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Regulatory Approaches for Addressing Reprocessing Facility Risks: An Assessment

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ABSTRACT

This report addresses methods for assessing the risks posed by a reprocessing facility, which have not previously been quantified relative to other fuel-cycle facilities. Reprocessing facilities can have higher potential source terms than other fuel-cycle facilities, which heighten the risk relative to the other facilities. This report explores the potential hazards that these facilities pose to the public, workers, and the environment by discussing literature on the regulation of these facilities and reviewing the experience of current operating facilities worldwide. It offers an overview of actual events and their consequences at these facilities. It also contains supporting information for assessing the feasibility, advantages, and disadvantages of undertaking detailed versus simplified quantitative risk assessments, for the range of events associated with large reprocessing facilities. The report gleans insights on regulating reprocessing hazards and risks from reports such as NUREG-1909 [Croff et al. 2008], and a white paper from the Nuclear Energy Institute [NEI 2008].

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EXECUTIVE SUMMARY

In 2008 AREVA NC Inc. [AREVA 2008] and other industry entities indicated to the Nuclear Regulatory Commission (NRC) their interest in commercial reprocessing in the U.S. In December 2008, the Nuclear Energy Institute submitted its white paper on reprocessing regulations [NEI 2008]. As a result, the NRC began assessing the need to expand its current regulatory regimes under Title 10 of the Code of Federal Regulations (CFR) Part 70 to encompass large spent fuel reprocessing facilities. In a series of SECY documents and their associated Staff Requirements Memoranda (SRMs), the NRC updated its planning for revising the regulatory framework for regulating reprocessing facilities. In SECY-09-0082 [NRC 2009] the NRC staff presents a gap analysis, as required by the Commissioners, associated with the development of the new framework. The staff presented twenty-three gaps in the current regulatory structure in SECY-09-0082 along with their descriptions, and priority.

This report relates to Gap #5, described in the enclosure to SECY-09-0082, that is concerned with risk considerations for a production facility licensed under 10 CFR Part 70. The risk assessment required by Part 70 involves an Integrated Safety Analysis (ISA) with a characterization of the likelihood and consequences of credible accident sequences. SECY-09-0082 notes that the existing requirements do not adequately address the increased risk posed by a reprocessing facility relative to that of other fuel-cycle facilities. Furthermore, it points out that reprocessing facilities can have higher potential source terms than other fuel-cycle facilities, which may heighten the risk of the former facilities.

A suitably performed risk assessment of a reprocessing facility potentially can characterize the associated risks of concern adequately. This is the general sentiment of the NRC Advisory Committee on Nuclear Waste and Materials (ACNW&M) (see NUREG-1909 [Croff et al. 2008]). SECY-09-0082 comments on the need to revise 10 CFR 70 to adequately address the unique hazards and risks related to these facilities.

Fuel-cycle facilities are distinguished from power reactors mainly by the diversity of their strongly interrelated inherent hazards, and by the large distribution and mobility of the hazards throughout the plant. A comprehensive identification and quantification of initiating events and scenarios is a challenge when performing a fully integrated Probabilistic Risk Assessment (PRA) for these facilities. A very consistent effort is deemed necessary to provide a realistic, accurate quantification of the risk; significant uncertainties generally are expected in the results.

Beginning in mid-2009, under contract with the Office of Nuclear Regulatory Research (RES), Brookhaven National Laboratory (BNL) prepared this report which provides information to support NRC's assessment of the feasibility, advantages and disadvantages of conducting detailed quantitative vs. simplified qualitative risk assessments for the range of accidents associated with reprocessing. For this report, BNL 1) explored the potential hazards that large reprocessing facilities pose to the public, workers, and the environment, 2) searched the literature for reports and other documents related, in particular, to risk assessments conducted for such facilities, 3) reviewed the experience of current operating facilities for example, in Japan, France, U.K., and elsewhere, and, 4) gleaned insights on regulating reprocessing hazards and risks from reports such as NUREG-1909 [Croff et al. 2008], and a white paper from the Nuclear Energy Institute (NEI) [2008].

This report is the product of a short-term, limited-scope study and did not fully survey the range of material existing in this subject area. Moreover, because some information was proprietary, some of the body of literature was unavailable. Nevertheless, enough material was available from which to make sound observations and offer supportive insights.

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ACRONYMS AND ABBREVIATIONS

ACNW&M	U.S. NRC Advisory Committee on Nuclear Waste and Materials
ACRS	Advisory Committee of Reactor Safeguards
AIChE	American Institute of Chemical Engineers
ANL	Argonne National Laboratory
BDC	Baseline Design Criteria
BNL	Brookhaven National Laboratory
BSL	Basic Safety Limits
BSO	Basic Safety Objectives
CFR	Code of Federal Regulations
DOE	U.S. Department of Energy
ER	Electrorefiner
EPA	U.S. Environmental Protection Agency
FCF	Fuel Cycle Facilities
FINAS	Fuel Incident Notification and Analysis System
FMEA	Failure Mode and Effects Analysis
GDC	General Design Criteria
HAZOP	Hazard and Operability Analysis
HSE	U.K. Health and Safety Executive
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
INES	International Nuclear and Radiological Event Scale
INPRO	International Project on Innovative Nuclear Reactors and Fuel Cycles
IROFS	Items Relied on for Safety
ISA	Integrated Safety Analysis
JAEA	Japan Atomic Energy Agency
LOPA	Layers of Protection Analysis (LOPA)
LWR	Light Water Reactor
MFFF	Mixed Oxide Fuel Fabrication Facility
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NFCIS	Nuclear Fuel Cycle Information System
NMED	Nuclear Materials Events Database
NRC	U.S. Nuclear Regulatory Commission
NRNF	Non-Reactor Nuclear Facility
NSC	Japanese Nuclear Safety Commission
OECD	Organisation for Economic Co-operation and Development
PHA	Process Hazard Analysis
PRA	Probabilistic Risk Assessment (or Probabilistic Risk Analysis)
PSA	Probabilistic Safety Assessment
PUREX	Plutonium-Uranium Recovery by Extraction
RRP	Rokkasho Reprocessing Plant
SAP	Safety Assessment Principles
SNF	Spent Nuclear Fuel
TAG	Technical Assessment Guide
THORP	Thermal Oxide Reprocessing Plant
TRP	Tokai Reprocessing Plant
TRU	Transuranic
UREX	Uranium Recovery by Extraction

1. INTRODUCTION

1.1 Objectives

In 2008, the Nuclear Regulatory Commission (NRC) began assessing the need to expand current regulatory regimes under Title 10 of the Code of Federal Regulations (CFR) Part 70 to encompass large spent fuel reprocessing facilities. Beginning in mid-2009, under contract with the Office of Nuclear Regulatory Research (RES), Brookhaven National Laboratory (BNL) prepared this report which provides information to support NRC's assessment of the feasibility, advantages, and disadvantages of conducting detailed quantitative vs. simplified qualitative risk assessments for the range of accidents associated with reprocessing. For this report, BNL 1) explored the potential hazards that large reprocessing facilities pose to the public, workers, and the environment, 2) searched the literature for reports and other documents related, in particular, to risk assessments conducted for such facilities, 3) reviewed the experience of current operating facilities for example, in Japan, France, U.K., and elsewhere, and, 4) gleaned insights on regulating reprocessing hazards and risks from reports such as NUREG-1909 [Croff et al. 2008], and a white paper from the Nuclear Energy Institute (NEI) [2008]. This report documents the observations of this study and supporting information.

This report is the product of a short-term, limited-scope study. It did not fully survey the range of material existing in this subject area. Moreover, because some information was proprietary, some of the body of literature was unavailable. Nevertheless, enough material was available from which to make sound observations and offer supportive insights.

The main observations from the international activities are limited and somewhat general. Accordingly, Section 1.3 gives a concise pertinent overview.

1.2 Background

In response to government- and industry-initiatives over the past few years, the NRC has been considering revising its regulatory structure for spent-fuel reprocessing. In a series of SECY documents and their associated Staff Requirements Memoranda (SRMs), the NRC updated its planning for revising this regulatory framework. In SECY-09-0082 [NRC 2009], along with their priority, the NRC staff presents a revised gap analysis, as required by the Commissioners, associated with the development of the new framework. The staff presented twenty-three gaps in the current regulatory structure in SECY-09-0082.

This report relates to Gap #5, described in the enclosure to SECY-09-0082, that is concerned with risk considerations for a production facility licensed under 10 CFR Part 70. The risk assessment required by Part 70 involves an Integrated Safety Analysis (ISA) with a characterization of the likelihood and consequences of credible accident sequences. SECY-09-0082 notes that the existing requirements do not adequately address the increased risk posed by a reprocessing facility relative to that of other fuel-cycle facilities. Furthermore, it points out that reprocessing facilities can have higher potential source terms than other fuel-cycle facilities, which may heighten the risk of the former facilities.

A suitably performed risk assessment of a fuel reprocessing facility potentially can characterize the associated risks of concern adequately. This is the general sentiment of the NRC Advisory Committee on Nuclear Waste and Materials (ACNW&M) (see NUREG-1909 [Croff et al. 2008]).

SECY-09-0082 comments on the need to revise 10 CFR 70 to adequately address the unique hazards and risks related to these facilities.

Fuel-cycle facilities mainly differ from power reactors by the diversity of their strongly interrelated inherent hazards, and by the large distribution and mobility of the hazards throughout the plant. A comprehensive identification and quantification of initiating events and scenarios is a challenge when performing a fully integrated Probabilistic Risk Assessment (PRA) for these facilities. A very consistent effort is deemed necessary to provide a realistic, accurate quantification of the risk; significant uncertainties generally are expected in the results.

1.3 International Regulatory Contexts

A brief synopsis of regulatory approaches to reprocessing facilities in other countries is provided here. It is based, in part, on private communications with officials and researchers in those countries and may not necessarily represent the stated positions of their respective governments.

In the United Kingdom (U.K.), the Health and Safety Executive (HSE) applies PRA to all facilities (nuclear, chemical, heavy industry, etc.). These PRAs must be suitable to the type of facility and sufficient to show that the numerical targets in the regulations are met. The targets are based on doses or risk to workers and the public; the same targets that apply to nuclear power plants and other facilities. In revising its Safety Assessment Principles (SAPs) for fuel facilities, in 2006 the U.K. removed the two targets on the release of radioactivity and plant damage, as they were considered as reactor-orientated and unsuitable for fuel facilities. The new SAPs are published at www.hse.gov.uk/nuclear/saps/index.htm. This page also explains the numerical targets. The U.K.'s SAPs are expressed in terms of Basic Safety Objectives (BSO) and Basic Safety Limits (BSL). The former is a level of risk considered as negligible (i.e., risks below the BSO do not require further regulatory attention), while the latter corresponds to a risk level analogous to a regulatory limit, viz. risks above the BSL would be considered unacceptable. The region lying between the BSO and the BSL is regarded as a "tolerable" risk region. In the latest revision of the U.K.'s SAPs, the BSO for the individual risk of an offsite fatality due to internally and externally initiated accidents that occur at a nuclear facility is set at 1×10^{-6} per year, while the corresponding BSL is 1×10^{-4} per year. (The U.K. guidelines do not distinguish between early- or prompt-fatalities and latent cancer fatalities). In addition, the U.K.'s HSE recently published a Technical Assessment Guide (TAG) on Probabilistic Safety Analysis (PSA)[1] [HSE 2009].

After the fire and explosion incident in Japan in 1997 ([IAEA 1999a] and [IAEA 2007]), safety reassessments of the Tokai Reprocessing Plant (TRP) were undertaken from 1998 to 1999; also various activities were carried out to enhance plant safety. From 2002 to 2003, the relative importance of safety functions at the TRP was evaluated by applying a PRA. Thereafter, from 2004 to 2005, a PRA also was applied to four representative accident scenarios to assess quantitatively the effectiveness of the hardware modification and operating procedure improvement which were implemented based on the PRA results. Since then, the Japan Atomic

[1] Some publications use the term Probabilistic Risk Assessment (or Analysis) (PRA), while others use Probabilistic Safety Analysis (PSA). These terms have the same meaning and are used interchangeably on this report.

Energy Agency (JAEA) has conducted a study estimating the component failure rates for a reprocessing plant based on the maintenance records stored in the TRP.

In France, there has been an effort to develop the necessary knowledge of existing risk-management tools, thereby allowing their use to assess the risk of nuclear facilities. In this context, the PRAs are of particular interest because of their significant role in the safety culture of the nuclear industry. There appear to be no distinctions in the approaches of the French nuclear authorities to various types of fuel-cycle facilities.

1.4 Organization of the Report

The report has seven chapters and two appendices. Chapter 2 briefly reviews previous work. Chapter 3 summarizes accidents at, and risk assessments for reprocessing facilities. Chapter 4 discusses applying qualitative versus quantitative risk methods. Chapter 5 summarizes the insights gained about the current regulatory situation and offers some observations. Finally, Chapters 6 and 7, respectively, contain the references, and acknowledgments. Appendix A concisely presents the accidents that have happened in reprocessing facilities worldwide. Appendix B gives a short description of electrochemical processing (pyroprocessing) and its literature.

As a guide to the reader, it is noted that there is no one-to-one correspondence between the sections of the report and the four tasks stated in Section 1.1. The aim was to provide an integrated, holistic view of the subject area, and thus, the report sections do not draw on the elements of inquiry given by the four tasks in isolation from one another.

2. REVIEW OF RECENT WORK

2.1 ACNW&M Evaluation of Potential Regulatory Changes

In 2008, the Advisory Committee on Nuclear Waste and Materials (ACNW&M) of the NRC published a report on the Background, Status and Issues Related to the Regulation of Advanced Spent Nuclear Fuel Recycle Facilities (NUREG-1909 [Croff et al. 2008]). In the context of this present paper, the ACNW&M define recycle as involving (a) reprocessing the spent nuclear fuel (SNF) to separate it into its constituent components, (b) refabricating fresh fuels containing plutonium and minor actinides, and, (c) managing and storing the gaseous-, liquid-, and solid-wastes generated along with spent fuel. The ACNW&M report describes the historical approach to the SNF recycle, reviews recent advances in technology, and evaluates the technical- and regulatory-issues that will need to be addressed to assure the viability of commercially reprocessing spent fuel.

The following are of particular interest to the work documented in this report: (1) The ACNW&M's evaluation of potential modifications to Part 70, perhaps including creating a new rule that would have to be considered for an effective, efficient regulatory process, (2) the impact of reprocessing on other NRC regulations dealing with nuclear wastes, and, (3) the complexity of some advanced approaches to reprocessing, such as the UREX process[1] or electrochemical processing that need more technical development and evaluation before an adequate regulatory framework including regulatory guidance can be established.

The ACNW&M report points out that fuel-fabrication facilities, which now are licensed under 10 CFR Part 70, utilize an ISA to assess the safety of the design and to identify equipment relied upon for safety. While using the ISA is an important step towards risk quantification and the expanded use of risk-informed regulations, the ACNW&M report indicates that the "Joint Subcommittee of ACRS and ACNW&M noted shortcomings in ISAs that would likely need to be addressed to expand its role in regulatory decisions involving reprocessing facilities."

In discussing ISA versus PRA for analyzing risk in reprocessing facilities, the ACNW&M report comments:

"The primary reason for using ISA rather than full scope PRA is that the consequences of likely accidents in or routine releases from fuel cycle facilities are believed to be small compared to the consequences of accidents at reactors, and does not justify the effort of doing probabilistic analyses. However, the effort required to prepare an ISA for complex SNF recycle handling liquids containing substantial quantities of concentrated cesium, strontium, and TRU[2] elements is likely to approach the effort that would be required to evaluate risks using a PRA. The Committee and the ACRS have previously advised [ACNW&M, 2002, 2006] that a regulation that utilizes PRA insights is preferable to one based on ISA because the latter has significant limitations in its treatment of dependent failures, human reliability, treatment of uncertainties, and aggregation of event sequences."

[1] There is a suite of UREX processes, each of which consists of a series of steps designed to remove specific groups of radionuclides to tailor products and compositions of the desired product and waste streams [Laidler, 2006].

[2] TRU stands for transuranic.

A related methodological issue discussed in the ACNW&M report is a best estimate versus a conservative approach:

"A companion issue to that of probabilistic versus deterministic approaches is whether analyses should be based on data and models that represent the best estimate of what might really occur with an associated uncertainty analysis to explore the effects of incorrect data or models, or should be based on demonstrably conservative data and models. Most regulations and license applications for fuel cycle facilities have used a conservative, deterministic approach. The Committee has letters on record pointing out problems with using this approach (see Appendix C [of NUREG-1909]). Some of the most important problems are that using very conservative assumptions can mask risk-significant items and most conservative analyses are not accompanied by a robust uncertainty analysis."

In overall terms, the ACNW&M report determined that the experience and lessons learned from licensing fuel fabrication facilities under 10 CFR Part 70 to some extent are applicable to reprocessing facilities; however, several features of reprocessing facilities may require additional regulation. In particular, ACNW&M indicates that "a new rule" could be formulated specifically for licensing reprocessing facilities, and they point to the development of a risk-informed performance-based framework for licensing new reactor designs (published as NUREG-1860 [NRC 2007]) as an example of an approach that "…may be advantageous because of its flexibility." In ACNW&M's view, an advantage of formulating a new rule is that it would "…avoid the need to write exemptions for rules already in place and would place all the regulations relevant to the recycle facilities under one part of the regulations, effectively leaving other parts of the regulations unchanged." The drawback to this approach would be the additional time and resources required to develop such a rule, although ACNW&M also stated that "….it is unclear whether the requirements for developing a new rule are significantly greater than those of other approaches."

The approach that ACNW&M implicitly recommends in considering a new regulation is the one set out in NUREG-1860 as a technology-neutral framework for a risk-informed and performance-based approach to licensing a future generation of nuclear power reactors. The ACNW&M report emphasizes the following aspects of this technology-neutral framework:

[The framework put forth in NUREG-1860] "…integrates safety, security, and emergency preparedness to establish a comprehensive set of requirements as a license condition. The approach focuses on the most risk-significant aspects of plant operations and uses the Commission's safety goals (separate goals would need to be developed for recycle facilities) as top-level regulatory criteria that designers must meet to ensure adequate safety. The approach eliminates the need for exemptions by implementing guidance to accommodate technological differences between designs."

The ACNW&M report details the following new activities and facilities that will require decisions about the appropriate licensing regulations.

- reprocessing fuels from light water reactors (LWR), and later, from other advanced reactors
- fabricating fuels to recycle TRU- or fission product-elements or fuels for some new reactor designs (e.g., graphite-moderated reactors)
- disposing of new types of wastes, such as cladding and TRU (GTCC) waste[3]
- extending the interim storage of intermediate-lived radionuclides (cesium and strontium), followed by *in situ* disposal.

Since there are uncertainties about future reprocessing technologies that may be developed and implemented, ranging from aqueous technologies, like PUREX (plutonium-uranium extraction) and UREX, to dry technologies like electrochemical processing, a technology neutral set of regulations supplemented by technology specific regulatory guidance seemingly offers a balanced, flexible approach to creating the regulatory framework for reprocessing.

Another feature of any new or modified rule for reprocessing identified in the ACNW&M report is that it should "…be consistent with Commission policies including the Commission's PRA policy statement [NRC 1995]. The latter states, in part, "The use of PRA technology should be increased in all regulatory matters to the extent supported by the state of the art in PRA methods and data, and in a matter that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy." The ACNW&M report reiterates "The Committee has gone on record repeatedly in letters to the Commission about the use of risk-informed decision making, starting in October 1997 and most recently in the letter of May 2, 2006." Appendix C of the ACNW&M report lists the committee letters related to risk-informed activities and PRA.

A new rule that licenses reprocessing also will need to specify limits on gaseous- and liquid-effluents generated during operation. The ACNW&M report comments on the need to formulate "ALARA requirements" for reprocessing facilities that establish design objectives and limiting conditions for radioactive material effluents. These requirements will be analogous to the current Part 50 Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as Is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents.

In summary, the ACNW&M report has two main conclusions: (1) It determined that no existing regulation in the United States is "entirely suitable" for licensing reprocessing facilities. Existing fuel-cycle plants, such as fuel-fabrication facilities, handle relatively small amounts of radioactive materials but future reprocessing plants likely will process much larger quantities of radioactive materials in solids, liquids, and gases distributed in various locations in the facility. (2) It concludes that compared to other fuel-cycle facilities regulated under 10 CFR Part 70, the possibility of larger source terms, the presence of both radioactive and chemical hazards, as well as the greater complexity of equipment and operations suggests employing

[3] GTCC waste stands for "greater than Class C" waste. 10 CFR Part 61.55 categorizes low-level waste from Class A (least hazardous) to GTCC (most hazardous).

correspondingly more sophisticated methodologies, such as PRA, to analyze the risk of reprocessing facilities. Following the conclusions of the ACNW&M report, should PRA be adopted as the method for carrying out risk assessments of reprocessing plants, several improvements and enhancements to the traditional PRA methods used in reactors are warranted for the latter facilities since they pose both radiological- and chemical-risks. Examples of these potential enhancements are briefly mentioned in Chapter 4.

2.2 NEI Report

The NEI [2008] proposed a licensing framework for a reprocessing facility that "…is modeled under the risk-informed and performance-based approach of Part 70 supplemented with provisions from Part 50." The framework is designed to implement a new part, labeled part 7x, under Title 10 of the CFR. The NEI proposes that the framework is technology neutral, and sufficient to encompass licensing of the different reprocessing technologies that industry is studying.

From a substantive and technical standpoint, the framework basically adopts the approach and requirements that mirror those in Part 70. It requires performing an ISA to identify facility accidents and items relied on for safety (IROFS), management measures to assure the availability and reliability of the IROFS, and other associated administrative requirements. It requires quantitative assessments of risk to a member of the public located outside the controlled area from high consequence accidents involving fission products to the extent practicable based on the availability of data to support quantitative analysis, and establishment of Technical Specifications for IROFS identified for such accidents.

In offering a rationale of the need for a new part beyond Part 70 to license a reprocessing facility, NEI identifies one regulatory argument, and one substantive technical one. The regulatory argument is derived from the fact that a reprocessing facility is considered a "production facility" under the Atomic Energy Act, and hence, is subject to 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities." NEI cites the portions of the existing Part 50 that refer to reprocessing facilities but argues, citing several NRC SECY documents such as SECY-08-0134 [NRC 2008b], that Part 50 mainly was used to license operating light-water reactor (LWR) power plants and it "….would not be effective or efficient to revise Part 50 to license reprocessing facilities." The substantive technical argument rests on the claim that reprocessing facilities would have a "…greater source term than other fuel cycle facilities." Additionally, the NEI report indicates that as a production facility, regulatory requirements such as Technical Specifications, which are lacking in Part 70, would have to be identified.

The NEI proposes an additional technical feature beyond Part 70, viz., for "…accident scenarios that could result in a high consequence event involving fission product releases to an individual located outside the controlled area, the ISA is to be supported by a quantitative assessment of the risk to the extent practicable based on the availability of data to support quantitative analysis including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components." Since a quantitative risk assessment generally signifies a PRA, the NEI proposal essentially is to perform an ISA (a qualitative analysis but might be supported by semi-quantitative assessments) and then a limited PRA that analyzes only a subset of potential accidents at the facility. The NEI mentions that the likelihood determination made in the ISA will be "…supported by quantitative analysis" for a "…relatively limited subset of IROFS." The NEI report also states that the Technical Specifications will be developed only for this limited set of IROFS that protect against accidents that might entail high consequences from releases of fission-products to an off-site individual.

In addition, the NEI proposal for Part 7x incorporates a set of baseline design criteria (BDC) for the reprocessing facility that are modeled on corresponding design criteria in 10 CFR Part 50 and Part 70. Appendix A of Part 50 specifies General Design Criteria (GDC) for nuclear-power plants, and Part 70.64 specifies BDC for fuel-cycle facilities licensed under Part 70. The BDC proposed in Part 7x are a combination of the Part 50 GDC and the Part 70 BDC.

3. ACCIDENTS AND RISKS ASSOCIATED WITH REPROCESSING FACILITIES

3.1 Summary of Accidents at Reprocessing Facilities

This chapter briefly summarizes accidents at nuclear-fuel reprocessing facilities throughout the world. It is organized into four main sections. Section 3.1.1 gives an overview of reprocessing facilities, and Section 3.1.2 summarizes accidents therein. Some reports published by international agencies related to the safety and regulation of such facilities, were identified while searching the literature for the accidents; Section 3.1.3 briefly describes them. The focus of this literature search was on the accidents, and not on the reports; hence, those included in Section 3 are publications that appear relevant, but were not obtained through an exhaustive search.

3.1.1 Overview of Reprocessing Facilities Worldwide

Table 1, shows past, current, and planned reprocessing facilities worldwide. Both current and projected reprocessing capacities shown in this table are a relatively small fraction of the total spent fuel generated by nuclear power plants worldwide, as expressed by the International Atomic Energy Agency (IAEA) TECDOC-1587 [IAEA 2008a]:

> "Currently about 10,500 tHM[1] spent fuel are unloaded every year from nuclear power reactors worldwide (Figure 1). This is the most important continuous growing source of civil radioactive materials generated, and thus need to be managed appropriately. Also, this annual discharge amount is estimated to increase to some 11,500 tHM by 2010. The total amount of spent fuel cumulatively generated worldwide by the beginning of 2004 was close to 268,000 tHM of which 90,000 tHM has been reprocessed. The world commercial reprocessing capacity is around 5,550 tonnes per year.[2] Projections indicate that the cumulative amount generated by the year 2010 may be close to 340,000 tHM with a corresponding increase in reprocessed fuel. By the year 2020, the time when most of the presently operated nuclear power reactors will approach the end of their licensed operation life time, the total quantity of spent fuel generated will be approximately 445,000 tHM."

Discussing spent fuel produced in the United States, the NRC [2011] indicates

> "As of January 2011, the amount of commercial spent fuel in safe storage at commercial nuclear power plants was an estimated 63,000 metric tons. The amount of spent fuel in storage at individual commercial nuclear power plants is expected to increase at a rate of approximately 2,000 metric tons per year."

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[1] tHM means metric tonnes of heavy metal (MtHM).

[2] Table 1 indicates that the world's current reprocessing capacity is 5,950 MtHM/year. The cause of the difference from 5,550 MtHM/year was not found in the IAEA-TECDOC-1587; however, this difference is not significant.

Table 1: Past, Current, and Planned Reprocessing Capacity in the World (in tHM/year) (from [IAEA 2008a])

Country	Site	Plant	Type of Fuel Processed	Operation Start	Operation Shutdown	Capacity Present	Capacity Future
Belgium	MOL	Eurochemic	LWR	1966	1975		
China	Jiuquan	RPP	LWR	?			25
	Lanzhou		LWR	2020			800
France	Marcoule	APM	FBR	1988	1996		
	Marcoule	UP1	GCR	1958	1997		
	La Hague	UP2	LWR	1967		1000	1000[3]
	La Hague	UP3	LWR	1990		1000	1000[7]
Germany	Karlsruhe	WAK	LWR	1971	1990		
India	Trombay	PP	Research	1964		60	60
	Tarapur	PREFRE 1	PHWR	1974		100	100
	Kalpakkam	PREFRE 2	PHWR	1998		100	100
	Kalpakkam	PREFRE 3A	PHWR	2010			150
	Tarapur	PREFRE 3B	PHWR	2012			150
Japan	Tokai-mura	JAEA TRP	LWR	1977		90	90
	Rokkasho-mura	JNFL RRP	LWR	2007		800	
Russian Fed.	Chelyabinsk	RT1	WWER-440, BN-350, BN-600 RR	1977		400	400
	Krasnoyarsk	RT2	WWER-1000	2025			1500
		Demonstrative facilities	VVER-1000 RBMK	2013			50-100
U.K.	Sellafield	B205	GCR	1967		1500	
	Sellafield	Thorp	LWR/AGR	1994		900	1000
	Dounreay	UKAEA RP	FBR	1980	2001		
USA	West Valley	NFS	LWR	1966	1972		
	Hanford	Rockwell	U metal	1956	1989		
	Savannah River	SR	U metal	1954	1989		
	Idaho Falls	R	U-Al alloy	1959	1992		
Total Capacity						5950	6525

[3] 1000 MtHM for each plant, with a cumulated maximum of 1700 MtHM for the La Hague site.

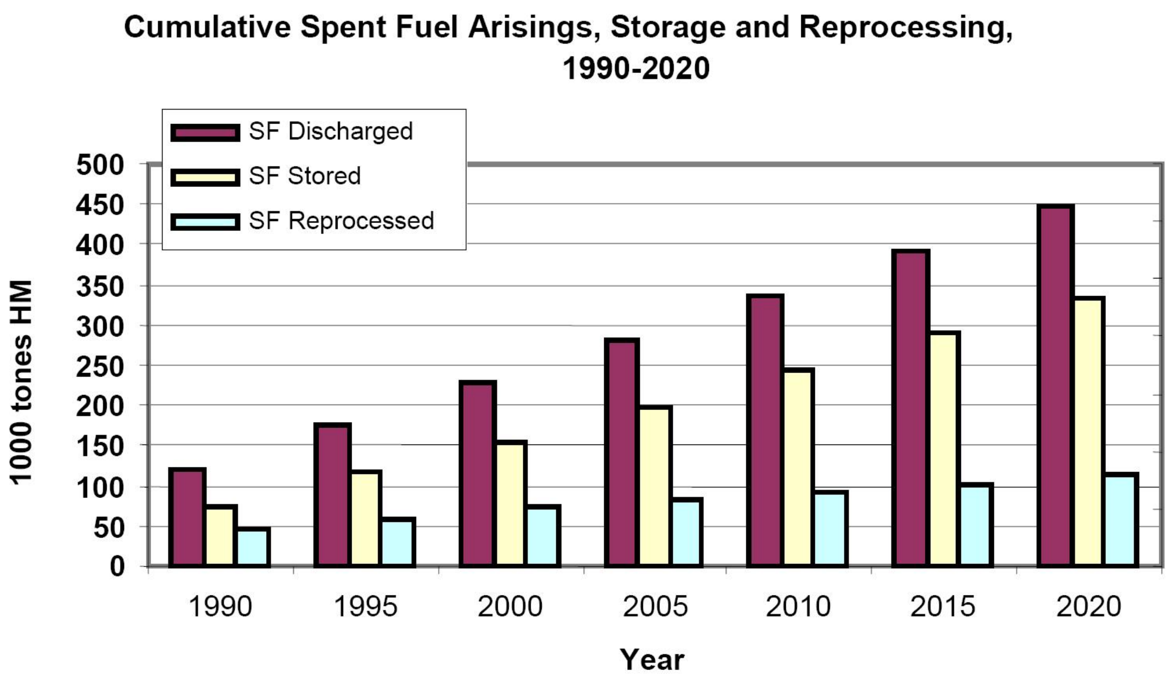

Figure 1: Trends in spent-fuel management (from [IAEA 2008a])[4]

Routine Releases of Radioactivity

Reprocessing facilities apparently used to release significantly more radioactivity during normal operation than other nuclear facilities. In particular, Schneider et al. [2001] stated, "Reprocessing operations release considerably larger volumes of radioactive discharges than other nuclear activities, typically by factors of several 1,000 compared with nuclear reactor discharges. In the U.K., Fairlie [1997] has estimated that about 90% of nuclide emissions and discharges from the U.K. nuclear programme result from reprocessing activities." On the other hand, a draft report by the IAEA [2007] pointed out:

> "With the state-of-art technology, it is now possible to design and operate fuel reprocessing plants with as low an environmental release as any other conventional chemical industry. For instance, it has now been demonstrated that the volume of waste generated in the reprocessing operations have been considerably reduced in recent years. The dose to the public due to waste discharged from reprocessing has also shown a steady decrease [IAEA 2005]. A combination of learning from experience and continuous improvements, modifying both plant and practice, such as introduction of automated operations has reduced the average employee radiation exposures at reprocessing facilities from over 10 mSv to 1.5 mSv per person pa over the past two decades. (By comparison, the average annual exposure for airline crew is about 2 mSv). Radiation exposure of the public has also been reduced, largely in line with the reductions in radioactive discharges. As the quantity of radioactivity being discharged has declined each year, the proportion of radiation exposure that is attributable to

[4] "SF Discharged" means SF generated.

> current discharges has declined. In the U.K. today, the average annual exposure of individuals due to radioactive discharges is less than 0.1 mSv. By comparison, the average annual exposure to individuals in the U.K. from natural background radiation is about 2.2 mSv."

3.1.2 Accidents at Reprocessing Facilities Worldwide

BNL conducted a survey of accidents at reprocessing facilities, and this subsection summarizes them. McLaughlin, Frolov, et al. [2000] reviewed in detail criticality accidents in nuclear facilities, including several at reprocessing facilities around the world. Discussing such accidents, the IAEA [2007] states

> …Of the nearly 60 criticality accidents, which have occurred since 1945, about a third occurred at nuclear fuel cycle facilities...Twenty of these accidents involved processing liquid solutions of fissile materials, while none involved failure of safety equipment or faulty (design) calculations. The main cause of criticality accidents appears to be the failure to identify the range of possible accident scenarios, particularly those involving potential human errors.

Some accidents occurred due to a chemical ("red oil") reaction. Hyder [1994a and 1994b], and Paddleford and Fauske [1995] describe the reactions of tri-n-butyl phosphate (TBP) with nitric acid and nitrates.

In addition, an annex to the report by Schneider et al. [2001] contains some events that occurred at the La Hague reprocessing facility; COGEMA[5] reported them to the French nuclear-safety authorities between 1989 and the end of the first half of 2001, and the safety authorities published them.

Table 2 lists the major accidents that were categorized using the International Nuclear and Radiological Event Scale (INES), as presented in the literature, as well as other events whose consequences were of different degrees of severity and that have not been classified using this scale yet. Events having relatively small consequences were not included in this table, but nevertheless occur in reprocessing facilities.

Cadwallader et al. [2005] discuss this situation when they state the following:

> Searching the DOE Occurrence Reporting and Processing System for [Idaho Chemical Processing Plant] ICPP events returned ~ 600 events in the past 14 years. Many of these events were personnel anti-contamination clothing or skin contamination events, and a number of false fire alarms (as well as false criticality, security, and evacuation alarms) were included in the 600 events. There were several power outage and voltage dip events, and a number of personnel safety issues, including lockout-tagout deficiencies, chemical overexposure, radiation overexposure, procedure violations, and industrial injuries. There was a lightning strike at the facility in June 1998, but there was little damage after alarm systems were returned to normal. Several small fires were

[5] COGEMA (Compagnie générale des matières nucléaires) is a French company, created in 1976 from the production division of the Commissariat à l'Energie Atomique (CEA), which is the French Atomic Energy Commission. In 2001, COGEMA became part of the larger group Areva; the subsidiary's name was changed to Areva NC in March 2006.

reported, in electrical distribution and other equipment, and two small fires with kerosene in the calciner apparatus. Other events included wrong casks and wrong fuel moved or stored, environmental contamination by chemicals (motor oil, diesel fuel, etc.), equipment failures, a few dropped items, and a bomb threat hoax. An ICPP worker was killed in an industrial accident when he was struck by a forklift truck in May 1991. These events are tragic and nonetheless endemic to many types of industrial facilities.

Table 2: Summary of Accidents in Reprocessing Facilities Worldwide
(The number in parenthesis after an accident is the number of resulting early fatalities, if available.)

INES		Kind of Hazard		
Level	Type of Event	Criticality	Radiation	Fire and/or explosion
7	Major accident			
6	Serious accident			Mayak PA[6], 9/29/1957
5	Accident with wider consequences			
4	Accident with local consequences			Windscale, 1973
3	Serious incident		THORP, 8/2004	La Hague, 1/6/1981 Siberian Chemical Enterprises, 4/6/1993 Tokai-mura, 3/11/1997
2	Incident			
1	Anomaly			
Classification not available		Mayak PA, 3/15/1953 Mayak PA, 4/21/1957 (1) Mayak PA, 1/2/1958 (3) Oak Ridge Y-12 Plant, 6/16/1958 Los Alamos Laboratory, 12/30/1958 (1) ICPT,[7] 10/16/1959 Savannah River, 9/13/1960 Mayak PA, 12/5/1960 ICPT, 1/25/1961 Hanford Works, 4/7/1962 Mayak PA, 9/7/1962 Siberian Chemical Combine, 1/30/1963 Siberian Chemical Combine, 12/2/1963 Mayak PA, 12/16/1965 Mayak PA, 12/10/1968 (1) Windscale, 8/24/1970 ICPT, 10/17/1978 Siberian Chemical Combine, 12/13/1978	Mayak PA, 8/19/1950 Mayak PA, 9/13/1950 Mayak PA, 9/20/1950 Mayak PA, 10/1/1951 (1) Mayak PA, 9/20/1952 Mayak PA, 11/6/1954 Mayak PA, 12/22/1955 Hanford Works, 6/18/1956 Hanford Works, 9/3/1963 La Hague, 10/2/1968 La Hague, 10/1/1976 La Hague, 1/2/1980 La Hague, 4/1980 La Hague, 1983 Sellafield, 1983 La Hague, 2/13/1990 La Hague, 3/11/1997 Karlsruhe, 1999 KARP, 1/21/2003 Dounreay, 9/2005	Hanford Works, 12/4/1951 Savannah River, 1/12/1953 Hanford Works, 1953 Hanford Works, 2/1958 Hanford Works, 1958 ORNL, 11/20/1959 Fontenay-Aux-Roses, 6/26/1962 Hanford Works, 11/6/1963 La Hague, 1/14/1970 Savannah River, 2/12/1975 Hanford Works, 1976 UTP, Ontario, 1980 Karlsruhe, 1985 Hanford reprocessing plant, 5/14/1997

[6] Mayak PA stands for Production Association
[7] ICPT stands for Idaho Chemical Processing Plant.

The scope of BNL's literature search is restricted to those events with serious consequences, that is, ones with any of the following characteristics:

1. Events involving criticality, fire, explosion, or substantial leak of radioactive material.

2. Events causing fatalities or injuries to people in the reprocessing facility's site.

3. Events involving off-site releases of radioactivity.

All the events identified in this study were grouped according to the following three main kinds of hazard:

1. Criticality. This group includes events with one or more criticality excursions that exposed workers directly to radiation.

2. Radiation. This group includes events in which criticality did not occur, but one or more persons were exposed directly to radiation from an accidental release of radioactivity onsite and/or offsite from the reprocessing facility.

3. Fire and/or explosion. This group includes events involving chemical exothermic reactions. Consequently, in addition to the fire and/or explosion hazard itself, radioactivity and/or hazardous chemicals may have been released onsite and/or offsite from the reprocessing facility.

The public is notified of the safety significance of events associated with sources of radiation, via the INES. International experts convened in 1990 by the IAEA and the Nuclear Energy Agency (NEA) of the Organisation for Economic Co-operation and Development (OECD) developed this scale. According to the INES User's Manual [IAEA 2009], events are classified on the scale at seven levels: Levels 4–7 are termed "accidents" and Levels 1–3 "incidents;" events without safety significance are classified as "Below Scale/Level 0." Events that have no safety relevance for radiation or nuclear safety are not classified on the scale.

Table 2 summarizes all the events identified in this study as a function of the type of hazard involved and the INES classification. It shows that most of them have not been classified according to this scale, or their classification was not found in the public literature. If early fatalities resulted from an accident, the number of deaths is included in parenthesis after the accident. Here, "early fatality" is defined loosely as a death happening within a few months after the accident. In addition, other consequences such as "latent fatalities" (those that may occur with some time lag following exposure, such as latent cancers due to radiation exposure) may have occurred in some events, but this information was unavailable to this study.

All early fatalities were workers at the reprocessing facilities. Accounting only for these fatalities, apparently the event at Mayak Production Association (PA) on 1/2/1958 (3 early fatalities) would be categorized as Level 5, "Accident with wider consequences," and the events involving one early fatality (Mayak PA, 10/1/1951; Mayak PA, 4/21/1957; Los Alamos Scientific Laboratory, 12/30/1958; Mayak PA, 12/10/1968) would be classified as Level 4, "Accident with local consequences," in the international scale.

Appendix A gives additional information about each event in Table 2.

3.1.3 International Literature on Safety and Regulation of Reprocessing Facilities

Some reports published by international agencies about the safety and regulation of this type of facility were identified while searching the literature for the accidents presented in Section 3.1.2. Most of them were published by the International Atomic Energy Agency (IAEA), and the Nuclear Energy Agency (NEA). The focus of the literature search documented in this report was on the accidents and not on the reports, so the reports included are publications that appear relevant, but were not retrieved via an exhaustive search. This section briefly addresses them.

The report IAEA-TECDOC-1221 [IAEA 2001] contains the results of a meeting of the IAEA Technical Committee in 2000 whose main objective was to compile information on the nature of the safety concerns and status of the regulations on nuclear-fuel-cycle facilities in IAEA's Member States. It states, "…Although some similar safety hazards may be posed at reactor and non-reactor fuel cycle facilities, the differences between them give rise to specific safety concerns at the non-reactor fuel cycle facilities that must be especially taken into consideration in the design and operation of these facilities…" It further points out, "The IAEA maintains a database of nuclear fuel cycle facilities in Member States in the Nuclear Fuel Cycle Information System [IAEA NFCIS]. This system reasonably represents the approximate number and worldwide distribution of nuclear fuel cycle facilities. The system may not necessarily be up to date in all respects as the IAEA relies on Member States to refresh the information periodically; however, the information is sufficiently representative to be used in assessing the relative magnitude and diversity of existing and projected fuel cycle facilities in Member States."

The International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) was launched in the year 2000, based on resolutions of the IAEA General Conference, and developed a set of basic principles, user requirements and criteria together with an assessment method, which taken together, comprise the INPRO methodology, for the evaluation of innovative nuclear energy systems. The results of this work were initially documented in IAEA-TECDOC-1362, "Guidance for the evaluation for innovative nuclear reactors and fuel cycles" [IAEA 2003], and in IAEA-TECDOC-1434, "Methodology for the assessment of innovative nuclear reactors and fuel cycles" [IAEA 2004]. INPRO prepared additional guidance in using its methodology to assess the sustainability of an innovative nuclear energy system (INS) in the form of an INPRO assessment manual. The resulting INPRO manual is comprised of an overview volume (IAEA-TECDOC-1575, Rev. 1) [IAEA 2008c], and eight additional volumes covering the areas of economics (Volume 2), infrastructure (Volume 3), waste management (Volume 4), proliferation resistance (Volume 5), physical protection (Volume 6), environment (Volume 7), safety of reactors (Volume 8), and safety of nuclear fuel cycle facilities (Volume 9). Volume 9 [IAEA 2008d] presents the safety issues related to design and operation of mining, milling, refining, conversion, enrichment, fuel fabrication, fuel storage and fuel reprocessing facilities. It further discusses adapting the INPRO methodology in terms of identifying indicators and acceptance limits of various criteria for these facilities. For example, with respect to occupational exposure criteria, it states:

> It is realized that the experience with respect to nuclear fuel cycle facilities in various countries has not been collated and harmonized to the extent that has been done for the reactor systems. For example, the limits for exposures vary from country to country (see Figure 2 [ICRP 1993]). Arriving at such limits, falls strictly under the purview of the regulatory body in the respective country, even though it is presumed that the ICRP limits in general form the guidelines for the regulatory process.

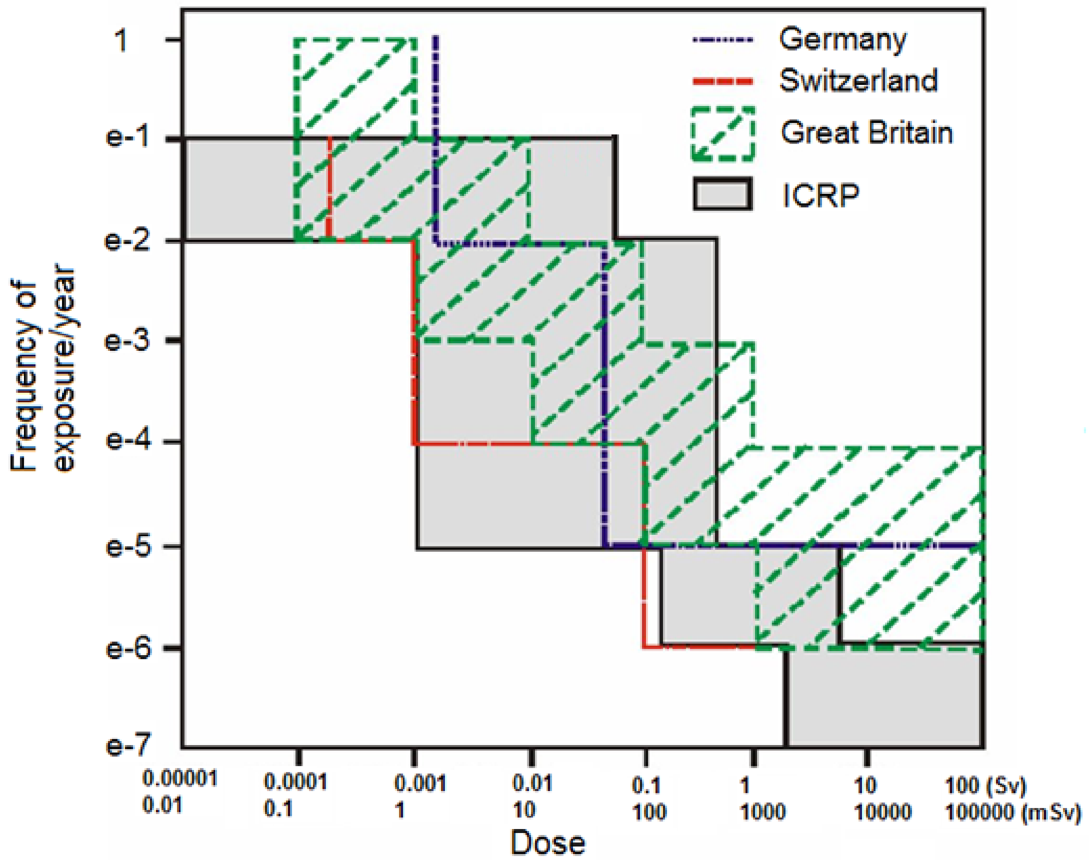

Figure 2: Variations in the frequency of occupational exposure with dose (from [ICRP 1993])[8]

The goal of Volume 9 of IAEA-TECDOC-1575, Rev. 1, is to provide guidance to an assessor of the safety of nuclear-fuel- cycle facilities in a country or region (or even on a global scale) that is planning to install a nuclear-power program (or maintaining or enlarging an existing one) on how to apply the INPRO methodology in this specific area. As part of Volume 9, Section 5.6, "Fuel reprocessing facilities" specifically addresses these facilities, and Section 6.2, "Safety-related RD&D[9] areas," states[10]:

> More research will be needed to bring the knowledge of plant characteristics and the capability of computer codes to model phenomena and system behavior for innovative fuel cycle installations to at least the same confidence level as for existing nuclear power plants. In addition, a method should be developed for quantifying the safety of such facilities.

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[8] 1 Sv = 100 rem.

[9] RD&D stands for research, development, and demonstration.

[10] The spelling in this quote was changed to conform to U.S. spelling.

> Further development of Probabilistic Safety Analyses (PSA) methods, including best estimate plus uncertainty analysis, and their supporting data bases are required and need to be capable of:
>
> - Assessing innovative nuclear designs implemented with lines of defense composed of inherent safety characteristics and passive, as well as active systems;
> - Assessing total risk from various states and considering both internal and most external initiating events;
> - Accounting for safety culture and human factors;
> - Accurately accounting for ageing effects; and
> - Quantifying the effects of data and modeling uncertainties.
> - Identify all important phenomena and try to computer simulate them.
> - Validate computer codes in all regimes of fluid and solid material behavior. Simulation may compensate for lack of operating experience if limited experimental results are available and they have been used to validate the computer code employed.
> - Justify scaling to commercial size installations, and
> - Obtain reliability data.

Further, the IAEA points out in its document IAEA-TECDOC-1587 [IAEA 2008a] that

> Safety requirements for reprocessing plants are reflected at national level in regulations and standards. However, there is a trend toward internationalization of safety standards for the nuclear fuel cycle in general and spent fuel management facilities in particular. This issue has been examined at the IAEA and a system of international safety standards for fuel cycle facilities is in development. A safety guide on spent fuel reprocessing facilities is also in preparation.
>
> Meanwhile, a Joint Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management has been agreed to, and entered into effect on 18 June 2001. The Joint Convention is the first international treaty relating to these areas of safety which is legally binding. It represents a commitment by States to achieve and maintain a high level of safety in the management of spent fuel and radioactive waste. The first Review Meeting of the Joint Convention was held in November 2003 [IAEA 1997] and the second in May 2006.

With regard to reprocessing, the scope of application of this Convention is "…Spent fuel held at reprocessing facilities as part of a reprocessing activity is not covered in the scope of this Convention unless the Contracting Party declares reprocessing to be part of spent fuel management."

In addition, the IAEA authored a report, "Safety of Nuclear Fuel Cycle Facilities: Safety Requirements," [IAEA 2008b] whose objective "…is to establish requirements that, in the light of experience and the present state of technology, must be satisfied to ensure safety, for all stages in the lifetime of a nuclear fuel cycle facility, i.e. its siting, design, construction, commissioning, operation and decommissioning. This publication is intended to be used by designers, operating organizations and regulators for ensuring the safety of fuel cycle facilities." The scope of this publication includes reprocessing facilities. Ranguelova, Niehaus, and Delattre[11] also discuss the safety of these facilities.

[11] The authors of this report were affiliated with the IAEA. This report is undated.

The NEA also carried out technical activities in this area. In particular, the book "The Safety of the Nuclear Fuel Cycle" [NEA 2005] discusses the safety of these facilities; however, due to the limited scope of this work, this publication was not included in this study. Furthermore, the NEA's Working Group on Fuel Cycle Safety (WGFCS) of the Committee on the Safety of Nuclear Installations (CSNI) organized the recent workshop "Fuel Cycle Safety – Past, Present and Future" [NEA 2007]. Representatives from Australia, Canada, France, the IAEA, Japan, the Russian Federation, U.K., and U.S. made presentations in the following sessions: "Legacy Waste Concerns," "Fuel Cycle Safety Communication Issues," "Ensuring the Safety of the Facility for the Future" (which included operational experience and research issues), and, "Regulation and Licensing of Nuclear Fuel Cycle Facilities."

Ueda [2005] describes using risk information for Japanese fuel-cycle facilities, indicating that studies on risk assessment methodology and related research conducted by Japanese organizations were based on methodologies, such as the probabilistic safety assessment (PSA) used for nuclear reactors, and methods of hazard analysis and risk evaluation for chemical plants.

3.2 Results of Risk Assessments

This section summarizes the results of some Probabilistic Risks Assessments (PRAs) of reprocessing facilities worldwide. The inclusion of the approaches and results herein does not necessarily endorse them.

As an introduction to this subject, a brief discussion by the IAEA in IAEA-TECDOC-1267 [2002] on the level of detail to use when conducting a PRA of a non-reactor nuclear facility (NRNF) is first presented:[12]

> Facility hazard can influence the depth of analysis, since it may be appropriate to analyze lower hazard facilities to less depth than higher hazard facilities (i.e., the depth of analysis is commensurate to hazard). Similarly, facility complexity can influence the depth of analysis, since it may be appropriate to analyze simple facilities to less depth than more complex facilities.

Thus, the concept of hazard-graded depth of analysis is appropriate for NRNF PSAs. This report seeks to provide a comprehensive guidance for assessing the risk of a high hazard NRNF for regulatory purposes. Table 3 illustrates the concept of a graded approach and provides some guidance as to how to reasonably apply reduced depth of analysis for facilities of lower hazard.

The following main observations were obtained from the information in Table 3: 1) A large reprocessing facility has a large radioactive inventory; 2) a detailed quantitative PRA is applicable for this kind of facility, and to some extent, to a medium-sized reprocessing facility, and, 3) this kind of PRA would include a Human Performance Analysis.

[12] The spelling in this quote was changed to conform to U.S. spelling.

Table 3: Facility and Analysis Ranking (from IAEA [2002])			
	Hazard Rank		
	Low (low activity inventory)	Medium (medium activity inventory)	High (large activity inventory)
Examples of Facilities[13]	Radioisotope Lab	Fuel Fabrication Facility	Research Reactor
	Small Calibration Facility	Waste Treatment Facility	Large Reprocessing Plant
	Hot Cell Facility	Low-Level Waste Storage Facility	High-Level Waste Storage Facility
	Depth of Analysis		
	Simple	Intermediate	Detailed
PSA Tasks	(qualitative to semi-quantitative)	(semi-quantitative)	(quantitative)
Familiarization	Simple, minor effort	↔	Detailed diverse review
Hazard Identification, Initiating Events Selection	Simple systematic or engineering evaluation	↔	Detailed systematic review (FMEA, HAZOP, etc.)
Undesirable End States		↔	Detailed development
Safety Measures Identification	Simple, minor effort	↔	Detailed identification
Safety Measures Information		↔	Detailed information
Event Grouping	Simple grouping	↔	Detailed development
Event Sequence Modelling	Simple modeling or engineering evaluation	↔	Complex modelling (Failure Tree Analysis, Event Tree Analysis, etc.)
Human Performance Analysis	Simple (judgment)	↔	Detailed analysis (Human Reliability Analysis, Task Analysis, etc.)
Consequence Analysis	Simple analysis	↔	Detailed analysis
Parameter Estimating	Few parameters, bounding case, qualitative frequencies	↔	Many parameters, best estimates
Sequence Quantification	Simple (dose, qualitative frequency)	↔	Complex (uncertainty analysis, sensitivity analysis, distributions)
Documentation	Basic	↔	Detailed

[13] Examples of facilities that may rank differently depending upon the facility's size (for instance, a smaller reprocessing facility or plant involving a lower activity inventory may be ranked as a medium hazard).

3.2.1 Rokkasho Reprocessing Plant

It appears the safety analysts in Japan regard PRA as an important aspect of risk evaluation of their reprocessing facilities, even before risk-informed requirements were promulgated by the Japanese Nuclear and Industrial Safety Agency. Takebe et al. [2007] noted that "…Performing PSA and utilizing the risk information for non-reactor nuclear facilities also has the same role in securing safe operation effectively and rationally as in the nuclear power plants. Taking into account the safety characteristics of reprocessing plants in which radioactive materials exist scattered in several chemical processes and storage facilities, we should evaluate risks of many events efficiently and systematically with various types, scenarios, frequencies and consequences in order to assess whole risk and its profile of the plant…" A simplified PRA method for the Rokkasho Reprocessing Plant (RRP) was developed [Shoji et al. 2005], incorporating previous detailed PRA results of some representative events. The intent of this work was to use the risk information in operating and managing the facility (e.g., classifying the components and systems in classes of importance to determine the terms of periodical inspections). In their preliminary results for an evaluation of hydrogen explosions in a Pu concentrate vessel, they found that the risk contribution is higher for a high-consequence, low-frequency explosion rather than a lower-consequence explosion at a higher frequency of occurrence.

Shoji et al. [2005] compared the so-called "risk index method" for the ISA, with their proposed "improved risk index method" (also called a simplified PSA) and a detailed PSA (Figure 3).

Takebe et al. [2007] point out that a detailed PRA for 15 selected events was first carried out, and the results "…showed us applicability of detailed PSA to RRP and quantified unavailability of utilities, electricity, cooling water and compressed air, those are commonly used in every processes. Some of the results were reflected upon the detail design works…In order to assess risk of many events efficiently and effectively seeking risk profile of the whole plant, a simplified PSA tool was developed because most part of events would have relatively simple sequence and low consequence and should not need detailed PSA. As the following step, using the simplified PSA, assessment works on 655 events are on the way."

Kohata et al. [2004] introduce the overall approach and results of the detailed PSA of RRP that included "…frequencies of occurrence of events, release pathways and the amount of activities released, damage probabilities of high efficiency particulate air (HEPA) filters, radiation exposure dose (mSv), risks coming from accidents (both social and individual risks)." They offer a conceptual comparison of the risk from reactor plants and reprocessing facilities (Figure 4), and observe that for practical purposes, the risk of the former can be discussed in terms of only core damage frequency without regard for the consequence; however, discussing the risk of the latter requires accounting for both frequency of occurrence and consequence.

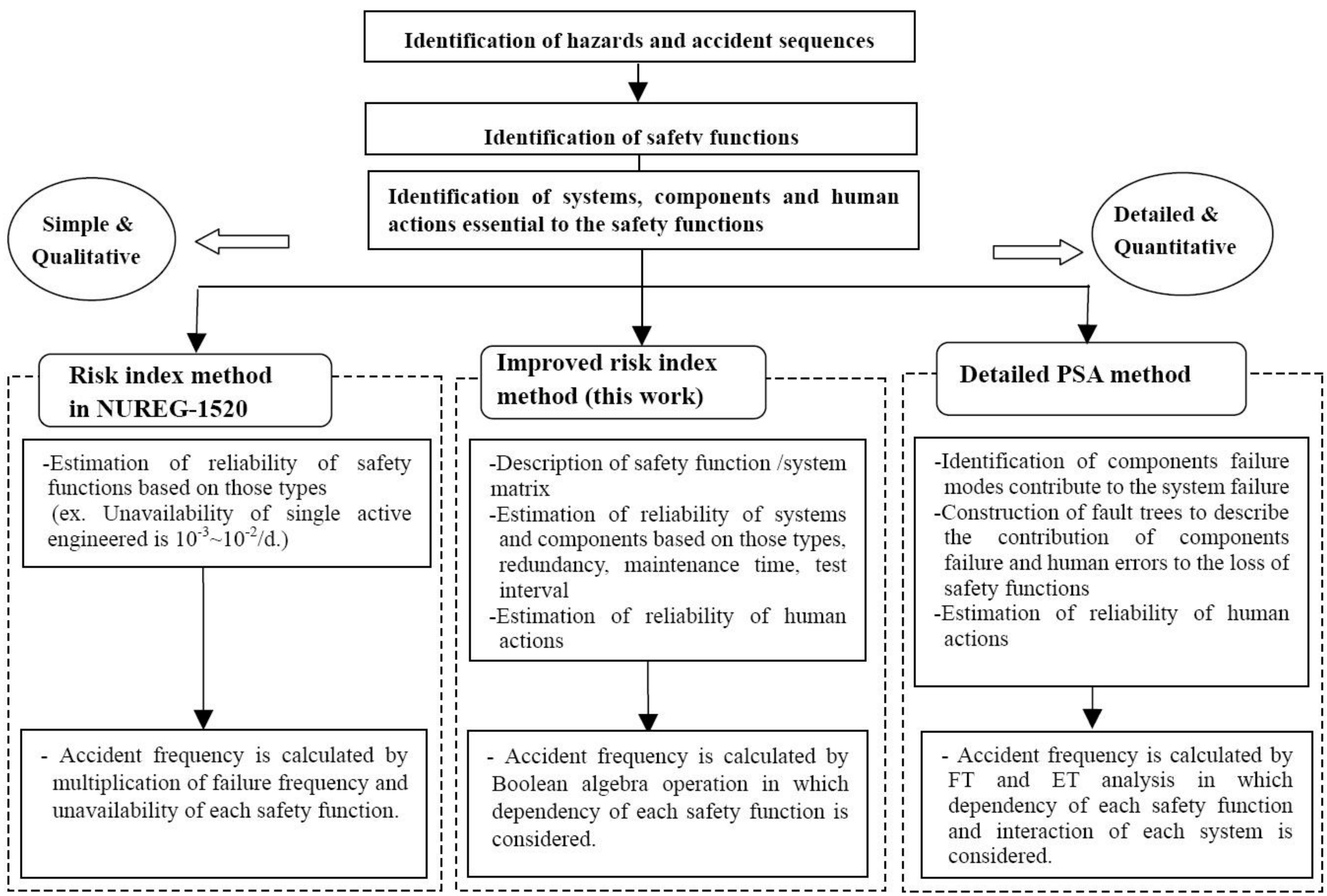

Figure 3: Approach for evaluating frequencies of accidents (from [Shoji et al. 2005])

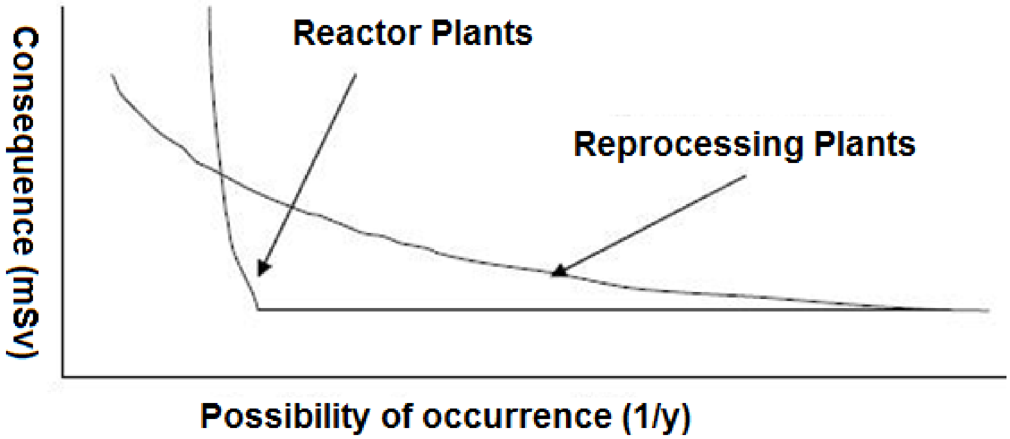

Figure 4: Conceptual comparison of risk (from Kohata et al. [2004])

Kohata et al. [2004] indicate that the usage of risk information from the detailed PSA will be as follows:

- "PSA can give useful information on how to ensure and enhance the safety rationally including operational procedures and maintenance scheme

- To perform PSA extensively and to utilize the knowledge obtained to successfully cope with risk-informed regulation that will be implemented in the not too distant future

- Knowledge obtained from PSA will be applied to:
    - Maintenance scheme including Allowed Outage Time (AOT) and Limiting Condition for Operation (LCO).
    - Operational procedures against events beyond design base etc.
    - Education and training of operators and staffs
    - Rationalization of total inspection scheme including mandatory periodic inspection, voluntary periodic inspection, in service inspection, etc. considering the importance

- Risk-Informed Regulation
    - A type of performance-based regulation
    - What are the suitable performance indicators suitable for reprocessing plants to diagnose plant operational status?
    - How to relate performance indicators to safety goal?
    - Development of in-house oversight process to provide clear and appropriate information to each level of personnel concerned"

Shoji et al. [2005] state that in the original "risk index method" the items relied on for safety (IROFSs) are assumed to be independent, but that the simplified PRA approach is an improvement compared to the original method due to, among other factors, the inclusion of "Evaluation for reliability of IROFS based on failure frequency or probability of individual systems, components and human actions, in consideration of dependency between each IROFS."

The simplified PRA applied to the RRP relies on information obtained from the detailed PRA that was performed for a limited set of accident scenarios. For example, Takebe et al. [2007] state:

> First of all, accident sequences and safety functions should be identified. Afterward, systems, components, and human action related to safety functions are identified from the design and operation information. As thus far explain, there is no difference from detailed PSA procedure…Analyst inputs prescribed failure rates for system, component and human action related to initiating events and safety function … selecting from a list "Database of system, component and human action" that have been set conservatively based on the published documents (IEEE-std 500, NUREG-1363, etc.). Human error rate was determined based on detailed PSA… Unavailability of such support systems as utilities, which related to many events, has been set … with simplified fault tree equation based on detailed PSA results.

3.2.2 Tokai Reprocessing Plant

Ishida et al. [2003] indicate that

> With the fire and explosion accident at the Tokai Bituminization Demonstration Facility in March 1997, JNC [Japan Nuclear Cycle Development Institute] had carried out the safety reassessment of the TRP [Tokai Reprocessing Plant] in 1999… The PSR [Periodic Safety Review] of the TRP has been carrying out to obtain an overall view of actual plant safety. As a part of the PSR, based on the results of the safety reassessments of the TRP, PSA methodology has been applied to evaluate the relative importance of safety functions that prevent the progress of events causing to postulated accidents…As evaluation methods, event tree and fault tree methodologies were selected by taking into account of the power plant PSA [NRC 1982], PSA specialist opinions and document [IAEA 2002]…PSA methodologies have been applied on all postulated accidents.

Concerning dependencies and common-cause failures, Ishida et al. [2003] state, "Dependent failures could be dominant contributors to the frequency of the postulated accidents and they should be taken into account in the analysis regardless of the selected modeling approach…" They also point out "…human reliability analysis was carried out based on the operation manual by using the Technique for Human Error Rate Prediction (THERP)…"

Ueda [2005] indicated that the Japanese Nuclear Safety Commission (NSC) and the Nuclear and Industrial Safety Agency (NISA) are planning to use risk information for nuclear-safety regulation, and outlined studies in Japan on using risk information for reprocessing facilities. He pointed out "…Studies on risk assessment methodology and related researches have been conducted by [Japanese organizations] based on methodologies such as the probabilistic safety assessment (PSA) used for nuclear reactors and the methods of hazard analysis and risk evaluation for chemical plants…" He presented the PSA results for some postulated events at the TRP including risk importance factors and frequency vs. consequence plots. He noted that a greater number and variety of events must be evaluated as possible major contributors to risk in a PSA for TRP than those for nuclear reactors. He considered this is because 1) radioactive- and nuclear-materials are processed throughout a facility with a variety of chemical and physical forms; 2) there is a wide distribution of potential hazardous sources, e.g., radioactivity, heating sources, and flammable and explosive materials, in many parts of a facility; and, 3) there is a wide variety of postulated events in many parts of a facility. For one of the postulated events, hydrogen explosions in a Pu purification process, Ueda gave a plot of "relative frequency" versus "relative consequence" of the related PSA accident sequences.

3.2.3 Thermal Oxide Reprocessing Plant

PRA was used during the design and the start of operation of the U.K.'s Thermal Oxide Reprocessing Plant (THORP), and was used subsequently in its Periodic Safety Reviews, and for the Magnox reprocessing plant [Vaughan 2009]. In particular, James and Sheppard [1991] discuss the risk of thermal runaway in a nuclear-fuel-reprocessing plant due to red-oil reactions using, for illustration, the uranyl-nitrate evaporator in the THORP plant. Their paper sets down the lessons-learned from previous incidents and discusses the research and development work undertaken to enhance understanding of the nature and kinetics of these reactions. They use their findings to analyze evaporator behavior and to identify scenarios that entail thermal runaway. Then, they outline their PRA approach for defining the frequency of red-oil hazards.

No other PRA studies of THORP were found in the public literature, partly due to commercial confidentiality and partly to security concerns. One issue recognized by the U.K. authorities is that PRAs cannot be split so easily into Levels1, 2, and 3 as the plant does not have a simple set of barriers. Two other problems are the changing nature of the hazard as the material moves through the plant and changes its physical state, and the need to account for much more human interaction as the process develops. In the U.K. analyses, this makes the form of the PRA somewhat different from that for reactors as the balance tends more towards fault trees, rather than event trees.

The U.K.'s HSE issued its Safety Assessment Principles (SAPs) for Nuclear Facilities [2006], and more recently published a Technical Assessment Guide (TAG) on Probabilistic Safety Analysis (PSA) [HSE 2009]. The purpose of the TAG is to "…provide an interpretation of those Safety Assessment Principles … related to PSA and to provide specific guidance to inspectors engaged in the assessment of PSAs and PSA related submissions … from Licensees, License Applicants or Generic Design Assessment (GDA) Requesting Parties…" This guide does not apply different approaches or methods for evaluating reactor and non-reactor nuclear facilities. However, it recognizes that a comprehensive PRA, including quantitative evaluations, must be carried out for reprocessing facilities, as reflected by the following quote from its Section 3.2, "Fault analysis: PSA – Need for a PSA – FA.10"[14]:

> The depth of the PSA for a given facility may vary depending on the magnitude of the radiological hazard and risks and the complexity of the facility. For example, for some facilities simplified analyses, or even qualitative arguments, application of good practice and DBA [Design Basis Analysis] may be sufficient to demonstrate that the risk is ALARP [As Low As Reasonably Practicable]. However, for complex facilities such as power reactors or reprocessing facilities, comprehensive PSAs that meet modern standards should be developed for all types of initiating faults and all operational modes.

[14] Fault analysis principle "FA.10," which is one SAP, was titled "Need for PSA."

3.2.4 La Hague

Simonnet [2004] reports that PRA was used in parts of the design of the La Hague reprocessing facility in France. No other PRA study of this facility was found in the public literature.

On the other hand, the French Atomic Energy Commission (Commissariat à l'Energie Atomique, or CEA) published a study on applying PRA to non-reactor nuclear facilities (NRNFs) [Bassi 2005]. It points out that "The bibliography analysis shows that the PSA [Probabilistic Safety Assessment] approach for NRNF is close to that currently adopted for the NPPs, but it has to be adapted due to the specificities of these plants..." As part of adapting the PRA method of power reactors to assessing the risk of an NRNF, Bassi proposes quantifying the risk using the "ARAMIS" approach [Delvosalle et al. 2006] that involves developing and quantifying a somewhat different version of fault trees and event trees. Bassi considers that "…ARAMIS … provides a semi-quantitative approach of the risk, potentially interesting for fuel cycle facilities, oriented towards supplying of whole risk management processes and regulatory demonstration, rather than towards an accurate quantification of the risk." This approach is semi-quantitative because it includes some qualitative aspects.

The French study recognizes that reprocessing plants are amongst the three most hazardous NRNFs, as Subsection 5.1.3, "Safety philosophy," stated,

> "Because chemical processes form an integral part of the nuclear fuel cycle facilities, insurance of safety requires the control of both the chemical and nuclear hazards. It's important to notice that the hazards vary from one facility to another, depending on the processes employed, the age, the output, the physical and/or chemical properties of the substances, and possibly the specific conditions [TECDOC1221]...Therefore, reprocessing plants, high activity liquid waste tanks, and plutonium handling plants are the most dangerous facilities, even if the nature of the dangers is globally the same in the whole fuel cycle."

3.2.5 Electrochemical Processing (Pyroprocessing)

SECY-08-0134 [NRC 2008b] notes that the two main methods of reprocessing of spent nuclear fuel (SNF) used to date are aqueous separations and electrochemical processing. The former employs solvent-extraction techniques for purification. The latter uses an electrochemical technique to purify spent fuel. Electrochemical techniques generate a fuel that is not as pure as aqueous reprocessed fuel, and consequently, this fuel currently is only suitable for the recycling of fuel in advanced burner reactors (ABRs) (fast-neutron reactors), where these "impurities" can be burned.

This subsection summarizes some observations related to the safety of electrochemical processing of SNF. In particular, NUREG-1909 points out the following:

- Electrochemical processing inherently is a batch process so that materials must be moved as solid physical objects in most of the various steps involved. The size of the batches is limited by criticality considerations. The large number of movements of highly radioactive objects containing fissile materials is likely to necessitate high equipment-reliability, low accident-likelihood, and a great need for nuclear-material accountability.

- There is no estimate of the amount and characteristics of failed or used equipment, such as electrodes and crucibles.

- Electrochemical processing per se does not use organic chemicals. This avoids the potential for accident scenarios involving organic chemical reactions (e.g., fire, red oil, resin explosions) and wastes from the cleanup of organic solvents and extractants.

Mariani et al. [1995] studied the criticality safety of the electrorefiner (ER) of the Fuel Cycle Facility (FCF) at Argonne National Laboratory West. They state, "Since the FCF ER is a complicated assembly of hardware, and the ER processes themselves are complex, the number of issues relevant to evaluation of the FCF ER process for criticality safety is substantial. The strategy to maintain criticality safety in the FCF ER process is summarized here, giving a few detailed examples of how the strategy is applied to static inventories, process items, and operations. A full account of the applied strategy has already been given [Mariani et al. 1993]." They also indicate, "In the absence of extensive statistical data on the operation of similar facilities, no formal PRA was performed. Consequently, the definition and classification of abnormal events required careful examination of the design and operation of the ER and required application of sound technical judgment. The classification of individual events as unlikely or extremely unlikely was based on the following technical issues:

a) Equipment and container designs,
b) Physical limits and controls,
c) Administrative controls,
d) Criticality control limits,
e) Process variations,
f) Sampling and analysis uncertainties,
g) Distinguishability of different material forms and containers, and
h) Number of steps or length of time required for the event to develop without notice.

According to the report by the Committee on Electrometallurgical Techniques for Department of Energy (DOE) Spent Fuel Treatment [NAS 2000], the Argonne National Laboratory's (ANL) Experimental Breeder Reactor-II (EBR-II) Spent Nuclear Fuel Treatment Demonstration Project began in June 1996 and ended in June 1999. Four criteria evaluated its success, addressing the process, the waste streams, and the safety of the electrometallurgical demonstration project. Criterion 4 required demonstrating that safety risks, environmental impacts, and nuclear-materials accountancy are quantified and acceptable within regulatory limits. One goal for meeting this criterion was to estimate the safety risks, environmental impacts, and material accountancy for the inventory operations. The Committee believed that this goal was met, based on ANL's safety analysis [Garcia et al. 1999].[15]

In previous reports, this Committee noted their concerns about the scale-up of the HIP[16] process. In particular, the National Research Council [1999] pointed out that ANL-West "…is working with an outside vendor to produce larger beryllia[17] crucibles needed to increase the

[15] This document was not found in the public literature.

[16] Salt-loaded zeolite is mixed with a borosilicate glass and consolidated at high temperature (850 to 900°C) and pressure (14,500- to 25,000-psi) in a hot isostatic press (HIP) to make the final waste form.

[17] Beryllia is another name for beryllium oxide (BeO), a white crystalline oxide. It is notable as it is an electrical insulator with a thermal conductivity higher than any other non-metal except diamond, and actually exceeds that of some metals. Its high melting point leads to its use as a refractory.

throughput of the cathode processor. The scale-up of beryllia crucibles continues to be a problem at the outside vendor. The larger beryllia crucibles are failing mechanically, apparently due to thermal stresses."

Appendix B briefly describes electrochemical processing and the associated literature.

3.3 Insights Gained from the Survey of Accidents and Risk Analyses

The following insights were obtained from the survey of accidents:

1. The most important conclusion is that accidents at reprocessing facilities can result in very severe consequences, up to, and including early fatalities and injuries of personnel, and substantial releases of radioactivity to the environment. The definition of high-consequence events in 10CFR70.61, "Performance requirements," includes, but is not limited to, those internally or externally initiated events that result in an acute worker dose of 1 Sv (100 rem) or greater total effective dose equivalent, or an acute dose of 0.25 Sv (25 rem) or greater total effective dose equivalent to any individual located outside the controlled area. The information available about these accidents was not enough to assess which ones met these criteria, if any. On the other hand, those involving early fatalities clearly exceeded them. In addition, those having a classification of 4, "Accident with local consequences" or higher in the INES scale also are significant accidents. Table 4 presents accidents with documented early fatalities and accidents satisfying this classification.

Table 4: Significant Accidents in Reprocessing Facilities Worldwide
(The number in parenthesis after an accident is the number of resulting early fatalities, if available.)

INES		Kind of Hazard		
Level	Type of Event	Criticality	Radiation	Fire and/or explosion
6	Serious accident			Mayak PA, 9/29/1957
4	Accident with local consequences			Windscale, 1973
Classification not available		Mayak PA, 4/21/1957 (1) Mayak PA, 1/2/1958 (3) Los Alamos Laboratory, 12/30/1958 (1) Mayak PA, 12/10/1968 (1)	Mayak PA, 10/1/1951 (1)	Tomsk, Russia, 4/6/1993

The accident at the Mayak PA facility on September 29, 1957 due to a chemical explosion may be considered a very high-consequence accident as it released a total activity of approximately 74 petabecquerels (PBq[18]) which were dispersed offsite over the territory of the Chelyabinsk, Sverdlovak and Tyumen regions.

[18] A PBq is equal to 10^{15} disintegrations per second (dis/s), and 1 Ci is equal to 3.7×10^{10} dis/s. Hence, 74 PBq is approximately equal to 2×10^{6} Ci.

2. Reprocessing facilities entail the risks associated with the traditional process industries, such as chemical toxicity and chemical reactions leading to fire and explosion, as well as the risks related to nuclear materials (such as radioactive contamination and criticality excursions). An accident may involve a combination of nuclear- and non-nuclear-hazards.

3. Many of the identified accidents occurred during the 1950s and 1960s, presumably because at that time, in comparison to current regulations and standards, there were minimal safety standards and/or regulatory oversight was relaxed or lacking. While accidents during the last two decades were fewer, serious events still occurred, such as those at Siberian Chemical Enterprises (Russian Federation, 1993), THORP (U.K., August 2004), and Tokai-mura (Japan, March 11, 1997). Hence, apparently there is a trend toward decreasing number and severity of consequences of accidents. This is likely due to a combination of implementing safer work practices, increased operating experience, better safety standards, and/or stricter regulatory oversight; it may also reflect the fact that several countries have shut down their main reprocessing facilities (Table 1).

4. Accidents of differing severities (from relatively mild to severe) have occurred in practically all countries having reprocessing facilities (Table 1). No known accidents at reprocessing facilities have occurred in Belgium and China.

5. Two of the identified events (Sellafield, 1983; Karlsruhe, 1999) happened during shutdown of a facility, indicating that there also is risk associated with the shutdown of a reprocessing facility.

The following conclusions were reached after studying the methods and results of risk assessments of reprocessing facilities:

1. A thorough approach, based on a comprehensive PRA that takes into account both radiological and chemical hazards, needs to be considered to assess the risk to receptors from reprocessing operations.

2. One Japanese PRA study used a simplified PRA for a reprocessing facility. A detailed PRA performed for a limited set of accident scenarios associated with this facility first was undertaken, and its models, data, and results were used to develop the simplified PRA. The Japanese researchers that developed and applied the simplified PRA approach to this facility consider it an improvement over the "risk index method" that is typically used for the ISA.

3. The comprehensive risk assessment includes a quantitative evaluation of the risk to receptors. Though failure data may be scarce for this kind of facility, the available data is supplemented with engineering judgment, as was done in the early days of applying the PRA method for nuclear power reactors, and still is used for some technical issues.

4. Since nuclear- and chemical-hazards are distributed throughout reprocessing facilities, approaches that are typically not used for the PRA of nuclear-power reactors, such as methods of hazard analysis and risk evaluation for chemical plants, are employed to identify, model, and quantify the risk of individual processes or the entire facility.

4. QUALITATIVE VS. QUANTITATIVE RISK METHODS

4.1 Introduction

SECY-09-0082 [NRC 2009] updated the progress towards developing a regulatory framework for licensing reprocessing facilities, including a regulatory gap analysis. In terms of risk considerations, Gap #5 indicated that (1) reprocessing facilities would have a higher source term, and thus, present a greater relative risk compared to fuel-cycle facilities licensed under 10 CFR Part 70, and, (2) the ISA methodology for risk assessment in Part 70 does not adequately address the risks posed by facilities with higher risk. The gap analysis recognized the ACNW&M's recommendation for a quantitative approach to risk, such as PRA, rather than ISA because of the limitation in the ISA's treatment of dependent failures, human reliability, uncertainties, and its aggregation of event sequences. However, the gap analysis also expressed some reservations about PRA use in reprocessing facilities, based on lack of relevant, reliable data.

Part 70 requires an ISA, and the risk index method used by most applicants and licensees involves an order-of-magnitude analysis. However, for a reprocessing facility, risk-informing the facility's design and operation through a quantitative risk assessment, in conjunction with the quantitative risk guidelines proposed by the staff in SECY-04-0182 [NRC 2004] and later updated in [NRC 2008a], could have major benefits, such as:

1. Significantly enhance the ability of the NRC's staff to better understand and categorize risk-significant issues. This clearly would be useful in the license review process, in ensuring that the risks posed by a new facility or by additions to an existing one are well characterized and understood.

2. Play an important role in the inspection process in evaluating the risk significance of the inspection findings, in a manner similar to the risk-oversight program for power reactors.

3. Identify risk-important sequences and components to support the allocation of resources to decrease the risk.

Some experience was gained in applying PRA techniques to facilities carrying out operations somewhat similar to large-scale reprocessing facilities. BNL conducted a limited PRA to analyze the risks of "red oil" explosions in the mixed oxide fuel-fabrication facility (MFFF) currently under construction at the DOE's Savannah River site in Aiken, SC. The MFFF uses aqueous processing based on the PUREX process to separate impurities in plutonium feedstock in manufacturing mixed oxide (MOX) fuel. The MFFF is being licensed under Part 70 using ISA methodology, but a limited PRA was carried out because it was felt that PRA methods could offer useful risk insights to NRC's staff reviewers in their analysis of some high-consequence events, such as red-oil explosions. The experience gained through this exercise indicated that while ISA is a useful starting point, it has several limitations, particularly in analyzing common-cause failures and human reliability. PRA has advantages in this regard. The experience of applying PRA techniques to MFFF showed that employing surrogate data from related facilities with similar equipment affords usable results for analyzing the safety of the facility. Over time, the issue of data should become less important for facilities, such as fuel reprocessing plants, as demonstrated by experience with data needs for reactor PRA over the last three decades. However, PRA techniques, developed mainly for operating light-water reactors, must be enhanced in several ways to facilitate their application to the safety of nuclear- and chemical-processes that are characteristic of reprocessing plants. Section 4.4 offers suggestions for these enhancements.

4.2 Approach

As discussed in Chapter 2, based on work carried out by the ACNW&M [Croff et al. 2008] as well as the NEI report [2008], the approach to conducting risk assessment of a reprocessing facility should be a technology neutral one since the overall regulation may need to accommodate different reprocessing technologies currently in various stages of development. While aqueous technologies such as the PUREX process are mature, other technologies are currently under development. Technology-specific regulatory guides could then be used to supplement the regulation. The ACNW&M mentioned the technology neutral approach identified in NUREG-1860 [NRC 2007] for risk-informed performance-based licensing of future power reactors (referred to in Chapter 2) as offering a valuable basis for developing a similar approach to reprocessing facilities.

At a high level, the following features of the technology neutral approach identified in NUREG-1860 are relevant:

1. The approach should be risk-informed and performance-based. Risk-informing is a philosophy that considers risk insights together with other factors to establish requirements that better focus the attention of the licensee and regulatory body on issues of design and operation commensurate with their importance to public health and safety. Risk insights are derived from a risk assessment of the facility; ACNW&M stated its preference for PRA over ISA, as pointed out in Chapter 2, above.

2. The facility design must encompass defense-in-depth and safety margins.

3. Technology-neutral risk-acceptance criteria need to be formulated to help in developing the licensing basis for the facility. Here, the frequency-consequence curve, discussed below, based on the one offered in NUREG-1860 is a possible candidate.

It should be noted that analogs of the reactor safety goals, known as quantitative risk guidelines, were developed for non-reactor nuclear facilities, such as fuel-cycle facilities, in SECY-04-0182 [NRC 2004], and later updated in [NRC 2008a]; they could also apply to reprocessing facilities, regardless of technology.

4.3 Insights from the Technology Neutral Framework of NUREG-1860

The provision of a technology-neutral framework that can be created for diverse technologies, using important probabilistic- and deterministic-criteria governing risk and performance, will facilitate developing a consistent, stable, and predictable set of requirements that are both risk-informed and performance-based. One important feature of the framework is developing a risk acceptance criterion that relates both elements of the risk, frequency, and consequence, posed by the facility or process regardless of technology. NUREG-1860 achieved this via constructing a frequency-consequence curve.

A criterion that specifies limiting frequencies for a spectrum of consequences, from very small to very large, can be denoted via a frequency consequence (F-C) curve. On the F-C plane, this curve provides an acceptable limit in terms of the frequency of potential accidents and their associated consequences. The objective of such a curve would be to establish the licensing basis, i.e., to identify the event sequences that must be mitigated via the design and operation of the plant. This objective involves first establishing criteria for ensuring that the frequency of occurrences of event sequences is inversely related to the consequences, e.g., high-frequency events such as minor transients should have low consequences, and high-consequence events like an explosion followed by a large release of radionuclides should have low frequency. Second, the objective involves establishing criteria that define the acceptable frequencies for different levels of consequences.

4.4 PRA for Reprocessing Facilities

PRA is a fairly mature technology for understanding the vulnerabilities, and predicting the risks posed by commercial nuclear-power reactors. However, fuel-cycle facilities also are chemical processing plants, which present a different set of challenges than power reactors. These differences include the nature and type of hazards they pose, the kinds of accidents that can occur, and the recipients of the risk. This is recognized in the 10 CFR Part 70 regulations governing the licensing of these facilities; in particular, the performance criteria in Part 70.61 specify both the radiological- and chemical-hazards posed by fuel-cycle facilities and establish limits on their consequences for the public, workers, and the environment as a function of the likelihood of events.

Furthermore, the risk assessments for facilities with nuclear- and chemical-processes like reprocessing facilities must account for several features that distinguish such facilities from reactors. These elements, analyzed below for enhancing traditional PRAs, are based on the following: (1) The insights gained from previous work on the risk assessment of possible red-oil excursion events in the proposed MFFF currently under construction at the DOE's Savannah River site, (2) a review of both domestic- and international-safety activities and databases (e.g., FINAS and NMED, mentioned by the end of this section) of non-reactor nuclear facilities, and (3) a review of the challenges faced by the ISA methods that are mandated by 10 CFR Part 70 regulations.

The risk assessment methods able to respond to the needs of the reprocessing facilities should account for the unique features of these facilities that make their risk profile different from those of commercial power reactors. Some of the major differences are noted below:

1. The hazards posed by the facility include toxic chemical- and explosion-hazards in addition to radiological hazards.

2. There is no analog of the reactor core as the main source of hazard in the facility; the source term for both chemical- and nuclear-hazards might be distributed throughout the plant, with the amount in each location varying depending on processing operations.

3. As noted in the report IAEA-TECDOC-1267, “Procedures for Conducting Probabilistic Safety Assessment for Non-Reactor Nuclear Facilities” [IAEA 2002], many current non-reactor nuclear facilities rely heavily on manual control during normal operation, as well as manual actuation to respond to faults and potential accident conditions. Future facilities, as reflected in the designs of proposed fuel-fabrication or reprocessing facilities, are transitioning to fully automated actuation, control, and monitoring.

4. There is a major reliance on operating systems for prevention and control, with less reliance on standby systems.

Due to these differences, the risk and reliability methods, data, and software tools developed for commercial nuclear power plants could be modified to become more suitable for use in the risk assessment of non-reactor nuclear facilities, such as reprocessing plants. In some cases, new tools or new databases may need to be developed.

The following two examples illustrate some of the major differences in the risk- assessment methods needed for these facilities versus those employed in the current PRAs of power reactors.

Example 1: An accident scenario starts with the introduction of some undesired processing chemicals from one vessel to another. Once the materials are introduced into the recipient vessel, a vulnerable condition can be created that will last for a period governed by the response time of the process-control mechanism. For an accident to occur, however, another condition (say, high temperature) may need to also occur during the time that the undesired processing chemical remains and is not cleared.

This example identifies several PRA modeling issues as follows:

1. The initiating event frequency is the frequency of occurrence of the vulnerable condition and its intensity (i.e., the frequency and amount of undesired chemical transfer to the improper location).

2. The period of vulnerability is a random variable governed by the process unit, and response time of the associated process control.

3. Since the systems are running continuously, the failure frequency (not the unavailability as usually is estimated in PRA codes) of the systems responsible for controlling the mixture temperature within the period of vulnerability must be estimated.

This example highlights three important issues relating to PRA models:

1. The chronological/temporal sequence of events is important. Hence, event A followed shortly by event B will not have the same impact as event B followed shortly by event A. This is equivalent to saying the failure of A will challenge B but not the reverse.

2. The system's survival probability within the period of vulnerability must be assessed, rather than the average system unavailability.

3. Both the duration and the chronological sequence of events need to be explicitly considered as a part of the model.

Example 2: A low set point on a relief valve could cause it to open and subsequently close under normal pressure variation during operation. The low set point caused by drift is such that it opens the relief valve on average once every 10 hours. Each time the relief valve opens, the unwanted materials that are transferred to the vessel in which operations are taking place accidentally can be transferred to another vessel. The probability must be calculated of a scenario wherein sufficient material is transported and accumulates in the second vessel leading to an undesired consequence. Enhancements to fault trees or other methods should be defined to evaluate such cumulative probabilities.

Several areas require further development to meet the needs for undertaking a risk assessment for fuel-cycle facilities. Enhancing the PRA tools to address more systematically the heavy reliance on process control and the use of operating systems is a high priority. The reliability and risk assessment of these areas require a dynamic approach, namely, treating time as an explicit independent variable. The objective is to develop tools and techniques for performing a reliability evaluation of the time-evolutionary path of a dynamic system.

This objective could be accomplished by methods to enhance the existing fault tree and event tree routines and their evaluation algorithms to address the risk and reliability issues associated with the fuel cycle facilities. The evaluation algorithms could include stochastic point process models including Markov and Semi-Markov models. These are considered as the preferred methods for quantification rather than the simple reliability equations currently used.

The approach needs to review and integrate the existing risk and reliability methods used in several industries, such as the nuclear-, chemical-, and aircraft-industries. The following are examples of such methods: PRA, ISA, Layers of Protection Analysis (LOPA), Process Hazards Analysis (PHA), Hazard and Operability Analysis (HAZOP), Event Sequence Diagram (ESD), and Petri nets. In particular, attention must be paid to the methods recommended in IAEA-TECDOC-1267, "Procedures for Conducting Probabilistic Safety Assessment for Non-Reactor Nuclear Facilities" [IAEA 2002].

The approach needs to determine the best, most suitable methods, and then identify the additional algorithms necessary for enhancing the existing methods to facilitate a more systematic evaluation of the issues emerging from risk evaluations of fuel-cycle facilities. They will include addressing the methodological needs of, and developing the tools for enhancing risk-assessment techniques. Also, identify those algorithms useful for integrating and transferring the modified fault trees and event trees to an equivalent Markov/Semi-Markov model; this will support the evaluation, and identify the needed parameters, along with the potential performance indicators that could aid in monitoring vulnerable states (also of importance for use in oversight program). Lastly, the types of models and potential sources of data needed for such analyses must be addressed.

Substantive reviews are needed of several important and available data sources, including the ones identified below:

1. Fuel Incident Notification and Analysis System (FINAS) database established by the Committee on Safety of Nuclear Installations of the Nuclear Energy Agency since 1992, now containing 90 events [NEA 1996].

2. Nuclear Material Events Database (NMED) maintained at Idaho National Laboratory (INL) for the NRC.

3. Russian statistical analyses on fuel-cycle facilities [Kolesnikov 2000].

4. Non-reduced (i.e., raw) data from Savannah River, Hanford, and other domestic DOE facilities.

5. Data from various experiments and accidents that will help in developing the knowledge base database for constructing fault trees.

6. Data from other large international fuel-cycle facilities.

These databases will help to identify the types of data needs and modeling requirements for assessing accident progression and conducting consequence analyses in fuel-cycle facilities. This includes identification of the end-states of accident sequences that involve failure of systems designed to control the releases of radiological materials or toxic chemicals that can lead to exposure of the public, workers, or the environment; the estimation of source terms resulting from the accident; and the calculation of consequences. The chemical source terms and their consequences demand particular attention, including mechanisms and amounts of releases, the transport of releases, and the resulting health effects models and standards for different chemicals. Detailed consideration is also needed of the radiological consequence models recommended in the report IAEA-TECDOC-1267 [IAEA 2002] (the chemical consequences were out-of-scope of this IAEA report, but are vital in NRC-regulated fuel-cycle facilities, as recognized in 10 CFR Part 70.61 criteria). Various governmental and industrial organizations that developed or adopted models for chemical release, transport, and consequence include the U.S. DOE (e.g., the models studied in the DOE's Accident Progression and Consequences, APAC, program [Chung 2002]), the U.S. Environmental Protection Agency (e.g., ALOHA [EPA 2007]), the chemical industry (e.g., the models recommended by the American Institute for Chemical Engineers, i.e., [AIChE 2000] and [AIChE 2001]), and those mentioned in previous NRC work (NUREG/CR-6410 [NRC 1998]). These models and approaches need to be reviewed along with chemical-exposure standards and limits developed by various industry bodies for their applicability and usefulness to fuel-cycle facilities. The potential interactions between nuclear- and chemical-hazards should be addressed as well.

5. SUMMARY AND OBSERVATIONS

5.1 Summary

This report discussed the current regulatory context for reprocessing facilities both nationally and internationally. As noted in SECY-09-0082 [NRC 2009], the existing requirements in 10 CFR Part 70 do not adequately address the increased risk posed by a reprocessing facility compared to that of other fuel-cycle facilities. Reprocessing facilities can have higher potential source-terms than other fuel cycle facilities, and accordingly, this may increase their overall risk. The present report concurs with this observation as a result of examining events that have happened in such facilities, and other risk information related to their hazards. In particular, as demonstrated in our review of accidents in reprocessing facilities, it is evident that the potential for health risk to the public is significantly greater than for other types of fuel-cycle facilities. Moreover, accidents at reprocessing facilities can result in very severe consequences, especially onsite including early fatalities and injuries of personnel. Accidents with documented early fatalities and accidents having a classification of 4, “Accident with local consequences” or higher in the INES scale are significant accidents; Table 4 presents them. The accident at the Mayak PA facility in September 1957 due to a chemical explosion released a total activity of approximately 74 PBq (approximately 2 x 10^6 Ci) which were spread offsite over the Chelyabinsk, Sverdlovak and Tyumen regions of the then Russian Federation.

5.2 Observations

NRC licensees and applicants currently implement varying degrees of ISAs, that is, from qualitative to semi-quantitative. Most of them use the order-of-magnitude approach described in NUREG-1520; Revision 1 of this NUREG was published in 2010 [NRC]. For relatively simple nuclear fuel-cycle systems, the ISA approaches may identify potential weaknesses in a facility’s design or operation, and enable the identification of IROFS. However, since the approaches do not incorporate inter-system dependencies, nor provide an integrated assessment of risk, they could miss some essential risk outliers in more complex facilities.

The Commission’s PRA policy statement issued in 1995 states in part, “The use of PRA technology should be increased in all regulatory matters to the extent supported by the state of the art in PRA methods and data, and in a manner that complements NRC’s deterministic approach and supports the NRC’s traditional defense-in-depth philosophy.” The authors of the present report observe that reprocessing facilities are distinct from other fuel-cycle facilities, which are less complex and do not represent a comparable level of risk.

Safety analyses of a reprocessing facility can benefit greatly from the systematic, disciplined procedures embodied in risk assessment methodology. As discussed in Sections 3.2 and 3.3, varying degrees of PRAs for reprocessing facilities already have been carried out in several countries. Notwithstanding the limited data available for PRAs of this type of facilities compared to that for power reactors, the safety analyses of these facilities can benefit from the potential understanding gained by uncovering potential weaknesses in design and identifying dominant contributors to the risk of a plant or facility, such as human errors and dependencies. Moreover, PRAs can indicate those areas of a facility that are not significant from a risk viewpoint. Consequently, in a reprocessing facility, regulatory attention and resources can be allocated according to the risk importance of operator actions, and of its structures, systems, and components. In addition, the approaches to uncertainty analysis commonly employed in reactor PRA studies can be very useful in expressing risks for reprocessing facilities.

5.3 Suggested Considerations

This report documented several accidents at reprocessing facilities, some of which entailed high-consequence events as previously discussed. Studies of these accidents might well offer more details to gain a better understanding of their causes and mechanisms of occurrence. The potential benefits from such studies could lie in identifying weaknesses in the design and/or operation of these facilities, and in establishing areas requiring additional regulatory attention.

In addition, as Volume 9 of the IAEA-TECDOC-1575, Rev. 1 [IAEA 2008d] points out, further development of PRA methods and their supporting data bases are needed to assess the risk associated with reprocessing facilities.

Some PRA studies of reprocessing facilities were carried out in Japan, and the U.K. The review of these studies documented in this report was brief for two main reasons: 1) The studies are not available publicly, so only brief articles were consulted, and, 2) the limited scope of this work. However, if obtained, a review of the original studies could be conducted to assess valuable information, such as the techniques, scope, data, and computer codes employed, and differences between them and a typical reactor PRA. In addition, insights might be obtained about these facilities' safety.

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[1] This document contains a note stating that it “is a working Document for the 'STOA Panel'. It is not an official publication of STOA. This document does not necessarily represent the views of the European Parliament.”

APPENDIX A: Brief Description of Events at Reprocessing Facilities

Tables A-1, A-2, A-3, A-4, A-5, A-6, and A-7, respectively, summarize accidents at reprocessing facilities in France, Germany, India, Japan, Russian Federation, U.K., and the USA. In some events included in the tables for the Russian Federation and the USA, the information about an accident did not specify the type of nuclear facility (i.e., whether it was a reprocessing facility or not); these events were included for the sake of completeness.

The International Nuclear and Radiological Event Scale (INES) is used for communicating to the public the safety significance of events associated with sources of radiation. The scale was developed in 1990 by international experts convened by the IAEA and the Nuclear Energy Agency (NEA) of the Organisation for Economic Co-operation and Development (OECD). According to the INES User's Manual [IAEA 2009], events are classified on the scale at seven levels: Levels 4–7 are termed "accidents" and Levels 1–3 "incidents" events without safety significance are classified as "Below Scale/Level 0." Events that have no safety relevance with respect to radiation or nuclear safety are not classified on the scale.

The tables in this appendix include the INES classification of each event. However, most of these events have not been classified according to this scale or their classification was not found in the public literature. In this case, the column with the heading "INES Level" contains "NA" (not available). Some of the unclassified events have relatively minor consequences, such as an inadvertent spill of radioactivity within a room of a facility with no injuries to workers, and no release of radioactivity from the room. However, it is clear that other unclassified events would have severe consequences and be categorized as accidents in this scale. For example, early fatalities have occurred as a result of an accident; here, "early fatality" is defined loosely as a death that occurred within a few months after the accident occurred. If an accident description does not have an indication of early fatalities, it does not necessarily mean that there were none; it may be due to lack of information about the consequences of the accident. In addition, other "latent fatalities" (those that may happen with a much longer time lag following exposure, such as latent cancers due to radiation exposure) may have occurred as a result of some of the accidents, but this information was unavailable to this study.

All early fatalities were workers of the reprocessing facilities. Taking into account these fatalities only, apparently the event at Mayak PA in the Russian Federation on 1/2/1958 (3 early fatalities) would be categorized as Level 5, "Accident with wider consequences," and the events involving one early fatality (Mayak PA, 10/1/1951; Mayak PA, 4/21/1957; Los Alamos Scientific Laboratory, 12/30/1958; Mayak PA, 12/10/1968) would be classified as Level 4, "Accident with local consequences," in the international scale.

Some tables contain more accidents than others; however, this does not necessarily reflect the safety record of a country or facility. This difference simply is considered to be mainly due to the availability of information from different countries about this type of accident.

Table A-1: Events at Reprocessing Facilities in France

Facility and Date of Accident	Summary	INES Level	Type of Hazard	References
Facility at Fontenay-Aux-Roses, June 26, 1962	Explosion (no detailed information was found about this accident).	NA	Fire / explosion	[Miura 2005]
La Hague, October 2, 1968	An abnormal rate release of gaseous iodine-131 (^{131}I) was detected at the UP2-400 factory. This high-level release (37,000 Bq/s) continued for eight hours, and then decreased for 15 hours before activity was under the authorized limit (370 Bq/s). The total amount of iodine-131 released was evaluated at 185 GBq (the current authorization for iodine is 110 GBq per year). This accident was caused by treating insufficiently cooled gas-graphite fuels. Also, a release of iodine-129 (^{129}I) was reported (7.4 GBq). There was no evaluation of the impact on public health.	NA	Radiation	[Schneider et al. 2001]
La Hague, January 14, 1970	During the chemical dissolution of the fuel, the temperature of the reaction increased sharply and an explosion occurred due to the emanation of hydrogen gas (H_2). The filters of the chimney recorded an activity of 5,900 GBq, mainly due to antimony-125 (^{125}Sb), 95%, and iodine-131 (^{131}I), 5%. The iodine-129 activity released was 2.7 times higher than the annual authorized limit (110 GBq). No measurements in the environment were undertaken after this accident, nor was there an evaluation of the impact on public health.	NA	Fire / explosion	[Schneider et al. 2001]
La Hague, October 1, 1976	A badly designed packaging of tritium (^{3}H)-rich wastes allowed the release of large amounts of tritium in the "Sainte-Hélène" stream: the activity reached 7,400 Bq/l in October 1976. Streams outside nuclear areas usually exhibit a level below 1 Bq/l. From 1977 to 1983, the mean annual tritium activity released was about 5,000 Bq/l. These unexpected high levels of contamination were due to this leakage. Re-packaging operations entailed the further release of 52,000 GBq of tritium. Many other radionuclides were detected over time in this stream, like strontium-90 (^{90}Sr), caesium-137 (^{137}Cs), and cobalt-60 (^{60}Co).	NA	Radiation	[Schneider et al. 2001]

Table A-1: Events at Reprocessing Facilities in France

Facility and Date of Accident	Summary	INES Level	Type of Hazard	References
La Hague, January 2, 1980	A one-meter long through-wall crack was discovered on the La Hague discharge pipe at about 200 meters from the shore. Fishermen received 3.5 times the annual authorized dose, namely 3.486 mSv, compared with the current European limit of 1 mSv/year.	NA	Radiation	[Schneider et al. 2001]
La Hague, April 1980	There was a loss of coolant resulting in gaseous releases (no detailed information about this accident was found[1]).	NA	Radiation	[Schneider et al. 2001]
La Hague, January 6, 1981	Graphite elements burned for 24 hours in a waste silo. The uranium metal caught fire following a mechanical shock during operations. The maximum measured level of air contamination, 700 Bq/m^3, was reached 10 hours after the fire began. The released activity was mainly due to caesium-137 and -134 (^{137}Cs and ^{134}Cs), and ranged between 740 GBq and 1,850 GBq. Strontium-90 (^{90}Sr) was detected in rainwater. A worker received in one day the annual admissible dose, 50 mSv. There was no assessment of the off-site health impact.	3	Fire / explosion	[Schneider et al. 2001]
La Hague, 1983	COGEMA said that strontium-90 contamination was caused by metallic waste stored in concrete pools that released radionuclides into the groundwater and nearby streams. At the end of 1999, the contamination still was detected: 20 Bq/l in 1991 and between 5 and 10 Bq/l later (maximum admissible concentration in drinking water: 36 Bq/l).	NA	Radiation	[Schneider et al. 2001]
La Hague, February 13, 1990	A routine replacement of a chimney filter led to the release of non-filtered, contaminated air for 10 minutes. An estimated 3.7 MBq of caesium-137 were released in the atmosphere. The radiological consequences were not evaluated.	NA	Radiation	[Schneider et al. 2001]

[1] The internet site http://prop1.org/2000/accident/facts5.htm briefly describes this alleged event. However, the site's reputation is not known.

Table A-1: Events at Reprocessing Facilities in France

Facility and Date of Accident	Summary	INES Level	Type of Hazard	References
La Hague, March 11, 1997	An independent laboratory measured a high level of dose- equivalent rate close to the discharge pipe at low tide. COGEMA confirmed the value of 0.3 mSv/hour, giving an individual the annual dose in three and a half hours. With this exposure and the strontium-90 (^{90}Sr) concentration reached inside the pipe, the vicinity of the pipe would have required classification as a "nuclear facility" in accordance with French legal framework on radioprotection.	NA	Radiation	[Schneider et al. 2001]

Table A-2: Events at Reprocessing Facilities in Germany

Facility and Date of Accident	Summary	INES Level	Type of Hazard	References
Karlsruhe, 1985	Fire with resulting gaseous releases (no detailed description of this event was found).	NA	Fire / explosion	[Schneider et al. 2001]
Karlsruhe, between March and April, 1999	Radioactivity escapes from the shutdown reprocessing plant through a defective ventilation system, briefly exposing 31 employees to contaminated air in three cases.	NA	Radiation	PortalU[2]

[2] PortalU is a cooperation of the German "Länder" and the German Federal Government. Its website is http://www.portalu.de/

Table A-3: Events at Reprocessing Facilities in India

Facility and Date of Accident	Summary	INES Level	Type of Hazard	References
Kalpakkam Reprocessing Plant (KARP), January 21, 2003	A leak in an isolation valve separating a high-level liquid waste tank from a low-level liquid waste tank resulted in an increase in the radioactivity level in the latter. Consequently, the six personnel who were involved in a transfer operation were exposed to a higher dose than the annual dose limit. There was no release of radioactivity to the environment.	NA	Radiation	[BARC 2003]

Table A-4: Events at Reprocessing Facilities in Japan

Facility and Date of Accident	Summary	INES Level	Type of Hazard	References
Tokai-mura, March 11, 1997	A fire and explosion contaminated 37 workers with radioactive material. The activity released to the environment was in the order of 10^{-3} GBq for alpha nuclides and several GBq for beta nuclides. The committed dose equivalent was of a magnitude of 10^{-3} to 10^{-2} mSv.	3	Fire / explosion	[JAIF 1997], [IAEA 1999a], and [IAEA 2007]

Table A-5: Events at Reprocessing Facilities in the Russian Federation

Facility and Date of Accident	Summary	INES Level	Type of Hazard	References
Mayak PA,[3] August 19, 1950	Azizova et al. [2005] characterized this accident as "…technology violation at the radiochemical plant." One person suffered acute radiation syndrome (ARS) as a result of the accident.	NA	Radiation	Azizova et al. [2005]

[3] Azizova et al. [2005] describe the Mayak Production Association (Mayak PA) as "...Mayak PA is located in the Southern Ural in the Chelyabinsk region, which is approximately 1000 miles east of Moscow. Mayak PA includes nuclear reactors, radiochemical and plutonium plants, and nuclear waste storage areas." González [1998] points out "Operation of uranium-graphite reactors for plutonium production and a reprocessing plant began in 1948."

Table A-5: Events at Reprocessing Facilities in the Russian Federation

Facility and Date of Accident	Summary	INES Level	Type of Hazard	References
Mayak PA, September 13, 1950	Azizova et al. [2005] characterized this accident as "…technology violation at the radiochemical plant." One person suffered ARS as a result of the accident.	NA	Radiation	Azizova et al. [2005]
Mayak PA, September 20, 1950	Azizova et al. [2005] characterize this accident as "…technology violation at the radiochemical plant." One person suffered ARS as a result of the accident.	NA	Radiation	Azizova et al. [2005]
Mayak PA, October 1, 1951	Azizova et al. [2005] characterize this accident as "…technology violation at the radiochemical plant." Three people suffered ARS, and one died as a result of the accident.	NA	Radiation	Azizova et al. [2005]
Mayak PA, September 20, 1952	Azizova et al. [2005] characterize this accident as "…technology violation at the radiochemical plant." One person suffered ARS as a result of the accident.	NA	Radiation	Azizova et al. [2005]
Mayak PA, March 15, 1953	The accident occurred in a plutonium-processing building. The plutonium had been recovered from irradiated uranium rods. The plutonium solution was transferred between two vessels, assuming that one vessel was empty that actually was not. There was a single criticality excursion in a plutonium nitrate solution in an interim storage vessel. As a result, one operator received an estimated dose of 100 rad, and another operator received an estimated dose of 1,000 rad. The accident caused no physical damage to any equipment.	NA	Criticality	McLaughlin et al. [2000]
Mayak PA, November 6, 1954	Azizova et al. [2005] characterize this accident as "…technology violation at the radiochemical plant." One person suffered ARS as a result of the accident.	NA	Radiation	Azizova et al. [2005]
Mayak PA, December 22, 1955	Azizova et al. [2005] characterize this accident as "…technology violation at the radiochemical plant." One person suffered ARS as a result of the accident.	NA	Radiation	Azizova et al. [2005]
Mayak PA, April 21, 1957	The accident occurred in a glove box in which an excess of uranium accumulated during the filtration of uranyl oxalate precipitate. The history of the criticality excursion is unknown. One operator died, and five others received doses estimated to be upwards of 300 rad.	NA	Criticality	McLaughlin et al. [2000]

Table A-5: Events at Reprocessing Facilities in the Russian Federation

Facility and Date of Accident	Summary	INES Level	Type of Hazard	References
Mayak PA, September 29, 1957	One of the cooling systems of the concrete high-level liquid waste storage tanks broke down permitting the tank to dry out and overheat. The chemical reaction of dry nitrate and acetate salts in the waste tank containing highly active waste caused a chemical explosion and a large release of radionuclides. The total activity dispersed off-site over the territory of the Chelyabinsk, Sverdlovak, and Tyumen regions was approximately 74 PBq. This area was later called the "Kyshtym footprint."	6	Fire / explosion	[Croff et al. 2008] and [González 1998]
Mayak PA, January 2, 1958	The accident involved an enriched uranyl nitrate solution that was being transferred from an experimental vessel to bottles. Rather than use a drain line as prescribed, three workers lifted the experimental vessel and began to move it (to pour the contents manually into bottles) when the criticality excursion occurred. A fourth worker, who was 2.5 m away, was also exposed to irradiation. Total neutron- and gamma-absorbed doses were estimated at 6,000 ± 2,000 rad for the three who lifted the vessel (who died in five to six days), and 600 rad for the coworker at 2.5 m.	NA	Criticality	McLaughlin et al. [2000]
Mayak PA, December 5, 1960	This accident occurred in a building where waste solutions were processed to recover plutonium. Recovery consisted of three successive stages of purification, and took place in two glove boxes. There were multiple criticality excursions associated with a plutonium-carbonate solution in a holding vessel. During the accident and the subsequent cleanup phase, five individuals received doses in the range 0.24 rem to about 2.0 rem. There was no contamination or damage to any of the equipment.	NA	Criticality	McLaughlin et al. [2000]
Mayak PA, September 7, 1962	The accident occurred in a building that housed operations associated with converting plutonium feed-material to metal. The metal subsequently was purified in several processes and then cast into ingots. In each of these steps, dry residues were generated that contained recoverable quantities of plutonium. The accident occurred during the chemical dissolution of some of these residues in a vessel of unfavorable geometry. Three criticality excursions associated with a plutonium-nitrate solution in a dissolution vessel occurred, but there were insignificant exposures.	NA	Criticality	McLaughlin et al. [2000]

Table A-5: Events at Reprocessing Facilities in the Russian Federation

Facility and Date of Accident	Summary	INES Level	Type of Hazard	References
Siberian Chemical Combine,[4] January 30, 1963	This accident occurred in a waste recovery line of a uranium-metal production building. The waste feed was a dry precipitate. The first step in recovering the uranium was a concentrated nitric acid dissolution process. Multiple criticality excursions associated with a uranyl nitrate solution happened in a collection vessel. Four people standing 10 m away from the collection vessel received radiation doses of 6 to 17 rad. No damage occurred to the vessel, nor was there any contamination of the surroundings.	NA	Criticality	McLaughlin et al. [2000]
Siberian Chemical Combine, December 2, 1963	This accident occurred in an enriched uranium reprocessing and purification facility. The combination of the unfavorable geometry of the holding vessel and the unplanned accumulation of much larger than expected quantities of organic solutions led to multiple criticality excursions associated with an organic solution of uranium in a vacuum-system holding vessel. The largest individual dose received was less than five rem. There was no damage to the equipment or radioactive contamination.	NA	Criticality	McLaughlin et al. [2000]
Mayak PA, December 16, 1965	This accident occurred in a residue-recovery area of a metal- and fissile-solution processing building. Multiple criticality excursions happened, associated with an uranyl nitrate solution in a dissolution vessel. Of the personnel in the area at the time of the accident, 17 received doses of 0.1 rem or less, 7 between 0.1 and 0.2 rem, and 3 between 0.2 and 0.27 rem. The process equipment was not damaged, and no contamination occurred.	NA	Criticality	McLaughlin et al. [2000]

[4] According to the NTI website (http://www.nti.org/db/nisprofs/russia/fissmat/putomsk/tomsk7.htm#SKHK), the closed city of Seversk, formerly Tomsk-7, is the location of the Siberian Chemical Combine (SKhK). The SKhK comprises several large facilities: The Reactor Plant (housing five plutonium-production reactors); the Isotope Separation Plant; the Radiochemical Plant; the Conversion Plant; the Chemical Metallurgical Plant; the Scientific Research and Design Institute; fissile-material storage facilities, radioactive-waste-management facilities, and several auxiliary facilities.

Table A-5: Events at Reprocessing Facilities in the Russian Federation

Facility and Date of Accident	Summary	INES Level	Type of Hazard	References
Mayak PA, December 10, 1968	The accident occurred in a building where various chemical and metallurgical operations with plutonium were performed. A vessel with unfavorable geometry was being used temporarily in an improvised and unapproved operation for storing a plutonium organic solution. Two independent handling operations with this same vessel and same contents less than one hour apart resulted in two prompt critical excursions. A weak excursion occurred between the two energetic ones, when no personnel were present. One person received an estimated absorbed dose of about 700 rem; another received about 2,450 rem and died about one month afterwards.	NA	Criticality	McLaughlin et al. [2000]
Siberian Chemical Combine, December 13, 1978	There was a single criticality excursion associated with plutonium metal ingots in a storage container. An operator received an estimated total body dose of 250 rad, and more than 2,000 rad to his arms and hands. Seven other people received doses between 5 and 60 rad. The equipment was not damaged, and no contamination resulted.	NA	Criticality	McLaughlin et al. [2000]
Siberian Chemical Enterprises,[5] April 6, 1993	The accident occurred during reprocessing of irradiated reactor fuel, and damaged both the reprocessing line and the building, resulting in the release of about 30 TBq of beta and gamma-emitting radionuclides and about 6 GBq of ^{239}Pu. No personnel were overexposed. The accident happened due to a chemical (red oil) reaction.	3	Fire / explosion	[IAEA 1998]

[5] The Siberian Chemical Enterprises (SCE) site is located in the Russian Federation. The countryside around SCE is relatively sparsely populated apart from the regional capital Tomsk and Tomsk-7 (now known as Seversk), which lie south of the SCE site.

Table A-6: Events at Reprocessing Facilities in United Kingdom

Facility and Date of Accident	Summary	INES Level	Type of Hazard	References
Windscale, August 24, 1970	The plant was being used to recover plutonium from miscellaneous scrap. A criticality accident happened with an organic solution of plutonium in a transfer vessel. A 150 mm diameter hole was cut through the concrete roof, and a line to the vessel was opened. Two people were in the plant at the time of the accident; one received an estimated dose of 2 rad, the other less than 1 rad.	NA	Criticality	[McLaughlin et al. 2000]
Windscale, 1973	Radioactive material was released accidentally into an operating area in the plant after an exothermal chemical reaction in a reprocessing tank.	4	Fire / explosion	[Croff et al. 2008] and [Schneider et al. 2001]
Sellafield, during November 1983	Highly radioactive waste liquor was discharged accidentally into the sea. The accident was due to a failure of communication between shifts, so that a tank that was assumed to contain liquid suitable for marine disposal, in fact contained highly radioactive material that was discharged into the sea creating an environmental hazard. This incident occurred during plant shutdown for routine annual maintenance.	NA	Radiation	[Lardner 1996]
THORP. The leak began before August 28, 2004, and remained undiscovered until April 20, 2005. Likely the leak was relatively small until January 2005.	On 20 April 2005, British Nuclear Group Sellafield Limited (BNGSL) discovered a leak from a pipe that supplied highly radioactive liquor to an accountancy tank in a part of the THORP at Sellafield, known as the ‘feed clarification cell’. Approximately 83 000 liters of dissolver product liquor, containing about 22,000 kilograms of nuclear fuel (mostly uranium incorporating around 160 kilograms of plutonium), had leaked onto the floor of the cell. The leak came from a pipe that was severed completely at a point just above where it enters a tank. The leak remained undetected for about eight months. THORP was shut down after the incident; consent to begin reprocessing operations again was granted on January 9, 2007.	3	Radiation	[HSE 2007] and [Tait][6]

[6] These slides have no date.

Table A-6: Events at Reprocessing Facilities in United Kingdom

Facility and Date of Accident	Summary	INES Level	Type of Hazard	References
Dounreay, September 2005	The accident occurred during decommissioning of the facility. Some radioactive reprocessing residues in the form of dissolved spent fuel, kept in underground tanks, are pumped to an area where they are mixed with cement, then stored in 500-litre drums. A robot carries out these operations, but on the day of the accident, 266 liters of radioactive material and 300 kgs of cement were spilled and solidified onto the floor of a treatment cell. There were no injuries to workers and no radiological release occurred from the cell itself.	NA	Radiation	[BBC 2005] and [HSE 2005]

Table A-7: Events at Reprocessing Facilities in United States of America[7]

Facility and Date of Accident	Summary	INES Level	Type of Hazard	References
Hanford Works, December 4, 1951	A fire occurred in the waste storage room of a small laboratory building. Apparently, nitric acid on rags used for cleaning and decontaminating parts that had been boxed for disposal spontaneously combusted. Firefighters used water to extinguish the fire. The fire burned for about four hours. Firefighting was hampered by the need for protection against inhaling plutonium. The fires in the building's air filters were difficult to extinguish; they had to be removed to prevent further release of Pu-239 to the atmosphere. The fire damage was not severe, but the contamination spread throughout the building was quite severe, so that it had to be abandoned.	NA	Fire / explosion	[Cadwallader et al. 2005]

[7] McGuire [1988] states that "The Nuclear Fuel Services reprocessing plant was plagued by many small releases into ground water, surface water, and air as well as unusually high occupational radiation exposures. However, the plant never had an accident of significance for offsite emergency preparedness." Since he does not provide information about these events, they are not included in this table.

Table A-7: Events at Reprocessing Facilities in United States of America[7]

Facility and Date of Accident	Summary	INES Level	Type of Hazard	References
Savannah River (facility code named TNX), January 12, 1953	This was a test bed facility built in 1951 to perform experiments using mockups of the prototype of the equipment for the tributyl-phosphate (TBP) process. One part tested was a nitric-acid evaporator to remove excess nitric acid from the plutonium stream. The TNX evaporator exploded due to a “red oil” reaction. Investigation revealed that TBP and a kerosene-like diluent, which were immiscible in the aqueous acid, nonetheless had transferred into the evaporator and then chemically reacted with the materials in solution, causing the explosion. This was a matter of concern since it might occur in production units.	NA	Fire / explosion	[Cadwallader et al. 2005]
Hanford Works, 1953[8]	During a process evolution, an acid solution and a caustic solution were mixed improperly. Metal waste supernatant was being pumped from a holding tank to a blend tank to prepare process feed for the TBP. Sixty percent nitric acid first was added to the blending tank, then the metal waste, and the agitator for the tank was started. A geyser of liquid quickly rose about 10 meters, and the prevailing wind carried the liquid onto several workers, contaminating them with about 4,000 counts per minute (cpm); they experienced stinging sensations from the liquid. Investigation revealed that the operator had activated the agitator switch; however, whilst it started to operate, it failed to continue. Areas near the blend tank contained a yellowish liquid that read 35 rem/hour at 6 inches. There were a significant number of reports of personnel contamination from maintenance and operations tasks.	NA	Fire / explosion	[Cadwallader et al. 2005]

[8] Cadwallader et al. do not give the exact date of this event.

Table A-7: Events at Reprocessing Facilities in United States of America[7]

Facility and Date of Accident	Summary	INES Level	Type of Hazard	References
Hanford Works (REDOX plant), June 18, 1956	An operator was in the control room of the plutonium-concentration facility when a continuous survey meter began to “break down’, that is, saturate counts at its given setting. The operator investigated and found the meter to be working correctly; the surfaces of the control room had become contaminated to greater than 7 million disintegrations per minute (dpm), off the scale of the instrument. Two spills of process solution (highly concentrated acidic plutonium product) were found on the floor behind the control panel. The solution had entered the control room by an instrument air-line. The skin of the operator who noted the survey meter’s “breakdown” was contaminated to over 40,000 dpm. Nasal smears read 30,000 dpm, and all of his clothing was contaminated. A rough estimate was that only about 350 cm^3 was spilled in the control room.	NA	Radiation	[Cadwallader et al. 2005]
Hanford Works (silver reactors), February 1958	Silver reactors were used to react iodine from the Hanford stack effluents. A silver reactor had been flushed with ammonium hydroxide and then with water to remove a plug of silver compounds that had accumulated after repeated regenerations. The ammonia compound allowed a rapid exothermic reaction.	NA	Fire / explosion	[Cadwallader et al. 2005]
Hanford Works (silver reactors), 1958[9]	Operators noted that after processing ammonia-laden off-gas, the silver reactors would experience exothermic reactions that would increase outlet temperatures.	NA	Fire / explosion	[Cadwallader et al. 2005]
Oak Ridge Y-12 Plant, June 16, 1958	This accident occurred in a building in a process designed to recover enriched uranium from various solid wastes. Multiple criticality excursions occurred from an uranyl-nitrate solution in a water-collection drum. Eight people received significant radiation doses (461, 428, 413, 341, 298, 86.5, 86.5, and 28.8 rem). At least one person owes his life to the prompt and orderly evacuation plans that were followed.	NA	Criticality	[McLaughlin et al. 2000]

[9] Cadwallader et al. do not give the exact date of this event.

Table A-7: Events at Reprocessing Facilities in United States of America[7]

Facility and Date of Accident	Summary	INES Level	Type of Hazard	References
Los Alamos Scientific Laboratory, December 30, 1958	The operations performed at the facility where the accident occurred were those chemical steps used to purify and concentrate plutonium from slag, crucible, and other lean residues remaining after recovery processes. A single criticality excursion was associated with a plutonium organic solution in an organic treatment tank. The accident resulted in the death, 36 hours later, of an operator; the estimated dose to his upper torso was 12,000 ± 50% rem. Two other persons received radiation doses of 134 and 53 rem. No equipment was contaminated or damaged even though the shock from the accident displaced the tank about 10 mm at its supports.	NA	Criticality	[McLaughlin et al. 2000]
Idaho Chemical Processing Plant, October 16, 1959	This accident occurred in a chemical processing plant that accepted spent-fuel elements from various reactors, among other items. The fissile material involved in the accident (34 kg of enriched uranium, in the form of uranyl nitrate) was stored in a bank of cylindrical vessels with favorable geometry. The initiation of a siphoning action, inadvertently caused by an air-sparging operation, transferred about 200 liters of the solution to a 15,400 liter tank containing about 600 liters of water. Multiple criticality excursions were associated with the uranyl-nitrate solution in a waste-receiving tank. Because of thick shielding, nobody received significant prompt gamma- or neutron-doses. During evacuation of the building, airborne fission products (within the building) resulted in combined beta and gamma doses of 50 rem (one person), 32 rem (one person), and smaller amounts to 17 people.	NA	Criticality	[McLaughlin et al. 2000]

Table A-7: Events at Reprocessing Facilities in United States of America[7]

Facility and Date of Accident	Summary	INES Level	Type of Hazard	References
Oak Ridge National Laboratory (ORNL) (Radiochemical Processing Pilot Plant), Nov. 20, 1959	A nonnuclear explosion involving an evaporator occurred in a shielded cell. Plutonium was released from the processing cell, probably as an aerosol of fine particles of plutonium oxide. No radioactive material was released from the ventilation stacks; no contamination of grounds and facilities occurred outside of a relatively small area of ORNL immediately adjacent to the site of explosion. No one was injured by the explosion, and no one received more than 2% of a lifetime body burden of plutonium or an overexposure to sources of ionizing radiation, either during the incident or the subsequent cleanup operations. The explosion is considered to have resulted from the rapid reaction of nitrated organic compounds formed by inadvertent nitration of about 14 liters of a proprietary decontaminating reagent.	NA	Fire / explosion	[King and McCarley 1961]
Savannah River, September 13, 1960	A valve corridor was highly contaminated by leaking coolant water that had been exposed to high activity waste from the PUREX process, and then had leaked through a defective waste-evaporator reboiler. Approximately 5,000 Curies was released with the water. Most of the radioactivity was contained in the building and was flushed to the waste-handling facilities. Minor amounts of liquid entered the seepage basins outside the building. A series of operating errors and miscommunications allowed the initial leakage water to flow back through open valves on steam traps and then out into the corridor. The measured radiation readings after the incident were as high as 400 rad/hour at 30 cm. Cleanup was costly and time consuming.	NA	Criticality	[Cadwallader et al. 2005]
Idaho Chemical Processing Plant, January 25, 1961	This accident occurred in the main process building where fission products were separated chemically from dissolved spent fuel. The uranium then was concentrated via evaporation. Multiple criticality excursions occurred from an uranyl-nitrate solution in a vapor-disengagement vessel. All employees evacuated promptly, and were exposed only to minimal doses (<60 mrem) caused by airborne fission products after leaving the building.	NA	Criticality	[McLaughlin et al. 2000]

Table A-7: Events at Reprocessing Facilities in United States of America[7]

Facility and Date of Accident	Summary	INES Level	Type of Hazard	References
Hanford Works, April 7, 1962	This accident occurred in a plutonium-waste chemical recovery facility, and involved: 1) Cleaning up the floor of a solvent extraction hood, 2) a product-receiver tank that could overflow into the hood, 3) a temporary line running from the hood floor to a transfer tank, and, 4) the apparent improper operation of valves. There were multiple criticality excursions associated with a plutonium solution in the transfer vessel. Three people received significant doses of radiation (110, 43, and 19 rem). The accident itself caused no damage or contamination, but precipitated the final shutdown of the plant.	NA	Criticality	[McLaughlin et al. 2000] and [Gamertsfelder et al. 1963]
Hanford Works (PUREX plant), September 3, 1963	The plant stacked 62 curies of I-131 over a three-day period, beginning on September 3. Several "short cooled" fuel elements were placed in the dissolver, releasing significant amounts of I-131.	NA	Radiation	[Cadwallader et al. 2005]
Hanford Works (PUREX plant), November 6, 1963	Resin in a plutonium-extraction column caught fire. The fire spread plutonium contamination throughout the building and some outside of the building.	NA	Fire / explosion	[Cadwallader et al. 2005]

Table A-7: Events at Reprocessing Facilities in United States of America[7]

Facility and Date of Accident	Summary	INES Level	Type of Hazard	References
Savannah River, February 12, 1975	There was a 'red oil' explosion in the Thermal Denitrator at the reprocessing plant. The cause was attributed to the exothermic reaction between TBP and uranyl nitrate.	NA	Fire / explosion	[Cadwallader et al. 2005]
Hanford Works, 1976[10]	An ion-exchange column underwent a chemical reaction and exploded. A worker, who was in front of the glovebox that housed the column at that time was injured by glass from the glovebox's shattered window, and contaminated with americium-241. The americium that had collected in the ion-exchange resin before the explosion was dispersed throughout the room during the explosion.	NA	Fire / explosion	[Cadwallader et al. 2005]
Idaho Chemical Processing Plant, October 17, 1978	The accident occurred in a shielded operation of a fuel-reprocessing plant in which solutions from the dissolution of irradiated reactor fuel were processed by solvent extraction to remove fission products and recover the enriched uranium. The history is unknown of criticality excursions associated with an uranyl-nitrate solution in a lower disengagement section of a scrubbing column. There were no significant exposures to personnel and no damage to process equipment. As a direct result of this event, the plant underwent an extended, expensive shutdown.	NA	Criticality	[McLaughlin et al. 2000]
UTP, Ontario, 1980	An explosion occurred with gaseous releases.[11]	NA	Fire / explosion	[Schneider et al. 2001]
Hanford reprocessing plant, May 14, 1997	Explosion (no additional information about this event was found).	NA	Fire / explosion	Miura [2005]

[10] Cadwallader et al. do not give the exact date of this event.

[11] Schneider et al. [2001] do not provide more information about this event, and no additional information was found. Since little is known about this accident, it is included in this table, even though it happened in Canada.

APPENDIX B: Brief Description of Electrochemical Processing (Pyroprocessing)

Section 6.2, "Pyroprocessing," of NUREG-1909 [Croff et al. 2008] gives an overview of the technology of electrochemical processing, and points out that there are many manifestations of electrochemical processing in the nuclear industry, several of which are directed at spent fuel recycle. As applied to reprocessing of SNF, electrochemical processing involves the use of molten salts and metals in an electrochemical cell to separate the SNF constituents. Electrochemical processing involves anodization (oxidation) of a metal feed material into a molten salt electrolyte and then reduction at a cathode.

The feed to electrochemical processing was originally intended to be metallic spent fuel, and the process lends itself best to reprocessing this type of fuel. As a consequence, the current Department of Energy (DOE) plans call for electrochemical processing to be used to reprocess metallic or possibly nitride SNF containing the transuranic (TRU) actinide elements after irradiation in a fast-spectrum transmutation reactor.[1] However, oxide fuels such as those from light-water reactors (LWRs) can be electrochemically processed by first converting them to metal through a head-end step that reduces the oxide to metal. This reduction is best accomplished using finely divided oxide, which can be prepared using voloxidation[2] to pulverize the oxide fuel. Process modifications are possible that separate uranium, plutonium, and other actinides from the remainder of the radionuclides. Figure A-1 [ANL 2002] presents the electrochemical processing flowsheet for oxide SNF under development by Argonne National Laboratory and other organizations such as Korean Atomic Energy Research Institute (KAERI).

The following are the major steps in this flowsheet:

- Oxide SNF is chopped into segments and voloxidized (not shown).
- Most of the oxides in the SNF are reduced to the metal.

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[1] Long-lived radioactive isotopes, especially actinides such as plutonium and neptunium but also selected fission products such as ^{99}Tc and ^{129}I, are converted to shorter-lived fission products or stable isotopes by fission and/or neutron capture from neutrons generated in a transmutation reactor. In this reactor, the TRU elements would be fissioned to produce energy and what is primarily a fission product waste, thus removing by transmutation the principal long-term heat-producing actinides from the wastes.

[2] The spent fuel is chopped (sheared) into segments using the voloxidation (volume oxidation) process. This process depends on the oxidation of the UO_2 spent fuel matrix to lower density U_3O_8 to break down the fuel matrix and release trapped gases from it.

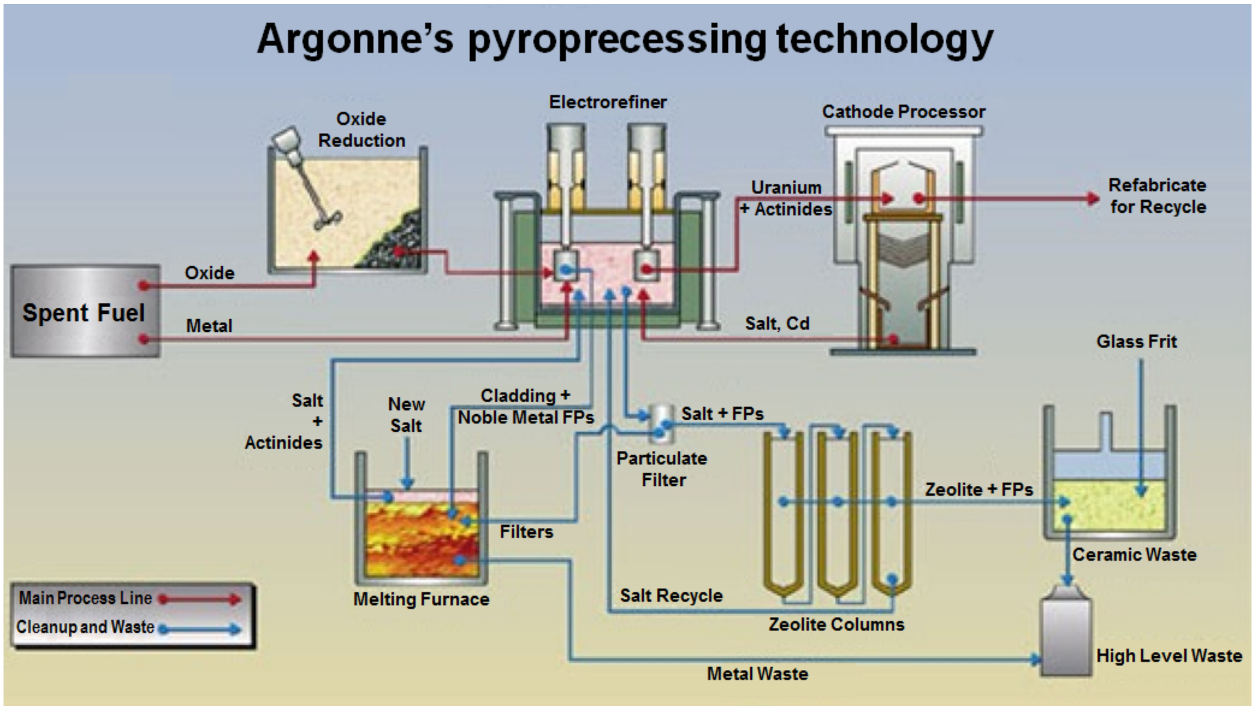

Figure B-1: Diagram of Electrochemical Processing Operations

- The metal from oxide reduction or metallic SNF, including the cladding in either case, becomes the anode in an electrorefiner (ER). The ER is essentially a crucible containing a molten electrolyte salt (a mixture of LiCl and KCl) atop a layer of liquid cadmium metal. The anode and two cathodes operating at different voltages are inserted into the molten salt. After operating for about 12 hours, the ER contains the following:
  - The anode contains elements that are stable as metals under the conditions in the ER (e.g., zirconium, technetium, iron, molybdenum).
  - One cathode contains most of the uranium as metal.
  - The other cathode (liquid cadmium) contains some of the uranium and rare earth fission products plus essentially all of the TRU elements as metal.
  - The molten salt contains most of the fission products that are stable as chlorides under the conditions in the ER (e.g., cesium, strontium, barium).
- The metallic products associated with all three electrodes also contain entrained electrolyte salt and cadmium.
- The cathodes are separately inserted into a cathode processor in which the entrained electrolyte salt and cadmium are recovered for recycle by vacuum distillation.
- The uranium metal is converted to an appropriate form, either hexafluoride for re-enrichment or oxide for direct reuse or disposal. The extent to which additional cleanup of the uranium might be necessary before conversion is not known.

- The TRU metal goes to an injection casting furnace (not shown) where it is refabricated into new fuel for a fast transmutation reactor.

- The metal left at the anode, including the cladding, is heated in a metal waste furnace to produce a solid metallic waste form having zirconium as the major constituent for LWR fuels and iron as the major constituent for stainless-steel clad fuels.

- The fission-product-laden salt is circulated through a zeolite ion exchange bed which incorporates the fission products into the zeolite matrix. The loaded zeolite is consolidated into a monolithic form by combining it with borosilicate glass frit and sintering it, which converts the zeolite to the mineral sodalite in a waste form called glass-bonded zeolite [NAS, 2000] [Kim, 2006]. Processes to improve the removal of fission products from the salt and recycle the salt are under development [Simpson, 2007].

A report by the Committee on Electrometallurgical Techniques for DOE Spent Fuel Treatment [NAS 2000] and a paper by Laidler et al. [1997] describe electrochemical processing technology for SNF in more detail.

Several reports by the International Atomic Energy Agency (IAEA) address electrochemical processing; in particular, the report “Guidance for the Application of an Assessment Methodology for Innovative Nuclear Energy Systems, International Project on Innovative Nuclear Reactors and Fuel Cycles manual — Safety of Nuclear Fuel Cycle Facilities,” [IAEA 2007] describes facilities of the nuclear fuel cycle, including electrochemical processing. An IAEA report [IAEA 2008] describes the technologies used for spent-fuel reprocessing, including electrochemical processing, and their implementation in several countries.

A report by the Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development [NEA 2004] gives a detailed description of the implementation of electrochemical processing in countries worldwide, including the USA.

Willit et al. [1992] present a comprehensive review of the literature on uranium and plutonium electrorefining in molten salts. It covers work published from 1943 to November 1991. Electrodeposition and electrodissolution at solid and liquid metal electrodes are discussed as well as mass transfer in liquid metal and molten salt phases. The journal “Nuclear Technology” dedicated the issue of May 2008 (Vol. 162) to electrochemical processing. Recently, a paper by Goff et al. [2011] also briefly describes electrochemical processing.

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11. ABSTRACT (200 words or less)
This report addresses methods for assessing the risks posed by a reprocessing facility, which have not previously been quantified relative to other fuel-cycle facilities. Reprocessing facilities can have higher potential source terms than other fuel-cycle facilities, which heighten the risk relative to the other facilities. This report explores the potential hazards that these facilities pose to the public, workers, and the environment by discussing literature on the regulation of these facilities and reviewing the experience of current operating facilities worldwide. It offers an overview of actual events and their consequences at these facilities. It also contains supporting information for assessing the feasibility, advantages, and disadvantages of undertaking detailed versus simplified quantitative risk assessments, for the range of events associated with large reprocessing facilities. The report gleans insights on regulating reprocessing hazards and risks from reports such as NUREG-1909 [Croff et al. 2008], and a white paper from the Nuclear Energy Institute [NEI 2008].

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