TopFuel

REACTOR FUEL PERFORMANCE 2012

Transactions

Manchester, United Kingdom
2 - 6 September 2012

TopFuel 2012 Gold Sponsor

Westinghouse

EUROPEAN NUCLEAR SOCIETY

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IAEA
International Atomic Energy Agency

AEN NEA

European Nuclear Society
Rue Belliard 65
1040 Brussels, Belgium
Phone + 32 2 505 30 54
Fax +32 2 502 39 02
E-mail ens@euronuclear.org
Internet www.euronuclear.org

ISBN 978-92-95064-16-4

These transactions contain all contributions submitted by 7 September 2012.

Plenary Session

TOP FUEL 2012

Back-end requirements that need to be taken into account in the fuel design phase.

Ph. Lalieux, D. Boulanger, M. Van Geet (ONDRAF/NIRAS)

1. General objectives of the long term management of spent fuel.

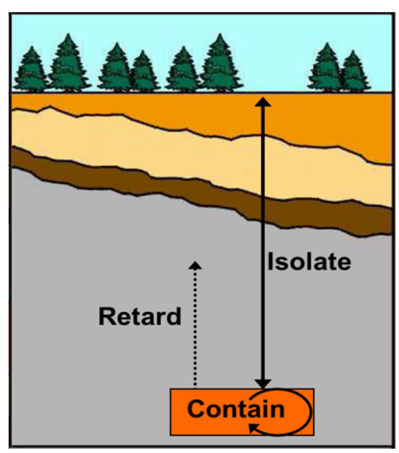
The management of spent fuel as a waste presents a unique specificity that is related to the associated very long time scales. These time scales are in the order several hundred thousands years[1].

Ensuring the long-term management of this waste means maintaining the radiation exposure to human and the biosphere below tolerable levels during such a period of time, whatever the characteristics and the properties of the materials.

The most promising solution to reach this goal is to rely on geological formations that have demonstrated stability over periods of times that by far overcome the million years. Geological disposal of radioactive wastes is based on the principle that the chosen rock environment will remain stable (from geochemical, geomechanical,… points of view) and largely unaffected by environmental change for up to several hundred thousands years.

For the disposal of spent fuel, three major safety functions must be fulfilled by the repository: isolation, confinement and retardation of the radioactive material (Fig. 1). Isolation means creating stable conditions for the durability and performance of the repository and reducing substantially the likelihood and consequences of inadvertent human intrusion. Confinement means retaining the radionuclides within a well-defined zone. Retardation means reducing the rate of the movement of radionuclides through specific materials due to the interaction (e.g. sorption) with immobile components.

Fig. 1 - Safety functions of the repository



A disposal facility requires the combination of man-made (the "Engineered Barrier System", EBS) and natural (the "host rock") barriers, each with specific function(s) to be fulfilled. These

[1] Such time scales also prevail for all other long-lived and highly active radioactive wastes. The present analysis is however focusing on spent fuel.

barriers must be sufficiently compatible both with each other, to not impair their respective functions, and with the contained waste form, to mitigate its unavoidable degradation and dispersion over time.
Various EBS - host rock combinations (forming the disposal system) are considered today, depending on the role each barrier can be given. Therefore, safety assessments must always consider the system as a whole.

To date, essentially three rock types[2] have proven their potential to host spent fuel: crystalline rock (granite), salt rock and argillaceous rock (clays).
Besides their geological stability, these rocks present mechanical properties that allow the excavation of large cavities and tunnels (though artificial reinforcement is required for argillaceous rocks). Their more or less significant capabilities of dissipating the spent fuel thermal load and of retaining radioactive material and retarding its migration to the biosphere will determine the properties needed for the associated EBS.

The EBS comprises essentially a metallic container that encloses the waste primary package(s) and has appropriate corrosion-resistant properties, as well as buffer/backfill materials added to attenuate the impact of the disposal environment on the container and ensure the contact between the container and the host rock.
The EBS will fulfill the confinement safety function, over a period of time that is dependent both on the thermal properties of the host rock and on its retardation efficiency. The main threat to the EBS confinement function is the container corrosion by deep ground waters or salt brines.

Granite host rock shows high mechanical strength and heat resistance, good sorption behavior and thermal conductivity but poor plastic behavior (no self-sealing of occurring cracks) and consequent limited retention ability (if fractured) [1]. It should be associated to an EBS that ensures a very long term (hundred thousand years) confinement of the radioactive material. Classical examples are the Finish and the Swedish disposal systems combining granite host rock with thick-walled copper-based containers with high corrosion-resistance [2].

Salt rock shows high heat resistance, thermal conductivity, plastic behavior and impermeability, as well as a good mechanical strength but very poor dissolution and sorption behaviors [1]. Both corrosion-resistant and corrosion-allowance concepts are considered for the EBS of disposal systems involving salt host rock. Large-scale underground research laboratories (URLs) have been used for developing these systems. Examples are the US Waste Isolation Pilot Plant (WIPP) and the German URL in salt dome at Gorleben.

Clay has poor mechanical strength, heat resistance and thermal conductivity but high plastic behavior and high isolation and retention ability [1]; it is associated to an EBS with good mechanical properties and thermal stability, able to ensure the confinement of the radioactive material over a relatively short (a few thousand years) period of time during which the host rock will be impacted by the waste thermal load, which might transitorily and temporarily modify its

[2] Other potential host rock types are being or could be considered and studied.

performance. Corrosion-allowance concepts are considered in several countries like France, Switzerland and Belgium, combining clay to thick-walled containers of low alloy or unalloyed steel [2,3].
It should be mentioned that clays play a role in most of the disposal systems considered today, either as part of the EBS (e.g. bentonite used as backfill material) or as host rock.

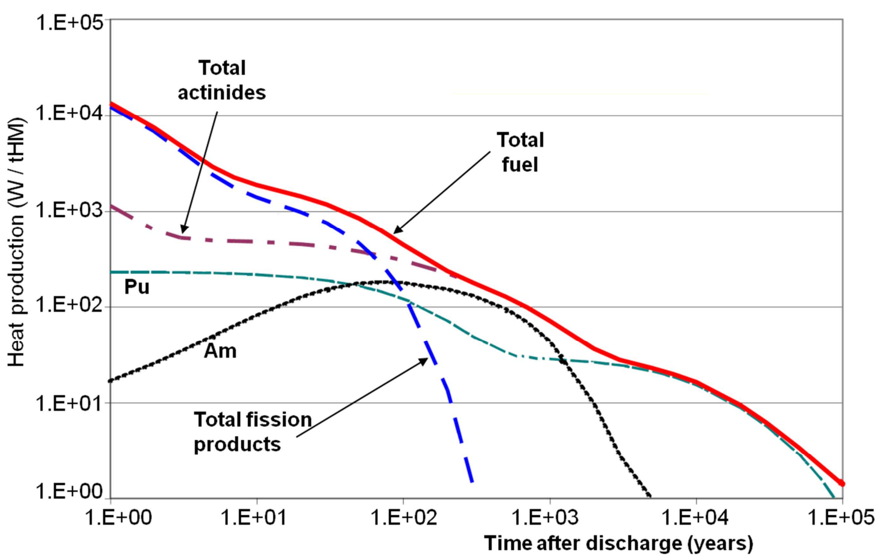
The different disposal concepts place thus a more or less important weight on the container as a barrier, depending on the host rock properties.
Whatever the time period over which radioactive material will be confined in the EBS, sooner or later all containers will breach due to corrosion and will start releasing their radionuclide inventory.

2. Key drivers for designing a disposal facility for spent fuel.

The impact of the spent fuel radiological inventory on the disposal system safety and feasibility is expressed in terms of thermicity, radiotoxicity and radiological risk, which are the drivers for designing a disposal facility.

The thermicity of the spent fuel is a major parameter for the design and the footprint of the system. It results from the contribution of a reduced number of fission products (mainly 137Cs and 90Sr and their decay products) and of most of the transuranic elements (plutonium, americium, curium). From about a hundred years after unloading from reactor, actinides (more particularly americium) contribute more than 90% to the thermal load of an irradiated UOX fuel assembly (Fig. 2).

Fig. 2 - Thermal inventory of irradiated UOX fuel (~50 GWd/tM) [4]



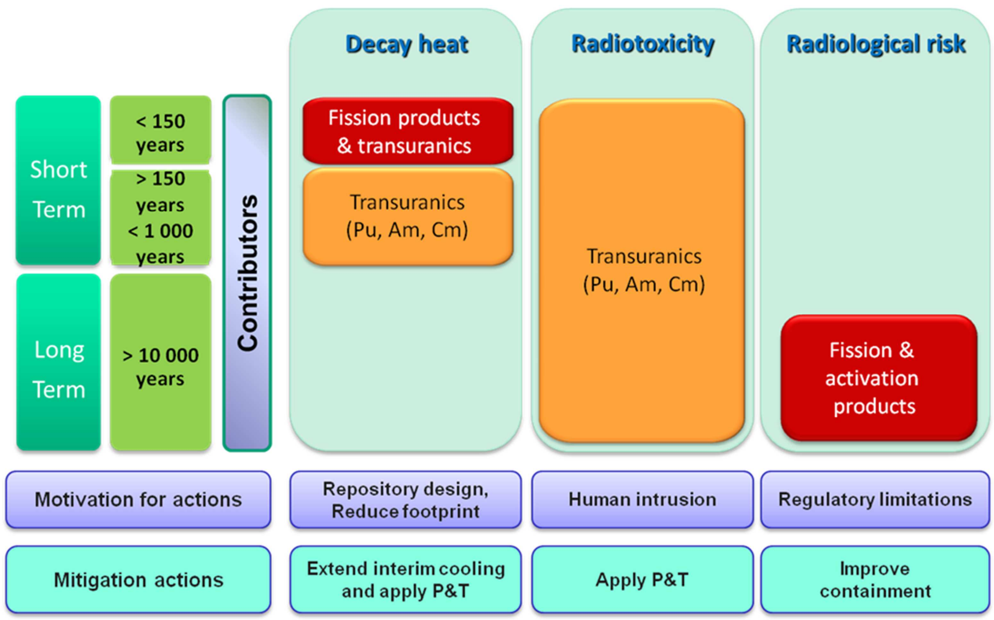
The spent fuel radiotoxicity, which is determined by its content of radiotoxic nuclides and represents its intrinsic radiological harmfulness, will be crucial for all scenarios involving an accidental contact of the radioactive material with the biosphere.

The radiological risk is a combined consequence of a disposal system's capacity to confine a radionuclide and the intrinsic radiotoxicity of this radionuclide. The radiological risk is the greatest concern for long term safety.

Transuranics largely contribute to the radiotoxicity of spent fuel but, on the basis of safety assessments carried out for disposal systems in clays, they only have a minor impact in terms of radiological risk [5]. The Radiological risk is mainly associated with a limited number of long-lived fission and activation products, whose mobility in the disposal system is significant (examples are 79Se, 129I, 36Cl).

Various ways are considered today with a view to mitigate the impacts of the radiological inventory of spent fuel. Fig. 3 positions these impacts over time, showing their major contributors and highlighting the main actions which may lead to their reduction (these mitigation actions do not remove the impact but moderate its effect on the disposal system).

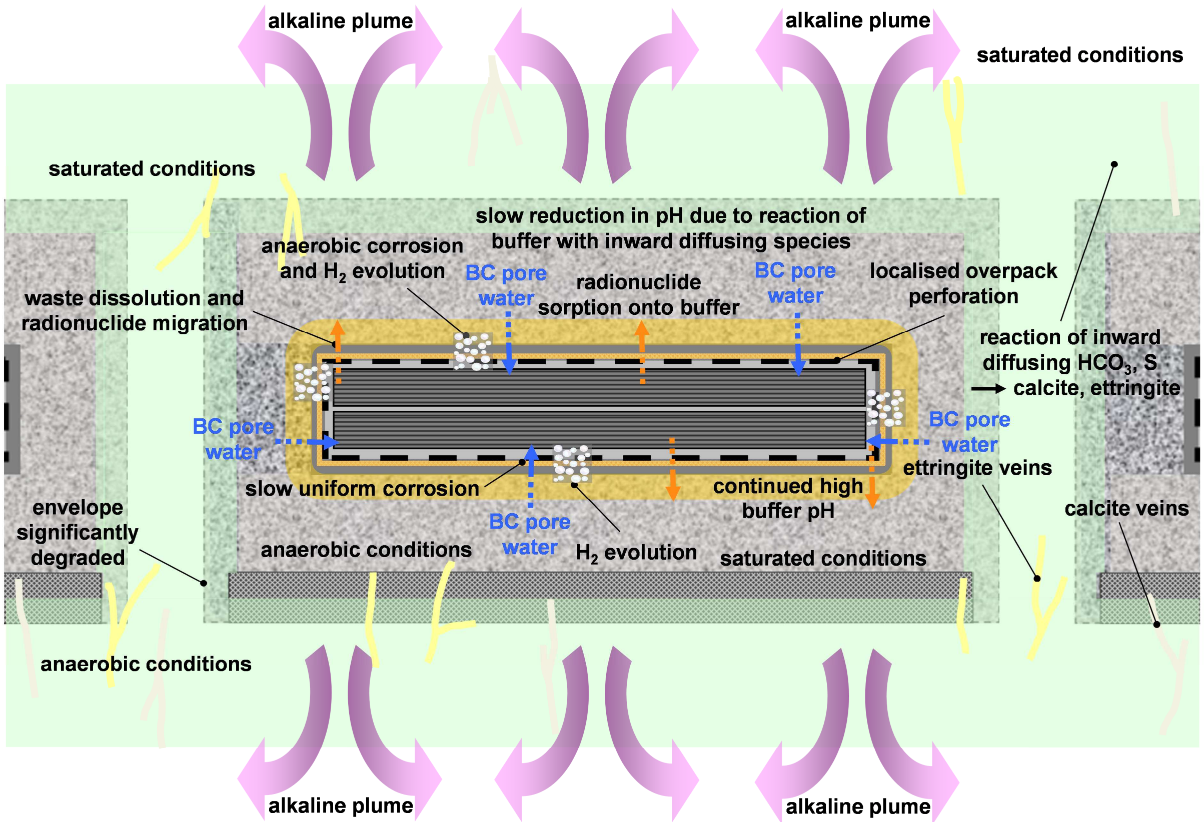
Fig. 3 - Drivers for designing a disposal facility



3. <u>Interactions of the spent fuel inventory with the disposal system</u>.

Once the EBS containment is lost (which corresponds to the container failure), the spent fuel inventory will start migrating to the host rock. To illustrate the complexity of the various interactions taking place in the EBS field, Fig. 4 gives an overview of the processes expected to occur after the container failure, in the ONDRAF/NIRAS concept for deep disposal of spent fuel (spent fuel is placed in a thick-walled carbon steel container (named "overpack") surrounded with a concrete buffer; an outer stainless steel (un-tight) envelope is placed around the buffer).

Fig. 4 –Overview of the major processes taking place in the EBS once containment is lost. (ONDRAF/NIRAS concept)



The radionuclides release from an irradiated fuel assembly will proceed in two consecutive steps: a rapid release of the "accessible part" of the assembly inventory, referred to as the Instant Release Fraction (IRF), and a slow, long-term release of the radionuclides embedded within the fuel matrix and in the assembly cladding and structure materials and which will be controlled by dissolution processes.

Practically (and conservatively), assessments of a disposal system performance consider the radionuclides located in the fuel-cladding gap of a fuel rod, at the fuel grain boundaries, in the High Burnup Structures (HBS), if present, and in the oxide (and crud) layers of the cladding and structures materials, to be part of the IRF. The contribution of critical (i.e. long-lived and mobile) radionuclides to the IRF is a key aspect of the long term safety of repositories.

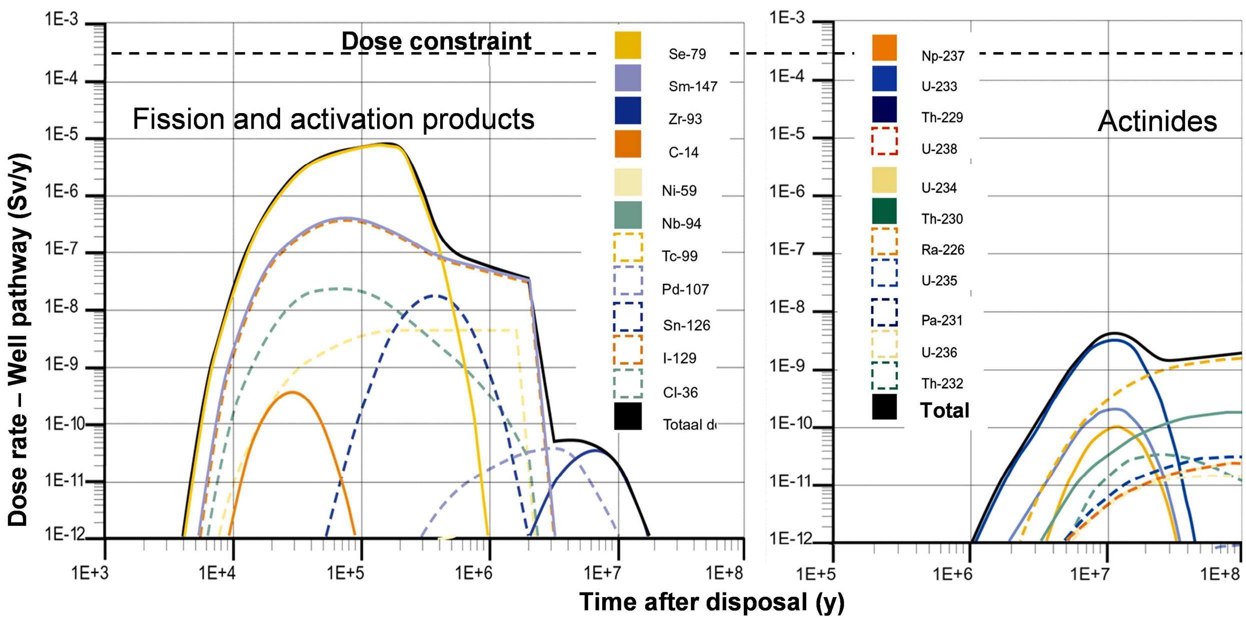
As regards the dissolution processes that will control the release of the "non-accessible part" of the assembly radioactive inventory, they will be dependent on the complex physicochemical conditions occurring in the breached container, mainly controlled by the characteristics of the penetrating water and by the radiations emitted by radionuclides.

The matrix of irradiated oxide fuels shows very slow dissolution kinetics in all classical disposal environments, which significantly contributes to the retardation of the radionuclides escape from the EBS. It is important to keep in mind that such characterization of spent fuel dissolution properties requires RD&D programs extending over significant time periods. The related Belgian experimental program was started in 2004 and basic conclusions allowing a first formal safety evaluation are expected by end 2013, which means that a decade was needed to confirm this major parameter (and its uncertainty) for the safety evaluations (the same applies for the other safety-relevant parameters). Note that for a license application, even more thorough and in-depth research is needed.

Once released from their respective emplacements in the waste, radionuclides will move out of the EBS and will become more or less mobile in the host rock. Groundwater is the main medium which can transport radionuclides by natural processes (like advection or diffusion) from the disposed waste position to the biosphere. The radionuclides transport processes will be governed by both the movement (velocity and paths) of groundwater through the rock and the specific behavior of the radionuclides as they interact with the host environment. This will depend on whether they are dissolved, in organic or inorganic form, transported as colloids, how they move through fractures and pores in the rock and are slowed down or immobilized by interaction with the rock components, etc.

Fig. 5 gives an example (Belgian SAFIR 2 concept [6][3]) of effective dose rates calculated for an individual exposed via a deep well, for the fission/activation products and for the actinides of UOX spent fuel in disposal. The highest calculated dose due to the fission and activation products is expected to occur beyond 100 000 years after disposal. As regards actinides, the peak dose is expected to occur around 10 million years (i.e., beyond a reasonable assessment timeframe).

Fig. 5 – Effective dose rate via a water well, calculated for UOX spent fuel (Belgian SAFIR2 disposal concept).



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[3] Calculations are being performed for the new Belgian concept [3]; preliminary results give similar global information.

4. Impact of the fuel fabrication and operation on the long-term repository safety.

The inventory in spent fuel of the contributors to the radiological impacts evolves both qualitatively and quantitatively with the as-fabricated characteristics of the fuel assembly and with its in-reactor operation conditions.

Utilities' approach aims at reducing the operating costs while maintaining high reliability and safety standards. Increase fuel performance, achieve higher burnups, minimize fuel failures etc., require developing new fuel designs with advanced materials for fuel, cladding and assembly structures as well as optimizing the reactor coolant properties (presence of additives for various purposes).

Using advanced materials for cladding and structures as well as making minor changes in the oxide fuel composition (like additives with some specific functions) could be challenging in terms of radiological risks as they could introduce new or notably higher contents of specific nuclides with low retention properties in the host rock. An example is given by the introduction of niobium in zirconium alloys for fuel rod claddings. The 94Nb activation product shows both a long half life and a significant mobility in clay, making it a critical nuclide for long term safety.
Note that amongst critical nuclides for long term safety, some are resulting from the activation of material impurities (an example is chlorine, producing 36Cl) for which only very conservative contents at fabrication (like upper tolerances) are available to waste managers. This may lead to notably over-predicted inventories and consequent over-predicted and unrealistic contributions to the radiological risk.

Besides classical UOX and MOX fuels, evolutionary fuel concepts are also being considered for thermal reactors, like those including "inert" matrices replacing the fertile 238U for hosting plutonium (or other transuranics), with the aim of stabilizing or burning down its inventory.
We already mentioned that the management of transuranics is not a key issue for long term safety. It will mainly concern scenarios of human intrusion into the disposal facility, whose probability is very low and whose impact is limited to the intruders themselves. It could however contribute to reduce the thermicity of the spent fuel, with an impact on the design and footprint of the disposal facility.
However, new matrices could not only introduce new materials (possibly leading to significant qualitative and/or quantitative changes in critical radionuclide productions), but also make lower contributions to the retardation of radionuclides escape from the EBS. New matrices would have to be analyzed in terms of their leaching resistance as well as their contribution to the IRF, thus necessitate extended RD&D.

The main factors that govern the IRF (i.e., the fraction of radionuclides positioned at accessible locations in the spent fuel assembly) are the fuel type (today, essentially UOX and MOX), the assembly power (temperature) history, the fuel burnup and, to a lower extent, the reactor coolant conditions, liable to affect the thickness and the composition of corrosion and crud layers developed at the surface of the assembly cladding and structure materials during in-reactor operation.

Changes in the fuel operating conditions could modify both the distribution of the radioactive inventory in the assembly and the fuel morphology (specific surface area), possibly increasing the fraction of fast-released radionuclides and the fuel leaching resistance.
Note that the RD&D work performed today on the dissolution rates of irradiated UOX fuel essentially addresses burnups below 55 GWd/tM (at the sampled pellet level). Significantly higher pellet burnups are expected for the future (up to 100 GWd/tM are already tested in commercial LWR).

Increasing the burnup of spent fuels will unavoidably increase their thermal load. In the Belgian concept for spent fuel disposal, the container surface temperature must be limited in order to not allow the development of localized corrosion. Significant evolution of the spent fuel thermal load could result in unacceptable temperatures and necessitate major modifications of the system design and/or footprint. Reduction of the fuel thermal load could also be reached by a substantial (above hundred years) increase of the intermediate storage time, with possible associated safety, security and societal issues.

5. Conclusions

Assessing and confirming the safety of a repository require the exhaustive knowledge of the major processes taking place in it and resulting from the various interactions between the spent fuel, the EBS and the host rock. The characteristics of a disposal system used as reference for a safety case are well defined and do not evolve within its framework, which might be not the case for the disposed materials. Revision of safety cases and licenses will be compulsory at each major change in the spent fuel characteristics.

Changes in materials and operating conditions of the nuclear fuel could necessitate additional RD&D which is time consuming and EBS/host rock dependent. Note that, most of the time, these changes occur much faster than the RD&D needed for supporting geological disposal safety cases. Without RD&D, however, conservative assumptions must be made in the safety analyses which may lead to a notable over-prediction of risks.

It seems obvious that some form of interaction between stakeholders in the various steps of the fuel cycle should contribute to optimize the management at all levels. Gaps in the knowledge of waste characteristics will unavoidably impact the risk predictions and consequently the cost of their management.

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ASSESSMENT OF FUEL DESIGNS FOR NEW COMMERCIAL NUCLEAR REACTORS WITHIN THE UNITED KINGDOM

J R JONES

Office for Nuclear Regulation

St James House, Cheltenham, GL50 3PR

ABSTRACT

The United Kingdom has recently started a process of licensing a new set of commercial nuclear reactor sites. This has not happened for twenty years and since that time, the industry has changed significantly. Previously, nuclear plants were built by a single government-owned body and individual sites were licensed by a government inspectorate.

The licensing practices have developed as a response to requirements, as detailed in this paper and changes have taken place in consultation between the power generating body and the licensing authority. This established relationship no longer exists. Potential power station sites are now owned by international utilities and designs are offered by reactor (and fuel) vendors from numerous countries.

Before the utilities commit themselves to a particular vendor design, they need to be confident that the design can be licensed in the United Kingdom. Both vendors and utilities are no longer necessarily familiar with United Kingdom licensing practices and principles and therefore the licensing authority has developed a pre-licensing design assessment process, in consultation with vendors, to provide confidence that UK requirements can be met.

The purpose of this paper is to illustrate the principles used within the United Kingdom to consider the suitability of safety cases supporting the licensing of operations at nuclear facilities. The application of these assessment principles to fuel designs is illustrated by considering some issues addressed during the recent Generic Design Assessment of the proposed fuel designs.

The United Kingdom safety case requirements fall broadly into two categories; a need to set operating rules, in the form of limits and conditions of operation, and a need to demonstrate that risks have been reduced to a level “As Low As Reasonably Practical”. These requirements are explained in this paper.

The issues used to illustrate the approach adopted by ONR include: the basis for protecting fuel against Pellet-clad Interaction; and the rational for the assessment against fuel limits in reactivity and loss-of coolant accidents.

1. Introduction

In the United Kingdom, current nuclear plants were built over an extended period by a single government-owned body and licenced by a government inspectorate. This established relationship no longer exists. Potential power station sites are now owned by international utilities and designs are offered by reactor (and fuel) vendors from numerous countries.

Before the utilities commit themselves to a particular vendor design, they need to be confident that the design can be licenced in the United Kingdom and therefore ONR has developed a pre-licensing design assessment process, in consultation with vendors, to provide confidence that UK requirements can be met [1]. In this context, ONR is not exercising its regulatory powers, but rather providing technical advice to the vendors on the licensing of the designs. The process has been completed for the fuel aspects two new designs and the findings are detailed in ONR assessment reports [2][3]. The purpose of this paper is to use this assessment as a way of illustrating UK regulatory requirements specific to the fuel and core.

The licensing arrangements in the United Kingdom were introduced in the late 1950s in response to an accident at a military facility in west Cumbria [4]. Legislation was introduced which required plant operators who handle nuclear material to apply for a site licence. The government established a Nuclear Installations Inspectorate to enforce compliance with the licence conditions and the Office for Nuclear Regulation (ONR) remains responsible for enforcement of the legislation to date.

The licence conditions relevant to fuel require: an *adequate safety case* justifying operations and *operating rules* defining the boundary of safe operation [5]. These requirements were originally addressed by performing *Deterministic* safety analysis to demonstrate safety margins between operation and plant damage. This includes transient analysis of Design-basis faults against fuel Design Criteria.

In subsequent years, a number of incidents outside the nuclear industry caused the UK government to form a larger regulatory body responsible for enforcement of health and safety legislation generally (The Health and Safety Executive). The associated legislation extended the duties of licencees; requiring them to consider whether additional measures to mitigate the risk inherent in their operations were reasonably practical [6]. This requirement has been reinforced by recent events.

The principles behind this approach are not unique to the UK and their impact on the design and licensing of new fuel and reactor cores is to some extent universal. This paper considers first the issue of Reasonably Practical safety enhancements and then discusses the application of Deterministic Analysis. Examples considered include the basis for protecting fuel against Pellet-clad Interaction and the rational for the assessment against fuel limits in reactivity and loss-of coolant accidents.

In making the decision on whether to sample a particular topic in detail and potentially to intervene, the regulator is required to follow a set of principles used for enforcement within the UK [7].

These require:

- Regulatory action be proportion to the risk;
- Consistency with other regulatory decisions;
- Transparency and accountability.

These principles are designed to ensure that assessment resources are correctly targeted and that licencees can have predictable interaction with the regulator.

In order to comply with these requirements, the inspectorate has issued guidance to its staff in the form of safety assessment principles [8] and technical assessment guides (for example [9]). The following discussion is based on this guidance.

2. The Test of Reasonably Practical

The UK law requires that the licencee consider measures that can be taken to eliminate or protect against a risk and to apply the test of whether the cost and trouble incurred is grossly disproportional to the incremental reduction in risk. Explanation of the thinking behind this approach is found in [10].

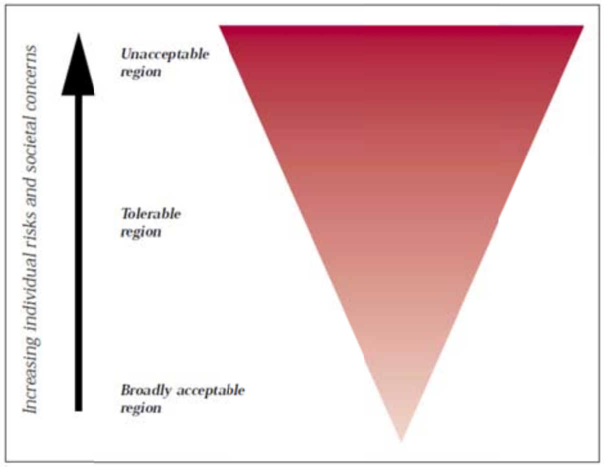

Fig 1. HSE Framework for the Tolerability of Risk

The framework is illustrated in Figure 1. The triangle represents increasing level of 'risk' for a particular hazardous activity (measured by the individual risk and societal concerns it engenders). As we move from the bottom of the triangle towards the top the need for mitigation is increased.

The dark zone at the top represents an unacceptable region. For practical purposes, a particular risk falling into that region is regarded as unacceptable whatever the level of benefits associated with the activity. Any activity or practice giving rise to risks falling in that region would, as a matter of principle, be ruled out unless the activity or practice can be modified to reduce the degree of risk so that it falls in one of the regions below, or there are exceptional reasons for the activity or practice to be retained.

The lighter region represents an area where the risk could be accepted but mitigation measures should be taken unless analysis of the balance between benefit and risk shows that it is not reasonably practical.

In developing a safety case, it is tempting to assign a value to the tolerable risk associated with a radiation dose; based on the risk of widely accepted in similar activities. However, ONR is likely to take a wider view. Risk can include consideration of the consequences of damage to trade and reputation. In a number of cases this has been the dominant risk [10].

The assessment of what is reasonably practical therefore becomes a qualitative rather than a quantitative process and the law regards relevant good practice as an illustration of an accepted balance.

An example of a consideration of fuel design against ALARP criteria occurs in the consideration of cladding failure in reactivity faults: A number of these faults can result in clad stress in excess of the threshold for stress-corrosion cracking in zirconium alloys, while simultaneously releasing reactive halide compounds from the fuel matrix. The resulting Pellet-Cladding Interaction (PCI) can lead to cladding failure.

PCI failures are likely to result in the release of volatile fission products into the primary circuit of the reactor, but in most fault sequences these will be contained and the main consequences would be contamination of the primary circuit and the fuel storage pond. It can be argued that some faults, occurring in intact primary circuits, would not require Design Basis analysis.

It may be possible to argue that the radiological consequences would not be sufficient in themselves to warrant substantial expenditure on safety measures. However, the general principals of good design practice (as set out in our safety assessment principles and IAEA standards documents [11]) require that the plant be designed so as to maintain the integrity of barriers to fission product release and ONR concluded that some protection was needed.

ONR is aware that changes in fuel pellet design proposed for the near future may increase safety margins for cladding stress. As a regulator, we will need to balance the need to encourage such changes with the need to see new designs well substantiated before adoption. This will inevitably involve the need for constructive dialogue between the regulator and the industry.

3. Deterministic Design-basis Safety Analysis

The overall philosophy underpinning a *Deterministic* safety-case limit is illustrated in Figure 2. The operating space consists of four regions:

> A region in which analysis has shown that there is a likelihood of significant damage to the plant;
>
> A region in which operation is likely to be benign, but which is too close to the physical limits of the plant to accommodate foreseeable changes and therefore can only be accepted in fault conditions;
>
> A region circumscribed by statutory limits for which operation may be permitted for a limited time while action is taken to restore normal operation; and
>
> Finally a region circumscribed by the normal operating limits in which operation is permitted.

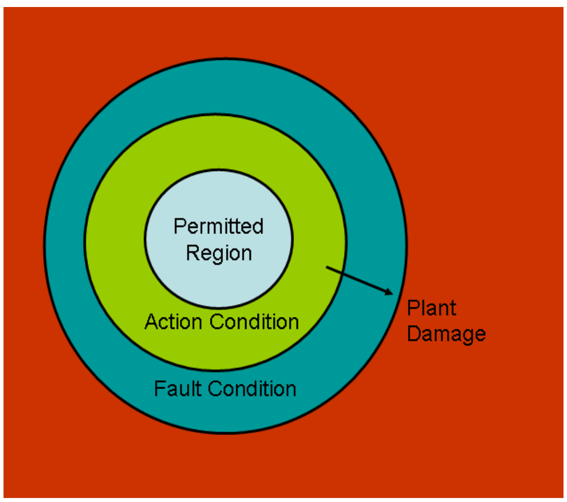

Fig 2. Target Model for Defining Operating Limits

Deterministic safety cases consider *reasonably foreseeable* fault conditions so as to identify the edge of the region of safe operation based on conservative assumptions. In principle, the transient analysis of these faults should be assumed to be initiated at the edge of a permitted action state and protection should ensure that the deviation in the plant state is arrested before the plant damage region is reached.

Fault studies are intended to be a robust determination of the adequacy of safety limits and therefore the analysis is required to assume that at least one of the safety measures intended to mitigate the fault fails to act. Moreover, design studies are required to take account of both random variation of properties and uncertainty in knowledge. The analyst is expected to apply the precautionary principle of assuming the worst conditions consistent with the current state of knowledge.

The appropriate allowance for uncertainty in modelling of operating conditions needs to take account of the magnitude of the hazard being protected and the likelihood of the event considered, but generally the level considered appropriate is a 95% probability at 95% confidence.

Often all parameters that can have a significant affect on safety margins are set to pessimistic values. However exceptionally, a selection of the most significant parameters is varied or the overall uncertainty is evaluated using a simulated response surface. In these cases, specific arguments are expected as justification and ONR will examine them. It will be necessary to demonstrate that correlation between the uncertain parameters has been adequately accounted for.

The use of more sophisticated analysis methods is usually an indication of limited safety margins and ONR may consider whether the reduction in safety margins is warranted.

It is often useful to retain a degree of margin for unforeseen events. For example, the assessment of the departure from nucleate boiling in a fuel assembly is generally based on a linear response surface taking into account the uncertainty in pressure and temperature and local power values, together with the uncertainty inherent in correlating the critical heat flux data. The method was originally developed by Owens [12], and determines a permissible ratio between the correlated critical heat flux and the predicted heat flux in the postulated transient. However, a good practice widely adopted is to include an allowance for unforeseen systematic effects in the permitted ratio. This practice has proven to be fortuitous because review of operational experience has identified a number of effects not included in the original analysis. In particular, the condition of the fuel in reactor can potentially influence the flow field; both through distortion of the design geometry and through changes in the hydraulic roughness.

Mitigation of the effects of assembly distortion has been subject to substantial amounts of work in the recent past, and ONR satisfied itself that both designs were addressing this issue adequately and were likely to provide fuel at least as good as current designs.

The topic of the effect of increased hydraulic roughness (induced by surface deposit) has received limited attention and therefore the ONR approach has been to ensure that measures are taken to provide confidence that the level of deposit will not be significant and that suitable surveillance measures will verify this.

The contingency margin used by the fuel designers has allowed ONR to accept existing safety analysis, provided that surveillance confirms the expected fuel performance.

4. Acceptance Criteria for Design Basis Faults

The dose resulting from a fault is generally assessed using conservative factors relating the number of fuel failures to the off-site release or on-site radiation exposure.

Two targets are apparent for each tranche of event frequencies: A dose level which is considered tolerable and one that is considered broadly acceptable.

In cases where risk is found to be tolerable, ONR would expect an optioneering study to determine whether anything could reasonably be done to reduce or eliminate the risk. The test being whether the risk is *As Low As Reasonably Practical* (ALARP).

If the risk was demonstrated to be broadly acceptable, the law does not exempt a licencee from considering whether the residual risk could reasonably be avoided, but the regulator is unlikely to focus effort on examining this aspect of the case, and would not generally expect the licencee to expend significant resource on this aspect of the design.

Often the simplest approach is to use fuel failure as a surrogate design limit. This has proved to be the case in control-rod ejection faults and consideration of this is a useful illustration of regulation in practice:

In recent tests, the possibility of fragmentation of high-burnup fuel has become evident and currently, the consequences of such an event are to a degree uncertain [13]. Rather than address the consequences, the practice recently adopted within the UK has been to set a limit on the peak radially-averaged fuel enthalpy at a level which will protect the fuel against cladding failure and hence prevent any release of fuel material. In this topic, the industry has benefited from work carried out by the US National Regulatory Commission [14].

In response to ONR queries, the fuel suppliers provided detailed analysis of the core response in such a fault and demonstrated that, for the control-rod insertion limits proposed, the fault is benign.

Since this fault is thought to have a low likelihood of occurring within the plant life, it would be tolerable to accept a small number of fuel failures as a consequence. However, there is the possibility that this could be avoided by a modest change to the control rod insertion limits, so ONR has requested that the designers consider whether this is reasonably practical. It is possible that a slightly tighter constraint on control rod insertion would avoid failures without affecting the ability to control the axial xenon distribution.

In circumstances of this kind, a judgement on the adequacy of a safety case is generally reached in consultation between specialist ONR inspectors and those responsible for the overall project within ONR.

5. Extent of Deterministic Analysis

In the UK, identification of *reasonably foreseeable* Design Basis faults is the responsibility of the licence holder. However, ONR do provide guidance to help limit the scope of the task [8]: The analyst is required to consider the likelihood of the fault and the magnitude of the hazard.

Faults with a return frequency of less than once in 100,000 years, and faults where the unmitigated dose would be insignificant, can be excluded. Quantitative guidance is provided [8] and this is illustrated for off-site dose in Figure 3.

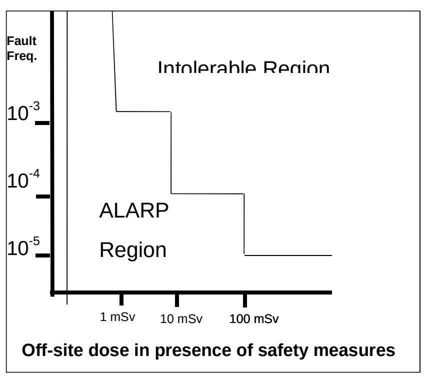

Fig 3. ONR Target Levels of Off-site Exposure from Postulated Accidents

The criterion for tolerability varies with the frequency of the fault; increasing as the fault frequency is reduced.

Measures taken to reduce the likelihood of a fault so that it falls outside the return-frequency targets, permit the analyst to relax the conservative assumptions used in analysis of a fault, but this does not necessarily justify neglecting the fault entirely. ONR would still require consideration of whether the risk from the fault was reduced to levels as low as reasonably practical and in some cases, this has required substantial amounts of work. To illustrate the judgements required, it is instructive to consider the Large-break Loss-of-coolant Accident (LBLOCA) where the reactor coolant depressurises rapidly and the reactor vessel is emptied.

Modern non-destructive inspection techniques made it possible for one of the suppliers to argue that for their design, LBLOCA need not be included in the Design Basis. However, ONR took the view that it's mitigation still needs to be within the design capability. This is because the unmitigated risk from such a fault would probably dominate the plant risk for proposed new designs. Measures to mitigate the fault are required and analysis is necessary. However, it is reasonable to do this on a better-estimate basis.

The fault provides multiple challenges for the fuel:

- If conventional conservative analysis of the shock wave associated with a prompt appearance of a displaced guillotine fracture is considered, then impact loadings on the fuel assembly spacer grids are likely to cause some buckling; reducing the flow area in some locations within the assembly. This shockwave can be mitigated by fitting restraints to primary pipework to limit pipe movement, but the restraints would impede inspection of the pipework. ONR accepted that if alternative arguments could be sustained, then no restraints would be required. One supplier followed the approach of demonstrating that more realistic analysis of the shockwave would confirm adequate spacer-grid strength. The other showed that fuel temperatures would remain low and some buckling would be tolerable. Both approaches were considered acceptable in principle.
- If the stored energy in the fuel pins is sufficiently high, then the fuel can exceed the critical heat flux early in the fault transient so that less of the energy is extracted during blow down of the primary circuit. This can lead to overheating of the fuel pins and potentially ballooning failures. Significant amounts of work have been done on fuel-clad ballooning in the past to demonstrate that a coolable geometry is maintained, but arguments still need to be made to demonstrate that this work remains applicable to new fuel and core designs. In the new reactor designs, extra margin can be had by optimising the rate at which the upper head blows down; to extend the period of useful two-phase cooling. One supplier was able to demonstrate that fuel temperatures would remain below levels likely to soften the cladding. The other relied on an argument that not withstanding some ballooning, the fuel would remain coolable. ONR used the detailed fuel ballooning model MATARE [15] to confirm that coolability was likely [16]. The analysis indicated a reducing level of flow blockage as heat flux increased and confirmed that for the LBLOCA transient with minimum levels of available safety injection, ballooning would not have an unacceptable effect on peak fuel temperature. This analysis indicated that fuel bundle blockages would be similar to those calculated for previous experiments e.g. [15] (illustrated in Figure 4) and allowed sufficient flow to limit fuel damage. ONR judged that this aspect of the design did not merit further regulatory action.

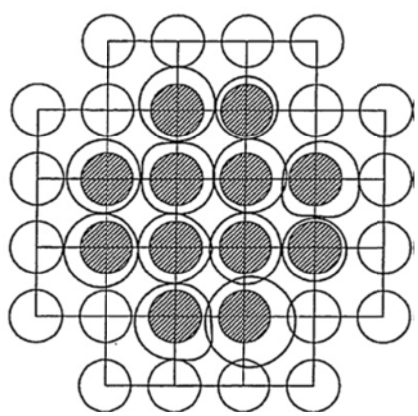

Fig 4. Calculated Flow Blockage for the MT3 experiment

- Late in the LBLOCA transient, it is necessary to realign the safety-injection pumps to extract water from the containment sump. In this case, there is a risk that debris from the sump will be carried into the reactor vessel and will restrict flow through the fuel. This is addressed by a complex system of sump screens and measures to limit the use of fibrous insulation within containment. ONR has requested further confirmation of the effectiveness of the proposed design and this remained an outstanding issue.

- Finally, it is recognised that not all events that might lead to an accident can be anticipated and measures are considered for circumstances where all installed safety systems have been defeated. This is an underling issue in the considerations being pursued as part of the response to the Fukushima accident.

In general, the development of a satisfactory safety case that balances the requirements of safe and economic power generation, is best achieved by a constructive dialog between the regulator and the potential licencee.

6. Conclusion

The intention of the system of regulation set out above is not to provide a prescriptive set of steps by which utilities can meet regulatory requirements, but a set of flexible guidelines which allow utilities to operate safely and to engage constructively with the regulatory body.

The aim is to provide a robust demonstration that the plant can meet the challenges presented by anticipated faults and that all reasonably practical measures have been taken to reduce the risk to a broadly acceptable level.

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European Nuclear Society
Rue Belliard 65
1040 Brussels, Belgium
Telephone: +32 2 505 30 50 - FAX: +32 2 502 39 02
topfuel2012@euronuclear.org
www.topfuel2012.org