Safety Assessment for Decommissioning of Research Reactors
International Project on Evaluation and Demonstration of Safety during Decommissioning of Nuclear Facilities (DeSa)

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Abstract. For the last three years about fifty experts from over thirty countries have been participating in the international IAEA project "Evaluation and Demonstration of Safety during Decommissioning of Nuclear Facilities (DeSa)". Their work has been focused on (i) the establishment of a harmonized safety assessment methodology for decommissioning; (ii) the development of recommendations for a regulatory review of such safety assessments; (iii) the development of recommendations on the application of a graded approach to performance and review of safety assessments, which ensures that the extent of the safety assessment is commensurate with the risks posed by the facility and the proposed decommissioning activities, and finally (iv) the application of the DeSa methodology to three test cases – inspired by actual facilities – with different complexities and hazard potentials – a nuclear power plant, a research reactor and a nuclear laboratory.
This paper provides an overview of the objectives and current status of the DeSa project activities, and their application to the development of safety assessment for decommissioning of a research reactor. The outcomes and lessons learned from the development and the review of the Research Reactor Test Case, together with the remaining five DeSa reports will be presented and summarized at the 4th Joint DeSa meeting in October 2007, where a potential follow-up project will be also discussed.

1. Introduction

In line with the International Action Plan on Decommissioning of Nuclear Facilities [1] in 2006 a new IAEA Safety Standard was published [2], that provides a set of key requirements for ensuring that decommissioning of facilities using radioactive material, thus not only nuclear facilities, will be conducted in a safe, adequate and reliable manner. For that, decommissioning shall be based on a decommissioning plan, which is supported – inter alia – by an appropriate safety assessment adressing occupational exposure, potential on- and off-site releases and related exposure of the public. During development of the decommissioning plan and its supporting documents the concept of graded approach shall be applied, in order to allow the presentation of balanced information and level of detail commensurate with the type and status of the facility and the hazards associated with the decommissioning of the facility. Thus, applying the graded approach allows optimization of the efforts needed for preparing for, performing and regulating decommissioning without jeopardising the safety.

At present a new Safety Guide DS 376 [3] is at a final stage of preparation to advise Member States on complying with the new Safety Reuirement [2] and preparing and reviewing safety assessment for decommissioning. This Safety Guide provides recommendations on how to perform a safety assessment and gives a methodology tailored to the special situation of facilities under decommissioning. Strong emphasis is herein laid on the application of the graded approach during safety assessment, thus helping operators and also regualtory bodies to optimize their effort and to focus on most relevant safety aspects of decommissioning. It is expected that this draft standard will be published in 2008.

One of the main contributions to the development of the draft Safety Guide DS376 was provided by the international project on *Evaluation and Demonstration of Safety during Decommissioning of Nuclear Facilities (DeSa)* [4], [5], [6] launched by the International Atomic Energy Agency (IAEA) in 2004. The main objectives of the project were to develop practical recommendations on how to harmonize safety assessment approaches and to establish an international forum for sharing lessons learned in this field. Since November 2004 about fifty experts have been working in several working groups [4] on:

- Establishing a harmonized and detailed safety assessment methodology for decommissioning,
- Developing recommendations for a regulatory review of such safety assessments and
- Development of recommendations on the application of the graded approach, which can be applied during both performance and review of the safety assessments for decommissioning.

In order to illustrate the application of the methodology and to demonstrate that the proposed methodology is fit for purpose, three real facilities from ongoing and planned decommissioning projects were selected for the purposes of the DeSa project. Reference safety assessments were performed for a nuclear power plant, a research reactor and a nuclear laboratory using the DeSa methodology. The selection took into account facilities of different type and complexity to ensure different levels of detail of the safety assessment.

With regard to this selection, it must be mentioned that research reactors belong to a wide group of facilites – some less and some more complex; due to the variety of the different types of research reactor there will be research reactors which can be regarded to be close to the complexity of a nuclear power plant. Thus, lessons learned on safety assessments for research reactors can be drawn not only from the research reactor test case, but also from the other two test cases of the DeSa project.

A short overview on the safety assessment and graded approach methodologies is first given in the following Section 2 and 3. The implementation of these methodologies to the Research Reactor Test Case as well as a description of the choosen research reactor, are described in Section 4.

2. Overview on the Safety Assessment Methodology

The DeSa project explored the experience of Member States in the development of safety assessment for various facilities under decommissioning. On this basis the project has recommended a harmonized approach that has also been refelected in the draft IAEA Safety Guide DS 376 [3]. The safety assessment approach consists of a sequence of seven main steps as presented in Fig. 1. These steps may be repeated in an iterative manner, depending on the compliance of the safety assessment results with the relevant requirements and criteria.

The safety assessment should be based on a defined framework by which all prerequisites, such as the scope and objectives of the assessment are clearly defined, followed by a description of the facility and the decommissioning activities, as defined in the decommissioning plan. These should be used to identify existing and potential hazards inherent to the facility and new hazards arising from the nature of the decommissioning activities to be undertaken. The relevant hazards are further quantified and associated consequences to workers and public evaluated, followed by engineering analysis of the safety relevant systems, structures and components. The resultant effective doses and risks associated with these hazards should then be compared with the relevant safety criteria, which are prescribed by the national legislation, to determine whether these criteria are met. On this basis the set of safety controls proposed to be applied during decommissioning needs to finalised.

Finally, the analysis and its results should be subjected to a review, independent from the developer but still in the responsibility of the operator, in order to provide confidence in the assessment methodology, data used, assumptions made, results obtained, conclusions and recommendations drawn. After that the safety assessment may be submitted to the responsible authority for regulatory review.

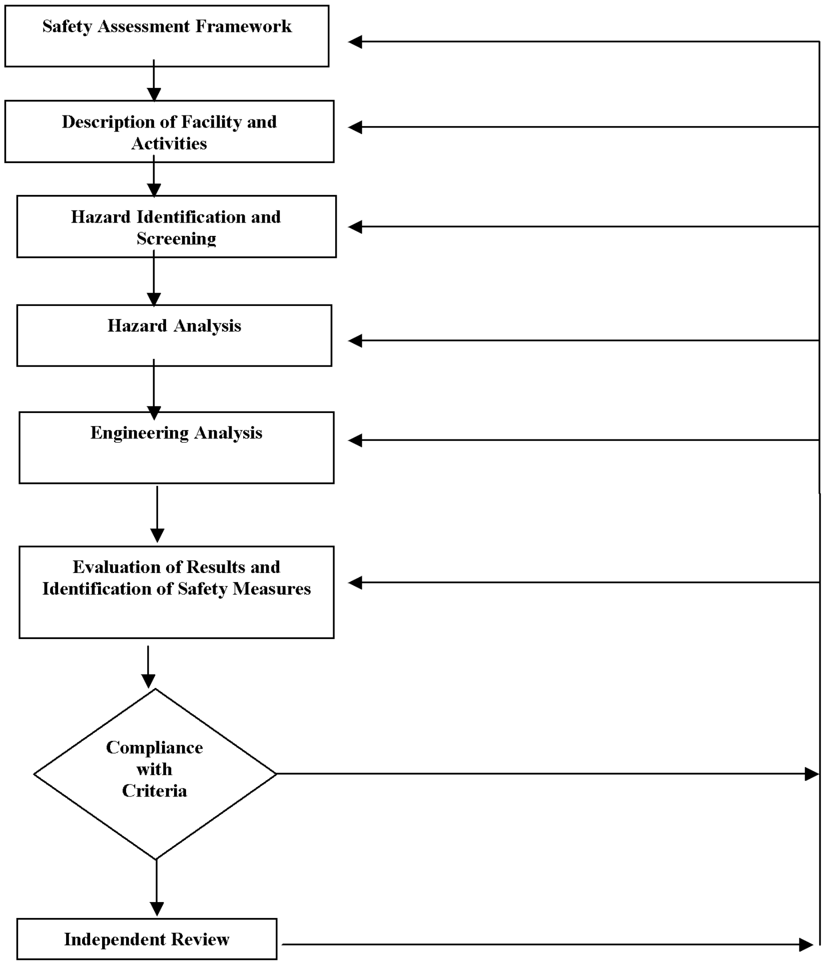

Fig. 1.

During the DeSa project this methodology was described in more detail to provide help in its implementation [7].

3. Overview of the Graded Approach

Special recommendations were elaborated during the DeSa project on how to apply the graded approach to ensure that the effort and resources put in the development and review of safety assessments for decommissioning are devoted to those aspects relevant to safety. A definition of graded approach within the safety assessment was developed as follows ([7]):

"Graded Approach with respect to safety assessments for facilities undergoing decommissioning means a process by which the level of detail of the analysis, the complexity of the approach, the documentation, and other issues necessary to demonstrate compliance with legal requirements and safety requirements are commensurate with:

(1) *The magnitude of any hazard (radiological or non-radiological) involved, associated with the facility or the work to be carried out,*
(2) *The particular characteristics of a facility, e.g.*
  - *type, size, source term, unique features and*
  - *level of ageing, safety culture,*
(3) *The requirements/demands by the regulator,*
(4) *The step within the decommissioning process, including decommissioning strategy (deferred – immediate decommissioning) and*
(5) *The balance between radiological and non-radiological hazard(s)."*

4. The Research Reactor Test Case – Overview and Results

As part of the DeSa project a working group was formed, which tested the proposed methodology on the safety assessment and the graded approaches by applying them to a volunteered research reactor. The research reactor was selected to represent a nuclear facility of small size and complexity in contrast to a nuclear power plant which was to represent the most complex facility.

In order to develop a realistic test case, the characteristics of a real research reactor were taken as an input for the Research Reactor Test Case. As far as deemed necessary assumptions were made to simplify the test case or to fill gaps of information. Special emphasis was made to ensure, that a consistent and realistic decommissioning project became subject of the test case.

The reference safety assessment for the research reactor, including the description of the facility, hazards, etc. are summarized in the DeSa Safety Report [7]. Depending on national requirements in a real decommissioning project, all this information may be contained in a decommissioning plan and the set of documents supporting it including the safety assessment report or may be gathered in one document.

4.1. Decommissioning of the Research Reactor – A Brief Overview

a. Description of the research reactor

Subject to the Research Reactor Test Case of the DeSa project is a homogenous liquid fuel research reactor of a maximum thermal output of 2.3 kW. At a thermal power of 2.0 kW the neutron flux at the central beam channel was $4 \cdot 10^{10} n/cm^2/s$. Fig. 2 shows a vertical cross section of the research reactor. The reactor was operated for 44 years and shutdown for decommissioning in 2001. The total generated energy was about 0.5 MWd. Within the last 20 years of operation the reactor was operated at a thermal power of 100 W.

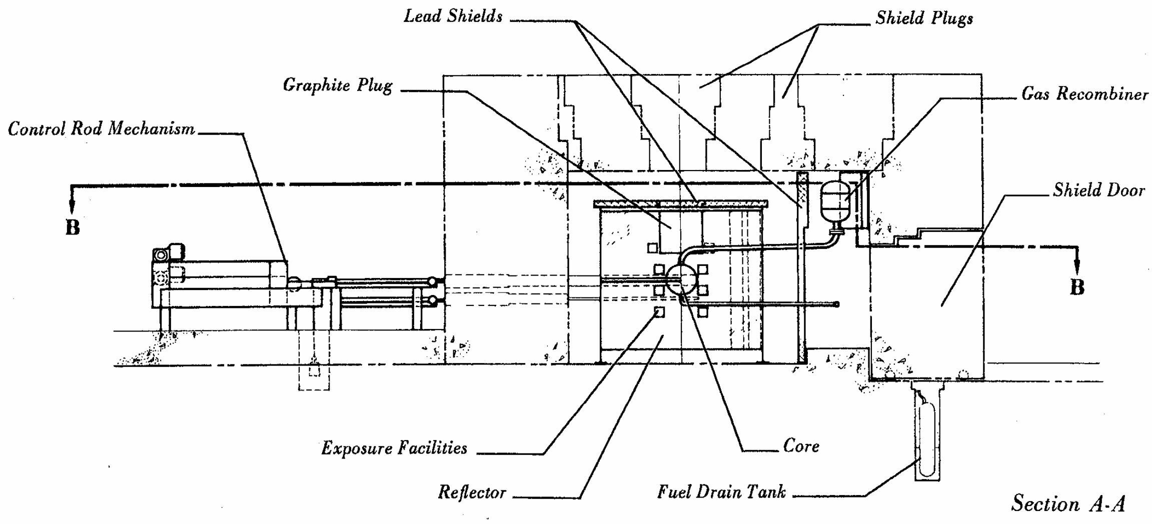

Fig. 2. Vertical cross section of the reactor

The research reactor was operated with a liquid fuel solution of 15.5 litres of uranyl sulphate dissolved in light water. The U-235 was enriched to 19.9 % with a total mass of 0.984 kg dissolved in the fuel solution. The primary core systems consisted mainly of a spheric stainless steel core vessel and a recombiner system to recombine hydrogen generated during operation. Both components were connected by pipes. The core vessel was surrounded by graphite stringers, which were nested in the cylindrical reflector tank inside the hexagonal biological shield made of concrete. Additional components, including a Ra-Be source were used to maintain the research reactor and to steer the reactivity.

The biological shield was errected in a reactor hall which is located at a larger site with other nuclear and non-nuclear facilities

b. Description of the decommissioning of the research reactor

After final shutdown and before the beginning of any decommissioning activities the liquid fuel was tapped off according to operational procedures. The neutron source was removed and disposed of and the core system was decontaminated by flushing with demineralised water. All these activities were performed under the terms and conditions of the operating license.

Subject to the safety assessment was the decommissioning of the research reactor and the preparation of the reactor hall for release for unrestricted use. The conduct of clearance measurements and the formal process of release of the reactor hall is not subject to this test case decommissioning project but to a follow-up project.

Main decommissioning steps are as follows:

- Removing of the recombiner system,
- Dismantling of control and safety rods,
- Dismantling of graphite stringers, core vessel and reflector tank,
- Dismantling of the other parts of the core system,
- Dismantling of the cooling system,
- Demolition of the biological shield,

— Cleaning and removal of radiological contaminations (if any) of the reactor building in preparation of the later clearance measurements.

The working steps are gathered in four working packages, which will be conducted in a sequential manner. The duration of the decommissioning activities is 17 months.

Only conventional tools are used for dismantling activities and no thermal cutting is foreseen. The cutting of the biological shield will be done by using dry wire cutting.

Existing contaminated or activitated components and other radioactive material are collected in waste containers which are transferred to other facilities at the site for further decontamination, storage, clearance or disposal. As these facilities are licensed accordingly, the scope of the safety assessment of this decommissioning project ends with the departure of the waste container from the reactor hall.

With respect to the decommissioning activities the radiological inventory can be regarded as low. After removal of the fuel solution and of the neutron source and after decontamination of the primary core systems the inventory is about 4 GBq mainly due to Cs-137 contaminations of the core vessel and the recombiner system and due to Co-60 and Eu-152 activations of other components. In addition alpha contaminations of less then 50 MBq were detected in the core vessel. Further activations of e. g. control rods, beam plugs, graphite reflector or the concrete of the biological shielding can – in agreement with the results of the safety assessment performed – be regarded of less importance to the safety during decommissioning.

The dose rates of the individual components are in general low, maximum values were detected for the core vessel (2 mSv/h at contact) and for the recombiner system (1.5 mSv/h at contact). No contaminations in the reactor hall except for one location due to some spills of radioactive liquids during operation are determined during the radiological characterization of the reactor nor are any contaminations outside the reactor hall expected and determined.

To ensure safety during decommissioning several protective measures are implemented. They consist of engineered systems (e.g. air monitoring systems, temporary housings and local ventilation systems during dismantling activities), of personal protective equipement and of procedural controls addressing preparation of work activities, control of work activities and specific tasks and to ensure that personal protective equipment provides the protection intended to deliver.

4.2. Conduct of the Safety Assessment – Examples for Grading

The safety assessment for the decommissioning of the research reactor was conducted according to the methodology presented in Section 2 and illustrated in Fig. 1:

(1) *Safety Assessment Framework*
As explained in section 2 this step is devoted to the collection of all information relevant for the conduct of the safety assessment. With respect to the Research Reactor Test Case, information about the context, objectives and scope of the safety assessment were given, as well as a description of the timeframes and the proposed end state of the decommissioning. In addition, the requirements and criteria to comply with, the safety assessment approach and the relation to operational safety assessment were explained.

As an example of grading, not all detailed requirements and criteria, which are relevant for complex decommissioning projects, shall be applied for the test case. Again due to the low radiological inventory simple assumptions on the climate at the site were made, instead of a detailled site specific investigation, which – depending on national requirements – might be required for a complex decommissioning project with high mobile inventory. As no liquid radioactive material (sufficient in volume) was handled during decommissioning and no groundwater or soil contamination were detected under and around the reactor no hydrological analysis was needed.

(2) *Description of Facility and Activities*
The research reactor, the site and the related decommissioning activities were described to such a detail appropriate for the radiological hazards and planned decommissioning activities and objective of the safety assessment. The reactor was briefly presented in Section 3.1. In real decommissioning projects the desription of the facility can be included in the decommissioning plan or in a detailed supporting document – depending on the national requirements.

The description in the Research Reactor Test Case took into account the graded approach: The level of detail of the description of the research reactor allowed an understanding of the decommissioning activities, but did not provide instructions on how to operate the research reactor. As another example and in line with the grading during the assessment framework no detailed description of the climate was provided. As the conduct of the clearance measurements of the reactor hall was not subject to the test case decommissioning project but to a later follow up, related descriptions on how to perform the clearance measurements were not part of the descriptions (and consistently no requirements and criteria are mentioned in the assessment framework). As no hydrological analysis was required not related descriptions were provided.

(3) *Hazard Identification and Screening*
In preparation of the hazard analysis, a hazard identification and hazard screening, i.e. the prelimiary analysis, was performed. The hazard identification was performed using a check list and the "What-If-Technique", as suggested by the DeSa methodology. This approach is appropriate to the radiological inventory, while e.g. the application of the technique of a hazard and operability study (HAZOP) is regarded to be oversized.

Based on the identified hazards, the relevant scenario for the workers and for the public under normal and incident conditions was determined. The corresponding doses were estimated using simple and conservative models for calculation.

As a result, under normal condition the relevant scenario for the worker was the removal of the recombiner and the core vessel, while for the public any discharge with air would have been far below the relevant criteria. Among the incident scenarios, the drop of the core vessel with resulting spil of alpha contaminations was the significant one, while for the public the significant scenario was a release of radioactive material due to fire. All other scenarios were expected to have no relevant radiological impact to workers and the public.

(4) *Hazard Analysis and Engineering Analysis*
As a feature of the methodology and at the same time as an example of grading only the identified significant scenarios were assessed in more detail. Again as an example of grading the selection of the models and the level of detail during calcuation were selected corresponding to the radiological inventory and the dose estimates of the preliminary analysis. Doing so, simple assumption e.g. on the dietary behaviour of the population in the vicinity of the facility were made.

Engineering measures were foreseen (see Section 4.1) in order to reduce the exposure of the workers. As part of the engineering analysis the measures to operate and maintain the related systems were described to provide evidence that the related safety functions will be available at the time needed. In addition, as they significantly contribute to the protection of the workers against inhalation the failure of dust masks as additional failure during the significant incident situation was analysed and the related doses were estimated.

(5) *Evaluation of Results and Identification of Controls*
The results of the hazard analysis on the exposure of the workers and of the public were

compared with the related criteria. Neither under normal nor under incident situation any dose limits were exceeded:

- The maximum effective dose for a worker, who is involved in all dose relevant decommissioning activities, is below 0.6 mSv for the whole decommissioning activities, i.e. for a period of 17 months. In case of the most conservative incident scenario the 50 years committed dose of a worker due to incorporation of spilled alpha contamination from inside the core vessel will be less than 25 mSv/a assuming the unlikely case that the foreseen personal protective equipement will provide no safety function.
- The dose of the most exposed individual of the public during normal scenario will be far below 0.1 mSv/a for the first year and below 0.01 mSv/a after completion of the decommissioning activities. In case of an accident the 50 years committed dose is far below 0.2 mSv/a.

Emphasis was also given to existing uncertainties influencing the safety assessment. With respect to the Research Reactor Test Case, the C-14 inventory of the graphite is not known but – as an example of grading – is was conservatively estimated on the basis of data orginating from another research reactor instead of requiring a dedicated chemical analysis.

To ensure that all criteria will have been met during the conduct of the decommissioning activities safety controls were identified during the safety assessment. They will have to provide the relevant safety functions and consisted of engineered systems like air monitoring systems and temporary covers to be used during dismantling of the biological shielding, and of procedural controls e.g. to prepare and control work activities or to ensure that personal protective equipment will protect the worker during decommissioning activities.

As an integral part of the safety assessment methodology a review of the safety assessment by an independent reviewer was perforemd by the Regulatory Review and the Graded Approach Working Groups of the DeSa project. The recommendations are being reflected in a revised version that will be presented and finalised at the 4th Joint DeSa meeting, from 29 October to 2 November 2007 in Vienna, Austria.

5. Summary and Outlook

The international project on *Evaluation and Demonstration of Safety during Decommissioning of Nuclear Facilities (DeSa)* was launched in 2004 to elaborate in detail a harmonized methodology for safety assessment. The methodology was developed and reflected in the new draft Safety Guide DS 376. Special emphasis has been placed on the elaboration of the recommendations on the application of the graded approach to optimize the effort of operators and regulatory authorities during conduct and review of the safety assessments and their results.

The recommendations on safety assessment methodology and graded approach were applied to three test cases in the DeSa project among which a real world research reactor was selected as an example of a facility with relatively low complexity. The Research Reactor Test Case showed that the methodology and the graded approach can be applied successfully.

The outcomes of the DeSa project are planned to be finalised by the end of 2007 as planned. The six DeSa reports were presented and lessons learned and conclusions were discussed at the 4th joint meeting of all DeSa participants, which was held in Vienna recently. On this basis the reports will be published in an IAEA Safety Report in 2008.
The second objective of the meeting was devoted to a future follow up project of DeSa: The success of the DeSa project is mainly based on the ability of the Member States of the International Atomic Energy Agency to provide a forum and opportunity for exchange of national experiences in decommissioning and to investigate common approaches to ensure safety during decommissioning.

The International Atomic Energy Agency appreciates this experience exchange among the Member States and therefore calls for proposals for a follow up project of DeSa and all Member States are encouraged to submit appropriate proposals based on their experience.

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