UO_2 and MOX fuel shows that the total activity, as well as the activity of fission products, is similar and that the main difference relates to the actinides and daughters, with a 3–4 times

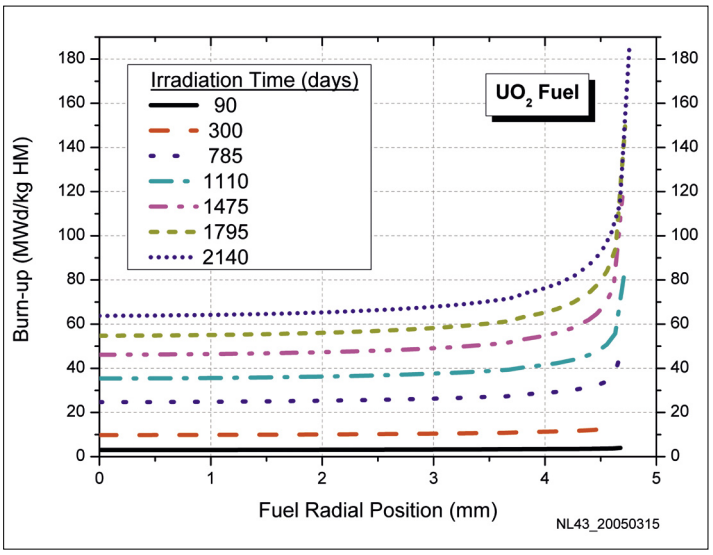

Fig. 1. The radial burn-up distribution in a fuel pin, as a function of irradiation time

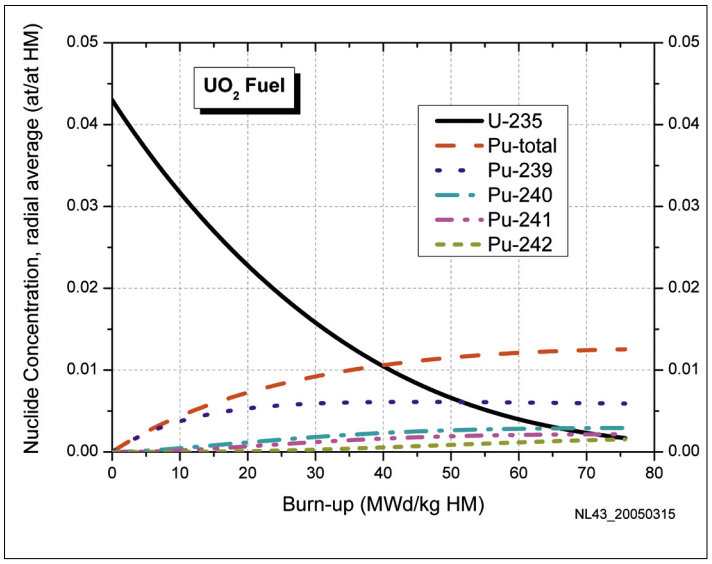

Fig. 2. Concentration of some U- and Pu-nuclides *versus* burn-up for UO_2 fuel

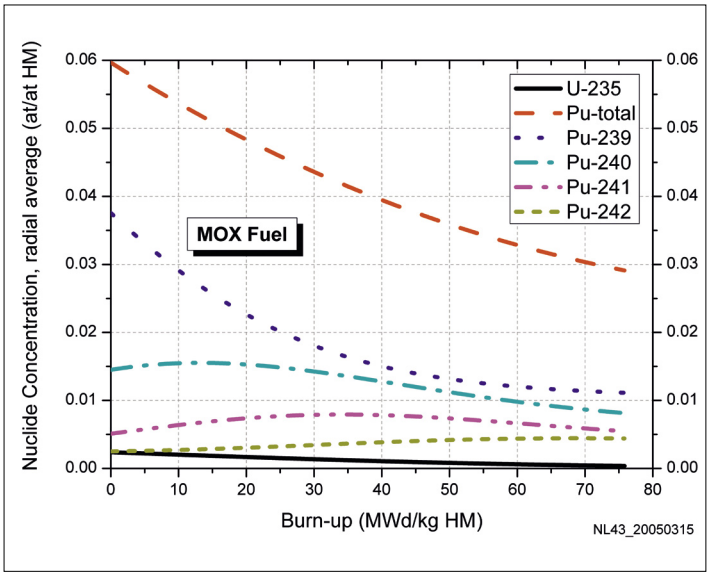

Fig. 3. Concentration of some U- and Pu-nuclides *versus* burn-up for MOX fuel

higher level for MOX fuel. The main reason for this is the much higher amount of the nuclide Pu-241. Worth noting is the buildup of Am-241, created by the decay of Pu-241, which is another reason for the higher MOX fuel activity. Furthermore, the activities after different decay times are shown in order to provide an impression of how the activity is reduced by the very different half-lives, which are specific to each nuclide. It can be seen that, for both UO_2 and MOX fuel, the activity after one, ten and one hundred years of decay is reduced to about 35, 10, and 1%, respectively.

Today, the trend is towards higher burn-up values for the fuel. This produces a higher activity and higher residual thermal power in the spent fuel elements. As a consequence, the high burn-up fuel has to be kept longer in interim storage facilities (storage pools) before it can be loaded into casks to be transported or to be placed in dry storage facilities. On the other hand, a higher burn-up value allows the fuel to stay longer in the reactor. This means that, for the same power output, fewer fuel elements are needed. Mainly economic factors (for fuel procurement as well as for waste treatment) speak for high burn-up. In order to reach high burn-up goals, safety questions, linked to the behaviour of the fuel and fuel cladding during irradiation and for waste treatment and storage, have to be carefully treated and positively answered as prerequisite.

Activation of Structural Materials

For the most part, fission neutrons produced in uranium fuel leave the outer surface of the fuel rods and enter the moderating water/steam between the fuel assemblies. Here they will be slowed down (*i.e.* 'thermalized'), mostly by elastic scattering with hydrogen nuclei, before either entering a neighbouring rod, causing fission and activation within the fuel matrix, or encountering the so-called in-core structural material components.

These components consist of all devices that are built into the core and belong neither to the fuel rods nor to the moderator, *i.e.* spacers for fuel assemblies, control rods filled with neutron poison, neutron sources and instrumentation tubes, various types of screws, pins, *etc.* and – in the case of a BWR – the fuel channels enclosing each fuel assembly. The main materials used for these components are zircaloy, stainless steels, and nickel base alloys. The control rods contain a neutron absorber (typically boron carbide/hafnium (BWR) or a silver-indium-cadmium alloy (PWR)).

Although significantly slowed down in the moderator, some of the neutrons leave the fission zone and propagate to outer core areas. This part of the neutron flux is called 'leakage' and will be reduced by the water

Table 1. Activity [Bq/kg HM] of important actinide and fission product nuclides in irradiated UO_2 fuel, initial enrichment 4.30% U-235, burn-up 60 MWd/kg HM

Decay Time [Years]	**0**	**1**	**10**	**100**
Fission Products				
Sr-90	5.13E+12	5.01E+12	4.04E+12	4.75E-11
Y-90	5.39E+12	5.01E-12	4.04E+12	4.75E+11
Zr-96	4.19E+13	8.04E+11	2.81E-04	0.00E+00
Nb-95	4.26E+13	1.81E+12	6.25E-04	0.00E+00
Ru-106/Rh-106	7.15E+13	3.48E-13	7.16E+10	0.00E+00
1-131	3.00E+13	6.69E-01	0.00E+00	0.00E+00
Cs-134	1.95E+13	1.39E+13	6.78E+11	6.78E+11
Cs-137/Ba-137m	1.70E+13	1.66E+13	1.35E+13	1.69E+12
Co-144/Pr-144	7.99E+13	3.27E+13	1.08E+10	0.00E+00
Σ FP	**3.13E+14**	**1.11E-14**	**2.23E+13**	**3.32E+12**
Actinides + Daughters				
Pu-238	5.07E+11	5.21E+11	4.89E+11	2.40E+11
Pu-241	6.57E+12	6.26E+12	4.06E+12	5.39E+10
Am-241	8.63E+09	1.89E+10	9.14E+10	1.98E+11
Am-242	5.15E+12	5.33E+08	5.12E+08	3.40E+08
Cm-242	3.77E+12	8.04E+11	4.24E+08	2.81E+08
Cm-244	2.36E+12	2.27E+12	1.61E+12	5.14E+10
Σ A+D	**1.84E-13**	**9.88E+12**	**6.25E+12**	**5.45E+11**
Sum of FP+A+D	**3.31E+14**	**1.20E-14**	**2.86E-13**	**3.86E+12**

Table 2. Activity [Bq/kg HM] of important actinide and fission product nuclides in irradiated MOX fuel, initial enrichment 4.24% Pu-fiss (4.46% Pu- + U-fiss), burn-up 60 MWd/kg HM

Decay Time [Years]	**0**	**1**	**10**	**100**
Fission Products				
Sr-90	2.27E+12	2.22E+12	1.79E+12	2.10E+11
Y-90	2.32E+12	2.22E+12	1.79E+12	2.10E+11
Zr-95	3.94E+13	7.56E+11	2.65E-04	0.00E+00
Nb-95	4.01E+13	1.70E+12	5.88E-04	0.00E+00
Ru-106/Rh-106	8.14E+13	4.04E+13	8.33E+10	0.00E+00
1-131	2.94E+13	6.54E-01	0.00E+00	0.00E+00
Cs-134	1.18E+13	8.44E+12	4.10E+11	3.04E-02
Cs-137/Ba-137m	1.32E+13	1.29E+13	1.05E+13	1.31E+12
Ce-144/Pr-144	6.83E+13	2.80E+13	9.30E+09	0.00E+00
Σ FP	**2.88E-14**	**9.67E-13**	**1.46E+13**	**1.73E+12**
Actinides + Daughters				
Pu-238	7.02E+11	7.77E+11	7.45E+11	3.68E+11
Pu-241	2.63E+13	2.50E+13	1.62E+13	2.15E+11
Am-241	8.64E+10	1.27E+11	4.16E+11	8.37E+11
Am-242	2.52E+13	7.22E+09	6.93E+09	4.60E+09
Cm-242	2.03E+13	4.32E+12	5.73E+09	3.80E+09
Cm-244	3.69E+12	3.55E+12	2.51E+12	8.04E+10
Σ A+D	**7.62E+13**	**3.38E+13**	**1.99E+13**	**1.51E+12**
Sum of FP+A+D	**3.64E+14**	**1.30E+14**	**3.45E+13**	**3.24E+12**

reflector zone enclosing the core and then by the thermal shield(s) and further water layer(s), before finally reaching the reactor pressure vessel (RPV; an approx. 20–30 cm thick steel layer) and the concrete layers beyond the vessel (*i.e.* the biological shield and the walls of the drywell). Other components inside the RPV that are irradiated over the years by the neutron flux are *e.g.* jet pump beams in BWRs or – lying in the vertical neutron streaming direction – lower and upper core plates and supports, the core barrel, the control rod drive system, steam separators and dryers (BWR).

By propagating through these different shielding layers, the neutron spectrum will change depending on the material currently being penetrated. If this material contains water (for example concrete), the spectrum will be more 'thermal'. If the material is steel, then the thermal neutrons will be absorbed primarily by the Fe nuclei, thereby causing the spectrum to be energetically much harder. Every penetrated shield reduces the neutron flux by one to several orders of magnitude, so it finally becomes negligible outside the walls of the drywell (BWR).

Neutron radiation interacts with structural material in two ways. Fast neutrons (neutron energy >1 MeV) lead to an embrittlement of metals by hitting atoms of the metal lattice and causing displacement cascades inside the material, thus reducing toughness. Slower neutrons will partly be absorbed by nuclei with a significant neutron capture cross-section, causing the activation of the material.

Embrittlement is an important factor in calculating the life expectancy of a nuclear power plant, since it will ultimately determine the reliability of the RPV as well as the performance and durability of in-core components [11].

The activation of structural material causes problems during plant maintenance, as the dose rates resulting from the activated materials necessitate cautious and complex handling procedures and interim storage of materials removed during and after the annual plant revision. Activated structural material waste typically accounts for approx. 6 vol.-% of the total annual radioactive operational waste of a plant. Finally, long-lived activation products from both operational waste and later arising decommissioning waste are relevant for the development and design of a final repository.

With respect to the handling of operational waste, Co-60 is the most important activation product from structural material as it regularly dominates the gamma dose rate. Co-60 is produced mainly *via* neutron capture by Co-59, a common impurity in most components. Besides Co-60, there are many other activation products, including Cs-137 (from fission of uranium impurities), Ag-110m, Ag-108m (from activation of neutron poison), Sb-125, Mn-54 (gamma emitters), Fe-55, Ni-63 (beta emitters) and actinides (alpha emitters, from activation of uranium impurities).

There are two ways to reduce the activation level of structural material: either by shielding of the component in question (as done by the water reflector and the thermal shield between the core and the RPV) or – if shielding is not possible – by using materials containing less impurities of nuclides with high neutron capture cross-sections. The second approach leads to the use of zircaloy rather than steel (and replacement of Co-containing alloys such as stellite) for structural material in high neutron flux areas such as the core itself. Zirconium has a very low neutron capture cross-section and much less cobalt impurity compared to steel, leading to much lower activity values.

For handling, conditioning, transport and storage of radioactive structural material waste during operation and decommissioning of the plant, knowledge of the nuclide inventory of each component is important. Measurements of dose rates and spectra, combined with neutron transport, burn-up and decay calculations, are performed to determine the nuclide inventory of each component. Codes used for this purpose must be able to simulate the significant spectral change in the neutron flux and the specific nuclide cross-sections at any location inside and outside the core. A typical combination of programs used to perform this task is MCNP, a Monte Carlo transport code, and GRSAKTIV [12] (including ORIGEN-X [13] as an update of the well-known ORIGEN code), an extended version of the common burn-up and decay code that can calculate not only the irradiation of fuel but also of structural materials by applying case-dependent cross-section libraries. Instead of using a time-consuming Monte Carlo code for calculation of the neutron source and the subsequent transport to the activation area, alternatives such as the one-dimensional deterministic code ANISN or the two-dimensional fuel lattice code BOXER [14] can be used.

Table 3 shows typical activities of key nuclides after 40 years of operation of a BWR and 5 years decay time for selected reactor components at different locations, either inside or outside the core (numbered (1) to (3) for core internals, RPV and bioshield). The reduction in total activity between the components covers up to 10 orders of magnitude.

Concepts for Radioactive Waste Treatment

As mentioned before, both fission and activation products as well as fuel particles may be released into the primary coolant water and, from there, dissipated into other systems. Reactor water clean-up (RWCU) systems, typically based on ion-exchange or evaporation processes, are operated to reduce nuclide contents in the circulating primary water to permissible levels, resulting in the generation of RWCU wastes (spent resins, evaporator concentrates) [3].

By law, man and the environment must be protected from emissions arising from radioactive materials. Within the Swiss regulatory framework on radioactive waste management, this general rule has been broken down into fundamental requirements related to the typical sequence of management steps: radioactive waste shall be in, or shall be brought into, a form that ensures safe storage and transport and, last but not least, suitable for disposal without subsequent destructive actions on the waste [15]. In practice, these objectives are met by technical measures including waste conditioning (waste treatment and packaging).

Waste treatment is designed to convert raw wastes, if appropriate, into chemically stable, solid, low-dispersible, water-resistant and leach-resistant waste products suitable for envisaged packaging options. A typical application is the solidification of

Table 3. Typical specific activity in [Bq/g] for BWR-components after 40 years of operation and 5 years of cooling (decay time) after reactor shut down

Nuclide	Core Internals (1) (steel)	RPV (2) (steel)	Bioshield (3) (concrete)	Drywell-Wall (concrete)
Co-60	4.3E+9	8.3E+5	3.6E+3	5.1E+0
Fe-55	6.4E+9	6.6E+6	1.2e+3	8.9E+0
Ni-63	1.8E+9	1.6E+5	4.5E+1	<1.0E+0
Cs-137	1.1E+5	9.5E+0	<1.0E+1	<1.0E+0
ΣAlpha	5.8E+3	9.1E+0	9.4E-1	2.5E-1
Total	1.3E+10	7.8E+6	5.3E+4	1.8E+2

liquid or dispersible solid raw wastes by mixing with an appropriate bonding agent, resulting in a homogeneous waste matrix after curing. Mechanical downsizing of solid waste (*e.g.* cutting of large items) may be required for subsequent packaging. Volume reduction (*e.g.* by compaction, incineration or melting) is attractive for keeping overall waste management costs low and minimization of organic and/or gas-producing substances (*e.g.* by oxidation processes) is favourable for disposal safety. Sometimes, waste pre-treatment is necessary for successful application of the fundamental treatment step; typical examples are sorting, drying/evaporation or chemical treatment (*e.g.* precipitation).

Packaging is intended to enclose and fix the waste product in a waste package with properties (*e.g.* dimension, mass, handling fixtures, nuclide inventory, radiation level) that ensure safe waste handling, storage, transport and disposal according to pertinent rules or design criteria. The basic packaging element is the container (*e.g.* a 200-l drum). Filler materials (such as cement grout) are frequently used to embed waste items and to eliminate voidage within the container. Fittings may be used for proper waste product emplacement (*e.g.* baskets) or radiation shielding (*e.g.* Pb liner). Temporary or permanent overpacking is an additional method of enhancing waste package performance if required.

A summary of conditioning methods applied or accepted so far in relation to typical raw waste streams from NPP operation in Switzerland and from reprocessing of Swiss spent NPP fuel in France (COGEMA) and Great Britain (BNF) is given in Table 4.

In principle, new conditioning processes at Swiss NPPs are implemented in three consequent steps:

- *Research and Development,* aimed at the selection of the most promising process within the spectrum of candidate methods. Aspects of technical realization, process performance (*e.g.* compliance of waste product and waste package with regulatory and waste management targets) and economics are important parts of the evaluation.
- *Prototype Testing,* in order to demonstrate the applicability, reliability and acceptability of the process under active full-scale conditions, to validate key properties of the waste package and of its components with respect to R&D forecasts and to consolidate envisaged QA measures. This action requires clearance by the pertinent Swiss authority (HSK), permitting the manufacturing of a limited number of waste packages under a well-defined type testing programme.

Table 4. Typical conditioning strategies applied to Swiss NPP and reprocessing wastes now and in the future.

	Raw waste	Pre-treatment	Treatment	Packaging	Site (period)
operational NPP wastes	ion exchange resins	dewatering, transfer into 60/150 l vessel	embedding in polystyrene	vessel grouted into 100/200 l drum	KKB
		drying	bituminisation	200 l drum	KKG
			cementation	200 l drum	KKL
		drying/thermolysis	cementation	200 l drum	KKM
	evaporator concentrates	drying	bituminisation	200 l drum	KKG
			cementation	200 l drum	KKL
	sludges	dewatering	cementation	200 l drum	KKB, KKM
	filter candles	dewatering	loading into tube	200 l drum/grouting	KKB
		dewatering	compaction	200 l drum/grouting	KKG
	spent reactor internals	[cutting]	loading into basket	200 l drum/grouting	KKL, KKM, KKG (2006–)
				thick-walled iron container/ drying	KKG (2006–)
	solid, incinerable	sorting/incineration	ash cementation	200 l drum	PSI (–2003)
	solid, compactable	sorting	compaction	200 l drum/grouting	PSI (–1988)
		sorting [blending]	supercompaction	200 l drum/grouting	KKL (1988–2001) ZWILAG (2002–)
	solid, neither incinerable nor compactable	[sorting]	[downsizing]	200 l drum/grouting	PSI
	solid, mixed	[blending]	plasma arc treatment in glass melt	150 l vessel, grouted into 200 l drum	ZWILAG (2004–)
reprocessing residues for return	liquid HLW	calcination	vitrification	180 l canister	COGEMA, BNF
	hulls and end pieces	dewatering	supercompaction	180 l canister	COGEMA
		dewatering	loading into basket	500 l drum/grouting	BNF
	sludges		bituminisation	200 l drum	COGEMA
			cementation	500 l drum	BNF
	techological waste	dewatering	supercompaction	180 l canister	COGEMA
	solid LLW		compaction	1.2 m^3 container/ grouting	BNF

- *Routine Production*, subject to clearance by HSK and requiring full implementation of the QA system. The QA system includes a QC programme (process and product control), conventions on record management (such as book-keeping in the 'Information System on Radioactive Materials' (ISRAM [16]), a nation-wide electronic database system) and, possibly, a waste characterization programme (WCP). A WCP may be required if a given raw waste stream is known or suspected to contain significant inventories of 'difficult-to-measure' disposal-relevant nuclides and if available information is not considered as being sufficient for deriving waste package inventories in a trustworthy manner.

Prototype testing may be bypassed in the case of small modifications of a well-established reference process.

According to the rules for the clearance procedures laid down in [15], the applicant has to provide statements that the waste package type in question is suitable for transport, interim storage, and disposal. The organization established for the planning of Swiss nuclear waste repositories, Nagra, is formally charged with the assessment and the certification of the disposability of waste package types.

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