UCRL-JC-132632
Preprint

ADVANCED RESEARCH WORKSHOP: NUCLEAR MATERIALS SAFETY

State Regional Education Center (SEC)
Aerodromnaya 4
St. Petersburg, Russia 197348
June 8–10, 1998

LESLIE J. JARDINE
Lawrence Livermore National Laboratory
Livermore, CA 94551

MIKHAIL M. MOSHKOV
V. G. Khlopin Radium Institute
St. Petersburg, Russia

January 28, 1999

Lawrence Livermore National Laboratory

Work performed under the auspices of the U.S. Department of Energy by Lawrence Livermore National Laboratory under Contract W-7405-ENG-48.

PREFACE

LESLIE J. JARDINE
Lawrence Livermore National Laboratory
Livermore, CA 94551 U.S.A.

The Advanced Research Workshop (ARW) on Nuclear Materials Safety held June 8–10, 1998, in St. Petersburg, Russia, was attended by 27 Russian experts from 14 different Russian organizations, seven European experts from six different organizations, and 14 U.S. experts from seven different organizations. The ARW was conducted at the State Education Center (SEC), a former Minatom nuclear training center in St. Petersburg. Thirty-three technical presentations were made using simultaneous translations.

These presentations are reprinted in this volume as a formal ARW Proceedings in the NATO Science Series. The representative technical papers contained here cover nuclear material safety topics on the storage and disposition of excess plutonium and high enriched uranium (HEU) fissile materials, including vitrification, mixed oxide (MOX) fuel fabrication, plutonium ceramics, reprocessing, geologic disposal, transportation, and Russian regulatory processes.

This ARW completed discussions by experts of the nuclear materials safety topics that were not covered in the previous, companion ARW on Nuclear Materials Safety held in Amarillo, Texas, in March 1997. These two workshops, when viewed together as a set, have addressed most nuclear material aspects of the storage and disposition operations required for excess HEU and plutonium (see Fig. 1, Opening Remarks). As a result, specific experts in nuclear materials safety have been identified, know each other from their participation in the two ARW interactions, and have developed a partial consensus and dialogue on the most urgent nuclear materials safety topics to be addressed in a formal bilateral program on the subject. A strong basis now exists for maintaining and developing a continuing dialogue between Russian, European, and U.S. experts in nuclear materials safety that will improve the safety of future nuclear materials operations in all the countries involved because of the positive synergistic effects of focusing these diverse backgrounds of nuclear experience on a common objective—the safe and secure storage and disposition of excess fissile nuclear materials.

ACKNOWLEDGMENTS

LESLIE J. JARDINE
Lawrence Livermore National Laboratory
Livermore, CA 94551, U.S.A.

MIKHAIL M. MOSHKOV
V. G. Khlopin Radium Institute
28, 2-nd Murinsky Ave., St. Petersburg, Russia 194021

This Advanced Research Workshop (ARW) generated a second timely important interaction on the topic of nuclear materials safety management for the disposition of excess weapons plutonium and high enriched uranium. The first ARW, held in March 1997 in Amarillo, Texas, was documented in a companion ARW volume. These two volumes form a sound foundation upon which to build future interactions and multi-lateral programs that can enhance the safety of the disposition of excess fissile materials no longer required by the United States and Russia.

The authors are grateful for financial support from the North Atlantic Treaty Organization (NATO) through a Disarmament Program Advanced Research Workshop grant (DISRM-ARW-970483), the United States Department of Energy (U.S. DOE), the Ministry of the Russian Federation for Atomic Energy (Minatom) and the Amarillo National Resource Center for Plutonium (ANRCP). Without this support, this ARW and the invaluable exchanges between ARW participants from NATO countries and Russia would not have been possible. The authors want to specifically thank Nancy Schulte, NATO, for her continued encouragement to organize and conduct this ARW.

The authors appreciate the outstanding assistance of the organizing committee, whose other members were K. L. Peddicord, Texas A&M University; Leonard N. Lazarev, V. G. Khlopin Radium Institute; and Fred Witmer and Paul Krumpe, U.S. Department of Energy. The endless encouragement and enthusiasm of Lee Peddicord are specifically acknowledged, as are the efforts of Leonard Lazarev, which included coordinating all aspects of the Russian technical participation in the ARW.

The organizing committee would like also to express their gratitude to the speakers from Belgium, England, Germany, France, Russia, and the United States who developed technical papers and presentations as well as the participants and others who attended the ARW and contributed to the discussions. To all those who made critical and other contributions, the authors express their sincere thanks.

The State Education Center in St. Petersburg provided excellent meeting and lodging facilities for the ARW. The staff of the V. G. Khlopin Radium Institute handled all the logistical arrangements for meeting, transportation, and lodging in Russia, which made major contributions to the success of the ARW. It is a pleasure to personally thank Phyllis Stephens from Lawrence Livermore National Laboratory for her innumerable efforts in handling the NATO country participant logistics, and coordinating these with the V. G. Khlopin Radium Institute staff in Russia, in addition to immeasurable organizational details flawlessly handled for the authors before, during, and after the ARW.

The ARW presentations and discussions were enhanced by the suberb quality of the professional interpreters, Cyril Flerov and Vova Khavkin; their work was outstanding.

Finally, the authors gratefully acknowledge the extensive work of Pat Boyd, Lawrence Livermore National Laboratory, the technical editor, for producing a manuscript of high quality appropriate to the importance of the ARW that this book represents.

OPENING REMARKS

LESLIE J. JARDINE
Lawrence Livermore National Laboratory
Livermore, California 94551 U.S.A.

With the ending of the Cold War and the implementation of various nuclear arms reduction agreements, the United States and Russia are actively engaged in the dismantlement of tens of thousands of nuclear weapons. As a result, large quantities of fissile materials, including more than 100 tons of weapons grade plutonium and significantly more high-enriched uranium (HEU) are becoming excess to the military needs in each country. To meet non-proliferation goals and to ensure the irreversibility of the nuclear arms reductions, these excess fissile materials must be placed and maintained over the next decades in safe, secure storage until the fissile materials are dispositioned as a nuclear reactor fuel, or in the case of plutonium, immobilized in a glass or ceramic solid form for direct geologic disposal. Figure 1 illustrates, functionally and in the format of a nuclear fuel cycle, the various nuclear facilities in Russia and the United States that are associated with these nuclear materials.

The strategic activities in Figure 1 used to maintain nuclear weapons in Russia and the U.S. will generate excess plutonium and high enriched uranium (HEU) requiring storage and disposition operations. The storage and disposition operations are of interest to this Advanced Research Workshop (ARW) on nuclear materials safety. The HEU disposition activities require that weapons components be converted to uranium oxides, then to UF_6, and finally blended down as UF_6 to non-weapons-usable low-enriched uranium (LEU) to make feed material for eventual use in conventional commercial light-water reactor fuel fabrication and afterwards reactors. The plutonium disposition activities require that stored weapons components be converted to oxides suitable for use in existing reactors as mixed-oxide (MOX) fuel or be permanently discarded without energy utilization as a stable glass or ceramic solid form in a geologic repository. Russia is pursuing the MOX fuel disposition approach in existing reactors which includes light-water reactors and possibly fast reactors, followed by proposed reprocessing and plutonium recycle after a period of spent MOX fuel storage. The United States is pursuing both MOX fuel and plutonium immobilization disposition approaches but will use direct disposal of spent MOX fuel with no reprocessing and plutonium recycle after a period of spent MOX fuel storage. It should be noted in Figure 1 that a geologic disposal repository is required for any and all approaches to both HEU and plutonium disposition. Geologic disposal needs to be explicitly considered and included in all future disposition decisions.

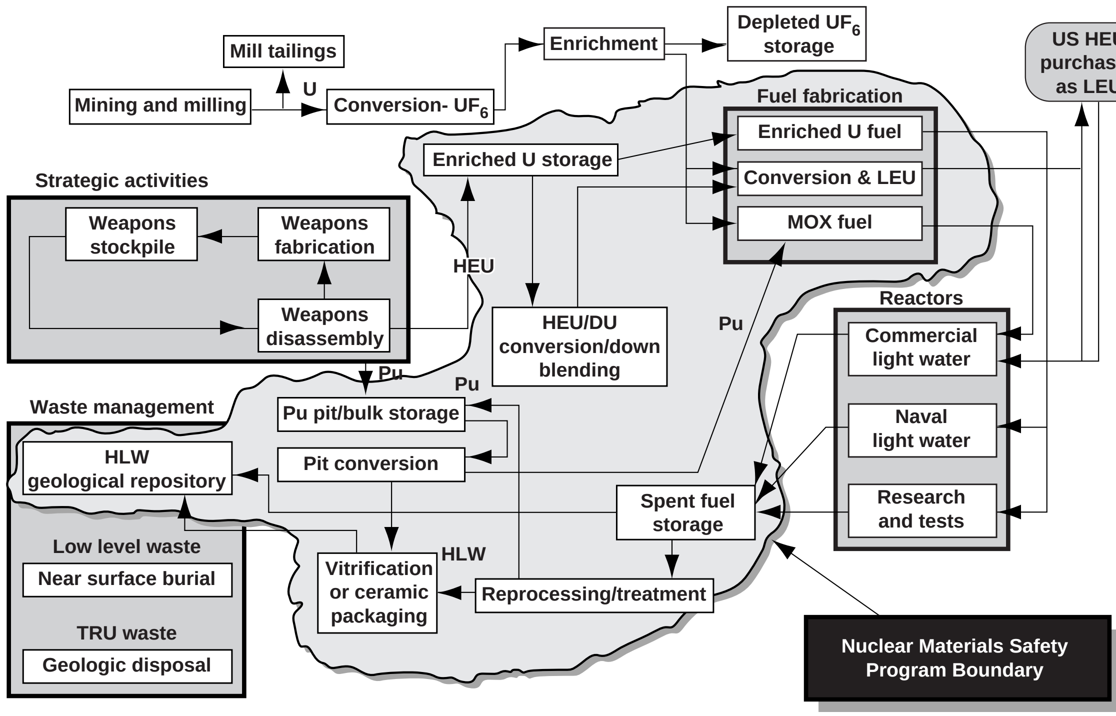


FIGURE 1. Flowsheet shows nuclear materials safety activities of interest to this ARW.

The boundary drawn in Figure 1 highlights the specific nuclear operations of interest to this ARW meeting. It also illustrates a proposed nuclear materials safety joint bilateral initiative between the U.S. and Russia that was discussed in the third day of this ARW. Some nuclear operations lie outside the boundary because they are:

1. Covered in other existing formal joint programs (International Reactor Safety Program);
2. Involve either low-level radioactive wastes or nuclear materials with very low risks of proliferation regarding the national security interests of Russia or the United States;
3. Currently sensitive or classified nuclear operations and can not be assessed jointly.

RUSSIAN VIEWPOINT ON THE SAFETY OF NUCLEAR MATERIALS

V. A. GUBANOV
Ministry of Atomic Energy (Minatom)
Ul B, Ordynka 24-26, Moscow 101000, Russia

1. Introduction

The topic of my speech is an all around review and assessment of safety problems faced by Minatom and significant safety projects to be carried out in the near future.

There are some differences between a narrow subject matter approach, which will be the subject of discussion during our Workshop, and approaches to safety analysis developed in Russia. We do not separate nuclear material management from technologies and facilities. Therefore, we develop safety analyses based on the assessment of technological safety, and nuclear materials are a component of this assessment. To go ahead, we should always analyze the statistics of equipment failures, the causes of troubles or accidents, including those with spontaneous chain reactions in the work with nuclear materials, and storage of nuclear materials.

2. Safety Records

How can we evaluate the year 1997 with respect to safety problems? This was not a typical year for Minatom. According to impartial assessments, Minatom achieved its best results in 1997. Our Ministry experienced the lowest level of traumatic injuries in the history of plant operations on the basis of sampling done for more than 40 years. We achieved the lowest level in Russia of traumatic injuries. At our plants, the lowest rate of worker death was achieved: 0.006–0.06 man per 1000 workers. As to radiation factors, the least collective dose was attained for all technologies, including fuel cycle. But, in 1997, we had a fatal accident caused by radiation. A research engineer from an institute in Sarov received an absorbed body dose of 5000 rad and died. This was a shocking situation for Minatom. Ten years after the Chernobyl accident, we were faced with the fact of death from radiation.

In 1997, we had two spontaneous chain reactions: at the All-Russian Research Institute of Experimental Physics in Sarov and at the Novosibirsk Plant of Chemical Concentrates. There were the situations when we could not expect that chain reactions might take place at such facilities.

I shall describe briefly the causes of these spontaneous chain reactions. In Sarov, the research engineer mounted an assembly of highly enriched uranium and reflectors from copper cladding. Strangely enough, he mounted an assembly for a Russian-American project and wanted to refine the parameters of this assembly. But, rewriting the parameters from one notebook to another, he make a mistake in numerical data. Thus, a wrong geometry was written in advance for the assembly. Then, working with the second hemisphere, he reached criticality in the whole system, received a dose of 5000 rad, and died in a clinic in Moscow. After this incident, much was done in the Ministry to prevent similar events by improving both the organization and techniques of work.

In Novosibirsk, a spontaneous chain reaction occurred in tanks with etching solutions. This spontaneous reaction resulted from the following situation. More than 15 years ago, the tanks were intended for the collection of drained waste waters and they had a nuclear-safe design. But later, as a result of some technical decisions, the etching solution streams changed and the procedures of nuclear safety assessment were violated. Branch industrial laboratories did not perform the nuclear safety assessment; this work was carried out by plant laboratories. The whole system was judged nuclear-safe, although the highly enriched solutions were already discharged in these tanks. Thus, a few tens of kilograms of uranium were accumulated, which caused the spontaneous chain reaction but fortunately without victims.

These two spontaneous chain reactions at Minatom facilities were most thoroughly studied, and serious conclusions were drawn that allow us to expect that in the future no uncontrolled chain reactions will occur in systems of this type. Unfortunately, again in Novosibirsk, an incident occurred on 28 May 1997, but without any radiation hazard. The matter was that in an unsuitable building in a stockroom a small amount of unregistered alkaline material was stored. This material ignited, and the fire was fought by ordinary means using fire extinguishers and water, which aggravated the situation and caused an explosion. Three men suffered, one of them died.

2.1 LESSONS LEARNED

Therefore, summing up the activities of Minatom in 1997, we had an ambivalent feeling. On one hand, the unbiased results which would suit us at any other time. However, two spontaneous chain reactions prompted us to draw the most important conclusion: the problems of nuclear safety, the observance of regulations and standards should be the focus of attention, since the origin of all incidents was the infringement of general safety rules.

After the Branch Conference on safety, we made a further important conclusion: today the main problem in our Ministry is the succession of generations for personnel engaged in safety management and the staff concerned with technology. Over a period when new generations succeed old ones, the first task is to provide incentives for young specialists and to improve professional training and retraining. A decision was made that all candidatures of specialists to be engaged in safety will be discussed and agreed with Minatom before their appointment.

Such are the results and lessons of 1997. As to general results and lessons learned, they are rather trivial. The main conclusion we arrived at is that it is necessary to carry out measures preventing accidents. Analyzing the incidents of 1997, we can see that the preventive measures prove to be more economic than the elimination of accident consequences. We must always remember the lessons of Chernobyl, and these lessons show that in Russia today the annual sum of social security payments to those who participated in the elimination of consequences from the Chernobyl accident and to the population affected by this accident is six trillion roubles (in old prices), i.e., $1 billion. So we spend annually $1 billion for social compensation. Thus, the prevention of accidents is a routine principle, which we must observe in Russia today.

3. Safety Programs

Now I shall touch upon some programs implemented in the field of safety, both individual projects and our priorities.

I would like to note that in 1997–98, two important events took place. The status of operating organizations was registered officially and legally. Minatom recognized 40 organizations that have the possibility to use nuclear energy as operators in its Branch. This is a matter of principle. In this way, a legal responsibility has been imposed on Minatom facilities. Formerly, the responsibility could be shared between the Ministry and facilities. Now a facility bears full responsibility for safety in law.

The second matter of principle adopted in 1997–98 is the introduction of licensing—not only for civil facilities but also the preparation of documents for the issue of licenses or temporary permits to facilities of the defense complex for a period up to three years.

These are two important actions carried out in 1997–98 in order to enhance the responsibility for safety of facilities.

3.1 SINGLE FEDERAL PROGRAM

The creation of a Single Federal Program for nuclear and radiation safety in Russia was also a significant project implemented in 1997. The government issued a decree that integrates all 17 programs for nuclear and radiation safety in Russia into a single program. This task was completed, and all 17 programs (including fuel and energy, radioactive waste management, safety of Russian atomic industry, development of nuclear material accountancy and control, remediation of territories, reduction of population exposure level, "Radon", the Urals Region, the Altai Region, protection of population from consequences of the Chernobyl accident, Children of Russia, measures for combined utilization of nuclear submarines, etc.) were integrated. The task was not only in the integration of programs, but also in searching for a mechanism coordinating the completion of all these works.

The concept of the single program was approved. All 17 programs were divided into five packages:

- The first package includes nuclear and radiation facilities, both civil and defense;
- The second package includes protection of population, remediation of contaminated territories, and radiation monitoring;
- The third package includes public health and environmental protection;

- The last two packages include new programs on legislative regulations and standards as well as fundamental problems of nuclear and radiation safety for which the Russian Academy of Sciences is responsible.

The general task for all packages was to reach a safety level corresponding to a socially acceptable risk. In this connection, the creation of an Interagency Council was proposed to determine annually the priorities within the framework of the Single Federal Program.

What is the further fate and prospects for this Program? At present, its fate is typical for a situation in Russia. All ministries and agencies agreed on this program except for two most important: the Ministry of Economics and Ministry of Finance. However, we hope for a favorable decision. The situation reveals the necessity of coordinated efforts and integrated approaches. We expect that the Single Federal Program for radiation and nuclear safety will be approved by the end of this year.

4. Priorities

What other key problems are being solved by Minatom at present? These are the determination of priorities within the framework of existing activities. Today the ecological problems are considered the most important.

Four tasks entering this package are now of top priority for our Ministry:

- Remediation of contaminated territories from the previous defense activity;
- Nature-protective measures;
- Management and disposal of radioactive waste;
- Decommissioning of nuclear facilities including submarines.

What are the priorities in the field of practice? These are again greater responsibility in each direction, appointment of scientific leaders and people responsible for every line of studies. This requires a funding increase, especially in the fields connected with decommissioning of nuclear submarines and facilities. Finally, there is accountability.

However, all these things also require organizational changes. It is proposed to organize an industrial concern in Minatom, which will deal with most of the above-mentioned problems. Therefore, a decree of the Russian Government was issued according to which all radiation facilities in Russia engaged in accumulation of radioactive waste are subordinated to Minatom and joined into the “Radon” concern. “Radon” will incorporate enterprises engaged in all nuclear activities connected with utilization of nuclear submarines. So, these priorities are established both at the level of the Ministry and for production facilities.

The priorities of different technologies are determined as well. Without any doubt, the problem of safe utilization of nuclear weapons is of prime importance.

The second significant question concerns the “Mayak” site where there are problems with high-level waste (HLW) vitrification, ceasing the intermediate-level waste discharge into Karachai Lake, protection of water basins, and waste removal from tanks.

The third priority is the activity of the Krasnoyarsk Mining and Chemical Combine, which is the removal of HLW from available tanks. This is a rather complicated task. There are 5.5 thousand cubic meters of liquid waste with activity of about 280 million curies in storage.

Hence, at the boundary of 1997/98, the priorities for technologies, territories, and sites were refined and more correctly formulated.

5. Other Programs

As to other important programs being now accomplished by Minatom, the following should be pointed out. Nowadays, due to forthcoming insurance problems, Minatom is involved in eight projects on risk assessment of different technologies. The future inclusion of enterprises in insurance operations has forced us to evaluate the risk of different technologies. We should be prepared to pay the tariffs, which could be imposed upon us by insurance companies before the Insurance Act of nuclear damage appears. We are carrying out a wide scope of works, one of which will be presented at this conference.

The second project of importance being executed by Minatom is the development of a system for accountancy and control of radioactive substances and radioactive wastes. A statement has already been prepared that is now being agreed upon with the subjects of the Russian Federation. We propose a three-level control over radioactive materials and wastes: the first level, Minatom will hold the database for the whole of Russia; the second level is that of agencies and subjects of the Federation; and the third level is that of enterprises.

Another important project now underway covers the description of all accidents which took place in Russia. These are the discharge of radioactive waste into the Techa River, the explosion at the "Mayak" plant in 1957, the accident in Tomsk in 1993, and the Chernobyl accident. We decided to analyze these accidents taking into account the efficiency of protective actions. I mention this project only because of the fact that in analyzing the Tomsk accident within the framework of this Workshop, we use an assessment made earlier at this site.

Thus, the direction of activities I am talking about are to a great extent connected with organizational aspects. It is high time to correct the management system of individual problems and to determine the personnel responsible for these or other directions of work.

6. Conclusions

In conclusion, I would like to say that a great scope of work is certainly connected with the problems of reasonable plutonium disposition. But I will not dwell upon this problem, as we present several reports that will review this topic in sufficient detail, and state the Minatom attitude toward these problems.

U.S. PERSPECTIVES ON NUCLEAR MATERIALS SAFETY

TERRY R. LASH
Office of Nonproliferation and National Security, U.S. DOE
1000 Independence Ave. SW, 4B-179
Washington, DC 20585 U.S.A.

1. Introduction

Cooperation between the United States and Russia on radiochemical operational safety began at a meeting in September 1993. Since then, efforts to better understand and develop internationally acceptable nuclear materials safety practices have been gaining momentum. The persistence of the organizers of this and the previous workshop is a testament to your devotion to this extremely important issue. Although our countries have been dealing with nuclear materials for decades, international interest in their safe management has grown as our defense-related interactions have become more open and transparent. This initiative is particularly timely as we prepare to engage in expanded cooperation to manage tons of excess weapons-grade nuclear material so that it will never again be used in nuclear weapons. Many of you here today were involved in the development of the weapons complex. We must now rely on you to help reduce the stockpile of nuclear material. Each of us is intimately aware of the potential dangers associated with the treatment, storage, and disposition of excess high enriched uranium (HEU) and plutonium (Pu). As professionals and experts in our fields, we also know that we possess the knowledge, expertise, and vigilance to ensure that the ultimate disposition of this material is carried out in a safe, timely, and transparent manner.

Our mutual non-proliferation goals are dependent on our success in safely managing nuclear materials. Significant bilateral and multilateral efforts are already underway to promote greater transparency in our nuclear weapon dismantlement process; to protect, control, and account for nuclear materials; and to improve the safety of older Soviet-designed nuclear power plants. These activities are integral to our mutual national security interests and they support broader global interests. After spending years amassing stockpiles of nuclear weapons in secrecy, we are undertaking ambitious efforts to destroy weapons in an environment of openness, transparency, and safety. As we begin to manage excess fissile materials from our strategic defense programs, it is imperative that safety be incorporated into all aspects of this mission. The U.S. National Academy of Sciences characterized surplus weapons plutonium as a “clear and present danger” to international peace and stability. Both the U.S. and the Russian Federation (RF) have embarked on well-defined initiatives to reduce this danger. Safety must be integral to this undertaking. We all realize that an accident in either of our countries might seriously jeopardize our disarmament objectives.

2. Background

Today I would like to outline the current U.S. perspective on nuclear materials safety. I believe the experiences gained through our past cooperative activities, including those related to nuclear reactor safety, can play an important role in the success of the nuclear materials safety initiative. At the same time, I would like to challenge each of you to work together to help better define what this initiative will become in the future.

As we all know, this effort was initiated because of concern over the need for better systems to process and manage radiochemicals safely. Over the past few years, this review of issues related to radiochemical processing has grown into a broader initiative to address nuclear materials safety throughout the nuclear complex. There are 14 functional areas involving nuclear materials. Four of these areas were addressed last year at the first Joint Advanced Research Workshop in Amarillo, Texas:

- Plutonium storage and transportation;
- MOX fabrication and transportation;
- Spent fuel storage;
- Geological disposal.

The proceedings from that meeting provide clear evidence that even after several years of joint cooperation much remains to be done.

Last year's workshop was a watershed in our collaborations on nuclear materials safety. That meeting was important because it represented the first organized, comprehensive interaction between the U.S. Department of Energy (DOE) and the Ministry of the Russian Federation for Atomic Energy (Minatom) concerning nuclear materials safety. However, the Amarillo Workshop did not deal with one very important component of the nuclear materials flowsheet (see Figure 1, Opening Remarks) that is of particular importance to me —nuclear reactors. This year, we are taking the next logical step by including a wider range of efforts in both countries dealing with reactor safety.

3. Nuclear Reactor Safety

Reactor safety is of particular interest to me because, for the past five years, one of my principal responsibilities within the U.S. Government has been overseeing the DOE cooperative activities with countries of the former Soviet Union to improve the safety of Soviet-era nuclear reactors. The United States, working in close cooperation with host countries and the international community, has supported this activity for nearly ten years. During this time, significant progress has been made to reduce the risk of an accident at operating nuclear power plants. This effort is significant because another accident of the magnitude of Chernobyl, or even a lesser event, would have overriding negative global impacts on the use of nuclear power and the use of nuclear technology. This would be over and above the serious regional, economic, social, and environmental impacts of such an event. This concern is also pertinent to all of our nuclear facilities, which are involved with the treatment, packing, storage, and transportation of nuclear materials.

Many of the same issues are applicable when addressing either the safety of operating nuclear reactors or facilities responsible for handling and processing nuclear materials. In either case, safety must be our number one and absolute priority. There is much to learn from our cooperative nuclear reactor safety activities as they relate to nuclear materials safety, the topic of this workshop. Our joint reactor safety efforts have focused on improving the plant's physical operating conditions, installing or upgrading safety-related equipment, developing improved safety procedures, establishing regional centers for training reactor personnel, and conducting plant-specific safety assessments that enable plant personnel to understand the nature of safety problems and how to solve them. In addition, this effort supports the development and maintenance of a strong, independent regulatory body responsible for licensing nuclear power plant operation. All of these cooperative activities help support a safety culture among plant owners, operators, and regulators. Together we strive to instill the highest regard for safety in the use of nuclear power. Each of these activities has a direct counterpart applicable to the safe management of nuclear materials and is grounded in the commitment to achieve a self-sustaining, uncompromising safety culture.

To support this nuclear material safety effort, the DOE has begun to develop a strategic plan that would allow us to allocate resources and work together with you and the international community to design approaches to address nuclear material safety issues. The capabilities and talents currently supporting our cooperative nuclear reactor safety activities, in combination with our growing cooperation within the weapons complex, provide the experience to point the way and provide the basis for successful implementation of a major nuclear materials safety initiative.

4. MPC&A Cooperative Activities

As I mentioned earlier, this initiative is also important because it supports non-proliferation goals by helping to assure safe, secure, long-term storage and disposition of surplus fissile materials while supporting transparent and irreversible nuclear reductions. The United States has entered into a partnership for nuclear material security with states of the former Soviet Union, in particular the Russian Federation. Nuclear experts from our countries are now cooperating to adopt effective Material, Protection, Control & Accounting (MPC&A) methods and technologies, as well as to develop comprehensive and self-sustaining MPC&A systems consistent with international standards. The primary objectives of the MPC&A activity are, first, to concentrate efforts on the most useful materials for nuclear weapons, namely, HEU and Pu, and, second, to install comprehensive, technology-based MPC&A systems that are consistent with international standards. In coordinating this program, we use proven MPC&A methods and technologies and transfer full responsibility for the long-term operation of upgraded MPC&A systems to our partners, after the completion of cooperative upgrades and provision of associated manufacturer guarantees. This transfer ensures the long-term effectiveness of improved MPC&A systems by establishing MPC&A training programs, strengthening national nuclear regulatory systems, and adopting national standards for MPC&A.

Significant progress has been made since this historic cooperation on MPC&A began in 1993. We have established trust and confidence, and we have installed fully operational MPC&A systems to improve security. With cooperative activities at over 40 sites in the Russian Federation, experience gained in implementation of this effort can have a positive impact on our future cooperation and planning for nuclear materials safety.

5. Nuclear Materials Safety Initiative

In contrast to the mature state of our nuclear reactor safety and MPC&A cooperation, bilateral and multilateral cooperation related to nuclear materials safety is just beginning to develop. This issue is receiving increased interest and support in the United States. Our interest is increasing now that efforts are underway to manage excess stocks of HEU and Pu. Based on my experience with our cooperative nuclear safety activities, we can not be successful in this critically important endeavor unless we make safety our number one priority when dealing with nuclear materials.

This scrutiny requires continued diligence and commitment to safety. In the fields of both nuclear safety and security, we are working together to meet or exceed the highest prescribed internationally accepted standards in these fields. Moreover, we must rely on an independent, strong regulatory system to ensure that these standards are maintained. The same can and should be said for all aspects of safely managing nuclear materials. Over the last few years, this group has worked to define and describe the various components of the flowsheet for managing nuclear materials. We are now at a point where we must work to develop a strategic plan for future cooperation.

Considering the size and complexity of this undertaking, we must focus our limited resources on establishing a strong safety culture while specifically addressing those components of the flowsheet that are the most risk sensitive. When considering prospective plans of action, we must remember that our mission is to reduce the risks associated with all aspects of managing excess weapons HEU and plutonium. We propose that we take an approach similar to that established by the nuclear reactor safety and MPC&A cooperative programs. That is, address near-term safety deficiencies first while, in parallel, establish a means for supporting a sustainable safety culture.

We currently envision an initiative with two primary components to address these objectives: the first I will call facility technical projects; the second, academic exchanges and training. As discussed at previous workshops, cooperative activities might be facilitated through a formal Memorandum of Cooperation to establish a foundation for future lab-to-lab and technical academic exchanges on nuclear materials safety. Furthermore, we should structure future activities such that they do not duplicate current U.S.-Russian activities, but rather complement ongoing efforts in MPC&A, reactor safety, waste management, and materials management. Our best approach would be to leverage past experiences by extending existing interactions among the DOE, the Minatom, and national laboratories, research institutes, and industrial sites as well as by establishing relationships between prominent universities in both countries.

I would now like to take a moment to address these two areas of cooperation in greater detail. Our top priority should be to minimize risk in the near-term.

5.1 FACILITY TECHNICAL PROJECTS

Through joint U.S.-Russian collaboration, the facility component of this initiative will establish improved safety methodologies with the goal of measurably improving safety at key facilities within a specified time frame. The strategy would entail sharing modern practices, methods, and technology to implement quality assurance programs and nuclear safety assessments that can be used immediately for problem solving and to address critical needs at priority facilities. We must begin by first establishing a baseline of the operational status and safety envelope for these facilities.

5.2 ACADEMIC EXCHANGES AND TRAINING

In addition, credentialed training programs for managers and personnel will be established to ensure that managerial and structural improvements are in place and sustainable long after this joint activity has come to a conclusion. Above all, our aim is to have safety-significant activities at key nuclear facilities systematically reviewed by highly qualified experts in order to contribute to increased levels of safety on a broad and continuing basis.

5.3 ESTABLISHMENT OF A SAFETY CULTURE

The ultimate success of the Nuclear Materials Safety initiative will depend on the creation of a sustainable and comprehensive safety culture within all segments of the nuclear sector. Whether in the United States or Russia, all responsible parties must have sound technical bases for decision making and be provided the authority to make decisions based on safety considerations. How do we go about ensuring that such a system is in place? I have already discussed one activity that will positively impact the long-term operation of the nuclear sector—training. We must ensure that the operators and managers of our nuclear facilities are adequately trained and that this training continues throughout the duration of their careers.

Another important contributor will be collaboration on new academic curriculum and degree programs that focus on safety. This will be accomplished through interactions between university colleagues in Russia and the United States. By incorporating safety at a fundamental level into the educational process, the graduating scientists and engineers who will become the future staff members, group leaders, managers and directors will have a basic understanding of and appreciation for safety. Some of this collaboration has already begun. I would like to commend the Russian nuclear engineering faculty for their innovative thinking and creative proposals for programs which could become the international standard for new academic endeavors in nuclear materials safety. By identifying these new approaches and introducing important new elements into higher education curricula, participating universities could become the recognized international centers of education in these important fields. From the perspective of the Department of Energy, these are especially noteworthy endeavors that will contribute significantly to nuclear materials safety and are a long-term investment needed to establish a permanent safety culture.

6. Conclusion

These are just some of the ideas we have been working on with our Russian counterparts. I am sure that as this Workshop proceeds, we will have the opportunity to better define the key aspects of the elements I have outlined this morning. We are embarking on a very ambitious and difficult journey, but one that has many rewards at the end of the day. I challenge each of you as we move forward to think creatively and to work cooperatively. For my part, I will continue to further this group's agenda, which is to support a safer nuclear materials infrastructure that will enable our countries to manage our growing stockpiles of excess nuclear material in a safe manner. Thank you again for your attention and I look forward to working more closely with you as we initiate what I believe can be an enduring and successful relationship in the safe management of nuclear materials.

THE RF REGULATORS' VIEW OF NUCLEAR MATERIALS SAFETY

A. M. DMITRIYEV
A. I. KISLOV
Russian Federation Gosatomnadzor (GAN)
State Inspectorate for Nuclear and Radiation Safety
34 Taganskaya Street, Moscow 109147, Russian Federation

1. Summary

Pursuant to the Federal Law on the Use of Atomic Energy, and on the basis of Executive Order No. 26 of the President of the Russian Federation, dated January 21, 1997, and titled "Federal Executive Agencies Authorized to Perform State Regulation of Safety in the Use of Atomic Energy," the Russian Federation (RF) Gosatomnadzor, together with the Russian Federation Ministry of Health, the Russian Federation Gosgortekhnadzor (State Committee for Oversight of the Safe Conduct of Mining Operations), and the Russian Federation Ministry of Internal Affairs, performs federal regulation of nuclear, radiation, engineering, and fire safety. Apart from mandated regulation of safety and oversight of safety assurance, the principal function of the RF Gosatomnadzor (GAN) is to license activity in the field of the use of atomic energy.

If we restrict the field of nuclear-materials safety from the standpoint of the most dangerous types of activity, we should consider the following:

- Storage of highly enriched uranium in metallic and other forms;
- Storage of plutonium in metallic and oxide forms;
- Spent nuclear fuel storage;
- Conversion of highly enriched uranium to hexafluoride and isotope dilution of it;
- Purification of weapons-grade plutonium;
- Plutonium conversion to oxide;
- Fabrication of nuclear fuel on the basis of uranium with different degrees of enrichment and of plutonium (oxide);
- Radiochemical processing of irradiated nuclear fuel.

1.1 REGULATORY DOCUMENTS

At the present time, the nuclear and radiation safety of the aforementioned activity is regulated by a series of documents, most of which were placed in effect in the 1970s and 1980s in what was then the Soviet Union. Most of the requirements and restrictions in those documents require revisions and amendments in connection with revisions of sanitation standards and regulations and changes in processes and organizational and legal arrangements. Furthermore, Russian law requires the introduction of many regulatory documents that do not yet exist.

On the basis of legislative requirements, the Russian Federation Gosatomnadzor and Russian Federation Ministry of Health have undertaken to drawn up federal safety standards and regulations. Among the most general documents, we should note the following:

- *Revised Radiation Safety Standards* (*NRB-96*), which meet international requirements and must be introduced in full in the year 2000;
- *Revised Basic Sanitation Regulations*;
- *Revised Regulations for Safe Transport of Radioactive Materials*;
- *General Regulations for Safety Assurance at Nuclear-Fuel-Cycle Enterprises*;
- *Nuclear Safety Regulations for Nuclear-Fuel-Cycle Enterprises*;
- *Processing of Spent Nuclear Fuel: Safety Requirements*.

Furthermore, the drafting of other regulatory documents pertaining to this area of safety is also planned. The Russian Federation Ministry of Atomic Energy has drawn up and continues to draw up industry-level regulatory documents on nuclear and radiation safety.

1.2 REGULATORY CONTROL THROUGH LICENSING REQUIREMENTS

In view of the incompleteness of the formation of the regulatory framework for safety, the Russian Federation Gosatomnadzor has been forced to incorporate appropriate regulatory requirements in the effective conditions of licenses issued to operating organizations. Here, use is being made of not only domestic experience and practice, but also the recommendations of international organizations (e.g., IAEA, WHO), provisions of international treaties that Russia has joined, and information conveyed within the framework of bilateral relations.

Of late, when several large projects involving an overall reduction of the level of nuclear confrontation are being implemented and planned for implementation, the regulatory framework often has lagged behind the introduction of technologies. Such instances are, of course, permissible only as exceptions.

An event that occurred during the design and installation of a mixing unit for flows of highly enriched weapons-grade uranium and low-enriched uranium in hexafluoride form at the Ural'sk Electrochemical Combine in Novosibirsk is instructive. At that facility, installation of a process to monitor the mixing of highly enriched and low-enriched uranium is being shaken down. A similar installation also is supposed to be in use later at the Electrochemical Plant in Zelenogorsk and at the Siberian Chemical Combine in Seversk.

During the design of the mixing unit, due allowance was not made for Russian *Radiation Safety Standards* (*NRB-96*) or Russian *Basic Sanitation Regulations* (*OSP-72/87*) concerning the permissible dose rates on the surfaces of piping carrying highly enriched uranium hexafluoride during initiation of reactions by the Cf-252 source and during determination of the flux density of hexafluoride by using Co-57 sources. The neutron source initiates such high radiation in the piping that the design shielding was unable to reduce the radiation intensity to permissible levels. A significant modification of the installation's design was required, with a change in the routing of pipes and a major upgrade of radiation shielding. The need for these changes caused a delay of approximately a year and a half in the startup of the permanent monitoring unit.

1.3 REVIEW REQUIREMENT FOR FOREIGN-MADE EQUIPMENT

Pursuant to the RF Gosatomnadzor Directive Document RD 03-36-97, *Conditions of Delivery of Imported Equipment, Products, and Assembly Components for Nuclear Installations, Radiation Sources, and Storage Facilities in the Russian Federation*, all foreign-made equipment used in production processes on nuclear installations operating within Russia shall undergo expert review for compliance with the requirements of Russian federal standards and regulations in the area of the use of atomic energy. This document describes the entire procedural process that defines interrelationships among a legal entity interested in using foreign equipment, expert organizations, and the Russian Federation Gosatomnadzor, which is responsible for the formal approval of the use of foreign equipment within the Russian Federation.

Since there are currently many proposals, particularly for fabrication of MOX fuel using foreign equipment and processes, the requirement of formal approval by the Russian Federation Gosatomnadzor should be borne in mind.

The process of vitrification of radioactive waste is now an important element in the radiochemical process of nuclear fuel in Russia. The failure of the vitrification furnace at the Mayak Combine has forced the Russian Federation Gosatomnadzor to reduce the rate of radiochemical fuel processing in order to limit the accumulation of liquid radioactive waste in storage tanks.

1.4 DECOMMISSIONING

The Russian Federation Law on the Use of Atomic Energy requires that during the design and before the construction of a nuclear installation, a decommissioning plan for this installation be put in place.

This situation should force the pursuit of only those installation designs that do not require excessive expenditures for decommissioning. This important provision will be strictly enforced by the Russian Federation Gosatomnadzor when permits are issued for the design and construction of new installations. This provision is especially important for installations that will process plutonium-containing nuclear materials.

1.5 NUCLEAR MATERIALS SAFETY ASSURANCE

We would like to note one other important feature of work with nuclear materials, a feature that is characteristic of Russia.

At the present time there are several major science centers in Moscow that work with nuclear materials, including highly enriched uranium and plutonium. This situation creates a big potential hazard, and the Russian Federation Gosatomnadzor will take steps to limit sharply the amount of nuclear materials with which work is done near large cities, and to step up measures to reduce the likelihood of possible incidents.

The critical-mass incidents that occurred in 1997 at the Novosibirsk Nuclear Fuel Plant and at the Scientific Research Institute of Experimental Physics in Sarov underscore the urgency and importance to Russia of efforts to raise the level of nuclear-materials safety assurance. The incident in Novosibirsk added new circumstances to the rich cache of Russian experience with critical-mass reactions: for the first time, a nuclear reaction resulted from neutron interaction between two adjacent pieces of apparatus containing high enriched uranium. These incidents revealed, among other things, the need to use so-called “configuration management,” which includes management of the system used to ensure the accuracy of design documents and monitoring of the actual state of facilities.

MEDICAL PROVISION OF RADIATION SAFETY WHILE HANDLING RADIOACTIVE SUBSTANCES

SVETLANA G. MONASTYRSKAIA
IBP-MPH
Zhivopisnaya, 46
Moscow, 123182, Russia

1. Introduction

The disarmament policy conducted at present by states possessing nuclear weapons after the cold war has both positive aspects that are aimed at promoting peace on the Earth, as well as a number of security problems that are common to mankind on the whole. Among these problems, the most complicated are the provision of highly reliable radiation safety for the personal and the population as well as environmental protection when dismantling nuclear weapons and disposal of the resulting fissile and radioactive wastes.

The vital nature of these problems is quite evident, as the world community has not had any experience in the practical solution of problems relating to the safety of broad-scale operations on dismantling nuclear weapons and their hypothetical medico-biological impact.

The Russian Federation (RF) Ministry of Health has made supplementary provisions to ensure the radiation safety of these operations. In a special RF Government Decree (1994), a comprehensive research and practical program titled *Radiation Safety and Medico-Hygienic Provision of Operations on Dismantling Nuclear Weapons Including Plutonium Disposal (1994-1998)* was approved.

The Federal Administration on Medico-Biological Problems and Emergencies (RF Ministry of Health) has been participating in elaborating and putting into practice some provisions of state-level regulation and control of radiation safety at plants and institutions relating to the RF Ministry of Atomic Industry.

At the Institute of Biophysics State Research Centre, elaboration of radiation-hygienic and medico-biological provisions to ensure a highly reliable national radiation safety system is underway. The system includes the safety aspects regarding personnel, population, and the protection of the environment at all stages of operations of dismantling of nuclear weapons and disposal of fissile and radioactive wastes.

The Program envisages theoretical and applied studies, research, and subsequently putting into practice the results obtained within the framework of medical provision of safety.

The present report stresses the following safety aspects while handling radioactive substances produced as a result of dismantling nuclear weapons:

- Radiation safety of personnel, the population, and the environment;
- Radiation safety in technology;
- Reliability and training of personnel;
- Disaster preparedness.

2. Radiation Safety of the Personnel, the Population, and the Environment

In Russia, radiation safety of the personnel, the population, and the environment is stipulated by appropriate legislation that specifies the basic radiation dose threshold values for personnel and population exposure as well as the conditions under which radiation energy is to be utilized.

The legislative and normative provision of radiation safety is promoted by RF legislation, RF Presidential Decrees and orders, and RF government statements and resolutions as well as by radiation safety standards. The latter documents include the permissible radiation dose values of radiation exposure for the personnel and the population, sanitary regulations, instructions on prevention of accidents as well as technological regulations.

Based on the Federal Law, *On Radiation Safety* (1996) in *Radiation Safety Standard* (1996) (RSS-96), three groups of standardizing parameters have been identified:

- Major dose threshold values;
- Permissible values of monofactor influence, e.g., threshold annual intake; permissible values of average annual volume activity (PVA), and permissible specific activity (PSA);
- Reference values (radiation doses and levels).

Instructions on handling radioactive substances have been stated in Federal *Basic Sanitary Rules* and branch sanitary–hygienic standards (e.g., sanitary norms and regulations, sanitary rules, sanitary rules and norms).

Radiation safety of handling radioactive fissile substances is ensured by observing the basic principles specified in radiation safety standards, which cover standardization, substantiation, and optimization.

These principles are observed under production conditions during all kinds of operations at all plants and installations for reprocessing radioactive substances (Medbioextrem Federal Administration services for sanitary–hygienic supervision radiation monitoring).

3. Radiation Safety in Technology

Medical assessment and provision of safety in technology is put into practice, first and foremost, by specialists in the independent appraisal of design materials and in building and restoration of different plants and installations (for instance, MOX-fuel production, fissile material storage). In this case, the ultimate decision on design feasibility is made by the RF State Sanitary Inspection Service.

Medical specialists' opinions are taken into consideration when choosing sites for new production buildings, outlining the area of sanitary protection and inspection, solving the problems regarding the minimization of the radioactive wastes released into the environment.

These solutions are based on sanitary instructions and regulations taking into account up-to-date technologies, safety requirements and methodologies.

In these documents, the basic medico-biological requirements for production technology, equipment, operating conditions for personnel, time limits preset for operations, ventilation, and accident prevention have been stated.

In practice, in the course of production of fissile materials, permanent sanitary assessments of technological safety, identification of operations to be carried out in working areas according to the degree of hazard as well as sanitary supervision of the working areas and the production environment, ventilation and gas purification systems, working places of the personnel, etc., are obligatory.

4. The Personnel Reliability and Training

Personnel reliability is one of the major constituents of safety in production operations. It depends both on the personnel training and medico-psychological features of a person, both of which affect the capability to carry out a certain kind of activity.

These two factors are taken into consideration in the process of choosing personnel and making the appropriate medical conclusions on their fitness.

Bearing in mind the peculiarities of operations when handling radioactive substances, special-purpose and psycho-physiological methods for selection and training of the personnel handling radioactive substances have been elaborated and are being put into practice.

Personnel reliability is promoted by the appropriate regimen of labor and rest, observing ergonomic requirements of the equipment as well as the provision of a highly operable man-machine system. Normal psycho-physiological atmosphere is very important while carrying out highly dangerous technological and repair operations.

5. Disaster Preparedness

Radiation standardization, forecasting, and control of the radiation situation in the vicinity of the accident site, and interdepartmental interaction in the post-accident period to mitigate the consequences of the radiation disaster (provided any items of nuclear weapons have been destroyed) are the most urgent problems to be practically solved in the future.

As a result of medical investigations, certain radiation factors affecting the personnel and the population as a result of an accident involving damage from nuclear weapons were determined and compared with the impact of factors related to weapons of mass destruction.

Recommendations on initial medical assistance in an emergency have been elaborated.

Consequently medical provision of operational safety for nuclear plants and installations in Russia is ensured by the common system of supervision and control of the personnel and equipment in the nuclear industry as well as by special measures elaborated by the Institute of Biophysics State Research Centre within the framework of State programs.

SUMMARY OF NUCLEAR MATERIALS SAFETY ARW IN AMARILLO AND ITS RELATIONSHIP TO THIS WORKSHOP

K. L. PEDDICORD
Department of Nuclear Engineering
Texas A&M University
College Station, TX 77843-3133, U.S.A.

LEONARD N. LAZAREV
V. G. Khlopin Radium Institute
28, 2-nd Murinsky Ave., St. Petersburg, 194021,Russia

LESLIE J. JARDINE
Lawrence Livermore National Laboratory
Livermore, CA 94551, U.S.A.

1. Introduction

A NATO Advanced Research Workshop (ARW) on nuclear materials safety management was held in Amarillo, Texas, in March, 1997. The Workshop focused on safety practices in various facets of the nuclear fuel cycle. Particular emphasis was given to the handling, treatment, storage and transportation of plutonium resulting from the disposition of fissile weapons materials. From the Workshop emerged a number of ideas for collaboration between the U.S. Department of Energy (DOE) and the Ministry of the Russian Federation for Atomic Energy (Minatom) for enhancing safe management of materials in the nuclear fuel cycle.

DOE and Minatom have been actively exploring and implementing initiatives for collaboration in a number of fields of mutual interest. The results of these interactions have led to significant advances in a number of technical areas and contributed to a deeper understanding of the practices and methods employed by the two agencies. Vigorous exchanges have taken place on the topics of nuclear waste management, nuclear reactor safety, and the disposition of excess fissile materials from disassembled nuclear warheads. These efforts have extended to other segments of the nuclear fuel cycle and now encompass the broad area of nuclear materials safety. This lays the ground work to address current issues of mutual interest and to embark on future even more substantive programs which can have a major impact.

2. Background

On April 6, 1993, an accident occurred in a chemical reprocessing plant of the Siberian Chemical Combine at Tomsk-7 (Seversk). The accident has been attributed to a reaction of nitric acid and tributyl phosphate solvent. More importantly, this accident opened up a new opportunity in which Minatom and DOE could exchange information and consider whether similar conditions might be present at DOE facilities. A technical team from DOE conducted a site review at Tomsk-7 in June, 1993. This was followed by the first joint U.S.-Russia meeting on radiochemical processing safety which was held in Hanford, Washington, in September, 1993 [1]. In November, 1994, the second joint workshop took place in St. Petersburg followed by a visit to Krasnoyarsk-26 [2]. The topic of the St. Petersburg meeting was radiochemical operational safety. In August, 1995, the third workshop occurred in Los Alamos, New Mexico, with the subject being non-reactor nuclear safety [3]. The next year in August, 1996, a program review and planning meeting was organized as part of Spectrum ‘96 in Seattle, Washington, to consider future technical exchanges. Most recently, the fourth U.S.-Russia workshop, held as a NATO Advanced Research Workshop, took place in March, 1997, in Amarillo, Texas [4].

In addition to the meetings described above, a number of joint projects conducted on a lab-to-lab basis were organized and carried out [5]. These efforts were based originally on the analysis and consequences of the Tomsk tank accident, but have expanded to consider other areas of safety. For example, specific lab-to-lab projects have dealt with (1) safety of anion-exchange processes in nitric acid media, (2) refinement and validation of consequence assessment methodologies for potential accidental release of radionuclides into the environment, (3) ensuring safety for underground repositories for long-lived radioactive materials, and (4) development of ceramics and glasses for the immobilization of excess weapons plutonium.

3. The Amarillo ARW

In March, 1997, the fourth workshop was held in Amarillo, Texas, with the broadened topic of nuclear materials safety management. This subject was considered in the context of the entire nuclear fuel cycle. The focus was on the non-reactor segments with emphasis on the disposition of weapons plutonium from disassembled nuclear warheads. It was recognized that an accident in either country could considerably delay and possibly disrupt the efforts to disposition fissile weapons materials in both countries. Materials safety is critical to achieving this mission. In addition, improvements in materials safety can contribute to better radiation protection of both workers and the general public.

The Amarillo Workshop had four objectives:

1. Continue to exchange technical information on nuclear materials safety management;
2. Continue the development of a joint Russia-U.S. safety initiative;
3. Continue to expand contacts between U.S. and Russian specialists to promote significant technical exchanges related to common safety issues;
4. Evaluate the participation of other countries with nuclear technology expertise.

The Amarillo Workshop also saw the introduction of the nuclear materials safety boundary [4]. This construct sought to characterize the entire nuclear fuel cycle and define the components in which nuclear materials safety plays a role. The program boundary also identified the specific areas in which exchanges are taking place between DOE and Minatom in order to avoid duplication of existing efforts.

At the Amarillo Workshop, 124 participants from six countries took part. The NATO grant enabled 24 specialists to travel from Russia. At the meeting, a total of 50 papers were presented. The workshop represented the first organized interaction between DOE and Minatom on the subject of materials safety. Technical presentations were made in the areas of:

- Nuclear materials safety management systems,
- Plutonium storage and transportation safety issues,
- MOX fabrication and transportation, and
- Other nuclear materials safety topics.

In addition to the paper presentations, a series of “break-out” groups were organized in which specialists from various countries could interact and help identify and define further interactions on the topic of nuclear materials safety. The areas addressed by the break-out groups were:

- Nuclear materials storage, transportation and handling safety;
- Mixed oxide fuel production, transportation and handling safety management;
- Spent fuel storage, transportation and handling safety;
- Geologic disposal, waste and environmental safety issues.

A number of recommendations were made by the breakout groups involving a range of topics and activities. The participants in the groups recommended that further visits and exchanges in nuclear materials safety management should be continued. Joint projects should be identified and conducted in several areas including:

- Training involving codes, simulators and practices dealing with nuclear materials safety;
- Experiments to obtain data for source terms, failure modes, and to verify monitoring methods;
- Assessments including safety analysis reports, safety methods, consequence modeling and studies of potential initiating events.

In addition to interaction between technical specialists, it was further noted the need for training and educational programs to achieve these objectives. It was also recommended that follow-up efforts should focus on safety management and safety technology development for weapons-related nuclear materials (exclusive of nuclear reactors, weapons dismantlement, waste disposal and other areas in which Memoranda of Cooperation exist between DOE and Minatom). However, it was further recognized that technical exchanges in the area of nuclear materials safety might benefit from a Memorandum of Cooperation between DOE and Minatom in this area and that efforts should be directed towards this accomplishment.

4. Relationship to the Current Workshop

The Amarillo Workshop contributed to the ongoing interaction between DOE and Minatom in the field of nuclear materials safety. It was recognized that not all of the topics could be covered in that meeting. In addition, because of similar formats and related objectives, there is a tie between the nuclear materials safety activity and the International Nuclear Reactor Safety Program that is already ongoing between DOE and Minatom. Also at Amarillo, the breakout groups recognized the need for training and educational initiatives to sustain nuclear materials safety. As part of this workshop, the university component will be addressed in greater detail. Finally, to establish a program which will be sustainable, it continues to be necessary to define and give substance to the elements of such a program. This workshop will continue with that effort.

Based on this background, the goals of the St. Petersburg NATO Advanced Research Workshop on Nuclear Materials Safety II are the following:

- Address topics on nuclear materials safety which were not covered in Amarillo including safety issues:
  —For vitrification operations and extrapolations for Pu vitrification,
  —Of plutonium processing required for immobilization or MOX fuels
  —Associated with safe shutdown and operation of Pu processing plants.
- Present the framework of the International Nuclear Reactor Safety Program;
- Address the university component of the nuclear materials safety activity;
- Further establish the basis of a joint program between DOE and Minatom on nuclear materials safety and the projects which would be part of this program.

Continued interaction among the specialists of Minatom and DOE along with other contributors will serve to establish a robust program to meet the needs in the area of nuclear materials safety, strengthen the concept of a safety culture, promote nonproliferation, openness and transparency between Russia and the United States, and facilitate the disposition of excess weapons materials in a safe and expeditious way.

Acknowledgments

This paper was prepared with the support of the U.S. DOE, Cooperative Agreement No. DE-FC04-95AL85832. However, any opinions, findings, conclusions, or recommendations expressed herein are those of the author(s) and do not necessarily reflect the views of DOE. This work was conducted through the Amarillo National Resource Center for Plutonium.

References

1. "Joint United States/Russian Federation Meeting, Radiochemistry Processing Safety," Sponsored by the U.S. Department of Energy and the Ministry of the Russian Federation for Atomic Energy, Hanford Washington, USA, September 24-25, 1993.
2. "Second Joint United States-Russian Federation Workshop on Radiochemical Operational Safety," Sponsored by the U.S. Department of Energy and the Ministry of the Russian Federation for Atomic Energy, St. Petersburg and Krasnoyarsk-26, Russia, November 10-19, 1994.

3. "Third U.S.-Russian Workshop on Non-Reactor Nuclear Safety," Sponsored by the U.S. Department of Energy and the Ministry of the Russian Federation for Atomic Energy, Los Alamos, New Mexico, USA, August 14-19, 1995.
4. K. L. Peddicord, Leonard N. Lazarev and Leslie J. Jardine, *Nuclear Materials Safety Management*, NATO ASI Series, Kluwer Publishers, Dordrecht, Holland (1998).
5. Fred Witmer, et al., "Russia-U.S. Joint Program on the Safe Management of Nuclear Materials," *Spectrum 98*, Denver, Colorado, USA.

COOPERATIVE EFFORTS TO IMPROVE THE SAFETY OF SOVIET-DESIGNED NUCLEAR POWER PLANTS

LAURIN R. DODD
Pacific Northwest National Laboratory
Richland, Washington 99352, U.S.A.

1. Introduction

The invitation to participants in the NATO Nuclear Materials Safety Advanced Research Workshop stated that the goal of a nuclear materials safety program is to enhance and improve nuclear materials safety and the safety cultures embedded in day-to-day operations associated with the storage and disposition of excess fissile materials from the dismantled nuclear weapons. The intent of this paper is to provide an overview of our cooperative reactor safety studies. It will be evident how that experience can be applied to enhance future cooperation related to use of mixed-oxide (MOX) fuel.

Cooperative efforts between the United States, Russia, Ukraine, and other host countries were initiated in 1992 to improve the safety of Soviet-designed nuclear power plants. The joint efforts originated from U.S. commitments made at the G-7 conference in 1992. Amid heightened international concern about the safety of Soviet-designed reactors, world leaders agreed to collaborate with host countries to reduce risks of operating these nuclear power plants.

The U.S. Department of Energy (DOE) developed and implemented a comprehensive, cooperative effort to improve the safety of Soviet-designed nuclear power plants that includes urgently needed safety work at 20 nuclear power plants with 65 operating reactors in these host countries. The work is conducted in cooperation with similar programs initiated by Western European countries, Canada, and Japan, as well as international organizations such as the Nuclear Energy Agency of the Organization of Economic Cooperation and Development, the International Atomic Energy Agency, and the European Bank for Reconstruction and Development.

The DOE manages U.S. involvement in the cooperative work. The Pacific Northwest National Laboratory provides technical leadership with assistance from other U.S. national laboratories, U.S. businesses, and host-country nuclear power plants and scientific organizations. U.S. specialists work in accordance with the guidance and policies of the U.S. Department of State and the U.S. Agency for International Development and in close collaboration with the U.S. Nuclear Regulatory Commission. U.S. specialists have established working agreements with organizations in the host countries including key government agencies, scientific institutions, engineering and design organizations, and the nuclear power plant operators. More than 50 U.S. commercial firms are providing equipment, training, expert services, and technology transfer.

Russian-U.S. bilateral nuclear disarmament initiatives on the reduction of nuclear missiles have resulted in a large inventory of surplus fissile materials recovered from the warheads of these missiles. Options have been identified and evaluated and plans developed by both the United States and Russia on how to dispose of this surplus fissile material. An option that is being pursued by both countries on the disposition of surplus weapons-grade plutonium is to burn it as mixed-oxide fuel in existing civilian nuclear power plants. In the April 1996 address to the Russian Security Council, President Boris Yeltsin included the following statement: "Nuclear materials are becoming available as a result of the dismantling of nuclear weapons. These should be used in peaceful activities for the benefit of all mankind." Russian policy for plutonium use focuses on its economic value, to be used as fuel for nuclear power plants. Accordingly, the Russian plan is to burn most of its surplus weapons-grade plutonium in civilian nuclear power plants.

An overview of the cooperative program to improve the safety of Soviet-designed nuclear power plants is provided in Section 2. The applicability of that work to safety improvements that would be desirable for VVER-1000s under a MOX fuels program is outlined in Section 3. A summary and some observations on plutonium disposition are provided in Section 4.

2. Improving the Safety of Soviet-Designed Nuclear Power Plants

The safety problems in Soviet-designed reactors have several underlying causes. The most significant cause is the historical isolation of the plant's nuclear operators, designers, and regulators. In the Soviet Union's closed society, these personnel could not exchange information or technology with the international nuclear community and had limited interaction even with their counterparts in Soviet countries. Another significant cause was a secondary emphasis on safety. Goals for low-cost production of electricity often outweighed safety goals. Regulatory bodies responsible for enforcing safety standards were ineffective, especially when their safety directives ran counter to production goals. The new governments of Armenia, Ukraine, Russia, and Central and Eastern European countries have invested millions of dollars toward improving the safety of their nuclear power plants. However, their economic and technical resources are insufficient to address many of the safety issues.

The results of joint projects are reducing the risk of an accident at operating nuclear power plants in these host countries. These cooperative efforts help protect Europe's public, economic, and environmental health and support a stable business climate for international investments. These efforts also help the host countries avoid a crisis that could destabilize their democratic governments.

2.1 OVERVIEW OF THE PROGRAM

Nine host countries are in the program—Armenia, Ukraine, Russia, Bulgaria, the Czech Republic, Hungary, Lithuania, Slovakia, and Kazakhstan. Joint projects in the host countries are correcting major deficiencies and establishing nuclear safety infrastructures that will be self-sustaining. Urgently needed safety work is being performed at 20 nuclear power plants with 65 operating reactors. The location of these power plants is shown in Figure 1. The efforts do not extend the operating lives of these reactors, but they do reduce risks until the reactors can be shut down or brought into compliance with international standards and practices.

2.2 REACTOR TYPES

Two basic designs, RBMK and VVER, were used to build 59 of the 65 reactors included in the cooperative safety efforts.

An RBMK is a graphite-moderated, water-cooled system. The Chernobyl reactor that exploded in 1986 was an RBMK. Fourteen RBMKs are in operation. Several problems are inherent in RBMKs. If several pressure tubes rupture simultaneously, the force will raise the reactor lid, releasing radioactive fission products. An RBMK is susceptible to power instabilities. Increased boiling can boost power levels, creating an uncontrolled reaction. The emergency core-cooling, fire protection, and electronic control-and-protection systems in the older RBMKs do not meet international standards.

FIGURE 1. Nuclear power plants participating in the cooperative effort to improve nuclear safety.

The VVER reactors are pressurized light-water-moderated and -cooled systems. Forty-five VVERs are in operation. Three principal VVER models exist. The earliest model, the VVER-440/230, has a limited containment system and virtually no emergency core-cooling system. Its backup safety, fire-protection, and electronic control-and-protection systems are inadequate. The VVER-440/213 is an enhanced version of the VVER 440/230 model. It has an emergency cooling system and a "bubble condenser tower" that provide a measure of containment. Fire protection and electronic control-and-protection systems are inadequate. The VVER-1000 is the largest and newest model that meets modern safety standards. It has an emergency core-cooling system and a containment building. However, its fire protection and electronic control-and-protection systems have shortcomings. Nineteen VVER-1000s are in operation.

In 1996, two Russian nuclear power plants, Bilibino and Beloyarsk, with neither RBMK nor VVER reactors, requested U.S. collaboration in developing and implementing safety improvements. The Bilibino plant in eastern Russia has four LWGR12 light-water-cooled, graphite-moderated reactors. The Beloyarsk plant operates a BN-600 liquid-sodium-cooled fast reactor. In 1997, the Asian nation of Kazakhstan requested U.S. collaboration in conducting joint safety projects. In November 1997, the U.S. agreed to collaborate, and work was initiated to improve the safety of Kazakhstan's Soviet-designed BN-350 sodium-cooled, fast breeder reactor as well as other related projects.

2.3 OBJECTIVES AND ACTIVITIES

The primary objectives of the Soviet-designed reactor safety program are to

- Improve the plants' physical operating conditions
- Install safety equipment;
- Develop improved safety procedures;
- Establish regional centers for training reactor personnel;
- Conduct in-depth safety assessments;
- Develop regulatory and institutional frameworks for plant design, construction, and operation that comply with international standards and practices.

The activities to reduce risk and improve safety are performed under the following projects organized into five areas:

- Management and operational safety projects;
- Engineering and technology projects;
- Plant safety assessment projects;
- Fuel cycle safety projects;
- Nuclear safety regulatory and institutional framework projects.

Management and operational safety projects increase the safety of day-to-day operations by training operators, establishing safe operating procedures, and transferring maintenance technologies and training. An example of a key accomplishment in this area is a full-scope simulator at the Balakovo nuclear power plant used to train control room operators (shown in Fig. 2).

FIGURE 2. Full-Scope Simulator, Balakovo Nuclear Power Plant.

Engineering and technology projects reduce operating risks by upgrading the physical safety systems of nuclear power plants. An example of a key accomplishment in this area is a system that displays key safety parameters of the plant in real-time on a console for the plant operators (shown in Fig. 3).

Plant safety assessment projects improve the abilities of designers, operators, and regulators to evaluate the safety of their plants through the use of internationally accepted

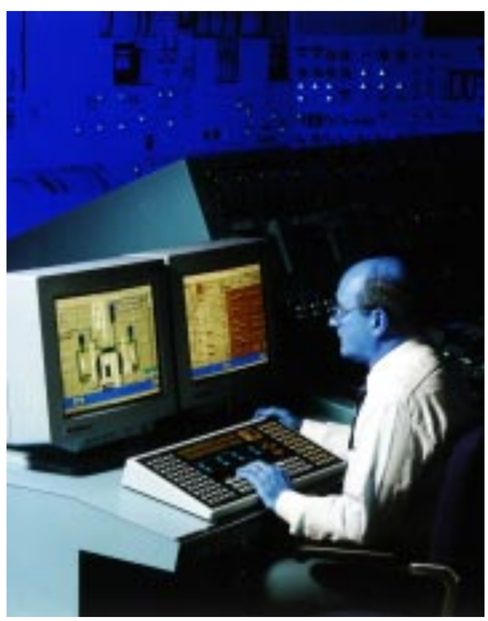


FIGURE 3. Safety Parameter Display System—Console.

methodologies and computer analysis codes. An example of a key accomplishment in this area is training technical personnel to use the COBRA-SFS computer code to perform thermal-hydraulic analysis (shown in Fig. 4).

Fuel-cycle safety projects improve the handling and storing of reactor fuel and the operation of nuclear fuel-cycle facilities. An example of a key accomplishment in this area is a dry-storage cask for spent fuel (shown in Fig. 5). Several types of modular dry-storage systems have been licensed by the U.S. Nuclear Regulatory Commission for spent fuel from commercial light-water reactors in the United States. This technology was transferred to Ukraine for storing VVER-1000 reactor spent fuel at the Zaporizhzhya nuclear power plant. The cask can hold multiple (24 VVER-1000) fuel assemblies in a double-sealed inert (helium) atmosphere in outside storage for a minimum of 40 years.

Nuclear safety regulatory and institutional framework projects address the need for effective regulatory systems in countries with Soviet-designed reactors. The U.S. Nuclear Regulatory Commission has the primary U.S. responsibility for this area of work. Figure 6 shows U.S. Vice-President Gore and U.S. Secretary of Energy O'Leary visiting Russia's Prime Minister Chernomyrdin in 1996 discussing cooperative efforts to establish effective regulatory frameworks.

Since the inception of these cooperative efforts in 1992, U.S. specialists have initiated more than 150 joint projects and have completed over 50 of these projects. The program initially focused on reducing risks at reactors with a high risk of having an accident. Many projects immediately reduced risks at reactors by decreasing the possibilities of equipment malfunction or operator error. As the program has evolved, more attention has been placed on VVER-1000 reactors, particularly in Ukraine.

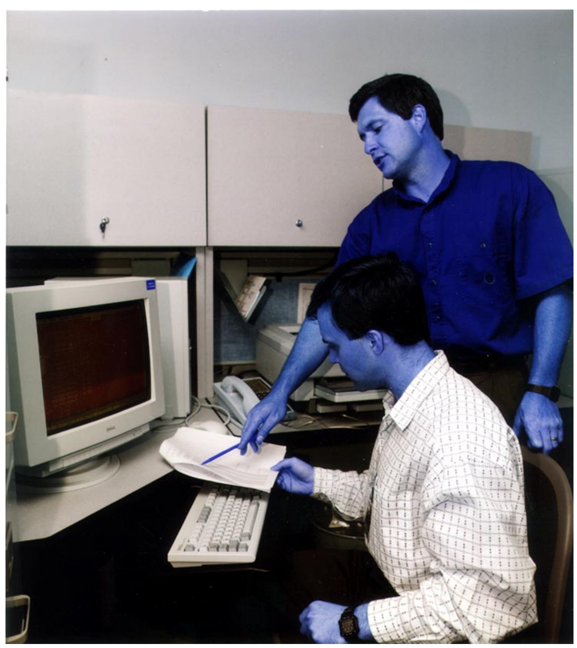


FIGURE 4. Host-country technical specialist training on U.S. computer codes for safety analysis.

FIGURE 5. Cask transporter for Zaporizhzhya nuclear power plant spent fuel dry storage system.

2.4 PERFORMANCE MEASUREMENT

The U.S. team has established end points defining the successful completion of projects in each technical area. A project reaches its end point when the host country, its nuclear support organizations, and its nuclear power plants can sustain safety achievements and build upon them to meet international nuclear safety practices. These end points are measurable, achievable targets. The U.S. team defined the end points by weighing several factors for each project: its safety impact, its cost-effectiveness, the time needed to achieve results, and the ability of the host country to sustain the safety achievements over a long period.

2.5 LESSONS LEARNED

Cooperative efforts between the United States and nine host countries were initiated in 1992 to improve the safety of Soviet-designed nuclear power plants amid heightened international concern about the safety of Soviet-designed reactors. Since then, many lessons have been learned.

These cooperative efforts have fostered an unprecedented atmosphere of teamwork between the countries involved. Developing joint work plans has required an extraordinary measure of openness and cooperation between the participants in the United States and the host countries.

It is absolutely essential that the designers, the operators, and the regulators of the nuclear power plants in the host countries be involved at the outset. Each participant plays a significant role in ensuring that the effort is successful.

FIGURE 6. High-Level U.S. and Russian decision makers discussing need for regulatory frameworks.

Concerns about the liability of U.S. industry working in Russia and the Newly Independent States were overcome for bilateral government-sponsored activities, and work is progressing on conventions that will protect private work.

Frequent and redundant communications with counterparts is helpful to avoid misunderstandings and keep projects on schedule.

Customs and tax issues can be a major impediment to meeting schedules. Resolving those issues consumes substantial resources.

Access to facilities without long approval times is required to keep projects on schedule.

3. Elements of the Cooperative Safety Program that Apply to Use of MOX Fuel

Russia has seven operating VVER-1000 nuclear power plants. These plants appear to be the most likely candidates for burning surplus weapons-grade plutonium. Activities that are part of the cooperative effort to improve the safety of these plants are outlined below. These and other activities would have to be undertaken, should the Russians proceed to using MOX fuel in these nuclear power plants. Russian VVER-1000 safety upgrades include: simulators, emergency operating instructions, training, safety assessment codes, and structure.

3.1 SIMULATORS

A nuclear power plant simulator is an effective training tool. Its computer programs mimic a variety of plant conditions, giving control room operators practice in responding to both routine and emergency situations. A full-scope simulator provides hands-on training by replicating a plant's control room. An analytical simulator uses a network of computers and computer display terminals. The computer graphic displays represent plant systems. Operators practice responding to various conditions by entering commands.

U.S. and Russian organizations are working together to develop control room simulators. A project is under way to provide Balakovo nuclear power plant with an analytical simulator and also to upgrade the plant's existing full-scope simulator.

3.2 EMERGENCY OPERATING INSTRUCTIONS

During an emergency, nuclear power plant operators must stabilize a reactor quickly to prevent damage to the reactor core and the release of radioactive materials. Symptom-based emergency operating instructions (EOIs) for rapid response were developed in the United States after the 1979 accident at Three Mile Island. Symptom-based EOIs now are used at all U.S. nuclear power plants and many others around the world. Previously, operators could not respond immediately to abnormal conditions. They first had to determine the cause of an emergency such as a leak in a steam-generator tube. They then followed procedures designed to correct that specific problem and contain its consequences. These procedures, still used at most Soviet-designed reactors, are called event-based emergency operating instructions.

The pilot plant for institution of VVER-1000 EOIs in Russia is Balakovo. The technology developed at Balakovo will be transferred to other VVER-1000 plants in Russia.

3.3 TRAINING

Well-trained, safety-conscious workers are essential for safe operation of a nuclear power plant. Under the Soviet system, reactor personnel often worked in isolation from their counterparts at other plants and from the international nuclear community, hindering the exchange of information, skills, and lessons learned.

The United States and Russia have established a nuclear training center at Balakovo nuclear power plant. The establishment of in-country training centers accelerates the transfer of skills and information to plant workers. Beginning in 1993, Balakovo instructors received extensive courses in the Systematic Approach to Training, a methodology adopted at all U.S. nuclear power plants after the Three Mile Island accident. The approach provides a standard framework for identifying training needs, developing course materials, and teaching. It combines classroom instruction with the hands-on use of equipment. The training technology developed at Balakovo will be transferred to other VVER-1000 plants in Russia.

3.4 SAFETY ASSESSMENT CODES

The United States is working with Russian specialists to develop in-country expertise in conducting plant safety assessments. The U.S. Nuclear Regulatory Commission transferred the RELAP-5 computer code to two Russian organizations, IBRAE and the Electrogorsk

Research and Engineering Center for Nuclear Power Plant Safety. RELAP-5 is used to create a computer model of a plant's thermal-hydraulic system, including the reactor pressure vessel, piping, and steam generators. The code then performs calculations that predict the progression of various emergencies involving the thermal-hydraulic system and the temperatures in the reactor core.

Although the transferred safety assessment codes currently are not being used at VVER-1000 plants, this technology eventually will be applied to VVER-1000 plants in Russia.

3.5 DEVELOPING REGULATORY AND INSTITUTIONAL FRAMEWORKS IN RUSSIA

U.S. experts are working with Russian authorities to develop a strong legal framework for regulating Soviet-designed nuclear power plants. The objective is to promote strong, independent regulatory bodies with the capabilities to effectively regulate nuclear activities.

In October 1995, representatives of the United States and Gozatomnadzor (GAN, the nuclear regulatory authority in Russia) agreed to work together in improving GAN's capabilities for regulating fuel-cycle facilities. U.S. experts provided training for GAN inspectors, and U.S. and Russian personnel have exchanged technical information and analytical tools. In addition, workshops on fuel-cycle facility quality assurance, safety analysis, inspection techniques, and criticality safety have been held for Russian regulatory personnel.

In addition to the VVER-1000 safety activities in Russia, a host of projects are taking place in Ukraine (e.g., safety parameter display systems, in-depth safety assessments) that are relevant to the safety of using MOX fuel in Russian VVERs.

4. Summary and Observations

An overview of the cooperative effort between the U.S. and nine host countries to improve the safety of Soviet-designed nuclear power plants was provided. Elements of the cooperative effort that apply to ensuring that MOX fuel can be burned safely in Russian VVER-1000 nuclear power plants were identified.

The following observations are offered to help put the subject of reactor safety and use of MOX fuel in Russian nuclear power plants in perspective.

- A Russian-U.S. bilateral program on disposition of surplus weapons-grade plutonium using the MOX option will be viewed as the United States endorsing the long-term operation of the reactors selected to burn the MOX. Accordingly, the United States will be held accountable for promoting higher levels of safety under these circumstances.
- The experiences gained through the cooperative program to improve the safety of Soviet-designed nuclear power plants can play an important role in the Russian plutonium disposition program and the success of the nuclear materials initiative.

- Use of MOX fuel in Russian VVER-1000 nuclear power plants will require modifications or adaptations to the safety improvements already made in these plants under the existing U.S.-Russian bilateral program.
- Consideration should be given to applying the plant safety upgrades incorporated in Ukrainian VVER-1000 plants to the VVER-1000 plants in Russia that may use MOX fuel.

RESEARCH ON NUCLEAR CRITICALITY SAFETY AND ACCIDENT RISK EVALUATION FOR NUCLEAR FUEL CYCLE FACILITIES

S. K. KOUZMINE
B. G. RYAZANOV
V. I. SVIRIDOV
V. V. FROLOV
State Scientific Center, Institute of Physics and Power Engineering (IPPE)
Bondarenko sq. 2, Kaluga Region, Obninsk, 249020, Russia

1. Nuclear Safety System

The nuclear and radiation safety system for nuclear industry facilities in Russia [1] was formed in the course of industrial development. The system is based on the long-standing experience in designing and operation gained by the plants, institutes, and agencies responsible for the reliable operation and safety of nuclear installations. The current safety system has the following elements:

- Technical measures directed towards the facility reliability and safety;
- Successive upgrading and reconstruction of installations and plants; timely shut down and decommissioning of those whose life time has expired and for which there is no further need;
- Improvement and introduction of regulatory documents;
- R&D and experimental investigations on substantiation and development of safety measures, assessment of their reliability, estimation of accident frequency and risk;
- Personnel selection, professional training, and access to work;
- Facilities made ready for accident management, elimination of radiation consequences, protection of the population and territories in case of accidents, including emergency conditions of man-induced nature;
- State inspection, industry and facility control of compliance with requirements, and implementation of nuclear and radiation safety measures;
- Follow-on of chief designers, scientific leaders of projects, and general designers at all the stages of nuclear facility operation.

Continuous improvement of all these elements based on the results of investigation is the most important factor in the efficiency of the safety system. It is extremely urgent nowadays, when the requirements level for population and environment safety have become more stringent not only under normal operating conditions but also under man-made emergency conditions. The organizational structure of the Minatom nuclear safety system is given in Figure 1. It is necessary to stress the specific importance of the operations carried out by the nuclear safety services in the nuclear industry facilities as well as the research and engineering work performed by the Nuclear Safety Division (NSD), Institute of Physics and Power Engineering (IPPE).

The design institutes, which develop the designs in close cooperation with the specialists from facilities and NSD, play a determining role in facility nuclear safety.

Nuclear safety requirements and standards for nuclear industry facilities are set up in the regulatory documents at various levels, i.e., rules, guidelines, and instructions. The structure of the document system is presented in Figure 2. These requirements and standards have been set up based on the deterministic approach; probabilistic methods are used only for the accident risk assessment as well as for special research in support of deviations from the rules and requirements. The basic and regulating document on nuclear safety is presented in the form of principal rules on nuclear criticality safety in the course of nuclear hazardous fissile material use, reprocessing, storage, and transportation. These rules impose the basic requirements to and principles of nuclear safety in the course of nuclear hazardous fissile material reprocessing, use, storage and transportation. They are in force both for facilities where nuclear hazardous fissile materials are fabricated, reprocessed, stored, and transported and for R&D and design institutes and separate departments.

The work on Nuclear Safety in Minatom of Russia is the responsibility of the ECSD. This department is responsible for:

- Nuclear safety in the facilities of nuclear power and industry;
- Compliance with the rules, requirements, and standards of nuclear safety in Minatom institutions;
- Formation and implementation of scientific and technical programs on nuclear safety;
- Development and introduction of nuclear safety principles, criteria, rules, and standards;
- Design coordination for nuclear safety portions and approval of final reports on nuclear safety;
- Qualification upgrading of specialists in nuclear safety;
- Nuclear safety experience sharing among the facilities, including foreign companies.

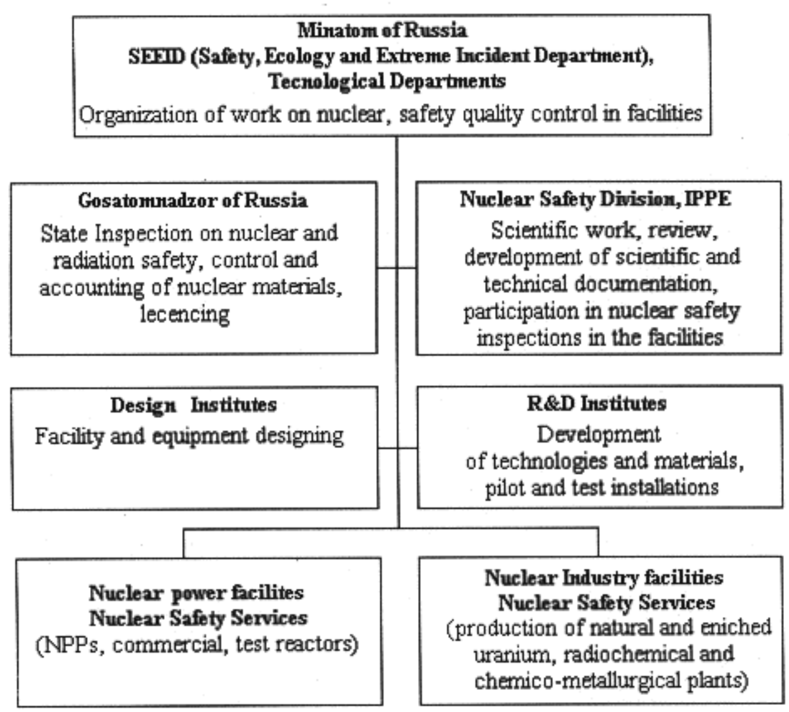


FIGURE 1. Nuclear safety system structure in the Russian nuclear industry.

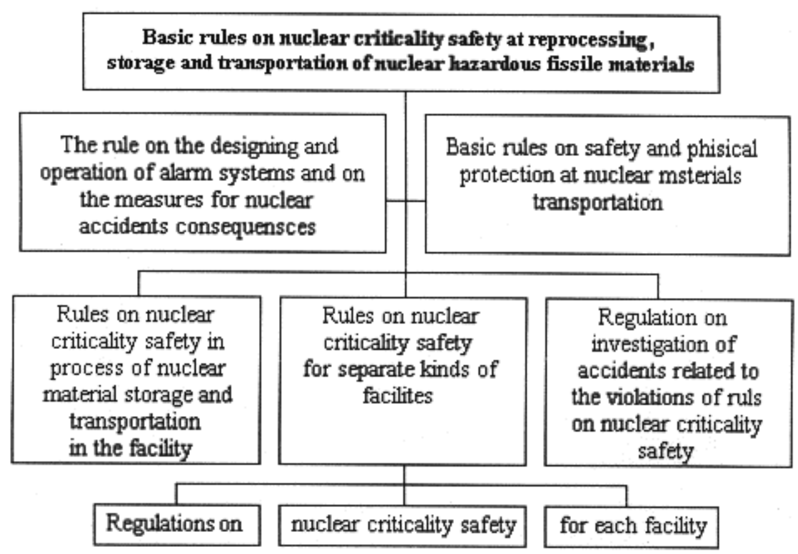


FIGURE 2. Nuclear safety rules for industrial nuclear facilities.

The Nuclear Safety Division of the State Scientific Center IPPE is the basic organization on scientific and technical issues of ECSD, Minatom, Russia. The principal objective of NSD, IPPE, is to solve the problem of nuclear safety in facilities from the scientific and technical point of view. This division is responsible for R&D work on nuclear safety, including:

- Development and improvement of experimental and computation methods and codes, calculations and measurements with the aim to determine critical safe acceptable parameters;
- Development and improvement of methods and tools to control parameters regulated by criticality safety;
- Analysis, guidelines, and R&D work on the assessment of nuclear safety of technological processes;
- Assessment of reliability of nuclear safety measures, spontaneous chain reaction (SCR) risk and consequences in the nuclear dangerous areas, plants, and during transportation of nuclear hazardous fissile materials;
- Development of regulatory documents on nuclear safety (e.g., rules, standards, guidelines, manuals, requirements for methods, tools of control over parameters).
- Conclusions on separate inquiries from the plants, institutes, and organizations dealing with nuclear safety;
- Reviews of designs from the point of view of their nuclear criticality safety, reviews of technical documentation of nuclear danger areas, plants, installation, and equipment; if necessary, reviews of technical documentation for methods, tools, and the system of control over nuclear safety parameters; final reports and conclusions based on the review results;
- Methodological development on nuclear safety, the assessment of consequences, and risk of SCR;
- Participation in nuclear safety inspections in the facilities, checking the plants in terms of their readiness for commissioning, and investigating causes of accidents;
- Organization of training courses for experts in nuclear safety. The nuclear safety services in the nuclear industry facilities are responsible not only for the work directly related to criticality safety but also for their participation in R&D and design work. This includes the development of new technological processes, equipment, storage facilities, methods and tools of control over nuclear safety parameters, emergency alarm systems that initiate the signal for SCR onset and measure the elimination of its consequences, reliability studies of nuclear safety measures, etc. The R&D and design work is carried out in close contact with the R&D and design institutes as well as with the NSD, IPPE, which provides these services with the required data and information on nuclear safety.

In the design institutes and departments, calculations and theoretical studies performed in support of nuclear safety or with the aim to optimize the design solutions are usually carried out with the help of techniques and computer codes, developed and verified in NSD, IPPE.

2. Experiments and Calculations Made in the NSD, IPPE

A great number of various experiments have been performed in the critical facility of NSD, IPPE, in order to substantiate experimentally the available nuclear safety parameters.

The first set of experiments, started in 1959, included simple sub-assemblies with uranium solutions in spheres, cylinders, and parallelepipeds. After that, models of real installations were studied. Then there were experiments with subassemblies with regular cylinders filled with uranium dioxide of high enrichment, with inserts filled with low-enriched uranium with H/U=7.8; with solution devices in the form of a parallelepiped which interacted, for example, through brick, water, and polyethylene; and with solutions poisoned with gadolinium.

All the experiments performed to date have been described in a great number of reports and the apparatus published only partially. The survey of critical experiments and a brief description of the results are given in the report [2]. The quality of the experiments performed in various years varies; however most of them can contribute at least the original composition of fissile material and core geometry to the experiment data libraries.

Since 1995, work on documenting and estimating the results of critical experiments has been carried out in the IPPE facility with the aim of including them in the International Handbook of Evaluated Criticality Safety Benchmark Experiments [3]. Currently, the following critical experiments have been included in the Handbook:

- Experiments with uranium and gadolinium solutions;
- Experiments with spheres filled with aqueous solutions of uranium nitrate with uranium enrichment of 10%;
- Experiments with the models of tubular equipment.

In 1998, the plan is to include into the Handbook the results of experiments to determine the effectiveness of heterogeneous neutron absorbers of various types and sizes (e.g., boron carbide and cadmium in the form of separate rods and groups of rods). Experiments with regular cylinders filled with uranium dioxide and located in the moderator in the form of usual and heavy water mixture will also be put in the Handbook. These results are at the review stage.

Based on the data obtained in critical experiments, the computer codes of various types and levels of complexity that are used in the NSD, IPPE, have been verified. A broad scope of calculations were made in the course of this verification.

The results of experimental studies and calculations were used for the preparation of two editions of the Criticality Safety Handbook [4,5] and industrial rules on criticality safety for several types of production.

In the NSD, vast experience has been gained regarding the application of computational methods for determining nuclear safety parameters, both for separate pieces of equipment of various types and for entire installations. For instance, the experts from this division participated in working out the criticality safety requirements for the RT-1 plant (Mayak), which is currently in operation, and used them in designing the RT-2 plant.

For the immobilization of excess weapons-grade plutonium and uranium, active work is under way on criticality safety for the storage of U and Pu compounds and the disposal of radioactive wastes that contain uranium and plutonium. Certain measures and parameters of criticality safety have been recently determined for PC^2 storage facilities at the Mayak Plant Siberian Chemical Integrated Works (SCIW), Krasnoyarsk Mining Chemical Integrated Works (MCIW). Criticality safety has been assessed in the underground repository for liquid radioactive waste (LRW) [6]. For the RT-1 plant, certain methods have been proposed on how to provide criticality safety under the conditions of reprocessing and implemented with the application of a verification process. The design of the installation for bituminizing LRW has been reviewed from the point of view of criticality safety.

From these requirements, NSD has gained experience in developing methods and tools for controlling humidity in the powdered forms of uranium and plutonium. The principle of operation of the device designed for this purpose is based on the effect of neutron moderation in the product under control, with the predominant contribution from the average neutron energy variation. Due to the design of the detector and the reflector an intermediate neutron spectrum is formed in the powder. The average neutron energy is shifted effectively towards the thermal region with hydrogen nuclei available in the powder under control and in this case this effect is registered with a very high reliability by the ^{3}He counter [7].

The device is a stationary control system designed for operation under production conditions. It can be installed either directly in the room where the finished products are distributed or somewhere outside. In high production environments, this process can be automated. The devices for measuring uranium dioxide humidity, designed and fabricated under the NSD supervision, have been in operation successfully for a long time at the Ust-Kamenogorsk and Elektrostal plants.

3. Assessment of Accident Risk and Reliability of Nuclear Safety Measures

The NSD is specifically involved with the assessment of accident risk for the Minatom nuclear power installations and nuclear material transportation. Similar work has been performed for several installations; for instance, the dissolution installation at the RT-1 plant and the installation for plutonium dioxide production at the MCIW radiochemical plant.

With a very stringent approach, the total intensity of the radiation risk for the personnel and population should be determined based on the frequency and consequences of all the potentially possible accidents at all the plant installations. However making use of this approach results in a tremendous scope of investigations, even when the conditions are simplified. So the most optimal way consists in developing the maximum risk level for a number of the most dangerous installations, thus eventually achieving an upper conservative risk estimate for the plant as a whole. In accordance with the indicated approach, the probabilistic analysis of criticality safety conditions has been carried out for the MCIW radiochemical plant. The analysis of criticality safety in the course of VVER-1000 spent fuel sub-assembly reprocessing at the RT-2 plant was performed based on the design materials and presented at the third U.S.-Russian meeting on non-reactor criticality safety in Los Alamos in 1995 [8]. The activity that started in Russia on insurance against damage as a result of nuclear facility operation also initiated work on risk assessment for the enterprises that are involved in the nuclear power complex of Minatom, Russia. In 1997-98, the NSD together with the experts from different institutions made an accident risk assessment to the public for the SCIW radiochemical plant including the accident risk to the public during transportation of uranium hexafluoride and VVER-spent fuel sub-assemblies. In this risk assessment not only the SCR was taken into account but also some other types of man-induced accidents (e.g., thermochemical explosion, railway accident) were considered as initial events. The results of this work have been discussed by many specialists in nuclear power, by representatives of insurance companies, medical staff, and ecologists. Currently the NSD is continuing with this work on accident risk assessment.

In order to determine how reliable the nuclear criticality safety measures are for installations, plants, institutions, and the nuclear complex in general, the NSD, IPPE, developed a special database, that uses a technique for determining indices of any danger of failure and corresponding software to determine the frequency of nuclear accidents. The technique has been verified by means of the database on nuclear accidents that have happened in the past at nuclear industrial enterprises. The results of nuclear accident frequency estimation were presented in a paper submitted to the international conference on nuclear criticality safety in Albuquerque [9]. The technique is universal and can be used to estimate the frequency of any accident (e.g., nuclear, radioactive, technological). In order to get the results, it is necessary to collect and classify the corresponding human mistakes and equipment failures that can result in accidents and assign them to the indices.

4. Conclusions

At the present time we can state that the current nuclear criticality safety system that exists in Minatom, Russia, meets all current requirements. The criticality safety of the installations that belong to the main process cycle, starting with the uranium enrichment up to waste disposal, has the required reliability. At present, specific attention is paid to the issue of fuel management; in particular, to uranium and plutonium removed from the nuclear weapons complex and to the modification and development of installations and storage facilities required for this purpose. The level of reasonable danger for production or the acceptable level of its risk were a subject for serious scientific investigations and discussions which resulted in the current recommendations formulated for the facilities of nuclear power industry by a number of authoritative international organizations. In these recommendations, consideration was given to the fact that those accidents which are rather rare but whose consequences are serious induce even stronger negative public response than do accidents which happen much more frequently but do not have such serious consequences.

Proceeding from the fact that severe accidents in the facilities of nuclear power industry evidently cannot be eliminated completely (i.e., their probability cannot be decreased to zero), the following main safety objectives have been formulated:

- Prevention of severe accidents and elimination of catastrophic destruction of facilities;
- Protection of people and the environment from the consequences of any accident, if it did happen, to ensure that the consequences of any radiation-hazard accident will be limited and will not exceed the limits accepted for the given type of accident;
- Facility safety should be so high that the majority of people would recognized it as absolute.

These criteria are used for designing and upgrading nuclear industry facilities. Unfortunately the complicated work to estimate the frequency values of various accidents (and their possible consequences), especially for severe accidents for all the facilities of nuclear industry, has not yet been finished. So far, these estimates have been initiated and completed only for separate nuclear installations. Some time ago, many estimations of this type were based on the deterministic approach. In order to predict risk, it is necessary to solve a number of both theoretical and experimental problems. Investigations performed in the NSD, IPPE, on nuclear criticality safety and assessment of accident risk for Minatom nuclear installations for the public can serve as a starting point for this work.

Reference

1. Ryazanov, B.G. (1996) "Basic Principles and Nuclear Criticality Safety System in Facilities of Nuclear Industry of MINATOM, Russia," *Russian-Chinese workshop on nuclear safety in spent nuclear fuel reprocessing*, Beijing Nuclear Design Institute, Beijing, May 24-31.
2. Gurin, V., Sviridov, V., Ryazanov, B. (1995) "Survey of Integral Nuclear Criticality Safety Experiments Performed at SSC RF IPPE." *Proceedings of the Fifth International Conference on Nuclear Criticality Safety*, Albuquerque, New Mexico, U.S., September 17-21, pp. 4.48-4.54.
3. "International Handbook of Evaluated Criticality Safety Benchmark Experiments," *NEA/NSC/DOC* (95) 03, September 1997 Edition.
4. Dubrovsky, B.G., et al. (1966 "Critical Parameters of Systems with Fissile Materials and Criticality Safety (Handbook)," Moscow.
5. Diev, L.V., et al. (1984), "Critical parameters of Fissile Materials and Criticality Safety (Handbook)," Moscow.
6. Kouzmine, S., Ryazanov, B.., Sviridov, V. (1997), "Criticality Evaluation of Deep-hole Disposal of Fissile Containing Liquid Radioactive Waste," *Proceedings of the Topical Meeting of the Criticality Safety Division—Criticality Safety Challenges in the Next Decade*, September 7-11.
7. Bulanenko, V.I., Frolov, V.V., Charychansky, V.V. (1984), "Neutron Control of Low-enriched Uranium Dioxide Humidity," *Atomnaya Energiya* **56**, 155-157.
8. Kouzmine, S.K., Ryazanov, B.G., Sviridov, V.I., (1995), "Analysis of Criticality Safety in VVER-1000 Spent Fuel Sub-assembly Reprocessing at the RT-2-plant," *Proceedings, III US-Russian Meeting on Non-reactor Nuclear Safety*, Los Alamos.
9. Ryazanov, B.G., Gorunov, V.K., Sviridov, V.I. (1995), "Numeric Estimation of Nuclear Criticality Safety Level in Nuclear Power Industry," *Proceedings, V International Conference on Nuclear Criticality Safety*, Albuquerque.

SAFETY AND THE FRENCH-GERMAN-RUSSIAN TRILATERAL MOX FABRICATION FACILITY IN RUSSIA—DEMOX

G. BRÄHLER
Siemens AG RBRefV
P.O. Box 110060, D-63434 Hanau, Germany

J. PIERRE
Cogema
2, rue Paul Dautier
BP 4 78141 Vélizy Cedex, France

EVGENI I. TYURIN
L. I. PETROVA
Specialized State Designing Institute, GSPI
8a, Novoryazanskaya str., Moscow, Russia

1. Introduction

The G7 Moscow summit in April 1996 on nuclear matters provided a political framework to deal with one of the most significant challenges facing the nuclear industry today—find a solution to the weapons-grade fissile material disposition issue resulting from the disarmament efforts by both the United States and Russia.

European experience has shown that the transformation of plutonium (Pu) into mixed-oxide (MOX) fuel is a very efficient, safe, nonproliferating and, under certain circumstances, economically effective solution.

In order to improve both the speed of the disarmament process and its economical feasibility, it is a common understanding that existing experience and technologies—to the extent possible—should be used. The French and German experience gained in more than 20 years of industrial MOX fabrication and irradiation and the New Hanau MOX Plant, which was nearly completely constructed but never put into operation, represent assets of high value.

Since 1992, both France and Germany have developed bilateral cooperation programs with Russia in order to assess the feasibility of recycling weapons grade plutonium in Russian reactors. These cooperation studies came to a similar conclusion; loading MOX fuel (made from weapons-grade plutonium) into Russian VVER-1000 and fast reactors, in particular the Balakovo units and the BN-600, is feasible.

Both programs also developed concepts for pilot plants to fabricate fuel for these reactors. They were designed for capacities of approx. 1 to 1.3 t Pu/year, representing the consumption of four VVER-1000s and one BN-600 reactor.

In autumn 1996, Russia, Germany, and France decided to combine their efforts in a joint initiative for the peaceful management of weapons plutonium in Russia. This cooperation was announced during the P8 expert conference at Paris, October 28–31, 1996.

Consequently, Cogema, Minatom, and Siemens have launched a joint project, which includes the design, construction, and start-up of a first MOX fabrication plant in Russia—the DEMOX project. The project has found broad international consent and attention, from the G7 Denver Summit to various studies on the subject of disarmament (Independent Scientific Committee, CSIS). Just recently, the G8 experts conference at London stated again that the Western countries have to help Russia in financing DEMOX.

2. The Experience of the Partners

Plutonium recycling is now an industrial reality in Europe. More than 20 reactors are already loaded with MOX fuel, representing more than 70 t of Pu processed in Europe.

Cogema is currently operating two MOX fabrication plants in France: the Cadarache plant (40 t of MOX/year) and the MELOX plant (120 t of MOX/year).

Using the experience gained during the operation of a first plant which had a capacity of 25 t/year, Siemens constructed a large MOX plant with 120 t heavy metals (HM)/year capacity. It has never started due to political factors.

Minatom has significant and broad expertise in both fabrication and irradiation of MOX fuel. Use of Pu as nuclear fuel started in the second half of the 1950s. Construction of a large industrial plant (A300 at Mayak) was well advanced at the end of the 1980s but was suspended together with the BN-800 program.

3. The DEMOX Project

3.1 BASIS

DEMOX will be developed using the considerable industrial experience gained by Cogema and Siemens in Western Europe in the field of MOX fuels and combining it with the significant expertise of Minatom.

Russia intends to start the program for the use of weapons plutonium by introduction of MOX fuel into the existing four VVER-1000 reactors and one BN-600 reactor. The aim of the DEMOX project is to build an industrial demonstration plant in Russia that is able to produce the required MOX fuel. This plant could be in operation as soon as 2002. The fuel requirements are:

- VVER-1000: 29 t HM/year including 1.07 t of plutonium;
- BN-800: 1.2 t HM/year including 0.24 t of plutonium.

The experience acquired during this period could be used to increase the capacity of the plant up to 5 t Pu/year, which would allow plutonium to be recycled in a greater number of reactors.

3.2 STATUS

An integrated project team based at Cogema's engineering offices and led by a Siemens project manager started work in March 1997.

The first phase was the Harmonization Phase, which started with the results of the two bilateral (Russian–French and Russian–German) feasibility studies.

The goal was to harmonize the Cogema process and technology, as represented by the A-MIMAS process in the MELOX Plant, and the Siemens design and equipment, as realized in the New MOX Plant Hanau that had been constructed but not commissioned, and whose equipment is available for DEMOX.

The result was the concept of the fabrication processes described in a set of documents, taking into account Russian basic conditions, as known from the feasibility studies. Generally speaking, the plant consists of one line from the Hanau plant, but with single equipment where the Hanau plant has double. Some of the equipment has to be adapted to fulfill the A-MIMAS requirements.

The consolidation phase in progress since December 1997 covers the validation with Minatom of the basic options and data. It also includes the preparation of the first Russian documents to start the project within Russia, including the justification of investment, a site evaluation, and a first cost estimation.

Russian experts will join the project team for the basic design work that will be performed during 1998 and 1999.

4. DEMOX Safety Design

It should be clearly noted that DEMOX will first satisfy Russian requirements and that the Russian operator will be responsible to the safety authorities. However, the experience of Cogema and Siemens will be of help and their existing facilities may serve as references.

4.1 BASIS

The following documents and activities will serve as basis for the safety design of DEMOX:

- Existing Russian regulatory documents, standards, and experience with safety design of Pu-handling plants and laboratories;
- Cogema and Siemens experience with safety design of industrial MOX facilities;
- Safety-related activities in bilateral studies 1992-96: Russian and Western European general safety principles are compatible;
- Design of the existing equipment in the Hanau Plant.

Concerning the application of the general safety principles to the conceptual and basic design of DEMOX, recent seminars have enabled us to compare Russian and Western European design measures concerning fire protection, criticality control, and radiation shielding.

Further work will be performed by Minatom, Cogema, and Siemens in order to define the measures to be taken in the DEMOX case, in these fields and in others.

4.2 GENERAL FEATURES

The general safety target in nuclear facilities is the protection of the population and the working staff from harm caused by radioactivity, both external radiation and incorporation (e.g., ingestion, inhalation).

A general approach for MOX plants considers the following main safety features:

- Nuclear safety requirements (due to the nuclear properties of material to be processed) include radiation protection of staff, radiation protection of environment and population, and criticality safety;
- Enhanced conventional requirements (due to increased risk from nuclear materials) are defined in two categories: (1) protection against impacts from inside the plant and (2) protection against impacts from outside the plant;
- Standard conventional safety requirements include environmental protection, industrial safety, and others.

Concentrating on the nuclear safety requirements, the radiation protection inside the plant (staff) is accomplished by shielding, size, and arrangement of radiation sources; removal of radiation sources before intervention, wherever possible; substantive alpha barriers (glove boxes, piping, transport containers) in connection with sub-atmospheric pressure; and by automation and remote operation.

Radiation dose measurement, personnel dosimetry, and medical supervision control the effectiveness of the protective measures.

Concerning radiation protection outside the plant, a barrier concept with progressive rates of vacuum from inside (e.g., glove boxes) to the outer atmosphere in connection with the ventilation system and treatment of off-gas by HEPA filters will achieve protection of the population. Emission of airborne radioactivity is monitored and the radioactivity of liquid effluents is controlled.

Criticality safety is a major concern in any fuel fabrication facility. Proper design of the hardware and process control together with administrative precautions prevent criticality accidents. Additional monitoring and alarm systems will be installed.

4.3 PARAMETERS AND DATA

The limits for radiation protection are as follows:

- Doses for operational staff
  20 mSv/year effective, with design basis below this value;
  50 mSv/year partial body.

- Contamination
  0.005 Bq alpha/m^3 in air;
  0.05 Bq alpha/cm^2 on surfaces;
- Doses for population 1 mSv/year.

The Pu isotopic compositions used for all radiation protection calculations is given in Table 1 and the Pu contents used in the calculations are given in Table 2.

There are barriers and ventilation systems (Fig. 1) for the prevention of incorporation and the spread of contamination inside the building and into the environment. The facility is divided into two main building areas: a process building for handling of Pu and an auxiliary building for inactive auxiliary functions (e.g., ventilation, electrical systems, automatic control systems).

Three barriers, each a combination of a static containment barrier (e.g., mechanical structure, glove box, or fuel rod) and a dynamic containment barrier (e.g., ventilation, air flow respiratory pressure), assure the safe enclosure of radioactivity. The air pressure increases at each stage, from the glove boxes to the working rooms, to the building or corridors, and then to the outer atmosphere.

TABLE 1. Radiation protection: isotopic composition of Pu for calculations.

Pu-x	**As % of (Pu-tot + Am)**
Pu-239	92.5
Pu-240	5.9
Pu-241	0.6
Am-241	1.0

TABLE 2. Radiation protection: Pu content in different products (%).

Product	**Pu content (%)**
VVER Master Mix	30
VVER Final Mix	3.7
VVER pellets, rods, assemblies	3.7
BN powder	20
BN pellets, pins, assemblies	20

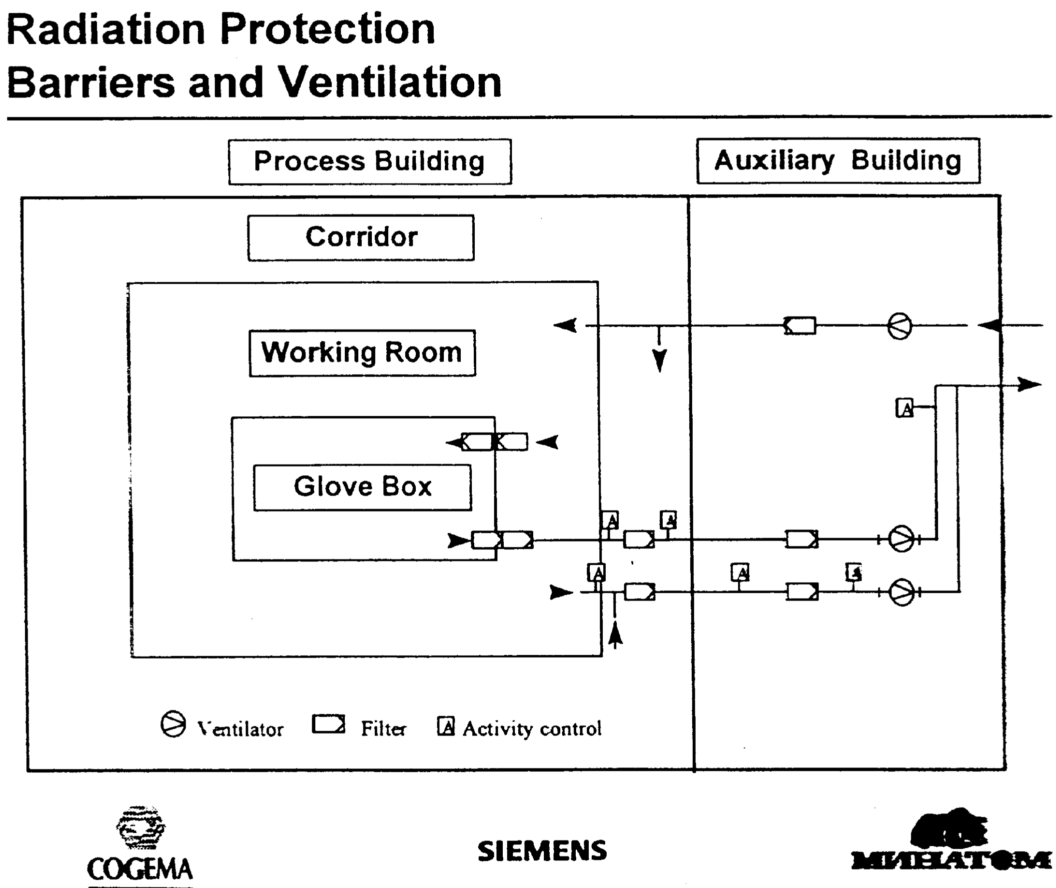


FIGURE 1. Radiation protection barriers and ventilation.

4.3.1 Criticality safety

For criticality safety calculations, the isotopic composition differs from that used for radiation protection design: Pu-239 in Pu-tot = 95% and Pu-240 in Pu-tot = 5%. Similarly the contents of Pu in different products applied are more conservative (*cf.* Table 3).

Generally the moderation in powder area is limited. For each unit, one makes a choice of the criticality control mode: (1) mass in process equipment (interlinks and administrative measures) or (2) shape and size in storage areas (hardware). The double contingency principle will be applied. Calculations will be done using fixed conservative values for densities of products, a water reflector, and as acceptance criteria a K-eff. below 0.95 or the safety factors given in Table 4.

TABLE 3. Criticality safety: Pu contents (%).

Product	**Pu content (%)**
VVER Fuel Master Mix Powder	35
VVER Fuel, Final Mix Powder, Pellet	5.5
BN-600 Powder, Pellet	30

TABLE 4. Criticality safety: safety factors applied to critical values.

Mass	2.1
Volume	1.3
Diameter	1.1
Slab thickness	1.1

4.3.2 Fire protection

The philosophy of fire protection is based fundamentally on the prevention of fire by choice of materials and by minimization of inventory of incendiary material. Once a fire occurs (going on in the scenario), the second measure is to prevent the spread of fire by the design of fire sectors and the use of fire-resistant barriers. Thirdly, one has to detect the fire, usually by means of a fire alarm system. Finally, no doubt, one needs organization, staff, and equipment for fire extinguishing by installed and mobile means.

SAFETY OF THE BELGONUCLEAIRE MOX FABRICATION PLANT

YVON VANDERBORCK
JEAN VAN VLIET
Belgonucleaire
4, Avenue Ariane-B 1200 Brussels, Belgium

1. Introduction

As a result of the end of the cold war and the subsequent agreements, one of the most pressing issues for the international community is to decide on the fissile material recovered from ongoing and planned dismantlement of U.S. and Russian nuclear weapons.

The weapons-grade plutonium (WG-Pu), in particular, should be transformed in a way that allows it to meet the spent fuel standard requirement, as defined by the U.S. National Academy of Sciences [1] and as retained in U.S. and Russian plans today.

Two options have emerged from the various investigations carried out. They have been confirmed during the last G7/P8 extended expert meeting of October 1996 in Paris and described in the final PEIS issued in December 1996 [2]; as clearly indicated in the Record of Decision of January 14, 1997, the options are:

1. Use it as nuclear fuel for electricity production in civil power plants and keep the remaining plutonium in association with the fission products in spent fuel in a “once-through” strategy;
2. Immobilize the WG-Pu with fission products to fabricate a highly radioactive but stable compound by vitrification, and to dispose of it in a deep geological repository.

The choice of using WG-Pu as nuclear fuel (i.e., MOX option) is natural considering that the technology is mature, proven, and timely and the technical, economic, and safeguards aspects are satisfactory. This paper describes the current experience in Belgium for MOX fabrication and the safety aspects related to the Belgonucleaire MOX fuel fabrication plant.

2. Belgonucleaire at the Heart of the Belgian Nuclear Industry

Belgonucleaire was created in 1957 by Union Minière to develop plutonium fuels and has been since its creation a pioneer in this field, at the heart of the nuclear industry.

Today, Belgonucleaire is part of the TRACTEBEL group. TRACTEBEL is the largest Belgian industrial group and has an international strategy.

ELECTRABEL, which is the private Belgian electrical utility company, is the major electrical operator in the Belgium unit. It operates 7 nuclear plants with a total capacity of 5500 MW, which represents 56% of the total electricity demand in Belgium. The group's key advantages are its human potential and its technical know-how. It has a staff of roughly 37,000.

TRACTEBEL has an equity (1996 financial data) of 9 billion USD and a balance sheet of 24 billion USD. The group had a turnover of 11 billion USD with a total cash flow of 2.5 billion USD and a net result of 1 billion USD.

2.1 TRACTEBEL'S INTERNATIONAL STRATEGY

Within its various specialties, the TRACTEBEL Group is active in about one hundred countries. The presence of TRACTEBEL in the CIS is well known, more specifically in the Russian Federation in the engineering studies and backfitting for safety and reliability (EU-TACIS-BERD) for VVER-1000 and RMBK and a reactor simulator for the Beloyark reactor. In Ukraine, TRACTEBEL made feasibility studies for a nuclear power plant. The company developed a simulator for an RBMK reactor in Lithuania and is very active in the construction and operation of electrical power and heat generation plants in Kazakhstan.

2.2 BELGONUCLEAIRE AT THE HEART OF NUCLEAR KNOW-HOW

The two main activities are MOX fuel supply and nuclear engineering. In nuclear engineering, Belgonucleaire specializes in MOX fuel design, MOX plant construction, and waste conditioning. It runs a 35 t/year MOX plant in Dessel, in the northeast of Belgium. As engineers, they have designed and built this plant, and also used their expertise to cooperate with Cogema to build MELOX.

3. Belgonucleaire's Experience with MOX Fabrication

This experience is described in [3] and [4].

3.1 AN EARLY BEGINNING

After an early beginning in 1957, and a period of R&D programs, Belgonucleaire now owns an industrial fabrication plant called P-zero (P0), which has been operating since 1973. The plant is situated in the Mol-Dessel nuclear site in Belgium.

In 1985, after the Electricitee d' France (EDF) decision to recycle its plutonium in their 900-MW pressurized water reactor (PWR) plants, it appeared that a large commercial market was building up for MOX fabrication for light water reactors (LWRs). Therefore, Belgonucleaire simultaneously decided to increase its production capacity up to 35 t of heavy metals (HM)/year and also to change its fabrication process in order to meet the commercial and technical requirements of the customers.

The MOX plant has received certificates from different customers such as Framatome, Siemens, Toshiba, Hitachi, and Japan Nuclear Fuel. It operates according to IAEA 50CQA rules and has recently been certified according to ISO 9002 (1994).

3.2 THE MIMAS FABRICATION PROCESS

The process developed by Belgonucleaire (Fig. 1) is called MIMAS for "MIcronized MASter blend." The MIMAS MOX pellets are composed of a solid solution of UO_2 and PuO_2 homogeneously dispersed in a UO_2 matrix. This result is achieved through two blending steps: (1) the primary (or master) blend is obtained by ballmilling and (2) the secondary (or final) blend is used to produce the specified plutonium content in the pellets.

The features of the MIMAS process include:

- Primary blend obtained by dilution of PuO_2 to about 30% Pu, prepared in advance of the pellet fabrication. The same primary blend can be used to produce pellets of various Pu enrichments;
- Two-step blending process allows high flexibility for Pu isotopic homogenization;
- Use of simple equipment, with a good reproducibility;
- Dry technology;
- Reduced scraps;
- Demonstrated economical viability shown by full capacity production;
- Industrial-level technology.

These advantages led Cogema to adopt the same process for the MELOX plant and, since 1996, also for CFCa (the Cadarache plant). The MIMAS process will thus remain for many years as the world leader for MOX production. Out of the 1400 tHM MOX fuel expected to be produced by the end of the year 2000, 1100 tHM will be MIMAS fuel.

3.3 PRODUCTION RECORDS

By March 1998, more than 182,000 fuel rods had been manufactured by Belgonucleaire with the MIMAS process in its Dessel plant in Belgium. This represents more than 350 tHM out of about 780 tHM of MOX fuel produced world-wide (Fig. 2).

This MIMAS fuel, fabricated by Belgonucleaire, is at present irradiated in 17 commercial reactors in 4 countries, namely 7 PWRs in France (EDF), 2 PWRs in Belgium, 2 PWRs in Switzerland, 4 PWRs in Germany, and 2 BWRs in Germany as well (Table 1). A fabrication for a Japanese BWR reactor was made last year. Considering the commitments over the coming decade, production will be pursued for Germany, Belgium, Switzerland, and Japan. Belgonucleaire today is the only MOX fabricator for BWR.

The P0 plant is a medium-size plant comprising two main fabrication lines laid-down in parallel. It is characterized by high flexibility in terms of fuel design (PWR or BWR), specifications (qualified for several designers: Belgonucleaire, Siemens, Framatome, and Toshiba/Hitachi, JNF) or PuO_2 characteristics (Pu content, isotopic composition). Table 2 shows the range of parameters already demonstrated.

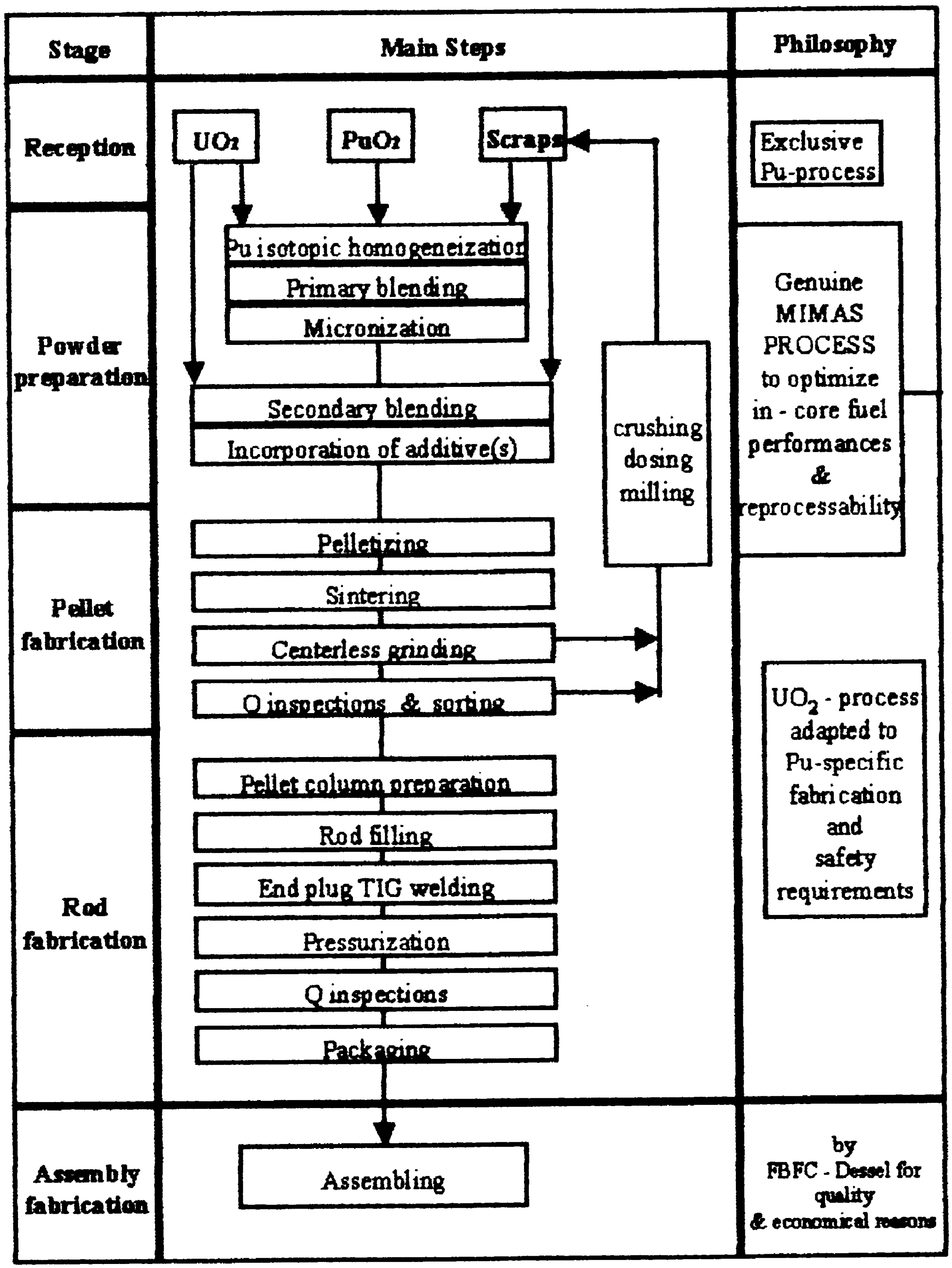


FIGURE 1. The MIMAS process.

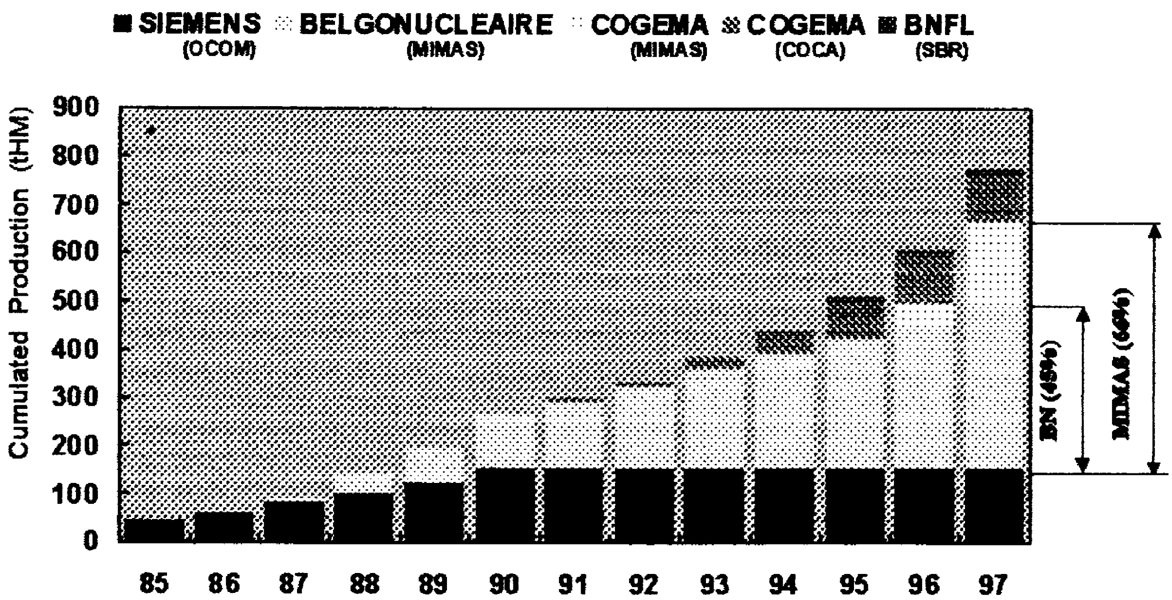


FIGURE 2. Cumulated MOX production for LWRs in Europe (at the end of 1997).

TABLE 1. LWRs loaded with MOX fuel from the P0 Plant, Dessel (March 1998).

Country	**Demonstration Program**	**Commercial**
BELGIUM	PWR BR3	PWRs DOEL 3 TIHANGE 2
FRANCE	PWRs CNA CAP	PWRs SLB 1 & 2 GRAVELINES 3 & 4 DAMPIERRE 1 & 2 BLAYAIS 2
SWITZERLAND	PWR BEZNAU 1	PWR BEZNAU 1 GOESGEN
GERMANY	–	PWRs GRAFENRHEINFELD BROKDORF PHILIPPSBURG UNTERWESER BWRs GUNDREMMINGEN B&C
SWEDEN	BWR OSKARSHAMN	–
THE NETHERLANDS	BWR DODEWAARD	–
ITALY	BWR GARIGLIANO	–

TABLE 2. MIMAS fabrication—range of tested parameters (March 1998).

Cumulated MOX tonnage	350 tHM
Fuel rod types	14x14, 15x15, 16x16, 17x17 (PWR) 9x9, 8x8 (BWR)
Qualified MOX fuel designers	FRAGEMA, SIEMENS, TOSHIBA/HITACHI/JNF, BELGONUCLEAIRE
Size of a fabrication campaign	4 to 29 tHM
Number of Pu contents per campaign	3 to 6
Pu tot. content in the pellet (over Pu tot. + U)	Up to 8 %
Am content (over Pu tot.)	Up to 17,000 ppm
Pu tot. content in primary blend (over U+Pu tot.)	25 to 35%
UO_2 material	Free-flowing AUC and TU2

4. The Existing P0 Plant—its Main Achievements

As already mentioned above, the plant has produced more than 350 metric tons of MOX MIMAS fuel for LWRs and 10 tHM for FBRs and has thus processed almost 20 tons of plutonium. The results of P0 are not only expressed in terms of capacity. They are also expressed in operating experience, such as the plant flexibility to accept several input products and fuel specifications, equipment operation records, etc. This knowledge is of the utmost importance in designing an efficient and safe MOX plant as we propose for U.S. and Russian needs to solve the issue of WG-Pu disposition.

4.1 SAFETY AND SAFEGUARDS AT THE P0 PLANT

We have in Belgium a special regulatory environment (Fig. 3) which has changed since 1955 and has reached today's level after the reform of the Belgian Federal State in the 1980s.

The emergency plan covers different types of accidental situations such as criticality, fire, contamination, stack discharge, and accident from external cause, and is set up according to national (federal) guidelines.

4.1.1 Operator exposure

The exposure of the plant operators is minimized. To reach this goal we have used hardware solutions like neutron and gamma shields on the glove boxes.

Software solutions are also used. They aim at moving the operator away from the equipment. Automatic inspection machines are used and, even if the visual control of the process remains important, most of the control cabinets are installed in separate rooms distributed along the plant close to the production rooms.

Recently, we have modified our production equipment for a better mechanization of all mass storage and we have developed new process equipment for visual inspection of the pellets and for opening the plutonium cans.

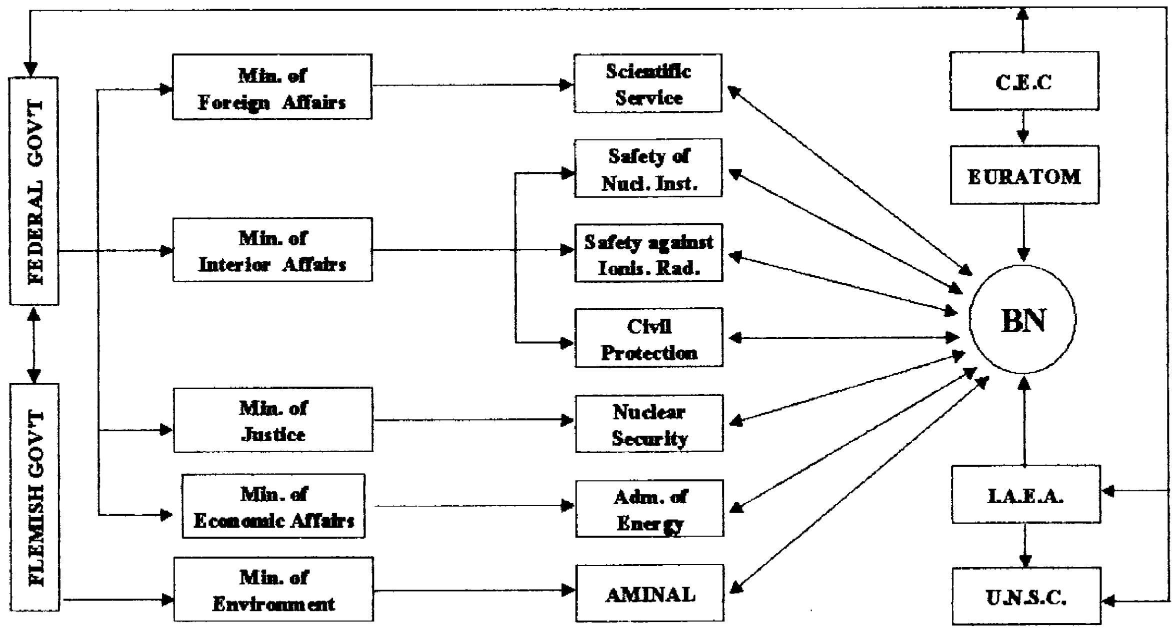


FIGURE 3. Regulatory environment.

4.1.2 Contamination inside and outside the containment

A triple confinement concept is used ranked towards outside: the glove boxes containing equipment, the fabrication rooms, and the confinement provided by the building itself. In addition, dynamic confinement is provided keeping in operation a sophisticated once-through ventilation system. The latter ensures air pressure cascades in such a way that any leak is directed from a less to more risky zone and filtration of the exhausted air. Obviously the blowers and filters are redundant, and the energy supply for running the fans is provided by an emergency generator in case of grid failure.

These features limit the contamination risk inside the containment. Among other features for limiting contamination is the specially designed plutonium dust collection system in the glove boxes.

4.1.3 External accidents

The location of the plant precludes any flooding. The risk of an external explosion due to road or water traffic as well as human or industrial activities has been evaluated and found negligible.

4.1.4 Internal accidents

Criticality. The risk of criticality is minimized down to an absolute minimum thanks to physical layout and mass limitations. The plant is completely dry, no moderator is allowed in the production rooms, and we use a dry process.

The transfer of material to or from one of these units is only performed when it can safely be done. A redundant fail safe instrumentation detects any increase of the fission rate should this kind of accident occur. Special alarm systems and escape ways are carefully designed.

Fire Risk Management. The risk of fire is especially considered, and advanced and modernized fire protection has been provided. The classical means of fire prevention like avoiding any combustible material and fire initiation sources are of course applicable.

Special attention has been also paid to fire fighting. It includes the partition of the plant in fire zones limited by fire-resistant walls. Each zone has its own ventilation subsystem able to isolate it from the neighboring zones. The ventilation system is designed to remove the fumes generated in the room impeding the access for the fire fighter crews and to maintain the room pressure in the damaged zone below that of the other zones.

As means used for fire fighting, let us note the existence of a carbon dioxide injection system with enough capacity to completely fill the larger rooms. Gas injection systems are also provided in the electric cabinets.

4.1.5 Safeguards

For safeguards, the existing rules have also been modified since 1957.

A permanent inspection by IAEA and EURATOM inspectors has been installed since 1978 with an accurate accounting system for the fissile materials under constant surveillance by guards and electronic devices. The physical protection measures and rules are determined by the Ministry of Justice and are confidential.

The Belgonucleaire accountancy system is based on the splitting of the plant in 400 work units for criticality and safety. All movements crossing the work unit boundaries are registered in near real time on the basis of shipper's data or calculated values or analytical results taking into account the radioactive decay of the materials.

This system is itemized and recognizes single items (e.g., bottles, cans), composite items localized at one spot (e.g., assemblies), inside glove boxes, or spread over different work units.

4.2 SECURITY AND PHYSICAL PROTECTION—PRINCIPLES

The main characteristic is prevention, including clearances for visitors, external workers, patrols escorts, compartimentation, and access paths.

Different electronic devices allow early detection and early notice to authorities. A series of successive physical barriers (including fences) are designed to delay any intrusion.

The measures related to physical protection are based on :

- Law of August 4, 1955;
- Royal Decree of March 14, 1956 (execution of the law).

The permanent control of these measures is performed by the Belgian Federal Nuclear Security Service (Ministry of Justice).

All these measures are coherent with IAEA recommendation INFCIRC/225/Rev.3, "The physical protection of nuclear materials."

5. The Belgian Rules for Licensing MOX Plants

5.1 GENERAL RULES APPLICABLE FOR AN INDUSTRIAL PLANT IN BELGIUM

- RGPT: General regulation concerning work protection;
- Other general regulations (e.g., fire, noise, electricity hazards);
- Belgian norms and standards.

5.2 LOCAL RULES APPLICABLE FOR AN INDUSTRIAL PLANT

- VLAREM: Flemish regulation concerning environmental protection.

5.3 RULES APPLICABLE FOR A MOX FACILITY

- RGPRI: General regulation concerning the public and work protection against hazards due to ionizing radiation;
- Specific licensing document issued as a Royal Decree.

5.4 LICENSING PROCESS OF MOX FACILITY

- Building license (local procedure);
- Operation license (Country's Governor + Communities);
- Special Commission (evaluation of the license application);
- Formal license granted by a Royal Decree;
- Special control body inspection after construction but before commissioning.

6. The Proposed P1 Plant for U.S. and Russian Needs

On the basis of our almost 25 years of safe experience on our P0 plant and with the return of experience of the Belgonucleaire participation to MELOX, we have designed a "third generation" P1 plant that will provide enhanced safety and flexibility with a capacity of 60 to 100 tHM. It will be able to fabricate PWR and BWR fuel and to answer the United States and Russian needs to solve the issue of WG-Pu disposition. This plant has three fabrication lines (instead of two, as in P0). It has been designed to withstand an earthquake and the crash of airplanes. The design is based on a MSK VI (0.03 to 0.09 g) earthquake related to the major credible motion in our country. Anyway the strength of the building is mainly fixed by the plane crash criteria and it can sustain a stronger earthquake than MSK VI.

The function of the building is to provide a suitable static confinement. Furthermore, the parts of the ventilation system playing an important role in the dynamic confinement of the plant are designed against earthquake and vibrations generated by the plane crash.

7. Conclusion

- MOX fabrication is a mature and safe technology;
- MOX irradiation is well experienced in Europe.

By its experience of more than 40 years, Belgonucleaire has acquired an expertise in MOX plant construction and operation, in MOX fabrication for BWR and PWR, and in MOX irradiation.

This global experience is offered to the United States and Russia to contribute to a safe disarmament by construction of a MOX plant in these two countries.

References

1. *Management and Disposition of Excess Weapons Plutonium.* National Academy of Sciences Report, National Academy Press, Washington DC (1994).
2. DOE EIS0229, December 1996.
3. Vanderborck Y., Vandenberg Cl., Haas D., "Disposition of weapons grade plutonium in MOX fuel. What does Belgonucleaire propose for implementing this solution in U.S. and in Russia," *Fourth Annual International Policy Forum*, Lansdowne, VA, February 11-14, 1997.
4. Haas D., Vanderborck Y., Vandenberg Cl., "Disposition of plutonium from dismantled warheads, Belgonucleaire's proposal," *ANS/ENS 1996—International Conference and Embedded Topical Meeting*, November 10-15, 1996, Washington DC.

ENSURING THE SAFETY OF MOX FUEL TRANSPORT

A. V. AFANASSIEV
A. L. LAZAREV
N. S. TIKHONOV
A. I. TOKARENKO
All-Russian Project and Research Institute of Complex Energy Technology (VNIPIET)
82, Savushkina Str. 197183, St. Petersburg, Russia

1. Principal Factors That Affect MOX Fuel Transport

The application of the mixed uranium-plutonium fuel in power reactors requires assurance of safe transport of semifinished items, fuel elements, and fuel bundles (FB). To research various aspects of safety, it is necessary to take into account that the thermal and radiation characteristics and criticality parameters of MOX fuel are higher than the characteristics of fuel on a basis of uranium dioxide.

Particularly, the minimum critical mass for Pu-239 is less than 1.5 times the appropriate parameter for U-235, whilst the critical masses for dry dioxides of the above isotopes differ from each other more than 3 times. Moreover, the mixed uranium-plutonium fuel is a powerful source of neutron and gamma radiation.

Sources of radiation in fresh fuel are plutonium isotopes, products of decay of the plutonium isotopes, and impurities of products of fission in the regenerated plutonium. As a result, the gamma and neutron radiation dose on a surface of fresh fuel bundles generated by fuel from weapon plutonium exceeds by more than an order of magnitude the appropriate dose capacity for FB from uranium fuel. Moreover, capacity of dose on a surface of FB with regenerated plutonium exceeds on an order of magnitude the dose capacity for FB with weapons plutonium.

Besides, surfaces of FB and the fuel elements can be polluted by plutonium. The level of heat release for FB made of weapon plutonium is essentially higher than that for FB made from uranium (in the latter, heat release is practically absent). For example, heat release of FB of reactor BN-600 reaches 20 watts and that in FB of reactor VVER-1000 reaches 130 watts. The heat release of fuel bundles with regenerated or recycled reactor-grade plutonium is several times higher.

For spent MOX fuel, the most essential thing is an increase of neutron radiation. Particularly, the levels of neutron radiation of FB with weapon plutonium and FB with the regenerated plutonium exceed the levels of radiation of FB with uranium approximately 4 times and 10-15 times, respectively.

Spent fuel bundles with fuel from power (i.e., reactor-grade) plutonium, which were stored for 1-3 years, are characterized by a heat release approximately 70% higher than those with uranium dioxides. The heat release from fuel bundles made of weapon plutonium is approximately 30% higher than that of FB with uranium [1].

All the above factors should be taken into account in the development of technology for the transportation and design of packages for MOX fuel.

According to the Russian document, “Main regulations of safety and physical protection of nuclear materials transport,” [2] IAEA Regulations for safe transport of radioactive material, [3] and other normative documents, the safety of transport should be ensured by the design of transport packages.

It follows from the above regulations that the transportation of fresh and spent mixed uranium-plutonium fuel should be carried out in transport packages of type B, which are supplied with devices for tightness testing, equipped with an effective radiation shield, and designed to ensure the removal of heat from the fuel.

2. Transportation of Fresh MOX Fuel

VNIPIET executed preliminary technological and economic studies to examine transport of various kinds for fresh MOX fuel in order to determine a direction for research on available transport casks and the development of new ones.

2.1. SEMIFINISHED ITEMS OF MOX FUEL

To manufacture fuel bundles with mixed fuel, it is necessary to ensure transport of semifinished items (powder or tablets). The analysis performed of the fleet of casks in Russia determined that, among packages for transportation of semifinished items of MOX fuel, packages TUK-29 and TUK-30 have the most acceptable parameters.

To ensure transport of MOX fuel in these casks, we must calculate and assess criticality, radiation safety, and thermal conditions, and probably modify the design of the packages and certify the packages.

2.2. VVER-1000 REACTOR FUEL ELEMENTS AND FUEL BUNDLES

At the moment, there are no transport packages in Russia that are suitable for the transportation of fuel elements and fuel bundles of reactor VVER-1000 with fresh mixed fuel. VNIPIET has performed design studies for such packages and appropriate auxiliaries. The work was conducted in two directions:

- Cask TK-S8 for transport of samples of fuel elements and fuel bundles (10–20 FBs/year);
- Cask TK-S9 for transport of industrial consignments of fuel elements and fuel bundles.

Package TK-S8 is the modernized variant of package TK-S5. TK-S5 packages are present in most of the enterprise manufacturers of FBs and are used for transport of regular FBs of VVER-1000 with uranium fuel. The modernization consists in supplying a cask with neutron shield and devices for tightness testing. The volume of a package is equal to two fuel bundles or two canisters with fuel elements. The weight of a package is about 5.5 tons. The package TK-S8 can be applied at the first stage of development of production of fuel bundles with MOX fuel, because 3 to 4 packages are enough for the transport of a typical fuel bundle.

Package TK-S9 for mass production for the industrial use of MOX fuel in VVER-1000 reactors is a result of new development. Volume of the package is one FB or one canister with fuel elements. Weight of the package is about 2.6 tons.

Transportation of fuel elements in both types of packages is carried out in a technological canister. Structure, size, and design of the canister are analogous to that of the fuel bundle. Capacity of the canister is up to 180 fuel elements or rods.

One step in the VNIPIET development of casks was research on a choice of materials for neutron shielding of packages. Of the materials obtained, the most advanced ones from the standpoint of a combination of the required properties were compounded on a basis of silocsan polymer and special superhighmolecular polyethylene. Now VNIPIET, together with a number of enterprises from the chemical industry, is carrying out work to refine the properties of these materials.

Technologies of operation with both types of packages are similar to each other. Design of both packages presupposes remote loading and unloading of fuel bundles and the possibility of unloading them under water.

Unloading of fresh MOX fuel bundles in nuclear power plants (NPP) can be carried out by various methods:

- Unloading of fresh fuel to storage according to a program which is in force now. Here, a warehouse should be equipped with an inverting transport machine, “hot” chamber for testing of FB, and protective baskets for storage and transport of FBs inside the NPP. This variant is more acceptable if the NPP becomes oriented to use MOX fuel;
- Unloading FBs directly to a pool for storage of FBs (with immersion of a package into the pool);
- Unloading FBs without immersion of a package into a pool using a protective reloading device. The two last variants can be applied for loading of sample fuel bundles.

2.3. TRANSPORTATION OF BN-600 REACTOR MOX FUEL BUNDLES

Presently, there are two types of packages for transportation of sample BN-600 FBs with mixed fuel: TK-S2 and TK-S3.

Package TK-S2 is intended for transport of one sample FB with mixed fuel made from weapon or power plutonium with enrichment up to 30%. The package is equipped with a device for tightness testing and neutron shielding. In principle, transport of 20-40 pieces of FB per a year is possible in this package (under the condition that FBs are not stored in the NPP, since the design of the cask is allowed by Federal Supervision for nuclear and radiation safety of the Russian Federation only for transport). It means that a delivered consignment of fuel bundles should be immediately removed from the cask and reloaded to an NPP basket or directly to a receiver for fresh FBs. Manufacturing of new TK-S2 packages is not allowed by Federal Supervision. Upgrade of the design of the packages or development of a new cask is necessary for the transportation of merchant quantities of FB.

Package TK-S3 is intended for transport of one regular FB with uranium fuel. The package has no neutron shield or device for tightness testing. As agreed with Federal Supervision, the transport of individual sample FB with mixed fuel from weapon plutonium with enrichment no greater than 26% is allowed in this package. Transport of only several individual FBs (10–15 pieces per year), without storage in NPP, is possible. The fuel bundle should not be radioactively polluted, since there is no opportunity for control of tightness of a package. Upgrade of this design or development of a new package is necessary for transport of industrial scale consignments of FBs.

3. Transportation of Spent Fuel

3.1. TRANSPORTATION OF VVER-1000 REACTOR SPENT FUEL BUNDLES

There is no transport package for transportation of spent FB (SFB) made of MOX fuel in Russia. The performance estimates for the TK-13 transport cask, which is now in use for transport of VVER-1000 SFB from uranium fuel, showed that the levels of capacity of radiation dose on a surface of a package during loading of SFB from weapon plutonium and SFB from power plutonium exceeds approximately (depending on time of storage) by 4 and 15 times, respectively, the capacity level of radiation dose for a package loaded by SFB from uranium fuel. Reduction of the capacity on the surface of a TK-13 cask requires the following increase of storage duration for spent MOX fuel:

- In case of use of weapon plutonium—about 20 years;
- In case of use of power plutonium—about 50 years.

If the duration of storage accepted at the moment (3 years) is not changed, then the transport of 2 or 3 SFB with weapon plutonium or one SFB with power plutonium in TK-13 is possible [1].

This means that the use of the available TK-13 cask will result in necessity of significant expansion of NPP storehouses and require huge expenses for transportation. The more rational solution to the given problem is development and manufacturing of new transport packing sets similar to TK-13.

3.2. TRANSPORTATION OF BN-600 REACTOR SPENT FUEL BUNDLES

According to results of preliminary analysis, the transport of BN-600 spent MOX fuel FB is possible in available wagon-containers, TUK-11.

The transport of several sample fuel bundles is postulated in baskets of the 14U type, which are available in Belojarskaja NPP and are in use for uranium FB transports. For this, an analysis for determination of the transport conditions, quantity of the transported FB, and substantiation of transport safety is necessary. By order of the NPP, such a work is now planned. For transport of regular MOX FBs, development of a new basket with an increased volume is expedient.

Since a supply of 7 available TUK-11 packages will expire in 2015, to ensure transport of the BN-600 reactor MOX SFB, development of a new transport packing set is expedient. In perspective, it would be used for transportation of SFB from BN-800 reactors as well.

4. Conclusions

At present, only preliminary technological, economic, and design studies of variants for transportation of fresh MOX fuel have been performed, whilst problems of transportation of spent fuel have still almost not been considered.

The safety of transport of MOX fuel requires careful analysis and substantiation. A use of available transport packages for MOX fuel transport will require essential upgrades and the development of new packages. This work would require considerable financial expense and a long time. Therefore, it should be started right now, so that there would not be a delay in deliveries of sample or regular fuel bundles to nuclear power plants. Besides, variants of the transport-technological methods of operation with fresh and spent MOX fuel in NPPs with reactor types VVER-1000 and BN-600 should be worked out.

References

1. Afanassiev, A.V., Ivanuk, A.I., Kulilov, V.I., and Tokarenko, A.I. (1997) "Experience of development of packagings for MOX-fuel transportation," Report at a Russian-Japanese seminar on problems of MOX-fuel, Moscow.
2. "Main regulations of safety and physical protection of nuclear fissile materials transport (OPBZ-83) (1984), Atominform, Moscow.
3. "Regulations for the safe transport of radioactive material," (1991), SS6 IAEA, Vienna.

SAFETY PROBLEMS FOR LONG-TERM UNDERGROUND STORAGE AND FINAL DISPOSAL OF NUCLEAR MATERIALS

T. A. GUPALO
V. P. BEYGUL
R. T. ISLAMOV
All-Russian Design and Research Institute of Production Engineering of MINATOM of Russia (VNIPIPT)
Kashirskoje shosse,33, Moscow, 115409, Russia

1. Introduction

The issues of long-term storage and final disposal of radioactive materials are important and inalienable components of the complex problem of nuclear material management; they are closely associated with ecological compatibility assurance for the whole process and nuclear weapons non-proliferation. Figure 1 shows a fragment of scheme for nuclear material management indicating the location of storage facility for spent nuclear fuel (SNF) from reactors of various purposes, ex-weapons plutonium and enriched uranium from dismantled nuclear warheads, as well as repositories for high-level wastes (HLW) and SNF in geological formations.

Reliability of the underground isolation of radioactive materials is ensured by a multi-barrier system that incorporates a complex of engineered barriers and mountain rock mass, their roles for long-term storage and final disposal of radioactive materials differing considerably.

For storage facilities with a specified term of radioactive materials storage, the major role belongs to engineered barriers: metallic containers, immobilizing matrix materials, isolating bentonite filling, etc. The mountain mass provides a reserve of isolation reliability in case of non-standard (emergency) situations of very low probability accompanied by releases of potentially hazardous amounts of radioactive isotopes dissolved in underground water, as well as a certain period for the liquidation of consequences without detriment for environment and population.

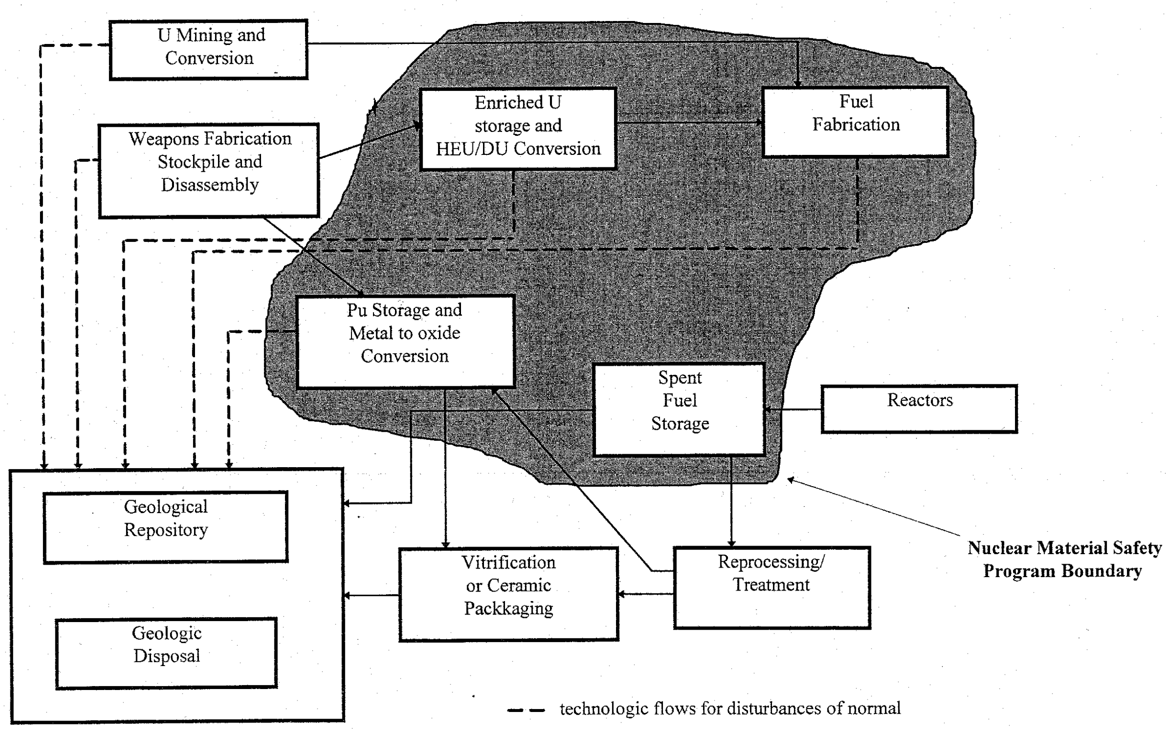


FIGURE 1. “Simplified” nuclear materials flowsheet to achieve the “spent fuel standard.”

For repositories of solidified HLW and SNF containing long-lived radioisotopes, the period of remaining potential ecological hazard can last several hundred thousand years. Engineered barriers play an auxiliary role, contributing to a temporary localization of isotopes being disposed of in the repository itself and its adjacent area. The most important are processes such as rock destruction, hydrogeological characteristics of the geological plot and its sorption properties as related to the long-lived isotopes contained in the HLW and SNF to be disposed of. For final disposal of wastes containing long-lived radionuclides, hard rock masses with low fracturing are the most suitable. According to the results of drilling deep and super-deep boreholes in various regions, stable properties of such massifs are preserved for periods long enough in the geological time scale.

As a criterion of safety of multi-barrier systems for long-term storage and final disposal of solidified radioactive materials, a probabilistic characteristic *r*, risk of release, is proposed for the most important of long-lived isotopes contained in radioactive materials (Fig. 2). The activity meant is that which exceeds its admissible level in water for group b population (AL_b) to the active flow exchange area. For most cases under consideration, it is plutonium that appears the most hazardous, critical isotope; therefore, to make the discussion more concentrated, safety issues will be considered here as applicable to plutonium.

With the proposed criterion applied when creating repositories for radioactive materials, the risk of death caused by the plutonium uptake by humans is actually excluded. This is due to the fact that areas are chosen for repositories located far enough from active flow exchange areas with low water inflows along the enhanced fracturing zones.

Optimum design solutions in the creation of multi-barrier systems for long-term storage and final disposal of radioactive materials should be chosen based on probabilistic safety analysis for each viable option, supplementing the latter with the cost estimates for the implementation of the option, i.e., using the complex criterion "risk-benefit."

The scheme for calculating *r* for plutonium for a storage facility for radioactive materials is shown in Figure 2. The probability calculation for possible non-standard situation occurrence and development is the basis for the evaluation of *r*.

In this case, the structure and characteristics of specific host rock mass, underground structure configuration and design, and strength properties of the supports are analyzed; mathematical modeling of stress-strained state of the massif contour zone is carried out, and strength characteristics are calculated for the whole profile of the workings' cross-section. This work includes also the accounting for dynamic impacts of possible intensity levels, analysis of measurements results obtained under natural conditions, and summarizing the archive and literature data. Figure 3 shows the scheme of probability calculation for the occurrence of major non-standard situations for underground storage facilities of solidified radioactive materials.

Safety of repositories for solidified radioactive materials is assessed with the assumption that for a period of several hundreds years, all potential non-standard situations will inevitably occur, and engineered barriers will be destroyed.

$$r(l = \mathbf{L}; \mathbf{T}_{xp}\ \mathbf{X}; \mathbf{y}; \mathbf{Z}) \leq \mathbf{E}$$

$r = \mathbf{P}_{\Sigma} / \mathbf{T}_{xp}$	***r* =** risk of plutonium release into the active flow exchange with admissible activity levels for group b population (AL_b) 1/year
$\mathbf{P}_{\Sigma} = \mathbf{P}_0 + \Sigma \mathbf{P}_i$	**L** = maximum admissible length of possible plutonium release in rock mass towards active flow exchange area, m
$\mathbf{P}_0 = \mathbf{p}_{0,0} \cdot \mathbf{p}_{0,1} \cdot \ldots\ldots \cdot \mathbf{p}_{0,j} \ldots$	$\mathbf{P}_0, \mathbf{P}_i, \mathbf{P}_{\Sigma}$ = probabilities of plutonium release; for normal conditions, for the i-th non-standard situation, total probability
$\mathbf{P}_i = \mathbf{p}_{i,0} \cdot \mathbf{p}_{i,1} \cdot \ldots\ldots \cdot \mathbf{p}_{i,j} \ldots$	$\mathbf{P}_{i,0}$ = probability of i-th non-standard situation
$\mathbf{P}_{i,j} = \int_{i,j} (\mathbf{X}, \mathbf{Y}, \mathbf{Z})$	$\mathbf{P}_{i,j}$ = probability of the j-row event in an accident of the i-th non-standard situation
	X,Y,Z = set of characteristics of radioactive waste, engineering barriers, rook mass
	$\mathbf{T}_{xp}$ = operation lifetime of storage facility, years
	E = maximum admissible risk value, 1/year (= 1×10^{-6} 1/year)

FIGURE 2. Criterion and principal relationships for safety assessment of radioactive materials.

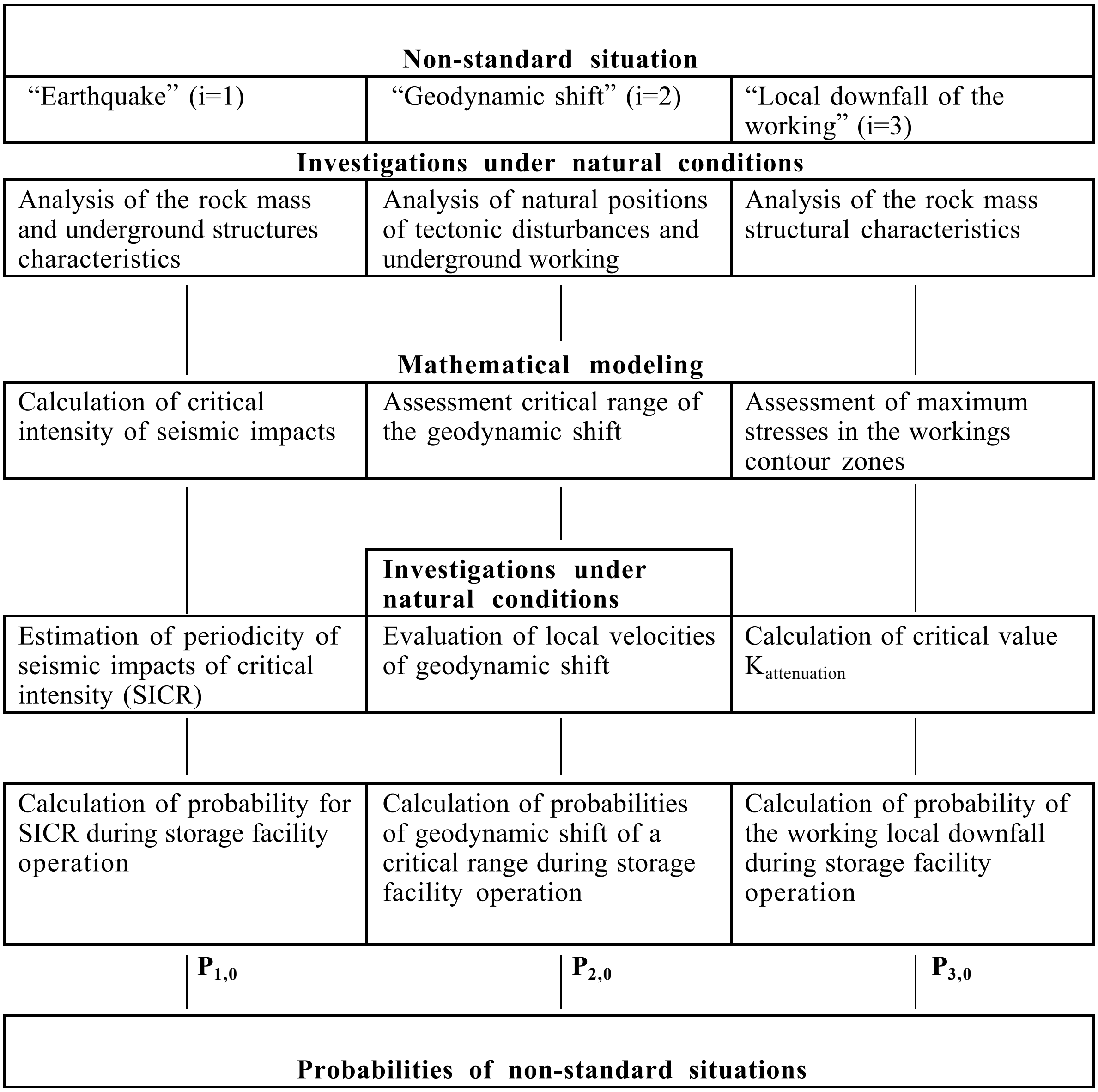


FIGURE 3. Scheme for calculating probability of non-standard situation occurrence for storage (disposal) facilities for solidified radioactive materials.

The scheme for safety assessment of repository of solidified radioactive materials is shown in Figure 4.

$$r(l = L; N_0\ T_{1/2}; R; V; Z) \leq E$$

where

L = minimum distance from the top boundary of the repository emplacement zone to the active flow exchange area along the tectonic disturbance zones, m

$\mathbf{N_0}$ = distribution of probability density of initial Pu activity in contaminated water at the outlet from the repository, fractions of $\mathbf{AL_b}$

$\mathbf{T_{1/2}}$ = Pu half-life, years

R = distribution of probability density of plutonium retention factor on rock and destruction materials of engineering barriers

V = distribution of probability density of underground water flow velocities in the tectonic disturbance zones, m/year

Z = set of rock mass characteristics taking into account the effect of the radioactive materials emplaced

E = maximum admissible value of risk, 1/year ($=1.10^{-6}$ 1/year)

FIGURE 4. Criterion for safety assessment of repositories for radioactive materials.

2. Conclusions

1. At any stage of nuclear material management (closure of plutonium facilities, vitrification processes, fabrication of MOX fuel, metallic plutonium conversion into oxide, etc.), the long-lived radionuclides appearing in case of any deviation from the standard process can be reliably isolated from the environment only by their final disposal in geologic formations.
2. For public acceptance of any projects within the framework of the “Nuclear Material Safety” program, the proof of non-proliferation and safety due to the use of underground storage facilities and final disposal of nuclear materials should be presented for each variant of violation from standard operation mode, based on the principle of “ecological risk-benefit.”

SAFETY ISSUES OF RUSSIAN EP-500 CERAMIC MELTER AND THE FEASIBILITY OF ITS USAGE TO VITRIFY PU-CONTAINING MATERIALS

G. B. BORISOV
A. S. POLYAKOV
O.A. MANSOUROV
State Research Center of Russian Federation A. A. Bochvar
All Russian Research Institute of Inorganic Materials (VNIINM), Moscow, Russia

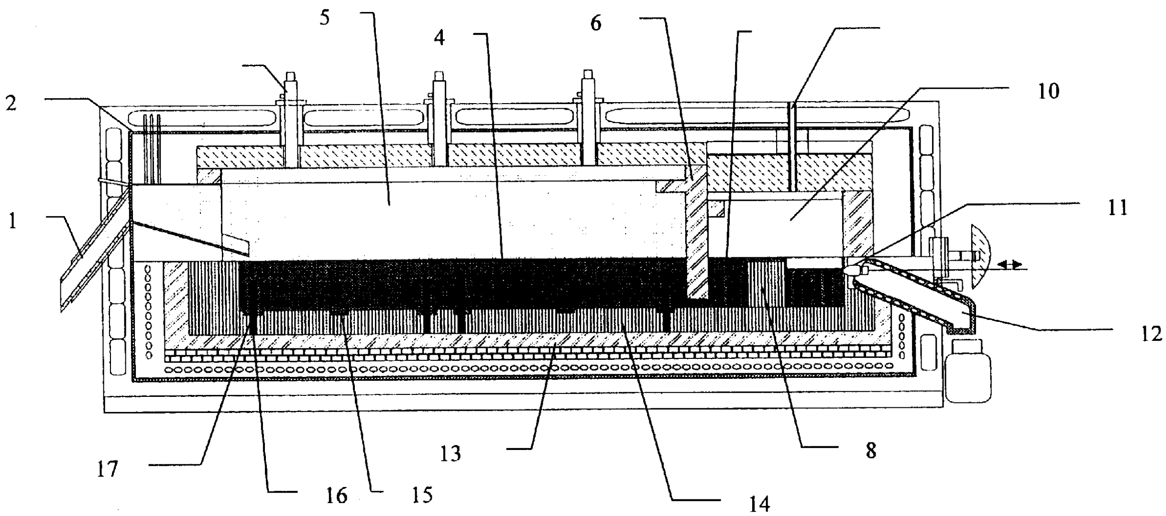
1. Summary

To resolve a problem of radioactive waste treatment in Russia, a complex for the vitrification of high-level wastes (HLW) in a ceramic melter, EP-500, was designed, constructed, and put into use during 1986-97 at the power unit, Mayak. This paper gives a short description of the facility and identifies worst-case and emergency situations that result from final failures of the melters, as well as ways and methods of increasing safety and reliability of the vitrification process that will be realized in the advanced melters, EP-500/3,4. By the way, from the standpoint of nuclear safety, we assess the applicability of the ceramic melter for immobilizing Pu-containing materials. We propose to use melters with SnO_2 electrodes, such as the "Vertical Super Melter" (VSM), which has a number of advantages in the field of nuclear safety to attain the same objectives.

2. Design Features of the Ceramic Melter EP-500

During the period 1986-97, a complex for HLW vitrification on the basis of two EP-500 ceramic melters were operated at the Mayak power unit (Fig. 1). For six years exploitation of a ceramic melter EP-500/1p, under conditions of remote operation, about 12,500 m^3 liquid HLW were transferred into glass-type matrices based on phosphate glass (Table 1). The total activity of radionuclides incorporated in glass is 285 million Ci on β-emitters and 2.5 million Ci on α-emitters [1]. Taking into account such parameters as great productivity (500 l/h) and prolonged operation life, without replacement of any units, the Russian melter does not have any analogues in the world.

The base of the melter is an electrically heated furnace, which is a sort of rectangular basin lined with bars made of cast refractory material of "Bakor-33" type and placed into a metallic water-cooled shell. The melter basin is divided into three areas: melting, flowing out, and collecting. At the bottom of the melter in each area, rod molybdenum electrodes are built onto a water-cooled current lead terminal. In the ceiling of the melting area, there are three water-cooled feeders, through which liquid HLW along with a glass-former (orthophosphate acid) and a depressant of ^{106}Ru volatility (ethylene glycol) are fed directly onto the surface of molten glass (Table 2).



LEGEND: (1) off-gas system; (2) metallic shell; (3) feeder; (4) molten glass; (5) melting area; (6) partition; (7) flowing out area; (8) threshold; (9) sensor of glass level; (10) collecting area; (11) pouring out unit; (12) pouring spout; (13) Shamot refractory; (14) Bakor refractory; (15) pit; (16) current supply; (17) electrode.

FIGURE 1. Overview of the Russian ceramic melter, EP-500/1p.

TABLE 1. Basic results of operation of the melter EP-500/1p.

Description	**Parameters**
Productivity	up to 500 l/h
Specific activity of reprocessed wastes	20-50 Ci/l
Volume of reprocessed HLW	12500 m^3
Amount of obtained phosphate glass	2200 t
Total activity of radionuclides incorporated in glass –α-nuclides –β-nuclides	 2.5 million Ci 285 million Ci
Number of containers with glass –cans -canisters	 4760 1587
Off-gas system	10^{-12}-10^{-15} Ci/l
Operation life	6 years

TABLE 2. Composition of industrial wastes fed into the melter EP-500/1p.

Elements including in initial solution composition	Range of concentrations, g/l	Range of evaluated compositions of glasses, % wt
Sodium	22.50-48.30	Na_2O - 21.40-26.80
Aluminum	12.50-20.00	Al_2O_3 - 12.50-19.80
Phosphor	29.00-63.70	P_2O_5 - 49.80-56.60
Iron	0.50-4.90	Fe_2O_3 - 0.51-2.69
Nickel	0.20-2.50	NiO - 0.18-2.00
Chrome	0.10-1.00	Cr_2O_3 - 0.05-0.80
Calcium	0.10-3.00	CaO - 0.07-3.00
Manganese	0.03-3.00	Mn_2O_3 - 0.02-2.10
Molybdenum	0.005-0.60	MoO_3 - 0.03-0.46
Lanthanum	0.03-2.40	∑РЗэ- 0.04-0.54
Cerium	0.03-0.30	
Neodymium	0.03-1.00	
Samarium	0.03-0.30	
Ruthenium	0.03-0.10	∑ platinoids -
Rhodium	0.01-0.10	0.04-0.54
Palladium	0.01-0.10	
Cesium	0.12-0.50	Cs_2O - 0.06-1.00
Uranium	0.71-9.10	UO_3 - 0.41-5.77
Plutonium	0.005-0.03	PuO_2 - 0.003-0.04
SO_4^{2-}	0.110-1.40	SO_4^{2-} - 0.05-0.61
Cl^-	0.08-1.20	Cl^- - 0.08-0.17
F^-	0.01-0.10	F^- - 0.05-0.07
HNO_3	35.00-129.00	
Voluminous ratio between wastes from regeneration of Transport - Power Facilities and VVER-440	5.5-11.5	

As molten glass builds up, it overflows from the melting area to the collecting area, from which it is regularly delivered through a pouring out unit to cans having volume of 200 liters each and installed on a step-by-step conveyer. In accordance with the existing transport, technological flow-sheet cans with glass, which being cooled, are sorted in canisters and subsequently transferred to a temporary repository. Figure 2 gives an overall scheme of the melter EP-500.

To identify the main factors of safe work of the advanced melters EP-500/3,4, six years of experience using the melter EP-500/1p are summarized; efficiency, technological deviations, and equipment failures, as well as violations of water-cooled system, off-gas system, control and measuring instruments that resulted in the temporary shut down of the vitrification process are studied.

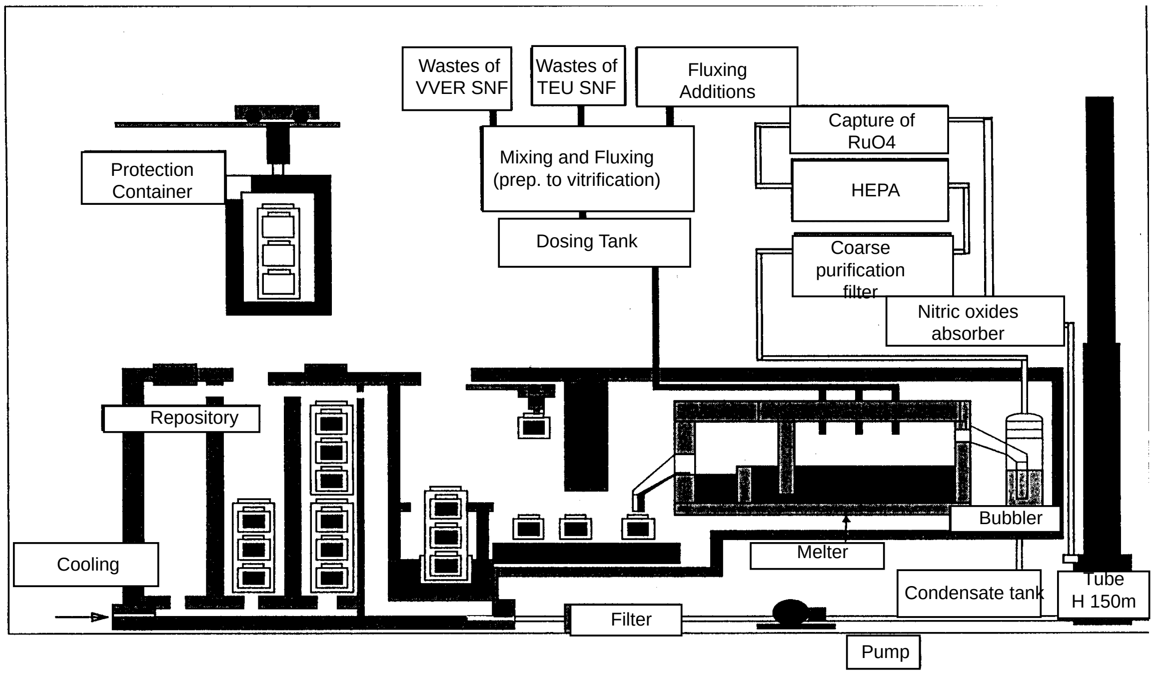


FIGURE 2. Flowsheet of HLW vitrification by a commercial melter, EP-500.

Emergency situations are analyzed that brought the final stop of the melter EP-500/1p. On the ground of reasons for deviations that occurred during the vitrification process and stand-by of equipment, a list of emergency situations and measures of prevention are compiled.

It follows from the analysis that the main factors limiting operational life of the melter and that forced the stand-by or eventual shut-down of equipment are the following:

- High temperature of glass melting and pouring (1200°C) in the first year of exploitation rsulting in destruction of one of two spouts of the pouring out unit;
- Dropping of glass onto elements of the step-by-step conveyer as a result of spraying of molten glass by overflows of cans and its overturning;
- Depressurization of water-cooled elements of a feeder that come in contact with molten glass or placed inside of the melter. Steel tubes of cooling of the shell and pouring out unit, current supplies of electrodes are of such elements. Depressurization was caused by formation of open microcracks in tubes and current supplies due to long-term operation under the conditions of thermal-cycle loadings and accompanied by oozing water through the microcracks and accumulation of it in the melter and a canyon. This one entailed radioactive contamination of a working room adjacent to the canyon after 5 years of use.

"Burn-up" of electrodes current supplies and elements of the pouring out unit were the main reasons for decommissioning two EP-500 ceramic melters as a result of the continued effect of great thermal and thermal-cycle loadings. The "burn-up" was accompanied by leaking and evaporating a considerable amount of water from cooling elements and eventual filling of tubes by molten glass.

The first melter EP-500 was halted in 1988 after the "burn-up" of electrodes current supplies of the melting area and leaks of radioactive glass from the melter to a working room. The second melter was decommissioned or halted as a result of the "burn-up" of water-cooled elements of the pouring-out unit. Thereto, a great amount of high-level glass was poured from the collecting area onto a vast zone of the step-by-step conveyer via the pouring spout. Otherwise, this accident did not lead to a change in the radiation situation in the working rooms.

3. Ways of Increasing Safety and Reliability of HLW Vitrification Process in the EP-500 Ceramic Melters

Two advanced ceramic melters, EP-500/3 and EP-500/4, presently are under construction. The first facility is scheduled to be put into operation in March, 1999.

To advance the safety and reliability of the HLW vitrification process in the new melters, five items are planned. The first four proposals will be realized in the advanced melters EP-500/3,4.

1. Installation of a turnable conveyer in lieu of the step-by-step conveyer.
2. Reduction of temperatures of glass melting and pouring on the account of boron additives specially included into an initial solution. This will allow a decrease in the melting temperature up to 850°C and thereby lessen corrosion of refractory, electrodes, water-cooled elements, and sensors of control and measuring instruments.
3. Development of a method of visual surveillance of conditions of refractory, electrodes, and other units of the melter during its operation and after decommissioning.
4. Development of noncontact devices on the basis of vibro-acoustic and acoustic-emission sensors to determine sizes of microcracks, local failures of cooling elements.
5. Development of systems and apparatuses for remote dismantling and disposing of the ceramic melters in order to avoid HLW surface accumulation.

It is necessary to note that removal of the EP-500 ceramic melter is the most important and unsolved problem of its exploitation after the end of its operational life. For instance, two decommissioned ceramic melters filled by highly radioactive glass with a total activity of ~ 2 million Ci presently are stored in a vitrification complex at the Mayak power unit. Therefore, there is an acute need to develop a system of remotely dismantling and disposing of the melter elements in standard containers for storage in a repository of solidified wastes.

4. Feasibility of Using EP-500 Ceramic Melters to Immobilize Pu-Containing Materials

On the basis of experience using the ceramic melters and present developments in this field, the present feasibility of Russian Pu-containing materials is assessed. Rich Pu wastes from the radiochemical and chemical-metallurgical production of plutonium are included in this category of wastes; for example, hard-to-dissolve residues after treatment by a mixture of nitric and fluoride acids. These wastes have a complicated chemical composition (Table 3).

To immobilize such waste types, a method of vitrification using the ceramic melters is proposed. Pu content in HLW fed to EP-500/1p did not exceed 0.03 g/l. Such a concentration in primary solution is 10 times less than the Pu concentration compared to its solubility of 0.1-0.2% in phosphate and borosilicate glasses, according to experimental data.

TABLE 3. Composition of hard-to-dissolve residues.

Type of residues	Content, %
Plutonium	1–13
Iron	2–15
Silica	4–40
Refractory oxides	1–70
Ca, Mg, La, Cu, Co, V, Ga	0.5-3.8

During the reprocessing of wastes with Pu content up to 0.03 g/l for 6 years of melter operation, about 600 kg of Pu was incorporated into glass and subsequently sent to storage. If we were to increase the concentration of plutonium in HLW up to the level of its solubility in phosphate glass (0.1–0.2%), than about 2.0-4.0 t of Pu could be immobilized into glass during 6 years of operating the melter. In this case, nuclear safety is provided by restricting the concentration of plutonium in the initial solution below its solubility in phosphate glass. Therefore, a considerable amount of plutonium can be incorporated into glass due to prolonged operation life and high productivity.

However, analysis of design specifications of the ceramic melter EP-500 and the vitrification process points out that the usage of the melter to immobilize Pu with concentrations above its solubility in phosphate glass is moot. First, plutonium oxides are apt to form precipitated agglomerates in molten glass. Their precipitation can bring to formation of Pu-containing sediments at the bottom of the ceramic melter. For glasses with Pu up to 0.3% treated for 200 h at 1000°C, the plutonium content in the bottom layer is 15 times higher than that in the top layer [2]. Second, the melter has a nuclear-dangerous shape, including pits and partitions where Pu-containing residues can be accumulated.

Contact of water or its vapors with the bottom sediments can occur in the case of "burn-up" of water-cooled elements of the current supply located in the proximity of molten glass and invasion of water inside the melter. This case can result in emergency circumstances. Ceramic melters without water cooling elements are being evaluated in Russia in order to immobilize wastes with concentrations of Pu above the limit of its solubility in glasses. For instance, a melter based on an electrical furnace with electrodes from glass-resistant and electrical conducting SnO_2 ceramics, which successfully passed the 0.5-year tests by vitrification of model wastes obtaining borosilicate glass (Fig. 3). It demonstrates the following basic advantages:

- Designed with no water-cooled elements in contact with molten glass;
- Possibility of melting glass at 1350–1400°C.

A vertical super melter, well-known for its high productivity and compactness, is presented in Figure 4. This apparatus is made of a vertical shaft with two tiers of symmetrically located SnO_2 electrodes and two pouring out units. Each tier of electrodes has a separate current supply [3]. The melter does not have water-cooled units. Due to the two-tier electrode location, it is possible to:

- Intensify molten glass agitation by supplying a different power level to each electrode;
- Supply the major power under the sole area of the melter to dissolve plutonium sediments.

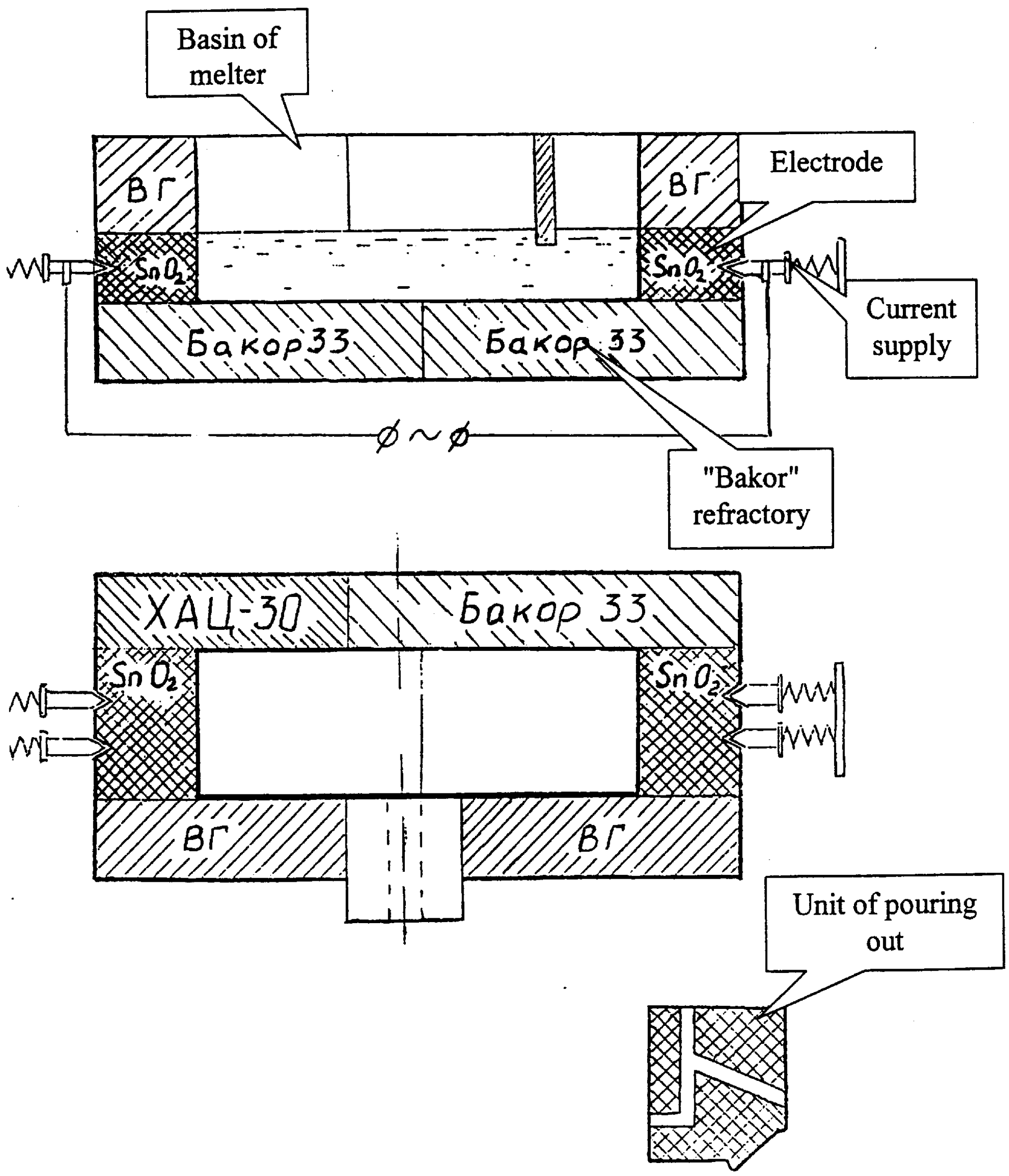


FIGURE 3. Pilot ceramic melter with SnO_2 electrodes for Pu-containing materials vitrification.

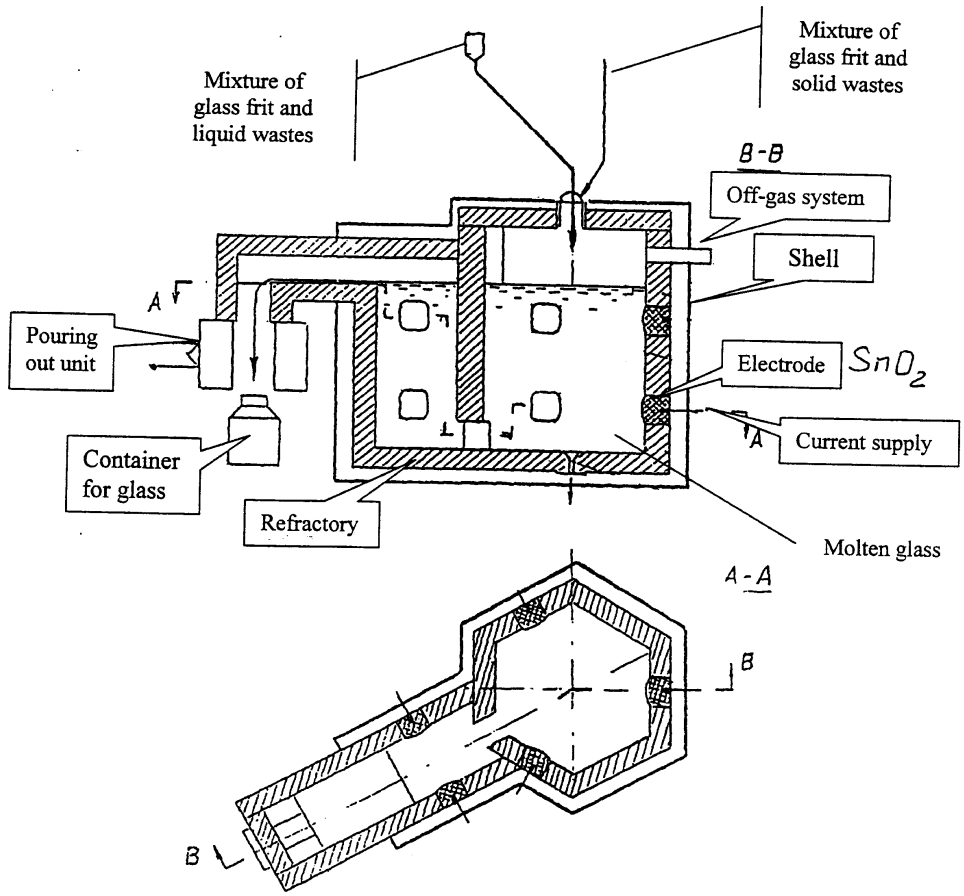


FIGURE 4. Design of the vertical ceramic melter for Pu-containing materials vitrification.

5. Conclusions

From the viewpoint of nuclear safety, the dubiousness of the use of the ceramic melters EP-500 to immobilize Pu-containing materials with high content (above the solubility in glasses) is depicted.

The use of the melter without water-cooled design units (for instance, VSM) to immobilize Pu-containing materials with high concentrations of Pu is proposed.

To increase environmental safety and reliability of the HLW vitrification process in the EP-500 melters in use in Russia, it is necessary to solve the problem of dismantling and disposing of decommissioned melters containing solid radioactive glass. On the basis of present developments in Germany, Belgium, etc., and vast experience, there is a need to develop systems and apparatus for remotely removing elements and parts of the melters.

References

1. Polyakov A.S., Borisov G.B., Moiseenko N.I., Dzekun E.G., et al., "Experience in Applying an EP-500/1p Ceramic Melter to the Vitrification of Liquid High-Level Wastes," *Atomic Energy*, **76** (3) pp. 183-188, 1994.
2. Borisov G.B., Polyakov A.S., Dzekun E.G., Gorbunov, V.F., "Scientific and Technological Aspects of Weapon Grade Plutonium Transformation into Glass-like Materials Fit for Long-term and Environmentally Safe Storage," *Proceedings of the 5th International Conference on High Level Wastes Management*. Berlin, September 12-16, 1995.
3. Kushnikov V.V., Matyunin Yu.I., Krylova N.V., "On the behavior of Alpha Radionuclides upon Solidification of High Level Liquid Wastes," *Atomic Energy*, **70** (4) pp. 239-243, 1991.

FRENCH VITRIFICATION PROCESS SAFETY ISSUES

PIERRE HUBERT
Cogema
2, rue Paul Dautier BP 4, 78141 Vélizy Cedex, France

1. Historical Steps

In France, laboratory research on containment matrices began in 1957, first with crystalline materials, and then with glass—which quickly proved to be more suitable for incorporating the forty-odd elements created by uranium fission (as well as additives and corrosion products resulting from fuel reprocessing) into a homogeneous matrix.

After the fabrication of highly radioactive glass blocks weighing a few hundred grams to assess their containment properties, the French research program then turned to process development. The first industrial vitrification facility, PIVER, began operating in 1969. Before it was shut down in 1972, PIVER produced 164 glass blocks, weighing a total of 12 tons, from 24 m^3 of concentrated fission product solutions containing 6×10^6 Ci. The facility resumed operation a few years later to vitrify high level waste (HLW) solutions arising from the reprocessing of fast breeder reactor fuel, producing 10 glass blocks of 90 kg each with very high specific activity. In 1989, PIVER was named a Nuclear Historic Landmark by the American Nuclear Society.

Faced with increasing demand, research was undertaken in the 1970s to develop a continuous process to obtain a final glass waste form by first evaporating and calcinating the feed solution in a rotary furnace, then melting the calcinate with glass frit in an induction-heated metal melter. The Marcoule Vitrification Facility (AVM) was commissioned in 1978 to vitrify fission product solutions from the French UP1 reprocessing plant.

The successful operating record and experience gained with AVM allowed the start-up of a commercial-scale, high level waste vitrification plant in France. These similar facilities, R7 and T7, are on line at the La Hague reprocessing plant. R7 was commissioned in 1989 and T7 in 1992.

2. Glass Acceptance Strategy

To demonstrate that the glass product is suitable for final disposal, a waste acceptance process was set up with the following three goals:

1. Develop and characterize a reference glass formulation and variations thereof;
2. Demonstrate that process variations can be detected and corrected without affecting glass quality;

3. Control the quality of glass by controlling process parameters in normal and abnormal situations.

The successful completion of the first two steps in the waste acceptance process resulted in the establishment of glass specifications accepted by the French nuclear safety authority (DSIN). Subsequently, the R7/T7 glass specifications were also approved by the regulatory authorities of Cogema's baseload customers in Germany, Belgium, Netherlands, Switzerland, and Japan.

The quality management system implemented by the Cogema Group at La Hague guarantees the quality of the glass product, thus completing the third stage of the acceptance program. The quality management system is in compliance with international standard ISO 9002. It comprises a Quality Assurance Program and a Quality Control Program that demonstrate the quality of the glass product as well as mastery of the conditioning process.

2.1 GLASS CHARACTERIZATION PROGRAM

Long-term studies performed by the CEA (French Atomic Energy Commission) resulted in the formulation of the so-called R7 and T7 glasses. Characterization studies have verified the main properties of the final products and have established the glass specifications. The vitrified waste specifications were subjected to peer review by an independent commission of nuclear specialists.

The principal criteria for the formulations were: mechanical and thermal stability, homogeneity, radiation resistance, high containment capacity, low volume, corrosion resistance, low leachability, easy fabrication (mastered technology), and flexibility with regard to the composition of waste to be conditioned.

Characterization testing was conducted in both inactive and active conditions to determine the principal properties of the reference glass composition. Inactive characterization testing focused on the physical, thermal, and mechanical properties of the glass, on its homogeneity, on the thermal stability of the glass in a temperature range of 500 to 1,200°C, on its leach resistance, and on glass volatility, i.e., weight loss as a function of temperature.

Active characterization testing was conducted on hundreds of active glass formulations using alpha doped and beta tracer glasses to determine radiation resistance, leach rates, and the thermal stability and volatility of the glass. In addition, tests were performed to assess glass quality sensitivity to variations in process parameters and to qualify a broad range of acceptable glass/waste compositions. A total of 90 glasses were characterized in this manner.

In parallel with glass sensitivity studies, tests on inactive full-scale prototypes were conducted to determine the sensitivity of the melter to variations in process parameters, such as different melting temperatures, variations in time periods between glass pours, and variations in glass-to-frit ratios or frit-to-calcine ratios.

A range of acceptable glass compositions was defined by the results of the sensitivity tests, and failure modes and effects analyses were performed to identify fault conditions that would impact glass quality, including its chemical composition, homogeneity, cracking rate, and propensity to crystallize.

2.2 R7/T7 GLASS SPECIFICATION

The objective of the glass characterization program previously mentioned was to provide a reference glass composition with a range of variation to provide for operational constraints and for actual fuel to be reprocessed. The glass characterization program resulted in identification of an optimum glass composition for HLW from oxide fuel (LWR-type reactor), along with an operating range of process parameters for the AVM-type technology.

This was formalized in the "Specifications of Vitrified Residues produced from reprocessing at UP2/UP3 La Hague plants." The specifications include "Guaranteed Parameters," i.e., those identified as key parameters in the process to ensure the glass canister's quality.

These specifications were subject to peer review by an independent commission of scientists and nuclear experts. They were also provided to ANDRA (French National Radioactive Waste Management Agency) for comment, and then submitted to the French regulatory authority, DSIN. The French safety authority approved the vitrified residue specification in February 1986.

The guaranteed or key parameters in Table 1 are listed with the data that refer to the chemical composition of the glass. Compliance with these guaranteed parameters, which the reprocessor is bound to respect, allows the declaration of the conformity of each glass canister. The Cogema Group also specifies the canister fill rate, the welding procedure, and the glass cooling procedure.

TABLE 1. Guaranteed parameters in vitrified residues from reprocessing.

Guaranteed parameter	**Chemical composition**
Maximum β/γ - activity:	
Cs-137	$<6.6 \times 10^{15}$ Bq/can. (180,000 Ci)
Sr-90	4.6×10^{15} Bq/can. (125,000 Ci)
Maximum actinide content	
uranium	< 4,500 g/can.
plutonium	< 110 g/can.
curium 244	< 90 g/can.
Non-fixed β/γ surface contamination	$< 3.7 \times 10^{4}$ Bq/m^2
Heat load (at time of transportation)	< 2 kW

2.3 INDUSTRIAL PROCESS

Results accumulated so far at La Hague show that HLW vitrification in France fully meets the criteria of a successful industrial process. Moreover, the resulting glass meets all required quality standards. Each facility, R7 and T7, is made up of:

- Receipt and adjustment units (13 tanks allow the receipt of active solutions);
- Liquid waste feeding, calcination and vitrification units: three lines, each implemented in individual cells, including two metering wheels (to feed adjusted fission product solutions and fines suspensions), a calciner to evaporate and to

calcine these solutions, a melting pot heated by induction in which the mixture of calcinate and inactive glass frit are melted and then the glass is formed. The La Hague facilities operate three vitrification lines. Each line is designed to produce 25 kg of glass/h, corresponding to one canister filled every 20-24 h. Currently each canister corresponds to more than 1.8 t of uranium reprocessed. Therefore, production is fitted to fuel characteristics (e.g., initial burn-up).

- Conditioning units: the filled canisters (Table 2) are cooled by air flow convection, welded (welding of the canister's lid), decontaminated by high-pressure water, and controlled by smear test before being stored in an interim storage facility.
- Off-gas treatment units: the off-gas produced on each line is treated in a dust scrubber, condenser, NOx absorber, safety column, then filtered through three stages of filters, and released through a 100-m-high stack. All the solutions issued from off-gas treatment are recycled in the HLW process to be vitrified.

Both R7 and T7 facilities are designed to produce 600 canisters a year, corresponding to 800 t of uranium reprocessed with reference solutions. It is worth emphasizing that these design figures correspond to the range of fuels that can be reprocessed, characterized from two reference fuels with, respectively, (1) burn-up of 33,000 MWd/t with 3.5% of U-235 (3-y cooling period) and (2) 45,000 MWd/t with 3.7% of U-235 (4-y cooling period). Ultimate wastes undergo vitrification after the cooling period. The process has proved to be highly flexible since fine particles from the dissolution step and alkaline effluents from the solvent regeneration steps have been incorporated in the glass matrix.

TABLE *2.* Main characteristics of the empty steel container.

Material (container and lid)	stainless steel
Height (with lid)	1,340 mm
Outside diameter	430 mm
Wall thickness	5 mm
Weight of empty container	90 kg

Through the years, various improvements have been achieved in both the process and the technology:

- Regarding maintenance procedures, various improvements have been achieved (for example, fewer maintenance operations are now needed due to a higher degree of automation in hot cells).
- Regarding active liquid waste, the main improvements are related to canister surface contamination. Due to a very efficient containment of gas generated by the pouring operation (i.e., by adding new perfectly designed devices), the volume and the radioactivity level of the canisters' surface contamination effluents are very low. Moreover, liquid effluents from the process off-gas treatments are totally recycled.

As a general consequence, radiological exposure of operating teams has been reduced.

2.4 OBTAINING VITRIFIED RESIDUE QUALITY

Cogema Group has implemented a quality management (or QA/QC) system for the final products (U, Pu) as well as for residues, the glass canister being one of them. This quality management system is in accordance with international standard ISO 9002. According to this standard, Cogema Group is able to demonstrate the quality of each residue as well as the mastery of the conditioning process. A Quality Assurance Program defined for the production of glass canisters encompasses:

- An organization of production facilities;
- A general organization of quality;
- A quality documentation;
- A quality control program;
- A treatment of deviations;
- A customer surveillance.

The quality control program includes three type of actions that pertain to the control of:

- Raw materials specifications compliance;
- Process functions affecting vitrified residue quality;
- The Quality Assurance Program.

2.4.1 Control of raw material specifications compliance

All raw materials are subject to the QA/QC system prior to acceptance. Raw materials are controlled for their compliance with procurement specifications. A conformity file is systematically established that documents the quality of glass frit and chemical reagents, the glass canister procurement, the smear test swab, and welding torches. In addition, Cogema is performing audits and inspections of the raw materials suppliers.

2.4.2 Control of process functions affecting the quality of vitrified residues

The basic glass properties required for vitrified residue depend on containment capacity, radiation resistance, vitreous state stability, and non-fixed surface contamination of the canister.

Containment capacity is a function of alterability and leachability that are linked to the glass chemical composition which should comply with the specifications previously defined. Glass chemical composition will be controlled as an essential process function.

Radiation resistance depends on the vitreous state of the product whose quality is assessed from the viscosity and homogeneity of the glass. A good vitreous state is achieved if the melting temperature is correct and the pouring rate satisfactory. Glass pouring will be followed as a second process function.

Vitreous state stability will be guaranteed if the temperature conditions supported by the glass after pouring are satisfactory. The process function glass cooling will be controlled.

The main properties affecting the safety of vitrified residue intermediate storage are the non-fixed surface contamination of the canister and canister tightness. These fourth and fifth process functions will be subjected to control.

With regard to the basic properties, the five process functions represent effective means to demonstrate the product quality on a real-time basis. Process control is performed both directly, by monitoring and measuring operating parameters at various stages in the process (flow rate, temperature, density, weight, welding parameters, canister transfer to storage), and indirectly, by corroborating operating parameters through analyses or comparisons of inlet and outlet materials balances.

2.4.3 Control of the quality assurance program implementation

An independent Quality Control structure is in charge of regularly inspecting the facilities. The frequency of inspection, their contents, and the inspection report format are defined in the Process Inspection Plan. This document is a technical support for the use of the Quality Control structure. It ensures the methodical verification of the parameters listed in the quality control program and of the process equipment related to residue quality. All process equipment is linked to official measuring chains of international metrology standards.

The verification made allows the data subsequently transmitted by the production unit in the Canister Quality Files to be validated.

Such a Quality File is attached to each glass canister and recorded. It contains all of the pertinent data relating to its production, including analytical results on the adjusted feed solution, the glass composition calculation sheet, and a description of processing operations for the corresponding glass batch. The Quality File is certified by Cogema Group's QC structure prior to shipment and disposal.

In addition, all the customers have entrusted an independent agency (Bureau Véritas) with the responsibility of performing an independent survey (on a permanent basis) to evaluate Cogema's measures of maintaining the specified quality. Bureau Véritas has to certify the conformity of each glass canister with the specification. Bureau Véritas is conducting inspections and quality audits to make sure that at any time the structure of the QA/QC set up by Cogema is appropriate and consistently applied.

Finally, Bureau Véritas checks all the documents pertaining to the production of each glass canister (including control parameters, Cogema QA certification of material, and so forth). On this basis, Bureau Véritas will certify that the glass canister conforms to the specifications. Bureau Véritas also checks all documents pertaining to the control of the glass canister at the time of loading into the transportation flask. On this basis, Bureau Véritas will certify that the glass canister meets the whole set of specifications and can be shipped.

3. Plutonium Vitrification

Plutonium vitrification has not been a subject of major research in France to date. A few fission product containment glass blocks were manufactured in the 1970s with a few percent of PuO_2 to dope them with alpha emitters in order to investigate alpha irradiation damage in the glass. This experiment proved difficult. Glass containing PuO_2 has been fabricated by Kernsforschungszentrum, Karlsruhe (KfK) in Germany, but plutonium vitrification has only been seriously investigated in the United States at Savannah River.

Broadly speaking, there seem to be two ways to vitrify plutonium.

The first route is through the incorporation of plutonium into borosilicate glass together with fission products. The intention here is to make recovery as difficult as possible. This way appears to be the safest in terms of non-proliferation imperatives but also the more technically difficult. It would be also preferable to get plutonium and fission products from the same country of origin for political reasons as it is necessary to have sufficient control over the materials.

The second path involves determining the optimum formulation for incorporating PuO_2 alone into a glass matrix. It should be noted that the maximum PuO_2 concentration in the glass would not only depend on its solubility in the glass, but could be limited by the risks of criticality and/or heat release in interim storage or after final disposal. The behavior of alpha emitting glass and its long-term behavior in repository conditions must also be investigated. This solution poses evident non-proliferation questions as the final product is more easily retrievable.

In any case, plutonium vitrification, contrary to fission products, is not a mastered technique. Reaching this goal will undoubtedly require time and money for R&D purposes.

4. Plutonium Conditioning

Incorporating plutonium into a ceramic matrix—in other words MOX—appears to be the best option because it answers all the non-proliferation issues raised by the plutonium disposition program. In any case, plutonium vitrification, contrary to the case for fission products, is not a mastered technique. Reaching this goal will undoubtedly require time and money for R&D purposes.

This solution will make the world safer for two main reasons:

- Moxification of weapons-plutonium diminishes the amount of weapons-grade fissile materials available to potential diversion. A 30 % MOX-loaded reactor has a neutral plutonium output, i.e. plutonium consumption equals plutonium production in the reactor. The difference lies in the quality of the plutonium contained in the fuel after irradiation.
- Moxification of weapons-plutonium offers a real advantage because Moxified plutonium undergoes an important isotopic degradation which makes the designing and building of an effective nuclear device using reactor-grade plutonium tremendously more difficult, if not impossible, than using weapons-grade plutonium.

The European and Russian positions rely on a long established commitment to MOX fuel. For instance, France, through Cogema Group, operates the following MOX fabrication plants: Melox (102 tHM tons produced in 1997), Cadarache (29 tHM tons in 1997); and, in association with Belgonucléaire, Dessel (37 tons in 1997) in Belgium. This program feeds almost all the 28 European reactors loaded with MOX fuel (15 in France, 8 in Germany, while Switzerland operates 3 and Belgium 2).

This unsurpassed expertise in MOX fuel led Cogema Group, Siemens, and Minatom to launch, at the end of 1996, an industrial initiative aimed at erecting a dedicated MOX fabrication plant for weapons-grade Pu in Russia. This contribution to the peace effort is based on the existing capabilities of Cogema Groups and Siemens. This proposal was officially praised by the world community. For example, the final communiqué of the last G8 Summit in Denver stressed the importance of this program as well as the joint U.S.-Russian "cooperation on the conversion of weapons plutonium." In that respect, Cogema Group is one of the prime contenders regarding the U.S. Department of Energy intention to build a MOX fabrication facility on its soil.

5. R&D Work

Still France is investigating the future with two programs aimed at studying the vitrification of other actinides than plutonium. To date, two directions have been favored by the scientific community: glass ceramic (titanite, zirconolite) and glass. It is a long-term endeavor with no immediate impact foreseen.

BRITISH VITRIFICATION PROCESS SAFETY ISSUES

C. J. THOMPSON
British Nuclear Fuels plc
Sellafield, Seascale, Cumbria CA201PG, U.K.

1. Introduction

British Nuclear Fuels plc (BNFL) provide a complete nuclear fuel cycle service with its sites at Springfields (AGR/Magnox Fuel Fabrication) near Preston and Sellafield (MOX Fuel Fabrication and Reprocessing) in Cumbria. BNFL also generates electricity using Magnox Reactors at Sellafield (Calder Hall) and Chaplecross in Scotland. This paper provides an overview of the Windscale Vitrification Plant (WVP) and reviews the major safety issues associated with vitrification operations. The practicalities of vitrification of Pu using the current WVP process are briefly discussed.

The Windscale Vitrification Plant vitrifies high level (highly active) liquid waste arising from reprocessing operations at Sellafield. The plant operates two identical vitrification lines with a current combined throughput of 350 product containers per year. A third line is currently under construction and will commence operation in the year 2000. The key safety function of the plant is to convert mobile material into a solid immobile form which can be more easily managed, stored, and transported.

2. The Vitrification Process at Sellafield

Highly active liquor (HAL) containing fission products arises from the first stage of chemical separation in the two Sellafield Reprocessing plants. This liquor is transferred via shielded pipebridges to the Evaporation and Storage Plant where it is concentrated by evaporation and then transferred to high-integrity, stainless-steel tanks. Due to the levels of radiogenic heat generated by the HAL, these tanks have a number of cooling coils through which water is passed to maintain the liquor at an acceptable temperature.

The Vitrification Plant processes HAL into a solid vitrified waste form that is then stored in a dedicated storage facility. The vitrification plant process is described below, considering each of the main cells in the plant in sequential process order:

- *HAL Cell*. Liquor is fed from the HAL Evaporation and Storage Plant to the Waste Vitrification Plant (WVP) HAL cell, which provides buffer storage of HAL liquor prior to feeding for vitrification. The cell contains two HAL feed vessels (10 m^3) and one high activity and one medium activity effluent vessel. Two high activity feed vessels permit one being used to feed liquor forward to the process whilst the other is being filled from B215. Feeds from B215 are interlocked to prevent overfilling; liquor feeds to the vitrification lines are controlled by automatic controllers on the feed systems.

- *Vitrification Cells*. Liquor is fed from the HAL cell to one of the two duplicate vitrification cells in the WVP. These house the main chemical process equipment, namely the calciner, melter, and primary off-gas vessels. The Vitrification Cell equipment evaporates the HAL liquor to dryness, mixes it with pre-formed glass (inactive) and other additives, and melts the mixture to form the glass product at a temperature of around 1100°C. The primary off-gas system removes dust, water vapor, and a proportion of the radioactive species that are entrained in the off-gas stream. HAL liquor is fed continuously into the Vitrification Cell and the glass product is poured in batches, approximately every 8 hours. Process conditions within the cells are generally manually controlled. All vitrification instrumentation is logged to a central control room; an interlock system is provided to trip the vitrification process if pre-set conditions are exceeded.
- *Pouring Cells*. The pouring cells contain the equipment for handling empty and full product glass containers. The pre-heated stainless-steel product containers are filled with glass from the melter, cooled to allow glass to solidify, and a lid is welded on the container to seal the product in the container. The product container is filled after two pours from the melter totaling 400 kg of product. Containers are left for two days to cool before welding on the lids.
- *Decontamination Cell*. The cell contains a high-pressure wash decontamination tank, a condenser for condensing water evaporating from the hot containers, and a decontamination fluid collecting tank. The product containers are sprayed with high-pressure water to decontaminate containers prior to surface contamination monitoring in the control cell. The in-cell equipment is controlled from an operating desk outside the control cell, and viewing is via closed circuit television.
- *Control Cell.* The cell contains a posting-in facility from the decontamination cell, an in-cell crane, a smear test machine to externally scrub the container, a pneumatic scrub transfer machine to allow swab monitoring, and a gamma gate for flasking out containers to the product store. The function of the cell is to monitor the external surface of containers prior to discharge to the product store; viewing is via closed circuit television.
- NO_x *Cell.* There are two nitrogen oxide (NO_x) cells, one for each vitrification line; each contains a single NO_x absorber column which receives the off-gas from the vitrification cell process. Their function is to remove radioactive species from the off-gas stream and to absorb NO_x compounds in the off-gas.
- *ESP Cell*. There are two electrostatic precipitators (ESPs), each in its own ESP cell. The electrostatic precipitator is the first stage of the secondary off-gas cleanup in which entrained particles of liquid droplets are removed from the off-gas stream. Only one ESP is on line at any one time; the other is isolated from the off-gas system by the ESP Seal Pots.
- *Wet Scrubber Cell.* The wet scrubber forms the second part of the secondary off-gas treatment. It removes active volatile contaminants from the off-gas stream, together with some of the remaining particulates before filtration. The wet scrubber operates continuously.
- *Filter Cell.* The filter cell contains banks of HEPA filters, associated ductwork, and ancillary equipment. The function of the filter cell is to filter the off-gas

from the vessel and cell ventilation systems before the off-gas is discharged to the environment.

- *Low Active Drain Cell.* The LA Drain Cell contains a 10 m^3 effluent tank which receives effluent from various sources in the vitrification process.
- *Vitrified Product Store*. The VPS consists of four heavily shielded concrete cells, each with its own inlet duct and outlet stack for cooling air; ventilation is by natural convection. Access to the cells is via concrete plugs in the floor of the main operating area. Each cell houses 200 vertical steel 'thimble' tubes held in a grid. Each thimble tube can hold a maximum of 10 product containers standing vertically, one on top of another. The thimble tubes stand inside a steel sleeve forming an annulus up which cooling air passes, indirectly cooling the containers. Product containers are transferred from the control cell to the vitrification product store in a product container transit flask, which is placed on a mobile gamma gate to allow the container to be lowered into one of the thimble tubes. The store has a total nominal capacity of 8000 product containers.

3. Demonstration of Safety

BNFL is permitted to carry out its operations at Sellafield under the conditions contained within its Nuclear Site License granted by the HSE under the Nuclear Installations Act 1965. This license contains a number of conditions which relate directly to the safe operation of nuclear plants and which have to be met in order for a plant to operate. One requirement is for the production and assessment of safety cases to justify safety during the design, construction, manufacture, commissioning, operation, and decommissioning phases.

Safety Cases are produced for all plants containing radioactive material on the Sellafield site and are reviewed on a regular basis. For each Safety Case, Radiological Hazards are assessed using HAZOP and Probabilistic Risk Assessment techniques and the results are compared against a set of safety criteria relating to both the work force and the public. Compliance with these criteria ensure compliance with current national and international radiological protection standards; for example, those of the International Commission on Radiological Protection (ICRP).

For criticality hazards, the double contingency approach is employed, and in some cases probabilistic techniques are employed to demonstrate that a limiting criterion can be met. However, the minimal amounts of fissile material in waste vitrified product mean that no detailed criticality assessments are required.

Key operating procedures and equipment identified in the Safety Case are referred to as Operating Rules and Safety Mechanisms.

An Operating Rule is a limit or condition that is necessary in the interests of safety and is designated only where a failure on its own would result in a numerical breach of specified criteria for either the public or the work force.

Safety Mechanisms are identified by the hazard analyses in two ways; either they are in support of an Operating Rule that has been identified in the analyses, or they can be shown to be a key instrument. A key instrument is one that if removed or permanently in the failed state would cause the estimated fault frequency to exceed the frequency target of a specified criterion.

Operating Rules and Safety Mechanisms thus provide the primary protection against the most severe accidents (major hazards).

4. Major Hazards Associated with Vitrification Operations

The Safety Case produced for the Windscale Vitrification Plant in 1994 included a detailed and comprehensive assessment of fault conditions in the plant using HAZOP and Probabilistic Risk Assessment techniques. The Safety Case identified a number of major hazards. These major hazards, along with the protective measures, Operating Rules, and Safety Mechanisms designed to prevent these hazards or to mitigate them are briefly described below.

4.1 DIRECT DOSE TO OPERATORS

The HLW feed to the Windscale Vitrification Plant and the vitrified product are highly radioactive, direct doses at 3-4 m from a product container exceed 40 Svh^{-1}. The Windscale Vitrification Plant has therefore been designed with appropriate levels of shielding and remote handling systems to minimize the potential for significant direct dose to operators.

The requirement to access cells for maintenance purposes and to transfer full product containers to the vitrified product store offers the potential for >1-Sv doses to operators under fault conditions. In order to prevent such incidents, entries to cells are controlled via inner and outer shield doors that are interlocked to gamma monitors. Additionally Operating Rules and Instructions place requirements on operators with regard to man entries to cells and prohibit the presence or introduction of active product containers to specified cells. Personal Alarmed Dosimeters are routinely worn during cell entries to provide an immediate indication of high dose rates.

4.2 LOSS OF COOLING TO HAL FEED VESSELS

Radioactive decay of HAL generates sufficient heat for the liquor to boil and release significant activity to the environment. To prevent this, the HAL tanks have been designed with a series of cooling coils through which water is continually passed. The consequences of the loss of cooling water and subsequent boiling are significant (>1-mSv aerial dose to the public).

Loss of cooling water could arise as a result of the loss of the water supply, the loss of the electrical supply for the pumps, or cooling could be lost as a result of the loss of temperature control. A series of engineered safety measures are in place to prevent this. The ventilation system (electrostatic precipitators, scrubbers, and filters) would also play a major part in mitigating the consequences of a boiling tank.

4.3 ACTIVITY IN COOLING WATER

In the event of a leak developing in a cooling coil in the HAL feed tanks, activity would get into the cooling water system. The cooling water passes to the cooling towers via a delay tank located in the cooling water monitoring rooms. Area gamma monitors are provided in the room along with automatic isolation systems (Safety Mechanisms) on the delay tank that would prevent water being pumped to the cooling towers.

4.4 PRESSURIZATION OF VENTILATION SYSTEM

The Off-gas and Vessel Ventilation systems maintain a depression in the liquor and liquid effluent storage vessels and clean gases arising from the vitrification process and liquor transfer system prior to discharge as gaseous effluent. Radiological hazards arise from two main types of faults, either loss of off-gas decontamination or by pressurization of the ventilation system. Equipment which confirms the safe operation of the ventilation system (HEPA filter, differential pressure gauges, electrostatic precipitator failure detection, and alarm systems) are designated as Safety Mechanisms.

5. Routine Doses

In contrast to the extremely high doses that could arise under fault conditions, routine doses to members of the WVP work force are very low. The average annual dose for the work force in 1997 was 1.01 mSv, and the maximum dose 5.67 mSv, both significantly less than the BNFL targets of <5 $mSvy^{-1}$ and 15 $mSvy^{-1}$, respectively. Dose is closely controlled and monitored in line with UK statutory requirements.

6. Routine Discharges

Routine aerial discharges from waste vitrification product and the vitrified product store are very low, being less than 3% of the current site authorized limits. Liquid discharges are minimal.

7. Extrapolations for Plutonium Vitrification

The safety implications of Pu vitrification will obviously depend on the nature of the process adopted, whether inactive or active material is used as a matrix, and how the Pu is introduced and incorporated in the system. Issues relating to the ability of current systems to cope with Pu as an additional feed component would need to be addressed to assess the viability of any proposals.

The following points relate to the suitability of the WVP process at Sellafield to be used as the starting point for a Pu vitrification process.

The WVP process at Sellafield is specifically designed to vitrify High Level Waste from reprocessing operations at the site. As such, the borosilicate glass was formulated to provide the optimum properties for the incorporation and immobilization of calcined HLW.

Very small quantities of Pu are carried over into the HLW stream and these are incorporated in the vitrified product. However, the mixed alkali borosilicate glass used in WVP can only accommodate ~3-5 wt% PuO_2 before glass quality and durability are impaired. The glass currently used is therefore unsuitable for immobilizing large quantities of Pu.

If the Pu were contaminated with other highly radioactive species, then calculations would have to be carried out to ensure that the extra heat generating radionuclides did not produce enough heat to increase the centerline temperature of the product package above the glass transition temperature (if this occurred then the product quality would again be compromised).

BNFL have no practical experience of vitrifying Pu (other than the trace quantities included in the WVP product) and have not carried out any experimental work on the topic to date.

There is pressure from the UK Nuclear Industry Regulator (the Nuclear Installations Inspectorate, part of the HSE) on BNFL to minimize levels of HLW stocks by vitrifying this material as soon as possible. BNFL are also concentrating on making use of the Pu resource by recycling Pu into MOX fuel.

8. Conclusions

The vitrification process at Sellafield provides a proven safe route for the immobilization and storage of high level waste arising from reprocessing operations.

Safety of the Windscale Vitrification Plant is demonstrated by a safety case, which is required by the UK Regulators to justify continued operation, key items of safety equipment, and key operating procedures identified within the Safety Case.

Routine doses to the work force and members of the public resulting from the operation of the Windscale Vitrification Plant are significantly below UK statutory limits.

The vitrification process currently employed at Sellafield is not amenable to the incorporation of significant amounts of plutonium.

DWPF VITRIFICATION SAFETY ISSUES

F. BERANEK
Westinghouse Safety Management Solutions, Inc.
1993 South Centennial Ave., Aiken, SC, 29803, U.S.A.

1. Introduction

High-level waste volume minimization currently requires the addition of organic compounds to precipitate Cs from the high volume of supernate. In addition, the alkaline nature of the waste requires pretreatment of the waste prior to introduction to the melter, which then calcines and vitrifies simultaneously. These pretreatment operations in the Defense Waste Processing Facility (DWPF) can release hydrogen, benzene, ammonia, and carbon monoxide, which could, under upset conditions, result in flammable mixtures. The design of the facility has accounted for accidents of this nature and provides defense in depth for prevention. Use of the DWPF process for direct vitrification of PuO_2 is not recommended due to the significant required design changes to ensure criticality safety. However, cans of PuO_2 in ceramic matrix, which are subsequently placed in a DWPF canister prior to high level waste glass being poured, appear to introduce no new significant safety issues.

Forty years of nuclear materials production at the Savannah River Site (SRS) has generated over 300 million liters of aqueous radioactive waste to date. This waste has been primarily generated from the two fuel reprocessing facilities as fission products from reactor irradiations. The current volume of waste is less than half the above due to processing through evaporators to remove excess non-radioactive water. The total activity stored in the tanks is over 600 million Ci (2.2E19 Bq).

Waste generated from operations in the reprocessing facilities was transferred to carbon steel underground storage tanks. Addition of NaOH minimized tank corrosion by making the waste strongly alkaline. Over time, the waste in the tanks physically separated into two components. A dense sludge consisting of metal hydroxides and hydrated metal oxides (primarily Fe, Al, Mn, and actinides) settled to the bottom. A supernate containing soluble salts such as Cs floated on top of the sludge. Because each phase contained highly radioactive materials, the final waste dispositioning process was required to vitrify both supernate and sludge.

2. Waste Pretreatment

This decision was made to treat each phase separately. The sludge would be jetted from the bottom of the tank, treated with caustic to remove the Al solids, washed with inhibited alkaline water and the remaining sludge transferred directly to the DWPF. The wash water would be processed at the low level waste Saltstone facility.

The supernate would be transferred to a dedicated tank and decontaminated by adding tetraphenylborate causing the Cs, K, and Na to precipitate out of solution. Also, sodium titanate would be added to adsorb Sr, U, and Pu. This mixture of solids would than be concentrated by filtration forming a precipitate slurry and low level waste solution. The solution would be processed into grout at the Saltstone facility while the precipitate slurry would be transferred to the DWPF.

3. DWPF Processing

Due to the alkaline nature of the SRS waste, directly calcining the material is impractical. Therefore, the waste material must be chemically processed in several steps prior to being added to the melter in which calcination and vitrification is performed in one vessel. Each step of the process was analyzed during the design process to determine required safety features. Hazards during normal and accident situations are identified and if potential consequences are too high, engineered safety equipment is designed into the facility.

The precipitate slurry transferred to the DWPF may have nitrite composition, which exceed levels that are compatible with the remainder of the process. Acceptable nitrite levels are assured by processing the precipitate through the Late Wash Facility. While the slurry is being processed, benzene and hydrogen are being released from normal tetraphenylborate decomposition and radiolysis caused by the incumbent radionuclides. If not controlled, these flammable gases could reach concentration levels in the vapor space, which could support combustion or deflagration. Prevention of such an occurrence is assured by provision of a safety class nitrogen purge system, which maintains the oxygen level at a low enough percentage to preclude combustion.

Excess organics are contained in the precipitate slurry transferred from the Late Wash Facility. These organics must be removed prior to vitrification in the 1150°C melter. The Precipitate Reactor receives the nitrite balanced slurry from the Late Wash Facility where it is heated and chemically treated to cause the release of benzene which is unavoidably accompanied by hydrogen. This benzene is transferred to another system to be purified and ultimately combusted. However, during the processing in the Precipitate Reactor, a continuous CO_2 purge is added to assure low oxygen concentrations in the vapor space and prevent deflagrable mixtures from forming.

The next two vessels in the process are the Sludge Receipt and Adjustment Tank and the Slurry Mix Evaporator. The sludge from the waste tanks enters the DWPF process at the Sludge Receipt and Adjustment Tank. The role of these vessels is to neutralize the sludge, remove Hg, adjust pH to improve rheology and add the glass frit. The safety hazards inherent in these processes are ammonia gas, which will form ammonium nitrate and condense in the ventilation system. Safety systems added to these processes include air/nitrogen purges to remove benzene and hydrogen and ammonia scrubbers to remove ammonia from the offgas preventing the formation of ammonium nitrate.

Finally, this mixture arrives at the melter where the waste is vitrified. In the melter, organic impurities are released which have been carried over from earlier processing and the gasses of hydrogen and carbon monoxide are generated. The temperature and airflow into the melter are designed to provide complete combustion of the gases before they reach the offgas system. If the design temperatures and airflows are not met, there are safety class interlocks, which stop the feed entering the melter.

4. Pu Disposition

With regard to using the DWPF for vitrifying PuO_2, it should be noted that the Pu-239 content in the design basis waste glass is only 1×10^{-2} wt%. This provides for the trace Pu that is within the waste. A canister of waste glass therefore contains less than 200 g Pu-239 and is surrounded by integral neutron poisons such as Fe and Mn. For this reason, DWPF process is inherently safe from a criticality perspective.

If the DWPF was to be used to vitrify PuO_2 directly (i.e., no prior process to make PuO_2 frit), the PuO_2 would presumably be added to the process in the Slurry Mix Evaporator. Economic feasibility of the process would require addition of PuO_2 in the tens of kilogram quantities in a batch. Masses of this magnitude are not inherently safe and the vessel is not geometrically favorable for preventing criticality. Therefore, the current design could not assure criticality safety due to the inability to assure homogeneous mixing and the potential for Pu to collect over time to form a critical mass.

The situation is similar for the melter. The melter would contain over 100 kg of Pu and is not designed for inherent criticality safety with that mass amount. Also, the ventilation system was not designed for the possibility of Pu offgassing. Major redesign of the DWPF vessels and ventilation system would be required.

Another risk that is exacerbated if PuO_2 powder is introduced is that of fire. Clearly the use of cold chemicals, the presence of electrical cabling and the generation of flammable gases lead to a defined fire risk. Unless buildings are designed with fire propagation as a design parameter or combustibles are kept at very low levels, fires can spread to areas where radioactive materials are present potentially resulting in a thermal, elevated release from the facility. If PuO_2 is introduced as a fine powder or stored in sealed cans, care must be taken to assure release due to a fire is prevented. The fine powder can be made airborne if disturbed and the storage cans may rupture causing a significant release. The potential source term for the DWPF would increase dramatically if PuO_2 were present.

However, an alternative process being discussed is called the Can-In-Canister (CIC) concept. In a facility separate from the DWPF, the PuO_2 would be immobilized in a ceramic matrix. This form would then be placed in a stainless steel can with dimensions much smaller than a DWPF canister. Approximately twenty of these cans would be placed in a rack, which is then inserted into an empty DWPF canister. The canister would then follow its design path, which is to be filled with molten glass thus, encasing the smaller cans within the highly radioactive waste glass.

This process, if the cans were designed properly, would introduce no new safety issues in the DWPF. There is no accident considered which could release PuO_2 from the cans in a manner that could harm the workers or public. The separate facility, which creates the PuO_2 ceramic, must deal with several hazards including PuO_2 powder and criticality.

5. Status

To date, DWPF has rendered immobile over 1,000,000 liters of sludge in over 1.4 million kilograms of glass. Observed gas generation (hydrogen and ammonia) has been within the anticipated ranges. No precipitate slurry has been processed to date so significant organic releases have not been observed. Currently, alternative processes are being evaluated for treating the supernate as higher than expected benzene generation rates were observed during initial processing.

MOL VITRIFICATION PROCESS (PAMELA) SAFETY ISSUES

JEF CLAES
Belgoprocess N.V.
Gravenstraat 73, B 2480 Dessel, Belgium

1. Introduction

Since 1985, Belgoprocess has operated the centralized waste management facilities for the Belgian nuclear activities, comprising nuclear power plants, fuel fabrication plants, nuclear research, radioisotope production, and medical and industrial applications of radioisotopes. The company is a subsidiary of the national radioactive waste management agency NIRAS/ONDRAF. The facilities are located on the nuclear site of Dessel and Mol. They comprise the former reprocessing plant of Eurochemic, the ownership of which was gradually transferred to the Belgian State beginning in 1976, and the former Department of Radioactive Waste of the research center SCK/CEN, which was integrated into Belgoprocess on March 1, 1989.

The Eurochemic reprocessing plant, erected by a consortium of 13 European member states of the OECD/NEA, was in active operation between 1966 and 1974. During these campaigns, 181.5 tons of natural and slightly enriched uranium fuels and 30.6 tons of highly enriched aluminum alloy fuels from material testing reactors were reprocessed.

As a result of these activities, about 900 m^{-} of liquid high level wastes (HLLW) were generated and stored. The main characteristics of these Low Enriched Waste Concentrates (LEWC) and High Enriched Waste Concentrates (HEWC) are summarized in Table 1. The vitrification of these wastes was a major objective of the Waste Management Programme of Eurochemic, to which it was committed in the frame of the transfer of the ownership to the Belgian State.

In 1979, Eurochemic concluded an agreement with the German company DWK for the erection of a vitrification plant on the Belgoprocess site, according to the PAMELA process. This process is basically vitrification in a joule-heated ceramic melter, operated at 1150°C with a continuous feed of the liquid waste stream with beads of an appropriate borosilicate glass, and a batchwise discharge of the glass product, using a bottom drain or an overflow drain. The top layer of the glass pool in the melter is the process zone, where drying, calcination, and melting reactions occur.

TABLE 1. HLLW vitrification program—main production data.

Parameter	Unit	LEWC 85/10/01 –86/06/12	HEWC 86/10/01 –91/09/05	Total period 1985-91
Total liquid waste fed to melter	m–	62.5	894.6	957.1
Total LEWC and HEWC fed to melter	m–	47.2	860.1	907.3
Alpha activity in feed	PBq	1.28	0.23	1.51
Beta activity in feed	PBq	278	164	444
Waste oxides	t	7.67	88.94	96.91
Pu	kg	4.0	9.7	13.7
U	kg	44	135	179
Ru	kg	70.3	37.6	107.9
Pd	kg	10.5	62.9	73.4
Rh	kg	15.9	8.6	24.5
Glass product	t	77.8	411.7	493.2
Waste oxide content	wt%	9.9	21.60	-
Containers filled - 60 l - 150 l	No. No.	567	934 700	1,503 700
Total weight filled containers	t	159.28	563.42	722.7
Time efficiency	%	88	93.16	-
Production efficiency	%	69	96.58	-
Alpha activity released	kBq	4.4	4.08	8.5
Beta activity released	MBq	3.4	0.6	4.0
Occupational dose	mSv	40	211	251

2. Safety Aspects Under Reference Operating Conditions

The PAMELA vitrification plant at Belgoprocess was designed and licensed for processing the LEWC fraction of the HLLW from Eurochemic. The decision also to vitrify the HEWC fraction in PAMELA was taken in 1986 and an additional nuclear license thereto was granted. These were the reference conditions for the operation of the PAMELA plant.

The safety aspects of the PAMELA vitrification process, the facility, and its operation are described in the safety assessment report as part of the licensing procedure. The table of contents of this safety report is given in Table 2, showing the extent and completeness of the process development, control and operation, the facility design and its reliability, as well as the quality assurance program of the resulting glass products.

TABLE 2. PAMELA Vitrification Facility—Safety Assessment Table of Contents.

1. Introduction **2. Characteristics of the PAMELA Plant** 2.1. Essential data of the HLLW 2.2. Relevant PAMELA process information 2.2.1. Vitrification process parameters 2.2.2. Emissions and radioactivity balance 2.2.3. Upper limits of radiation exposure 2.2.4. Product parameters **3. PAMELA Vitrification Process** 3.1. HLLW transfer and storage 3.2. HLLW metering and vitrification 3.3. Containment and transport of vitrified products 3.4. Off-gases from PAMELA and their treatment 3.4.1. Dust and aerosol removal from the off-gases 3.4.2. Removal of nitrogen oxides 3.4.3. Lead vapour removal 3.4.4. Fate of tritium 3.4.5. Fractional release of off-gas components 3.5. Treatment of secondary wastes 3.6. Decontamination and maintenance 3.6.1. Remote maintenance procedures 3.6.2. Direct maintenance procedures 4. Main Auxiliary Equipment and Utilities 4.1. Steam 4.2. Cooling systems 4.3. Electrical power supply 4.4. Instrumentation and control 4.5. Building ventilation systems 4.6. Sampling of solutions 4.7. Chemicals and other necessary additives 4.8. Waste water disposal **5. The PAMELA Plant Building** 5.1. Location 5.2. Dimensions 5.3. Structure **6. Safety Analysis**	6.1. General safety philosophy 6.2. Radiation protection equipment 6.2.1. Dose rate monitoring 6.2.2. Air monitoring 6.2.3. Contamination controls 6.2.4. Personnel exposure surveillance 6.2.5. Exhaust air and off-gas monitors 6.3. Nuclear safety 6.3.1. Containment and confinement 6.3.2. Shielding characteristics 6.3.3. Shielding design 6.3.4. Exposure to atmospheric releases 6.4. Conventional safety 6.4.1. Fire protection 6.4.2. Natural events 6.4.3. Emergency escapes 6.4.4. First aid **7. Accident Analysis** 7.1. System selection 7.2. HLLW buffer tank analysis 7.3. Analysis for ceramic melter 7.3.1. Heat removal 7.3.2. Leakage risk in the ceramic melter Annex 1. Efficiency of the Off-Gas Treatment Annex 2. Calculation of Radiation Exposure due to the Emission of Radionuclides from the PAMELA Plant via the EUROCHEMIC Stack to the Atmosphere Annex 3. Quality Assurance and Control Programme Annex 4. Decontamination Programme Annex. Radiation Protection Equipment Annex 6. Quality Control for the Final Vitrification Product Annex 7. Ventilation System of the PAMELA Plant Annex 8. Behaviour of the PAMELA Plant in Case of Loss of Electrical Power Annex 9. Container Handling

The following safety aspects can be considered as key issues:

- The large radioactivity inventory, which requires
  - Containment and confinement of the radioactivity in the facility in order to avoid spread of contamination
  - Heavy shielding provisions for the operators and the environment
  - Remote operation of the facility, including the replacement of plant components
  - Appropriate decontamination and maintenance capabilities
  - Proper design of the ventilation systems.
- The high quality requirements of the final glass products, which implies
  - Development of a typical glass and final product composition that meets the acceptance criteria for long term storage and disposal
  - Continuous control and monitoring of all process steps, in particular the waste feed stream, the thermal and chemical processes occurring in the glass melter, and the packaging of the final glass product.
- The use of a melting process at high temperatures, requiring
  - High reliability of the melter, including its confinement.
  - Highly efficient off-gas treatment.

Note that criticality is not specifically addressed in this reference safety assessment. On one hand, this is due to the low plutonium (Pu) concentrations in the HLLW, the resulting waste feed stream and all downstream liquid, gaseous, and solid waste streams, as is explained in section 4. On the other hand, the total Pu mass in the facility and in particular in the largest equipment components (e.g., the waste feed tank and the melter) is far below the level of any criticality concern. Also recall that the HLLW storage tanks are part of the former Eurochemic reprocessing plant and that these tanks have been considered in the safety analysis of the storage of the HLLW.

3. Operating Experience with the PAMELA Vitrification Plant

3.1 GENERAL INFORMATION

The PAMELA facility was erected between 1981 and 1984. It went into active operation in 1985 and was operated by a mixed Belgian-German crew under the nuclear operating license of Belgoprocess.

The PAMELA vitrification process development as well as the operating experiences have been presented at several international conferences and symposia on radioactive waste management [1–5]. Presentations were also made at the NATO Advanced Research Workshop "Disposal of Weapons Plutonium—Approaches and Prospects," held in St. Petersburg, Russia, May 14-17, 1995 [6–7].

In particular, Mr. Demonie presented a preliminary evaluation on the feasibility of the PAMELA process for the vitrification of HLLW with high concentrations of Pu, addressing some specific aspects related to this particular application.

The HLLW were successfully vitrified in two successive campaigns during the periods 1985-1986 and 1987-1991. The operation resulted in the production of 493 tons of glass product packaged in 1,503 canisters of 0.060 m– and 700 canisters of 0.150 m–. These canisters are in safe intermediate storage conditions at Belgoprocess until a geological disposal site is available around 2030. The total radioactivity inventory amounts to 4.44^{17} Bq of beta emitters and 1.51^{15} Bq of alpha emitters.

The facility was operated continuously at an average waste feed rate of about 16 liters/h during the LEWC campaign and about 25 l/h during the HEWC campaign. The average throughput was respectively 420 and 300 kg of glass product per day with a waste oxide content of 40 kg in the LEWC and 66 kg in the HEWC glass product.

The total operating staff numbered 44 during the LEWC campaign and was gradually reduced to 34 during the HEWC campaign. In addition, technical, radiation protection, and waste management support was made available by the Belgoprocess organization. The occupational exposure amounted to 251 man.mSv for the total period of active operation. The maximum individual annual dose was 20 mSv. These exposures were mainly due to intervention activities for repairing remote handling equipment.

The annual atmospheric releases of radioactivity through the stack were limited to 1.4 kBq of alpha emitters and 0.7 MBq of beta emitters. These releases correspond to 0.005% of the authorized limit for alpha emitters and 0.01% for beta emitters. Moreover, they represent only a fraction of the total inventory of 5.6^{-12} alpha emitters and 9.1^{-12} beta emitters in the vitrified HLLW. The main operating results of the two campaigns are also summarized in Table 1.

3.2 MELTER BEHAVIOUR, LIFE TIME, REPLACEMENT, AND DISMANTLING

The ceramic melter is the heart of the vitrification plant. The experience with and the understanding of operating phenomena with the ceramic melter is extremely valuable in the transition process from design and development to industrial proven technology.

The bottom of the ceramic melter used during the LEWC campaign had a horizontal shape. Small amounts of noble metals (Ru, Pd, and Rh) in the HLLW accumulated on the bottom because of their low solubility and high density. This viscous layer formed a preferential conductive pathway and could not be discharged from the melter through the central bottom drain. The resulting loss of power and hence throughput led to corrosion damage of the lower electrodes, caused by excessive high local current densities and to corrosion of both drain systems as well. Additional development work was carried out at the nuclear research center in Karlsruhe, FRG, to modify the bottom design of the melter for vitrification of HLLW from reprocessing commercial spent fuel with a higher content of noble metals. This work also included improvements of the bottom and overflow drains to accommodate for the blockages, that were experienced during operation.

The first melter had to be shut down after 34 months of operation. Within nine weeks, the 20-ton melter was fully remotely exchanged, using the equipment that had been installed in the melter cell as part of the basic facility concept. This was an extremely valuable operating experience, confirming the industrial level of the technology.

A similar positive experience was gained with the dismantling of the second melter, which was shut down at the end of the second vitrification campaign in 1991.

3.3 EVALUATION

The operating experience with the PAMELA vitrification plant at Belgoprocess has shown that this process has reached a fully developed and reliable industrial level. It has proven its ability to produce high quality final glass products within the narrow specifications as required for long-term storage and disposal. In particular, the outstanding records for occupational doses and radioactivity releases meet completely the fundamental radiation protection objectives for both workers and the public.

4. Vitrification of HLLW with Higher Plutonium Concentrations: Safety Considerations

4.1 OBJECTIVE

This section deals with qualitative and semi-quantitative evaluations from the perspective of a vitrification operator on safety issues related to operating conditions, whereby HLLW with higher Pu concentrations would be processed. This scenario is considered together with the use of MOX fuel in commercial nuclear power plants as options for disposition of surplus weapons-grade Pu.

4.2 AVAILABLE EXPERIENCE

The actual operating experience with vitrification at industrial levels in Western-Europe is associated with the reprocessing of commercial spent fuel. In the case of HLLW from Eurochemic, the concentrations of fissile materials in the waste and the glass product are listed in Table 3. For Pu, they are on the order of 0.01 to 0.1 g/l for the waste feed and 0.01 to 0.1 g/kg for the glass product. These concentrations of Pu are comparable with those of the COGEMA glass product specifications.

TABLE 3. Uranium and plutonium concentration in the HLLW and in the glass products.

	LEWC		**HEWC**	
	Feed g/l	**Glass product g/kg**	**Feed g/l**	**Glass product g/kg**
Uranium	0.93	0.57	0.16	0.33
Plutonium	0.08	0.05	0.01	0.02

U, Pu mass in HLLW solutions			
	LEWC	**HEWC**	**Total**
	47.2 m–	860.1 m–	907.3 m–
Uranium	44 kg	135 kg	179 kg
Plutonium	4.0 kg	9.7 kg	13.7 kg

The development of the PAMELA vitrification process, the definition of the required glass composition, the design of the vitrification facility, and the characterization of the final glass product have not considered operating conditions with higher Pu concentrations.

In such a case, it is obvious that *criticality risks* in the subsequent process steps, including the glass canisters and the ultimate disposal concept, the influence on the *glass composition, and quality* and *safeguards* are the main issues that need further investigation.

4.3 PU CONCENTRATION LIMIT IN THE HLLW

The highest acceptable concentration of plutonium in HLLW for vitrification is the lowest value to be derived from the following boundary conditions:

a) International consensus on the level of difficulty to recover Pu from the glass product as essential disposition objective;

b) Criticality analysis of the disposal concept, including site capacity and configuration of the glass canisters in the disposal site;

c) Criticality analysis of an individual glass canister, with emphasis on its long-term Behaviour;

d) Criticality analysis of the vitrification facility, including the ceramic melter and the upstream liquid waste plant components. If criterion (d) is the determining factor, criticality can be controlled by:
   - Operation of the plant under acidic HLLW conditions to exclude precipitations and limitation of the size of the batches in the vitrification plant in terms of total Pu mass and/or concentration;
   - The use of neutron poison (B, Ga, Cd ...);
   - Critical safe geometry of the equipment components.

The feed chain to the melter is equipped with vertical cylindrical vessels of 3.4 and 0.15 m–. The input vessel of 3.4 m– is provided with a pulsation system for homogenization. Liquids are transferred by air lifts. Mixing of Pu nitrate and HLLW requires additional equipment. Borated austenitic stainless steels are appropriate construction materials.

The ceramic melter as used in PAMELA has a glass volume of about 0.340 m– corresponding with a glass content of about 800 kg. The glass used in the LEWC and HEWC campaigns has a boron concentration of 4.1 and 7.8 wt%, respectively.

A Pu concentration below 6.9–7.3 g/l HLLW is a recommended critical safe concentration limit. The presence of ^{240}Pu in the Pu and the presence or addition of chemical poisons in the HLLW constitute conservative elements.

The Pu content of the melter depends upon the concentration of calcined residue in the HLLW and of waste oxide in the glass. As an example, Table 4 shows the Pu concentration in the glass for different concentrations of calcined residue in the HLLW and waste oxides in the glass, for 6 g Pu/l in the HLLW. According to this example, the maximum Pu content in the melter would be 12.8 kg. The maximum content of a 150-l canister with 400 kg glass would be 6.4 kg. The total HLLW volume required for 1,000 kg Pu is 167 m–, resulting in 23 m– of final glass product.

TABLE 4. Plutonium concentration in the glass product with 6 g/l Pu in the HLLW.

g/l calcined residue in HLLW	**Waste oxide content in the glass product (g/kg)**			
	80	**100**	**140**	**160**
60	8	10	14	16
80	6	7.5	10.5	12
100	4.8	6	8.4	9.6
120	4	5	7	8

The criticality analysis of the melter will have to confirm the validity of the assumptions above. Referring to the experience with noble metals in the PAMELA plant, the solubility of PuO_2 in glass needs further investigation to avoid any build up in the melter.

The criticality control requires reliable sampling, analyses, and inventories. The procedures as applied in reprocessing facilities remain valid.

4.4 GLASS COMPOSITION AND GLASS PRODUCT QUALITY

The composition of the glass used in the vitrification campaigns at Belgoprocess is given in Table 5. The differences in concentrations of the main constituents SiO_2 and B_2O_3 between the two campaigns is due to the high Al content in HEWC (MTR fuel—U/Al alloy). When considering the PAMELA process for vitrification of HLLW with higher Pu concentrations, the formulation of the glass composition will have to be verified on the basis of the composition of the reference HLLW and in particular of the waste oxide content. Special attention needs to be drawn to the confirmation of the solubility of PuO_2 in the glass product and hence to the potential subsequent process limitations. In this preliminary glass development Program, the feed rate of HLLW into the ceramic melter of a given size and geometry will also be determined.

TABLE 5. Composition of the glass for LEWC and HEWC vitrification.

	Glass frit type	
Chemical components	**LEWC wt%**	**HEWC wt%**
SiO_2	58.60	45.50
B_2O_3	14.70	33.00
Al_2O_3	3.00	-
Li_2O	4.70	4.50
Na_2O	6.50	9.70
MgO	2.30	-
CaO	5.10	6.50
TiO_2	5.10	-
Sb_2O_3	-	0.80

The characterization of the glass products with higher Pu contents in comparison with glass products from reprocessing of commercial spent fuel is a very important element in the feasibility study of the disposition option. The physico-chemical stability of the glass product and its resistance to increased radiation damage resulting from Pu over very long periods of time requires confirmation. As an example, the influence of He formation from the decay of Pu on the glass product and its containment is a topic for further investigation. This aspect is essential for the assessment of potential reconcentration processes after disposal and the derived criticality limitations.

4.5 SAFEGUARDS

Because of the low concentrations of fissile materials in the HLLW from reprocessing, and since vitrification is an immobilization process, the operation of the PAMELA plant was exempted from control by EURATOM and the International Atomic Energy Agency (IAEA). When higher amounts of Pu are processed, the safeguard concept and procedures applicable to reprocessing will have to be implemented. This involves additional analytical controls, process verification, and enhanced security measures.

5. Conclusions

The experience gained during six years of active operation of the PAMELA vitrification plant has proven the feasibility of the process and the industrial reliability, based upon state-of-the-art technology. Excellent results with the quality of the final glass products, the occupational doses, and the atmospheric releases have been recorded. Additional development work, carried out at FzK, FRG allows us to introduce further improvements of the ceramic melter design into the process.

When considering the vitrification of HLLW with higher Pu concentrations, a number of detailed investigations are required to confirm the validity of this option. From the viewpoint of the operator with the available know-how of the Eurochemic reprocessing plant, these investigations especially relate to a criticality study of the total vitrification plant and to the glass composition and glass product quality.

The rather small ceramic melter allows for a total Pu content up to 15 kg with subcritical Pu concentrations in the HLLW of 6.9 g/l. The reference plant design requires modifications to include the Pu/HLLW mixing step and to accommodate for critical safe geometries in the HLLW feeding components of the facility and for the neutron-absorbing construction materials that would be recommended in the criticality study as a mean of increasing the upper Pu concentration limit. These considerations do not take into account any other limitation of the Pu concentration in the glass product, resulting from the management of the waste products downstream the vitrification.

The selection and formulation of the proper glass composition involves a specific development program. This program also determines the process parameters and the throughput of the vitrification plant.

Finally, the knowledge of the impact of higher Pu concentrations in the glass product on its quality, in particular the long-term behavior and stability, is of utmost importance.

References

1. Wiese, H. and Demonie, M., "Operation of the Pamela high-level waste vitrification facility," *Nuclear Engineering and Design* 137 (1992) 147-151.
2. De Goeyse, A., De, A.K., Demonie, M. and Van Iseghem, P., "Geological Disposal of Spent Fuel and High Level and Alpha Bearing Wastes," *Proceedings of a Symposium, Antwerp (Belgium), 19-23 October 1992,* IAEA-SM-326/62.
3. De, A.K., Wiese, H. and Demonie, M., "Improvement of Vitrification Operation by Optimization of the Borosilicate Glassfrit at the Pamela Plant," *Spectrum '90 - Nuclear and Hazardous Waste Management International Topical Meeting, September 30-October 4, 1990,* Knoxville (USA).
4. Ewest, E. and Wiese, H. (1988), "High Level Liquid Waste Vitrification with the Pamela Plant in Belgium," *Nuclear Power Performance and Safety* IAEA-CN-48/177.
5. Claes, J. and Meyers, H. (1989) "The Radioactive Waste Management Programme associated with the Decommissioning of the former Eurochemic Reprocessing Plant," *International Conference on Radioactive Waste Management, Kyoto (Japan), October 1989.*
6. Demonie, M. "On the Feasibility of Vitrifying High-Level Liquid Waste containing High Amounts of Plutonium," *NATO Advanced Research Workshop, May 14-17,1995*, St. Petersburg, Russia.
7. Trauwaert, E; and Demonie, M. (1995) "Plutonium handling and vitrification: main process steps and their cost evaluation," NATO Advanced Research Workshop, May 14-17,1995, St. Petersburg, Russia.

SAFETY PROBLEMS OF PLUTONIUM MANAGEMENT AND ITS IMMOBILIZATION IN CRYSTAL MINERAL-LIKE FORMS

E. B. ANDERSON
B. E. BURAKOV
E. I. ILYENKO
V. G. Khlopin Radium Institute
28, 2nd Murinskiy Ave., St. Petersburg, 194021 Russia

1. Introduction

In considering the alternatives of weapons plutonium management, it is necessary to estimate the safety for the population and environment. A great deal of attention has been given to this problem in recent years.

Safety is defined as the absence of factors causing damage to man or the environment. It is a probabilistic value which quantitatively can be expressed in terms of the size of a risk level by an estimation of the probability of different kinds of events and the possible damage from each event (probabilistic analysis of risk).

All the events, depending on the possible frequency of their occurrence, can be divided into a number of categories: anticipated, unlikely, very unlikely, and incredible (Table 1).

TABLE 1. Qualitative relative probability classification.

Relative Probability Category	Est. Annual Likelihood Occurrence	Description
1. Incredible	$\leq 10^{6}$	All accidents not included in other categories. Frequency of less than once in a million years
2. Extremely unlikely	10^{-6}–10^{-4}	Accidents that will probably not occur during the life cycle of the facility. This class includes the design basis accidents. Frequency between 1 in 10,000 y and once in 1,000,000 y
3. Unlikely	10^{-4}–10^{-2}	Accidents that are not anticipated to occur during the lifetime of the facility. Natural phenomena of this probability class include: Uniform Building code-level earthquake, 100-y flood, maximum wind gust, etc. Frequency between one in 100 y and once in 10,000 operating years
4. Anticipated	10^{-2}–10^{-1}	Incidents that may occur several times during the lifetime of the facility (incidents that commonly occur). Frequency is one in 100 operating years

The consequences of events can be catastrophic, heavy, moderate or harmless. It is natural that the largest risks have the most probable incidents with catastrophic or heavy consequences. As is known, some variants of manipulation with weapons plutonium (Fig. 1) are considered now.

From the position of estimations of risk level for man and the environment, the safest way of manipulating weapons plutonium is its immobilization in stable forms with subsequent burial in geological formations.

Besides, increased requirements of safety are imposed on a repository, as it is a subject of danger for the present and future generations of mankind. So, the International Commission on Radiologic Protection (ICRP) recommends a possible dose commitment from a repository within the limits of 1-3 % from the regulated annual dose limit of 1 mSv, corresponding to a risk 10^{-6} year^{-1} (the number of fatal diseases of cancer per one year).

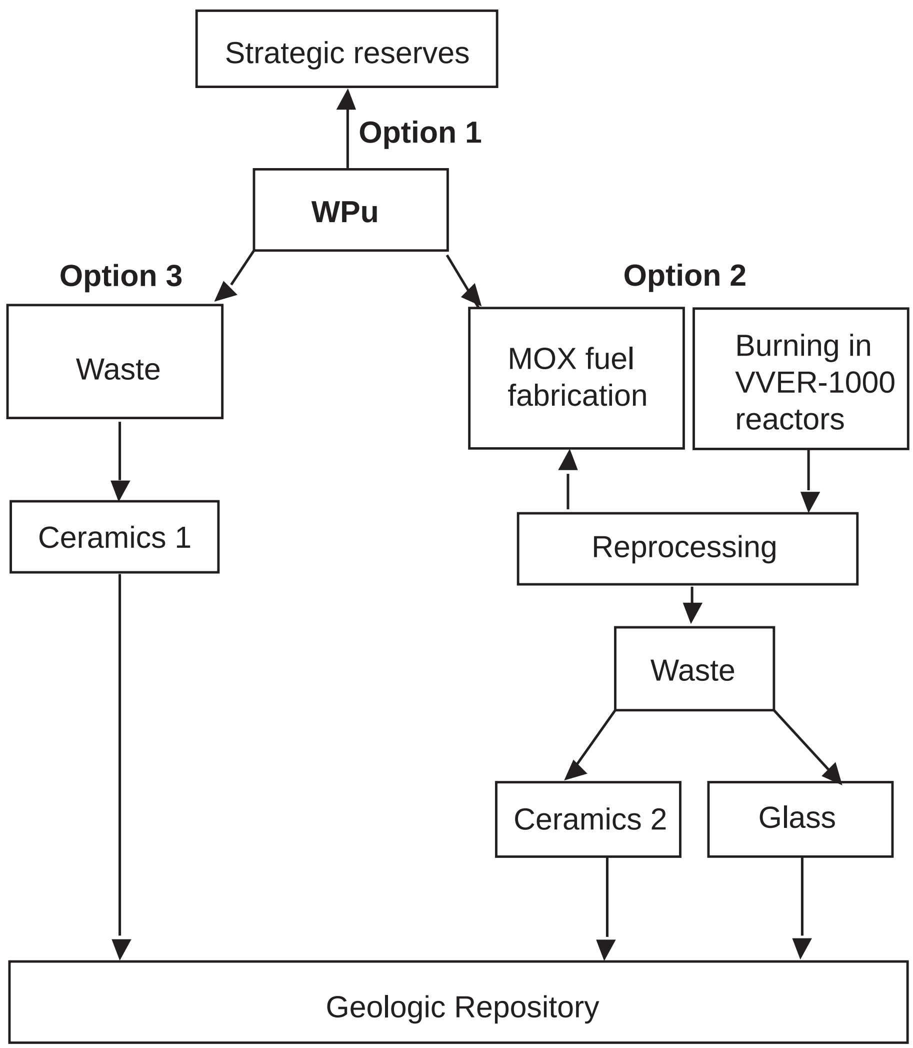


FIGURE 1. Variants of manipulation with weapons plutonium.

Recently, T. A. Gupalo (VNIPIPT) executed probabilistic calculations of a risk of plutonium release from a repository into a zone of active water exchange. As a criterion for estimation of safety, he used the allowable value of "release risk," which is equal to 10^{-6} year^{-1} by analogy with the allowable level of individual risk accepted in most countries. The period of risk analysis is accepted as equal to ten periods of plutonium half-life, , 250,000 years. The calculations were made in view of all the irregular situations that can take place during this period. He has shown that the "risk of release" of plutonium with a concentration exceeding the allowable one into a zone of active water exchange from vitrified plutonium (0.5 vol%) makes 1.2×10^{-9} year^{-1} (deep borehole) and 1.7×10^{-6} year^{-1} (mine working, depth 500 m). They are rather small and acceptable values. And the release of plutonium from ceramic matrixes must be much lower, taking into account that the leaching of plutonium from crystal ceramics (for example, "SYNROC_C") is lower by three orders of magnitude than from glass.

The interesting data concerning the risk for man and the environment from atomic engineering were reported by M. Dreicer et al. [1]. In a calculation for the population of 10 billion men for 100,000 years, the collective dose from all the stages of the Nuclear Fuel Cycle (NFC) makes 13 man·Sv/ (TW·hour).

Thus, the contributions in the dose formation will make:

Source	**%**
From NPP	17
From processing	79
From burial	1

For the same conditions, the risk from irradiation will make:

Cases per TW hour
0.65 fatal cases of cancer
1.56 unfatal diseases
0.13 changes of heredity
2.34 Total cases of harmful influence

Proceeding from these data, and also from the 1996 IAEA data that the electric power output was 2300 TW·hour, it is possible to calculate the number of fatal cases from the NFC as follows:

> If 1 TW·hour is expected to produce 0.65 fatal cases, then 2300 TW·hours is calculated as 1495 cases per 10^{10} population.

Thus, the risk from the NFC (the quantity of probable deaths per one year) will be 1.5×10^{-7}, and from the stage of waste burial: 1%–1.5×10^{-9}, i. e., the same value that is accepted by T. A. Gupalo for the risk of plutonium release from a deep borehole.

It is necessary to note that in itself the intention to immobilize plutonium (as well as other most biologically dangerous elements such as transplutonium elements and long-lived fission products) in a superstrong matrix on the basis of crystal mineral-like ceramics is caused first of all by reasons of long-term safety. It is known that the crystal state of a material is the steadiest in terrestrial conditions. It is fair also for a substance consisting radioactive elements. It is shown both by the presence in nature of crystal minerals holding radioactive elements in their structure during millions and even billions of years, and by numerous experiments on the synthesis of artificial minerals carried out in the recent years in the USA, Japan, Germany, Russia, Australia, and other countries just for the purpose of creating superstrong matrices for immobilization of the most dangerous long-lived radionuclides.

In Russia, besides the Radium Institute, research on the synthesis of crystal ceramics is carried out mainly for the immobilization of actinide-containing wastes at the RPA “Radon”, the Bochvar VNIINM, and some other organizations. The research activities are at the stage of laboratory experiments with a use of simulators of actinides. As a rule, ceramics of the titanate composition of the SYNROC type are synthesized, or some other phases (for example, monazite). Two methods of synthesis used are the “cold” crucible or hot pressing.

At the Radium Institute, the research and the experiments on ceramics, begun about 10 years ago, have essentially advanced in the last few years. The work is substantially related to the performance of the contracts, at first with Great Britain (BNFL), and then with the United States Department of Energy and Lawrence Livermore National Laboratory (LLNL), and also with participation in the linkage-grants assigned by the North Atlantic Treaty Organization for connections and partnerships between laboratories. Various methods are used for the preparation of initial materials and the synthesis of ceramics for immobilization of actinides in different forms of crystal mineral-like ceramics including: monazite, garnet, zircon, zirconium dioxide, etc.

In 1997-98, in connection with the contract with LLNL, experiments on the synthesis of ceramics for immobilization of surplus weapons plutonium are being carried out. For these purposes, the ceramic used is mainly zirconium on the basis of zircon ($ZrSiO_4$) and zirconium dioxide (ZrO_2). Samples with a 10% Pu content are synthesized.

2. Technology of Synthesis

We shall try to consider the basic stages of the technology of synthesis developed at the Radium Institute, from the viewpoint of safety problems.

The process for the manufacture of ceramics for immobilization of high-level wastes (HLW) as a rule is characterized by the following features:

- Use of dispersed powder materials as an initial precursor;
- Application of hot pressing for solid-phase synthesis at rather high temperatures $T \geq 1300$ °C;
- Multistage process of precursor preparation and the process of ceramics synthesis.

The production of ceramics for utilization of weapons plutonium as a whole inherits all the features of the process of HLW immobilization; however, it requires a number of additional rigid restrictions:

- Nuclear safety should be assured at all the stages of manufacture – from the precursor preparation to the synthesis of a final product;
- Number of stages should be reduced to a minimum;
- Application of hot pressing causes doubts from the point of view of safety. The replacement of hot pressing by sintering or melting is desirable;
- Necessity of ensuring a complete introduction of plutonium into the crystal lattice of the "host phase" as a solid solution, avoiding the formation of free separated phases of plutonium.

It is important also to note that the process of immobilization covers not only metal (weapons) plutonium, but also Pu-containing solutions and precipitates of complex chemical structure, accumulated as a result of manufacture of a nuclear weapon.

Thus, safety problems of the process of plutonium ceramics manufacture are exceptionally multifaceted. We have tried to consider them at two basic stages:

1. Preparation of the initial precursor;
2. Synthesis of ceramics.

3. Preparation of the Initial Precursor

3.1 PU-CONTAINING SOLUTIONS

In our practice, two ways of intermediate (preliminary) stabilization of Pu-containing solutions are used:

1. Gelation with obtaining a solid homogeneous gel, which further is calcinated at a temperature 400-500 °C with obtaining amorphous zirconium hydrosilicate (AZHS) [2]. AZHS differs by a high chemical stability and has a high capacity not only in relation to plutonium, but also to other elements, including neutron absorbers. AZHS is an initial material for the synthesis of ceramics on the basis of zircon $(Zr,Pu)SiO_4$ and zirconium dioxide $(Zr,Pu)O_2$.
2. Oxalate co-precipitation of plutonium with zirconium and gadolinium (neutron absorber), which allows us to ensure the maximum complete separation of plutonium from the solution as an insoluble deposit. The co-precipitated oxalates of Pu, Zr, and Gd can be palletized for direct synthesis of ceramics on the basis of cubic modification of $(Zr,Pu)O_2$ by the method of sintering or are subjected to plasma calcination with the purpose of obtaining a powder of solid solution $(Zr,Gd,Pu)O_2$ [3]. The simple calcination can be used for sintering the $(Zr,Gd,Pu)O_2$ powder into a strong steady ceramics on the basis of cubic $(Zr,Pu)O_2$ modification or (with addition of SiO_2) a biphase composition $(Zr,Pu)O_2/(Zr,Pu)SiO_4$. Thus, the plasma calcination of co-precipitated oxalates of Pu, Zr, Gd, . . . At the stage of preparation of the precursor allows us to avoid the problem of formation of plutonium free separated phases at the final stage of ceramics synthesis.

3.2 PU-CONTAINING DEPOSITS OF COMPLEX CHEMICAL COMPOSITION

The given material is characterized not only by chemical, but also by phase heterogeneity. The various phases have unequal reaction ability, i. e., during the synthesis of ceramics the preservation of the unreacted free phases of plutonium is rather probable. To increase the reaction ability of the precursor, it was offered to use an additive of surplus quantity of metal zirconium powder with a subsequent synthesis of polyphase ceramics on the basis of zircon/zirconium dioxide: $(Zr,Pu)SiO_4/(Zr,Pu)O_2$, as well as other phases of non-radioactive elements – impurities.

The other way to prepare the precursor is the addition of surplus quantities of yttrium, aluminium, gadolinium, gallium (in metal and oxide forms) for the further synthesis of ceramics based on garnet/ perovskite: $(Y,Gd,Pu,\ldots)_3(Al,Ga)_5O_{12}$/ $(Y,Gd,Pu,\ldots)AlO_3$ by the method of melting. At melting in a cold crucible, the use of the additives of metal powders provides the electric conductivity of the precursor necessary for successful induction heating.

3.3 METAL (WEAPONS) PLUTONIUM

The redundant weapons plutonium, according to the policy of the Minatom of Russia, cannot be buried as waste and should be used as nuclear fuel. Now creation of ceramic plutonium fuel is considered as an alternative to MOX fuel. The given ceramics is universal, i. e., can be used as a steady matrix of fuel and as a host phase for burial of actinides in deep geological formations.

We are studying the possibility of applying ceramics on the basis of the cubic modification of zirconium dioxide $(Zr,Pu)O_2$, taking into account that PuO_2 not only forms wide varieties of solid solutions with ZrO_2, but also is capable of stabilizing the cubic modification of zirconium dioxide.

The preparation of an initial material for the synthesis of ceramics in the case of weapons plutonium is not limited to any strictly certain methods. Plutonium metal can be transformed into oxide and used directly in the precursor, or is transferred into solution and co-precipitated as oxalates with zirconium. Another possible variant is a low-temperature (t~1000°C) melting of plutonium with a powder of zirconium metal with a subsequent oxidation of the alloy and obtaining a solid solution of $(Zr,Pu)O_2$.

4. Synthesis of Ceramics

Preliminary experiments that have been carried out at the Radium Institute have proved the possibility of a successful synthesis of plutonium-doped ceramics without the use of hot pressing. Now we are considering an application of methods of sintering and melting:

- Sintering—for synthesis of ceramics on the basis of $(Zr,Pu)O_2$ and $(Zr,Pu)SiO_4/(Zr,Pu)O_2$ for immobilization of Pu-containing solutions and weapons plutonium;
- Melting—for obtaining ceramics on the basis of garnet/perovskite for immobilization of plutonium precipitates of complex chemical composition.

In the case of melting, the reduction of the stages of ceramics manufacture seems an essential advantage in comparison with technologies of plutonium re-extraction from the precipitates with a subsequent processing by a separate scheme.

5. Conclusion

Thus, the production of ceramics is not now less safe than the manufacture of glass and can be considered as the most promising technology for immobilization of plutonium in the near future. It is necessary to take into account also that in itself ceramics are the safest and steadiest form for fixing of plutonium for a long period of time compared to the duration of geological processes.

References

1. M. Dreicer, V. Tort, M. Thieme, *Kerntechnik*, **62**, (10) pp. 34-39, 1996.
2. B. E. Burakov, K. B. Helean, E. B. Anderson, R. C. Ewing, (1997), "Amorphous Zirconium Hydrosilicate (AZHS) – a Prospective Material for Plutonium Fixation," *Transactions of the International Conference Plutonium Futures – the Science*, Santa Fe, New Mexico, USA, August 25-27, p. 21, 1997.
3. B. E. Burakov, K. B. Helean, V. A. Korolev, R. C. Ewing, E. B. Anderson, L. B. Shpunt, E. E. Strykanova, "Synthesis of actinide-doped Zirconia by plasma calcination," *Mat. Res. Soc. Symp. Proc.*, **506**, pp. 95-100, 1998.

SAFETY ISSUES OF U. S. CERAMIC PROCESS FOR EXCESS PLUTONIUM IMMOBILIZATION

CHIN W. MA
LESLIE J. JARDINE
GUY A. ARMANTROUT
Lawrence Livermore National Laboratory
Livermore, California 94551, U.S.A.

1. Introduction

The objective of this paper is to discuss the safety issues associated with the immobilization of excess weapons plutonium in ceramic form in the United States. The U.S. government has recommended a dual-track approach to dispose of excess weapons plutonium. According to this approach, about 33 metric tons of pure Pu will be fabricated into mixed oxide (MOX) fuels which will be burned in commercial nuclear light water reactors; and up to 17 metric tons of impure Pu will be immobilized into ceramic form which will be permanently disposed of in a geologic repository. It should be noted that a portion of the 33 metric tons of pure Pu may also be immobilized into ceramic form depending on the future decision of the U.S. government.

This paper first provides a brief description of the U.S. plutonium immobilization using ceramic processing. A systematic approach for identifying and solving possible technical safety issues is introduced in Section 3. Technical and other broad safety issues specifically associated with the ceramic processing for excess plutonium immobilization are summarized in Section 4. A brief conclusion is given in Section 5. It should be emphasized that the results are preliminary, since at this moment the details of the facility and equipment designs are not yet available.

2. U.S. Ceramic Process for the Immobilization of Excess Plutonium

This section provides a brief description of the following three topics: the proposed ceramic form to be used to immobilize the excess plutonium, the basic steps of plutonium immobilization, and the similarity between the ceramic process and the mixed oxide (MOX) fuel process.

2.1 CERAMIC FORM

The baseline chemical composition of the ceramic form used for plutonium immobilization is a titanite-based crystalline ceramic. Table 1 provides the elemental baseline composition, which indicates that the ceramic form consists of about 65 weight percent (wt%) ceramic precursors and about 35 wt% actinides. This ceramic form has the following special features:

- Based on chemically equivalent natural mineral analogs. These natural minerals have effectively immobilized actinides for hundreds of millions of years in wet environments. This is an indicator that this ceramic form is a durable geologic form which can be utilized for immobilizing excess U.S. plutonium;
- Contains about 20 wt% of neutron absorbers, HfO_2 and Gd_2O_3, which are used for criticality control;
- Contains about 25 wt% of U-238 to form the stable pyrochlore phase. The U-238 will also dilute U-235 (a fissile material which is a decay product from Pu-239) in the future to minimize the possibility of a criticality;
- Can accommodate various kinds of plutonium feed materials.

TABLE 1. Baseline chemical composition of ceramic form.

Composition	(wt%)	Remarks
TiO_2	35.9	Ceramic precursor
CaO	10.0	Ceramic precursor
HfO_2	10.6	Ceramic precursor, neutron absorber
Gd_2O_3	8.0	Ceramic precursor, neutron absorber
UO_2	23.7	Actinide
PuO_2	11.9	Actinide

2.2 BASIC PROCESS OF PLUTONIUM IMMOBILIZATION

The process for immobilizing excess plutonium consists of three phases as depicted in Figure 1.

2.2.1 Phase 1—plutonium conversion phase

The main objective of this phase is to convert plutonium feed materials into plutonium oxide for immobilization in Phase 2. The feed materials include clean and impure oxides, clean and impure metals and alloys, and various fresh fuels. The basic steps include:

- Receive and remove containers or fresh fuel cladding
- Convert metals to oxide using the HYDOX process
- Remove halides in impure oxide using halide wash
- Reduce the size of oxide powder to about 100 μm by milling.

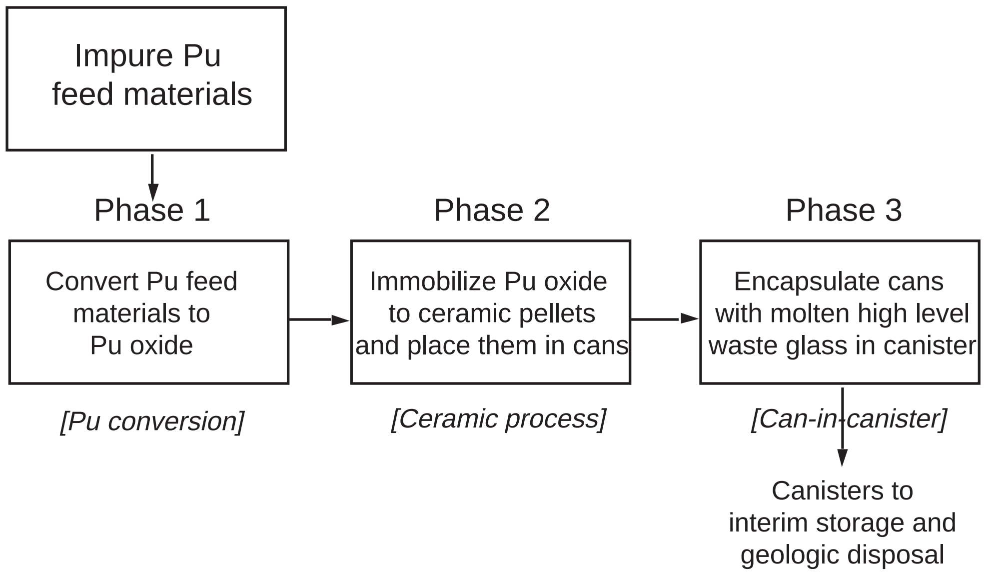


FIGURE 1. Three phases of U.S. plutonium immobilization.

2.2.2 Phase 2—ceramic process phase

The main object of this phase is to mix plutonium oxide with ceramic precursors to form ceramic pellets and place these pellets into cans. A more detailed description of this phase will be given in Section 2.3.

2.2.3 Phase 3—can-in-canister phase

The main object of this phase is to load cans containing ceramic pellets into canisters and fill these canisters with vitrified high level waste.

The end products of the plutonium immobilization process are these canisters. These canisters will first be sent to an interim storage facility with final disposition in a geologic repository.

2.3 BASIC STEPS OF CERAMIC PROCESSING

The main process of plutonium immobilization in the United States is the mixing of plutonium oxide with ceramic precursors to form ceramic pellets. These pellets will potentially immobilize the plutonium for hundreds of millions of years. The ceramic processing consists of five basic steps depicted in Figure 2.

- *Step 1—Dry milling.* In the first step, the PuO_2 and UO_2 powders are fed into a dry milling machine and the particulate size is reduced from about 100 μm to less than 20 μm. This small particulate size is needed later for the actinides to completely react with the ceramic precursors during the sintering process.
- *Step2—Dry blending.* In the second step, the PuO_2 and UO_2 powder are mixed uniformly with the ceramic precursors in a dry blending machine. This uniform

mixing is needed later for actinides to uniformly react with the ceramic precursors during the sintering process.

- *Step 3—Granulation*. In the third step, binder and lubricants are added and mixed with the ceramic materials, as a result the particulate size is effectively larger than 100 µm. The larger size is required produces flowable materials for pressing. It should be noted that water could be used as part of the binder and lubricants.
- *Step 4—Cold hydraulic pressing*. In the fourth step, the ceramic materials are pressed into pellets using a hydraulic press at room temperature. The pressure is about 1.5×10^7 Pa.
- *Step 5—Sintering.* In the final step, the pellets are placed in a sintering furnace and heated to about 1350°C for 4 hours (15 hours total cycle time). The final products are ceramic pellets. Each pellet contains about 50 g of plutonium, and has a diameter of about 6.5 cm and thickness of about 2.5 cm.

A proposed plant arrangement for the ceramic processing operation is depicted in Figure 3. The equipment is vertically arranged to aid powder transfer, and the total height of the arrangement is about 10 to 15 m tall.

The ceramic pellets will then be placed into a can. Each can holds about 20 pellets containing about 1 kg of plutonium. These cans will be loaded into a canister. Each canister holds 28 cans containing about 28 kg of plutonium. Finally molten vitrified high level waste will be poured around the cans and fill the canister. Figure 4 provides a view of the pellet, can, and canister.

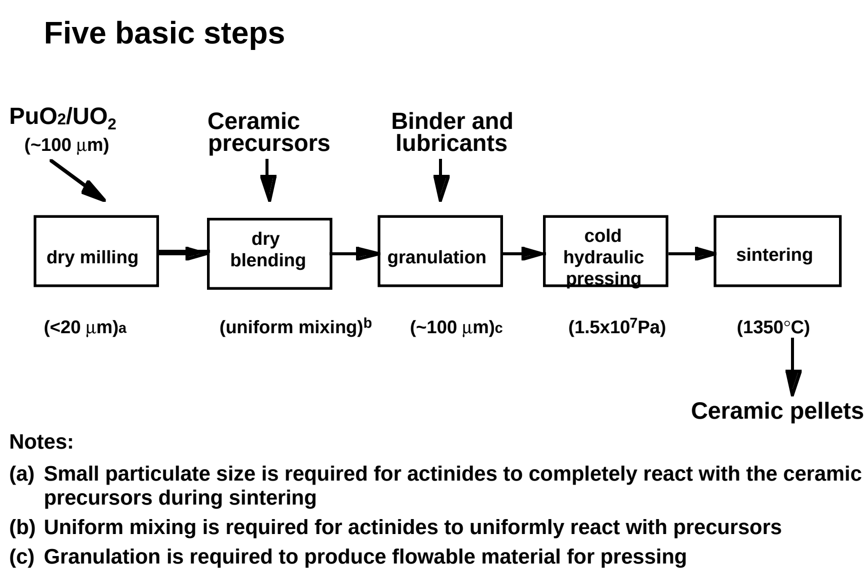


FIGURE 2. Basic steps of U.S. ceramic process.

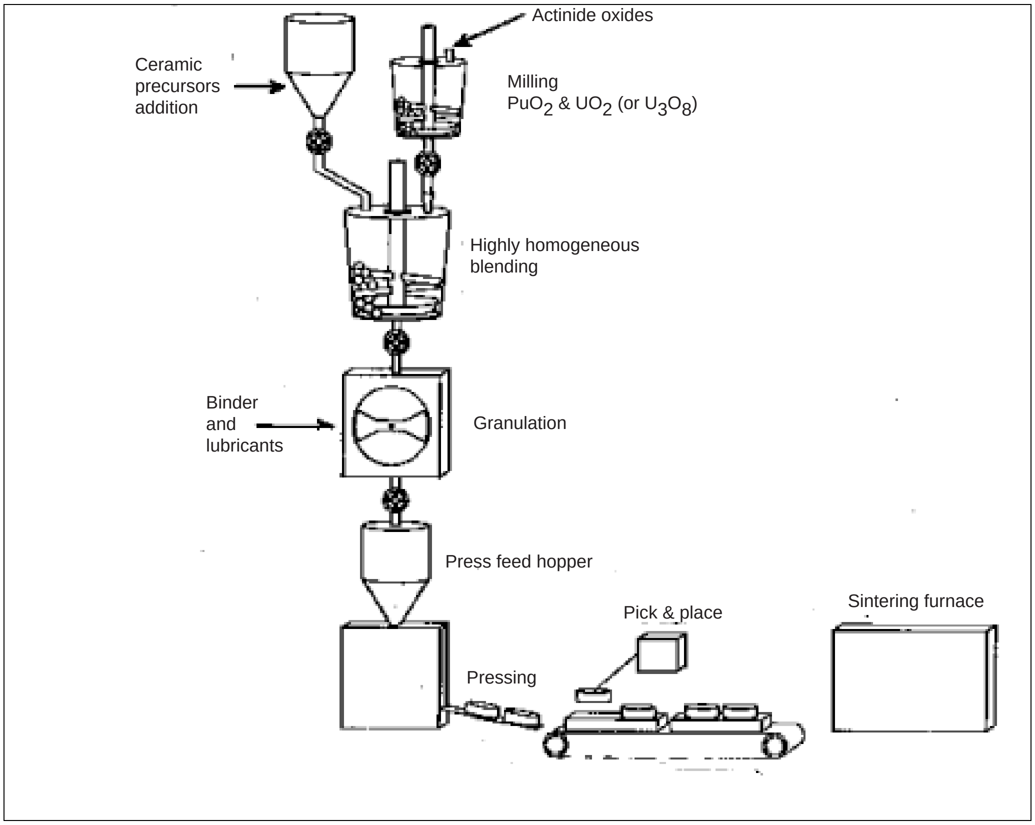


FIGURE 3. Proposed ceramic process plant arrangement.

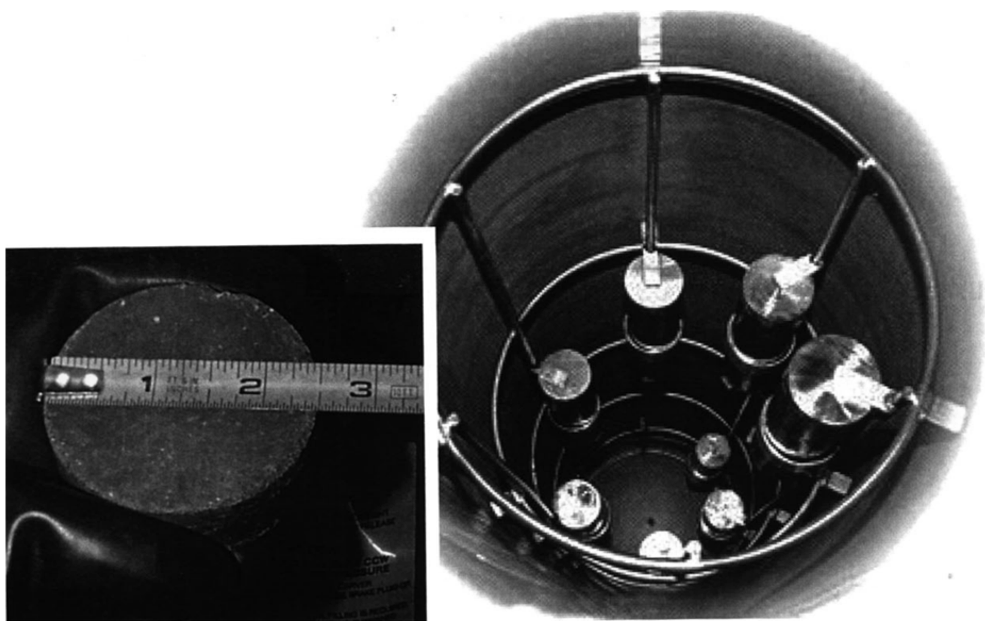


FIGURE 4. Can-in-canister immobilization: PuO_2 mixed with ceramic material forms a "puck" (L); pucks sealed in cans, embedded in canisters to be filled with vitrified HLW (R).

2.4 SIMILARITY WITH MOX FUEL PROCESS

The ceramic process described above is similar to the MOX fuel process. Actually, as far as safety is concern, several conditions and parameters of the ceramic process appear to be less stringent than those of the MOX fuel process. These similarities and differences are summarized in Table 2 and discussed below:

- *Milling*. Both processes involve milling. The milling requirement of PuO_2 particulates for the ceramic process is less than 20 μm, which is comparable to the 10 μm nominal requirement for the MOX fuel process. For plutonium oxide, particulate less than about 3 μm (corresponding to an aerodynamic equivalent diameter of 10 μm) are respirable. Thus under similar conditions, the potential inhalation dose associated with a spill accident of plutonium oxide powder for the ceramic process is no worse than that for the MOX fuel process.
- *Pressing*. The pressing pressure of the ceramic process is about 1.4×10^7 Pa which is about 30 times smaller than the pressing pressure of the MOX fuel process. Therefore, from the operational point of view, the ceramic process has a less severe condition.
- *Sintering*. The ceramic process uses air or argon gas in sintering while the MOX fuel process uses 4% hydrogen and 96% argon gases. Therefore the chance of explosion is not a concern for the ceramic process. In addition, the sintering temperature of the ceramic process is also less severe than that of the MOX fuel process.
- *Tolerance in pellet dimension*. The MOX fuel process has a very tight control in pellet dimension; the tolerance is 0.025 mm. After sintering, the pellets require grinding to meet the high standard. The ceramic process on the other hand has a large tolerance in dimension, and pellet grinding is not needed. As a result, the chance of creating respirable particulates is further reduced in the ceramic process.

MOX fuel facilities have been operated safely in several European countries for many years. The above comparison shows that the ceramic process and the MOX fuel process are similar in many ways; thus the ceramic formation processing facility can be expected to operate as safely.

TABLE 2. Similarity between ceramic process and MOX fuel process.

Item for Comparison	**Ceramic Process**	**MOX Fuel Process**
Milling	< 20 μm	< 10 μm
Pressing	$\sim 1.4 \times 10^7$ Pa	$\sim 4 \times 10^8$ Pa
Sintering	air/Ar (1350°C)	H_2/Ar (1650°C)
Tolerance in pellet dimension	~2.5 mm; no pellet grinding	~0.025 mm; requires pellet grinding

3. Systematic Approach of Identifying and Solving Safety Issues

After a brief description of the U.S. ceramic immobilization process, this section will describe a systematic approach of identifying and solving safety issues. In general, three types of hazards and accidents are considered in a safety analysis:

1. Operational hazards and accidents
2. External accidents
3. Conventional industrial hazards and accidents.

3.1 OPERATIONAL HAZARD AND ACCIDENT ANALYSIS

The hazard and accident analysis associated with normal operations is briefly discussed. For illustration, simple examples associated with the U.S. ceramic process will be used. The hazard analysis can be divided into four parts:

3.1.1 Part 1—hazard identification

The first part in the analysis is to identify hazards associated with the operations. First, the facility is separated into different areas according to their function (called functional areas). For example, a ceramic process facility can be separated into shipping and receiving area, ceramic process area, storage area, etc.

Second, the hazards in each functional area are identified using the help of a check list. Various typical and common hazards are provided in the check list for reference, such as radiological material, toxic material, combustible material, energy source, etc.

3.1.2 Part 2—accident event characterization

The second part in the analysis is to characterize the accident event. This is accomplished by the following four steps:

1. *Develop accident scenarios*
   The existence of a hazard merely implies that there is a potential for an accident. Something has to go wrong before an accident could occur. Therefore one has to develop various accident scenarios for each hazard. These scenarios usually involve human error and/or equipment failure. For example, an operator drops a container containing plutonium oxide, and the container cracks, causing the release of airborne PuO_2 particulates.
2. *Estimate the source term qualitatively*
   In the above case, the source term means how much airborne PuO_2 particulates are released to the atmosphere due to the drop of a container, what is the respirable fraction of the airborne PuO_2 particulates.
3. *Estimate the probability of occurrence qualitatively*
   In the above case, it means what is the likelihood for the operator to drop a container and the likelihood for the container to crack.
4. *Estimate the consequence qualitatively*
   In the above case, the consequence means the radiological dose to the workers and public due to the inhalation of the respirable airborne PuO_2 particulates.

In these four steps, one tries to identify and develop all reasonable accident events, whether large or small. The estimations need only be qualitative; quantitative evaluations will be performed later if necessary.

3.1.3 Part 3—hazard controls

The third part in the analysis is hazard control, which includes the prevention, mitigation, and detection of an accident. Prevention features include engineering design and administrative control. In the above case it means designing a safe transfer and handling system as well as operators training.

Mitigation of an accident means reducing the consequence after an accident has occurred. In the above case several mitigating features could be effective, such as restricting the container transfer height, having a robust container design, and installing high efficiency particulate air (HEPA) filters.

Hazard control also requires an accident detection system. In the above case it requires an area airborne monitoring system for the detection of airborne plutonium particulates.

3.1.4 Part 4—hazard screening

The hazard analysis may identify many potential accident events. Some may be minor and some significant. For those minor accidents, one can stop here and no further analysis is required. If the hazard analysis indicates that the risk of an accident event is significant, then a detailed quantitative accident analysis will be performed for that accident. The hazard screening allows the major effort of safety analysis to be concentrated on those accidents that are significant.

The accident analysis follows basically the same approaches described in Parts 2 and 3 except that it is quantitative and it requires more detailed information on facility designs and operations. The accident analysis includes development of scenarios, evaluation of probability of occurrence, calculation of source terms, and calculation of consequence. Based on the analysis, a set of safety requirements is developed for accident controls. Because of the time limit, the details will not be discussed here.

3.2 EXTERNAL ACCIDENT ANALYSIS

In addition to operational accidents, external accidents also require detailed study. Examples are earthquake, extreme wind, flooding, aircraft crash, external fire, etc. Because of the time limit, the details will not be discussed here.

3.3 CONVENTIONAL INDUSTRIAL HAZARDS

Conventional industrial hazards involve other kinds of safety issues, which are not unique to a nuclear facility. An example of conventional industrial hazard is an operator falling from a ladder. It should be noted that in some cases, industrial hazards receive less attention and thus cause more problems. Because of the time limit, the details will not be discussed here.

4. Summary of Safety Issues

Based on the pre-conceptual designs of the U.S. ceramic process, a summary of the technical safety issues is given below. It should be emphasized that the results are preliminary and generic because the facility and equipment design details are not yet available.

- *Criticality*. Plutonium is a fissile material, thus it requires criticality control for various operations, such as transferring and milling plutonium oxide, blending plutonium oxide with ceramic precursors and their granulation, sintering of ceramic pellets, interim storage, and permanent disposal. Criticality control involves control of factors and parameters such as mass, moderators, reflectors, geometry, and neutron absorbers.
- *Radiation*. Another safety issue for the U.S. ceramic process is radiation which includes gamma, alpha, and neutron radiation from plutonium and Am-241. Automation and remote handling of materials as well as adequate shielding are required to protect workers from this radiation.
- *Dust control*. Small plutonium oxide particulates may become airborne and result in inhalation dose. Particulate size distribution at various stages and aerosol transport are subjects for careful study. Equipment and process designs can be used to control airborne particulate, such as setting up various pressure control zones in ventilation design, installing HEPA filtration systems, and installing area airborne particulate monitors.
- *Spill of Pu oxide powder*. Spills of plutonium oxide powder by human error or equipment failure are always a safety concern. Examples are dropping a container containing Pu oxide, or leaky seals in the Pu oxide process system. Adequate protection systems to protect against accidental spills or leakage are mandatory.
- *Fire*. Fire protection to prevent and control a facility fire is always an important safety issue. Items for hazard control are combustible materials, ignition sources, fire-fighting water as moderator for criticality, leakage of hydraulic press oil, etc.
- *Worker radiation exposure during repair and maintenance*. The equipment designs in the U.S. ceramic process are highly automated and remotely controlled. They require regular maintenance and repair. Reducing exposure to workers from background radiation during repair and maintenance demands careful planning.
- *Emergency planning*. Emergency planning in case an accident occurs must be carefully prepared. The response of the Fire Department and the possible evacuation of workers and public are important safety issues.

In addition to the above mentioned technical safety issues, other broad issues related to safety are safety regulations and safety management. For example, safety regulation needs to set up the accident dose guideline. Safety management needs to define management responsibility and addresses operator training, job planning, etc. These issues are all closely related and together they form an integral part of safety.

5. Conclusions

Based on the above discussion, some conclusions are given below:

- The U.S. ceramic process for plutonium immobilization is relatively simple. Most of the equipment is standard, such as milling, blending, pressing, and sintering machines. It is expected that equipment failure and human error during operations will not be frequent.
- Most processes are dry, therefore criticality control will be straight forward.
- The ceramic process is similar to the MOX fuel process that has been in operation for many years, thus providing substantial safety experience.
- The timing is right for the nuclear materials safety initiative. Both the U.S. and Russia are starting to design and construct MOX fuel and plutonium immobilization facilities. It is suggested that safety analysis should be initiated as early as possible, so that safety concerns can be incorporated into facility and equipment designs in the early stages.

SAFETY PROBLEMS RELATED TO THE OPERATION AND SHUTDOWN OF RADIOCHEMICAL PRODUCTION

YU. A. REVENKO
YU. P. SOROKIN
N. N. SERGEYEV
MCC-K-26
53, Lenin Str., Krasnoyarsk Region, Zheleznogorsk, 662990 Russia

First allow us to express our thanks to the organizers of this second seminar, which is a natural continuation of the first seminar, for the opportunity to exchange information with our Russian and foreign colleagues on matters of safety assurance during work with nuclear materials.

In our paper, I shall briefly describe our radiochemical plant and our conceptual approach to safety assurance, and point out some of the problems related to further cooperation between Russia and S1Zh.

Radiochemical production was launched in 1964, and has run accident free to date. The plant is located within a rock massif composed of solid rock belonging to stable and extremely stable formations. The rocks are characterized by an extremely low water content. The radiochemical plant was designed to process irradiated standard uranium slugs from a nuclear reactor in order to separate uranium and plutonium.

The plant includes a uranium-slug processing complex and a complex for processing and temporary storage of medium- and high-level liquid waste and for preparing it for underground burial. The process flow diagram for the uranium slugs involves dissolving them and processing solutions of uranium, plutonium, and uranium fission products by using the normal Purex process, but with the difference that a noncombustible diluent/extracting agent that is safe from the standpoint of fire and explosion was chosen. The plant's process flow diagram makes use of sorption processing of process products by employing anion and cation exchange resins, processes of boiling-down, oxalate and alkaline precipitation, and electrochemical reduction, and so forth.

In the medium- and high-level waste-processing complex a considerable amount of radioactive suspensions and sludges containing, according to a rough estimate, approximately 600 kg of uniformly distributed chemical compounds of plutonium has been accumulated as a result of more than 30 years of basic production. At the present time, a body of research, planning, design, and practical work is under way on the extraction of sludges from storage tanks and on processing them in a department specially formed for this purpose. Sludge hoisting and processing are among the basic premises underlying the concept for decommissioning the radiochemical production facility. This work may take more than 10 years.

The plant possesses a duly issued license for types of activity related to processing of nuclear materials and the equipment of temporary storage facilities for medium- and high-level liquid waste, and to the handling of radioactive substances as individual units of the temporary storage facility for medium- and high-level waste are decommissioned. It holds a temporary permit from the Russian Federation State Academy of Sciences for all types of activity.

1. The Conceptual Approach to Safety Assurance at the Radiochemical Production Facility

The conceptual approach to safety assurance during work with fissile materials is based on the following premises:

- Accident prevention;
- Prevention of unregulated handling of nuclear materials;
- Accounting and monitoring.

The comprehensive safety-management system includes the activation of the following safety systems and special measures:

- Process safety;
- Nuclear safety;
- Radiation safety;
- Fire and explosion safety;
- Activation of the special emergency quick-reaction system;
- Observance of the requirements for accounting, control, and physical security of fissile materials;
- The operation of the personnel-training and -certification system;
- Environmental safety.

Each of these systems includes its own specific task, and coordination of work on the management of these systems is assigned, by combinewide and plantwide orders, to the chief engineer and his/her deputies, and is set forth in job descriptions.

2. Process Safety

- Ranking of manufacturing processes and equipment by their danger;
- Imposition of special requirements on processes and equipment related to nuclear safety;
- Implementation of the manufacturing process according to the plan of the general design organization;
- Introduction of new technologies only after processes are modeled and the recommendations of scientific organizations have been made;
- The use of process equipment as intended;
- Observance of the requirements of process standards and rules;
- Observance of the standards and requirements for safe operation of equipment;
- The completeness of the representation of the consequences of proposed deviations from the normal course of the manufacturing process;

- A clearly organized system of personnel actions (to be taken by the operator of the nuclear installation) in case of deviations from the normal course of the manufacturing process;
- Monitoring of the course of the manufacturing process, and the presence of systems to signal deviations in the process and in hardware (interlocks that prevent the development of any deviation, detection of the initial events of accident situations);
- Implementation of a set of accident-prevention measures;
- Implementation of a set of scheduled preventive-maintenance measures;
- Analysis of deviations in the manufacturing process from the standpoint of safety, and close review of materials of the analysis together with personnel;
- Observance of the prescribed procedure for starting and shutting down production, and the execution of permit documents.

3. Nuclear Safety

The organization of nuclear-safety work is built on observance of the requirements of the "Basic Industry Nuclear-Safety Regulations," and also in accordance with the requirements of the "Regulation on the Nuclear-Safety Service and the Organization of Work to Assure Nuclear Safety at the Radiochemical Plant" and a number of instructions and regulatory documents of the enterprise that deal with nuclear-safety assurance at the radiochemical production facility and nuclear-materials storage and transportation.

The main areas of work on nuclear-safety assurance are based on observance of the following:

- Substantiation of nuclear safety during the implementation of processes and the use of equipment, with a finding by the State Science Center of the Physical Energy Institute;
- The actions of the Nuclear-Safety Service, which is headed by the deputy chief engineer;
- Performance of expert review and the consent process with the manager of the service responsible for engineering decisions related to a change in technology and the intended function of equipment in sectors where there is a nuclear hazard;
- Performance of weekly inspection checks of the observance of nuclear-safety requirements by personnel, with preparation of corrective orders;
- The work of the Nuclear-Safety Service according to the plan and measures to improve the state of nuclear safety, and holding of regular meetings;
- Instruction and training of personnel in sectors where there is a nuclear hazard, and of managers and specialists, and certification of them;
- Analysis of the state of nuclear safety, and preparation of reports twice a year;
- A regular annual check of the state of nuclear safety at the plant by a board of the combine, and once every 3 years by the board of the State Science Center of the Physicotechnical Institute and the Committee for Safety, Ecology, and Emergencies;

- A periodic comprehensive and targeted check of the state of nuclear safety by a board of Russian Federation Gosatomnadzor [the Federal Inspectorate for Nuclear and Radiation Safety]. It should be noted that thanks to the well-honed system for nuclear-safety assurance, there has not been a single instance of a self-sustaining chain reaction at the radiochemical plant in its entire operating life (more than 34 years), and there have been no cases in which safe parameters were exceeded with respect to the mass or concentration of fissile material in equipment.

4. Radiation Safety

Work to assure radiation safety is organized according to the Standard Regulation on the Radiation-Safety Service of an Enterprise, and is based on observance of the requirements of the *Sanitation Standards for the Design of Enterprises and Installations of the Atomic Industry* (*SNP-77*), the *Radiation-Safety Standards* (*NRB-96*), and the *Basic Sanitation Regulations for Working With Radioactive Substances and Other Sources* (*SSP-72/87*).

The head of the Labor-Protection and Safety Service is the manager of the Radiation-Safety Service. The Radiation-Safety Service works according to the plan and measures for improving radiation safety at the plant.

During operation, to assure radiation and general safety procedures, allowance was made for the requirements of regulations for operating personnel, the layout of equipment and service lines, sealing of equipment, remote control during the manufacturing process, and mechanization of repairs that are imposed on radiochemical production by federal and industry regulatory documents.

Radiation (or biological) protection of the manufacturing process, three zonation, and hardware treatment canyons ensure radiation levels mandated by standards and regulations, and fall within the limits set by environmental-protection authorities.

The radiation situation in production is monitored by special laboratories. The scope of radiation monitoring is defined by the standards of the enterprise and meets the requirements of radiation-safety standards.

5. Accounting, Control, and Physical Security of Nuclear Materials

Accounting and control of nuclear materials are handled according to the requirements of the enterprise's normative documents.

Fissile-materials accounting is handled in quantitative and weight terms, and there also is an accounting of nuclear materials as physical assets. The nuclear-materials accounting system makes it possible to assure the integrity of nuclear materials at storage sites and in production, the proper documentation of all operations involving the movement of nuclear materials, acquisition of data on remaining nuclear materials at storage facilities and in unfinished production. The journal type of accounting is used at the radiochemical plant. Nuclear materials are inventoried monthly by standing commissions.

To assure the safety and physical security of nuclear materials, the security of data, and the organization of security, restricted-pass access, and the intrafacility regime, the plant has an operating Safety and Documentation Group, which is an independent structural unit.

The radiochemical plant, like the production unit of the combine, where basic manufacturing processes involving nuclear materials are carried out, has several security lines.

The most important elements of the security of special fissile materials are the technical means of controlling their unauthorized movement that are in place at the plant. The physical-security systems are now undergoing technical re-equipment according to the new concept.

6. Environmental Safety

To prevent emissions or discharges of hazardous substances, all systems of process units are equipped with gas-purification assemblies that act as barriers, to reduce emissions to safe levels, not only in normal operation but also in case the manufacturing process is disrupted.

Gas and aerosol emissions of radionuclides and hazardous chemicals from the radiochemical plant are monitored continuously.

The calculated external irradiation doses do not exceed 0.02 mS/year at a distance of approximately 1 km to the northeast of the plant, and for other directions these figures are even smaller. These values are significantly below those permitted under *NRB-96* (1 mS/year).

The general goal of the handling of radwaste of different levels and composition is to eliminate the harmful effect of ionizing radiation on human health and the environment. At the plant this goal is attained through the fulfillment of the following set of tasks:

- Development and implementation of methods of extracting high- and medium-level waste from the tanks of storage facilities;
- Development and implementation of conditioning methods for waste, including waste containing plutonium;
- Ensuring the safe storage of accumulated and conditioned waste; and
- Ensuring reliable burial of radwaste.

7. Fire and Explosion Safety

The plant's fire-safety system fully meets the requirements of the *Fire Standards and Regulations for the Design of Industry Extraction Facilities*, which the deputy minister for medium machine-building approved on June 30, 1986, and which were consented to by the heads of main administrations and the Main Administration of Fire Protection of the USSR Ministry of Internal Affairs.

Fire protection is assured by a set of organizational and technical measures, by observance of process regulations in the use and storage of both radioactive substances and other materials, and by meeting the requirements of fire-prevention and fire-control operational documents. The plant's fire protection is provided by a militarized fire department of the Russian Federation Ministry of Internal Affairs, and also by a volunteer fire brigade.

The fire and explosion safety of processes is assured by process parameters and scheduled preventive measures, and has been confirmed by findings of VKAKhZ.

8. Emergency Preparedness

The design of equipment, the layout of enclosed areas, and the positioning of equipment in them take into account the requirements of the "echeloned barrier" system:

- Manufacturing processes are implemented with sealed equipment—the first barrier. The cask (container) is the first barrier for the plutonium dioxide in storage;
- Safety barrier 2 consists of the enclosed areas in which all possible sources of harmful substances (hardware treatment canyons, pipe galleries, etc.) are located;
- Barrier 3 consists of repair areas;
- Barrier 4 is the mine workings of the plant;
- Barrier 5 is the rock massif in which the mine workings of the radiochemical plant are located.

Prevention of the escape of radioactive substances and chemicals in aqueous solutions into the environment is accomplished by setting up processing of aqueous process and drain solutions in the plant's process flows. The plant uses monitored discharge of the water used to cool apparatus, by employing automatic interlocks in the event of a hardware leak. The content of radionuclides in harmful chemicals and waste water is determined by collecting and later analyzing samples.

The handling of gas and aerosol emissions consists in maintaining reduced pressure in enclosed areas where possible sources of harmful substances are located, relative to the areas of the second and third safety barriers. This decision prevents the entry of radionuclides into industrial areas in case there is some leakage from process equipment.

With respect to assuring the reliability of electric-power supply, the plant's main consumers are classified as category I loads. A gas-turbine power plant is used as a self-contained emergency power source; it is an independent power supply for the special group of category I loads whose continuous operation is essential for non-emergency shutdown of basic production in the event of loss of voltage from the power system.

An aggregate specialized civil-defense team has been assigned to perform rescue work and other urgent work during accident-recovery operations. In the event of an accident, a personnel-evacuation plan is put to work at the enterprise, and measures are arranged for decontamination of the grounds, structures, and equipment.

A special emergency team has been formed and is in operation to isolate and correct emergency situations. The technical equipment of the civil-defense forces is satisfactory. The preparedness and professional skills of the personnel of the emergency team and an analysis of the training sessions and classwork demonstrate their ability to handle the assigned tasks.

During the life of the enterprise, there have been no instances of emergency situations, and no outbreaks of fire.

9. Personnel Qualifications

To maintain the preparedness of the plant's personnel to take actions in an emergency, a set of measures have been drawn up and put in place to ensure that the personnel maintain a constant state of training. Training schedules are drawn up annually, and training sessions are held for operational personnel.

Personnel employed in sectors where there is a nuclear hazard receive instruction, have their knowledge tested, and participate in accident prevention and control training sessions according to the requirements of the *Basic Industry Regulations.* (*PBYa-06-00-96*) and industry safety regulations.

10. Conclusion

Inasmuch as safety in nuclear-materials handling is a common task for all countries that possess nuclear materials, we believe it advisable to continue cooperation in the following areas:

- Exchange of information on incidents and accidents that have occurred;
- Work to improve the management of safety systems;
- Exchange of instruction programs and advanced-training programs for personnel;
- Exchange of information on the technology for decontamination and processing of accumulated waste and on conceptual approaches for decommissioning of radiochemical production facilities; and
- Improvement of the nuclear-materials security, control, and accounting system, and computerization of nuclear-materials accounting methods.

THE PROBLEM OF FIRE AND EXPLOSION SAFETY IN RADIOCHEMICAL PRODUCTION PROCESSES

YE. R. NAZIN
G. M. ZACHINYAYEV
Military Academy of Chemical Protection
Moscow, Russia

G. F. YEGOROV
Institute of Electrochemistry of the Russian Academy of Sciences
Moscow, Russia

1. Introduction

The problem of the fire and explosion safety of radiochemical production processes is a comprehensive one; its main components are shown schematically in Figure 1.

In principle, each specific manufacturing operation must be analyzed to determine potential hazards in the gas or condensed phase. If they are present, experimental research or calculations are carried out, and safe conditions for conducting manufacturing operations are recommended on the basis of the information obtained. To make a probabilistic assessment of the occurrence of accident situations and their aftermath, one must determine the conditions under which a potential hazard may be realized, as well as the characteristics of combustion, decomposition, or explosion processes.

Analysis of radiochemical production operations shows that many of them are potentially hazardous, either because of the formation of combustible gas–vapor–air mixtures or because of intense oxidation processes in mixtures of combustible substances with oxidizers, with the release of gaseous products and heat. Work experience from radiochemical production operations shows that potential hazards may be realized.

The evaluation of the fire and explosion safety of gas and vapor–air mixtures does not pose any serious complications, and there are standard methods of determining fire-hazard characteristics and reliable engineering methods of calculating them. To date, it does not appear to be possible to use computational methods to assess the characteristics of the thermochemical decomposition of industrial mixtures of combustible substances with oxidizers or their detonation capacity. Experimentation presently remains the sole source of information on their dangerously explosive properties.

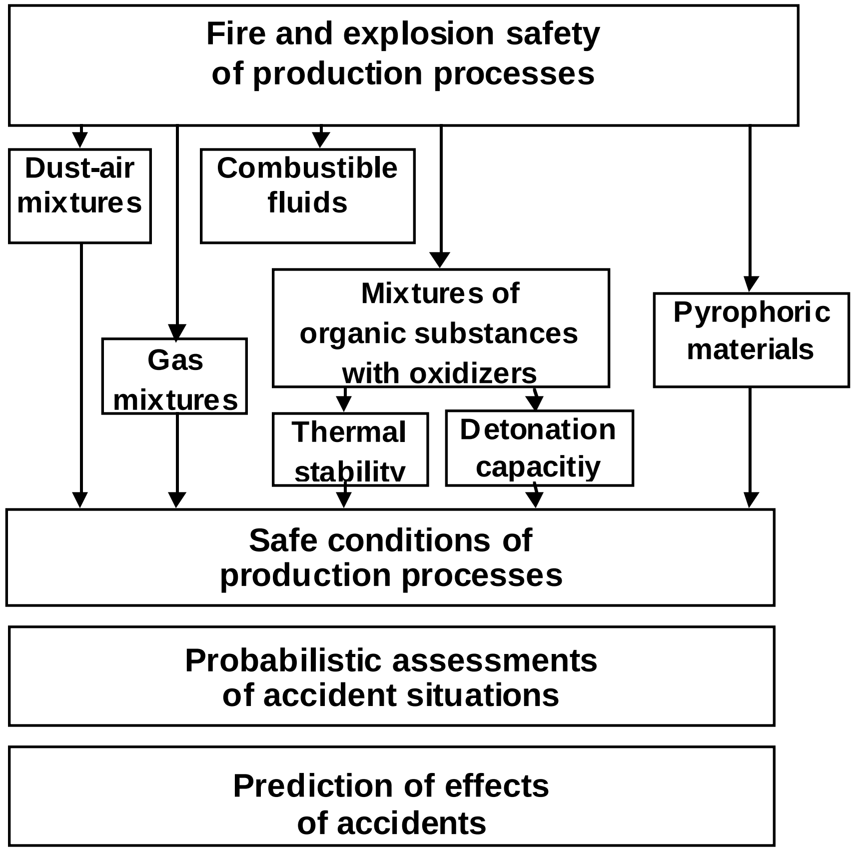


FIGURE 1. Basic components of the problem of the fire and explosion safety of radiochemical production processes.

The mixtures of combustible substances with oxidizers that are used in manufacturing operations may be classified as extremely weak explosive compounds, and their initiation under production conditions is unlikely.

Exothermic processes of oxidation in mixtures of organic substances with oxidizers pose the greatest real danger. This is true above all of mixtures used in extraction and sorption processes. The potential hazard of these mixtures increases in the presence of radionuclides, both as a result of the formation of highly reactive products of the radiolysis of the organic components of mixtures, and as a result of the possible uncontrolled heating of mixtures due to the heat of radioactive decay of the radionuclides.

At the present time, in Russian radiochemical production processes solutions of tributyl phosphate (TBP) in hydrocarbon diluents are used most extensively in extraction processes, while the anionite VP-1AP is used in sorption processes. Available information on the thermochemical decomposition of these reagents with nitrate oxidizers makes it possible to gain some idea of the potential danger of the processes that occur in them.

The experimental results provide grounds to believe that the processes of interaction of components of extraction and sorption mixtures, accompanied by gas evolution, occur in two stages, depending on the temperature.

For source mixtures used for extraction and sorption, gas evolution in the first stage, which takes place in the temperature range 70–100°C, is due to the oxidation of admixtures and/or products of the chemical degradation of the extractant and sorbent. The rate of gas evolution peaks and then decreases to low, roughly constant values. To date there are no data on whether heat is evolved during oxidation processes in this stage.

The second stage of interaction of the components of the source extraction and sorption mixtures begins at temperatures above 120–130°C. It is due to the oxidation of extractant or sorbent molecules directly, and generally is accompanied by exothermic effects and energetic gas evolution.

The occurrence of "secondary" oxidation processes, accompanied by exothermic effects and energy gas evolution, depends on the ratio between heat input in the reaction zone and heat removal from it.

In open vessels, where there are significant heat losses to evaporation of the aqueous phase, the conditions for the occurrence of exothermic processes are unfavorable. This apparently is why exothermic processes and a sharp increase in the rates of gas evolution were not observed in numerous experiments on the heating of two-phase mixtures of extractant and anionite with HNO_3 in concentrations up to 14 mol/L of HNO_3 at temperatures up to 100–110°C (up to the boiling point of HNO_3).

Weak self-heating in the organic phase of TBP, taken separately, with extracted HNO_3 was observed at temperatures above 140°C, but to date no systematic research has been conducted in this area.

In open vessels the anionite VP-1AP in nitrate form and mixtures of it with nitric acid decompose energetically in the form of a thermal explosion only after all or a large part of the aqueous phase is removed. The nature of the decomposition of these mixtures is shown in Figure 2.

The "starting" temperatures of a thermal explosion are 220–240°C for the anionite in nitrate form, ~150°C with the hexanitrate complex, and 130–150°C for mixtures with HNO_3 with a concentration of 3–12 mol/L. Irradiation lowers the "starting" temperatures of thermal explosion; in particular, for a mixture of anionite VP-1AP in nitrate form with 7 mol/L of HNO_3 irradiated with a dose of 5×10^8 rads, the "starting" temperature of thermal explosion is ~115°C.

In closed vessels heat removal from the reaction zone by evaporation of the aqueous phase is significantly lower, and losses of the oxidizer with gaseous reaction products are lower, so that the conditions for the occurrence of exothermic processes appear to be more favorable than in open vessels.

With weak heat exchange with the environment, experiments in closed vessels showed that in one- and two-phase mixtures of TBP with HNO_3 exothermic processes begin at temperatures above 120°C. They already take place when the HNO_3 concentration in the organic phase is 1.6 mol/L, which corresponds to an equilibrium concentration of HNO_3 in the aqueous phase of 2.1 mol/L. As we see from Figure 3, with increasing HNO_3 concentration in TBP the amount of self-heating increases from 20–40°C to 160–180°C; in mixtures with an HNO_3 concentration above 2.8 mol/L, the amount of self-heating increases slightly. It turned out that, practically in the absence of HNO_3, extracted uranyl nitrate is an effective oxidizer of TBP at a relatively low temperature (~150°C) (Fig. 4).

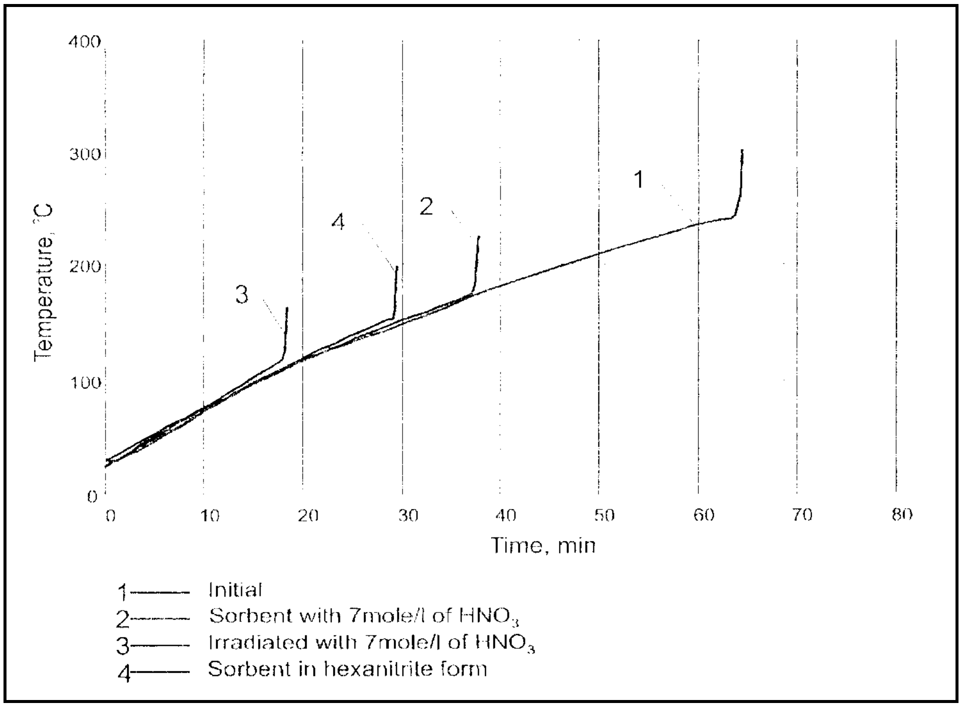


FIGURE 2. Change in temperature of air-dried samples of sorbent VP1-AP in nitrate and hexanitrate form during heating in open vessels.

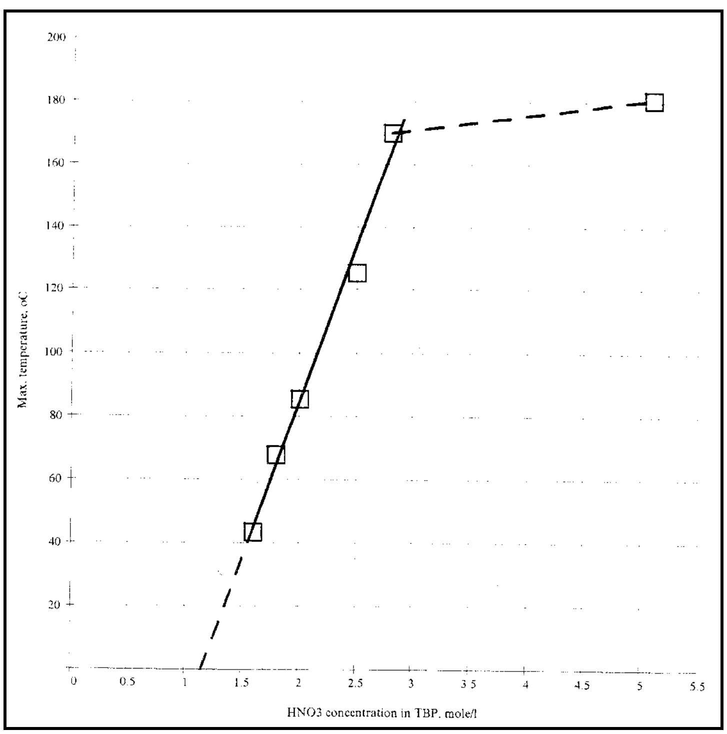


FIGURE 3. Maximum self-heating temperature vs. concentration of extracted nitric acid.

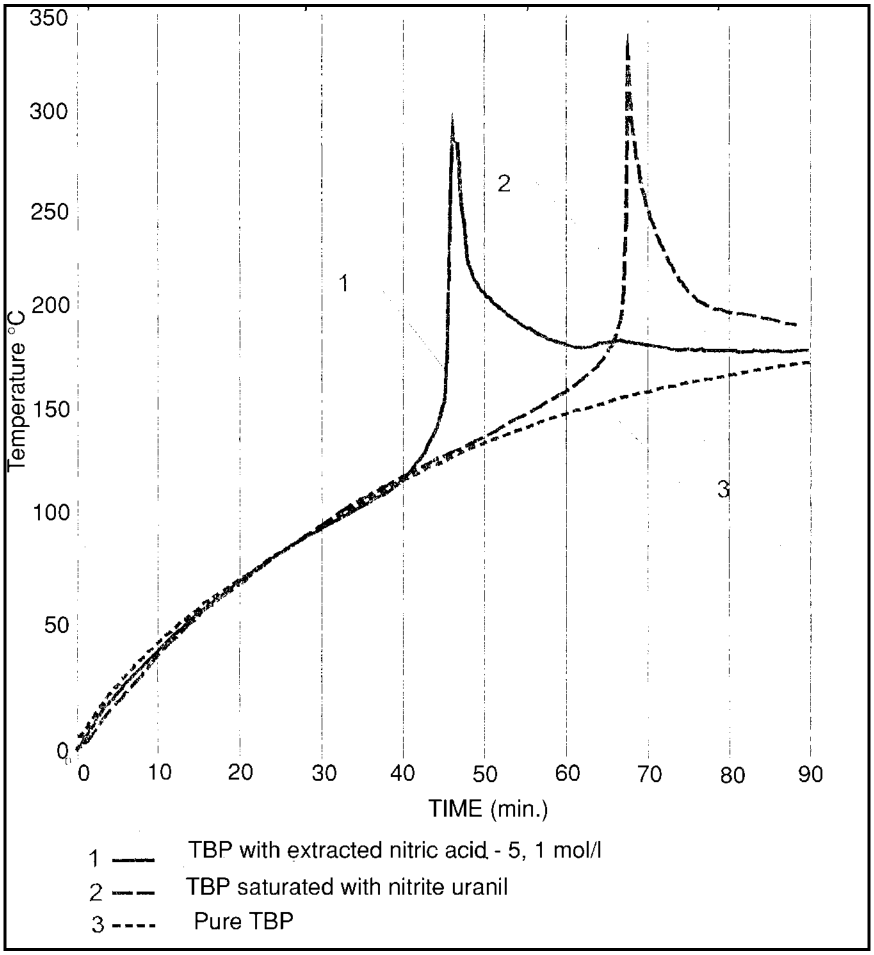


FIGURE 4. Change in temperature of the organic phase of "TBP–extracted HNO_3 (5.1 mol/L)" and "TBP–extracted uranyl nitrate" mixtures. The specified temperature of the thermostat wall is 230°C.

Measurement of the temperature at different points in the two-phase "extractant–nitric acid" mixtures showed that exothermic processes begin mainly in the organic phase, as Fig. 5 shows.

In closed vessels air-dry anionite VP-1AP in the NO_3 form decomposes under the conditions of a thermal explosion at the same temperatures as in open vessels. In mixtures of the anionite in the NO_3 form with HNO_3 in the presence of an aqueous phase over the anionite layer, the exothermic effects during heating in closed vessels were much less pronounced than they were in open vessels (Fig. 6), but a large amount of gaseous products is released, resulting in high pressures in the vessels.

Thus, according to available experimental data, in extraction and sorption mixtures dangerous exothermic processes may occur at practically all HNO_3 concentrations used in production operations. The decisive factor in their occurrence is the heating of the mixtures to the "starting" temperatures of the exothermic processes.

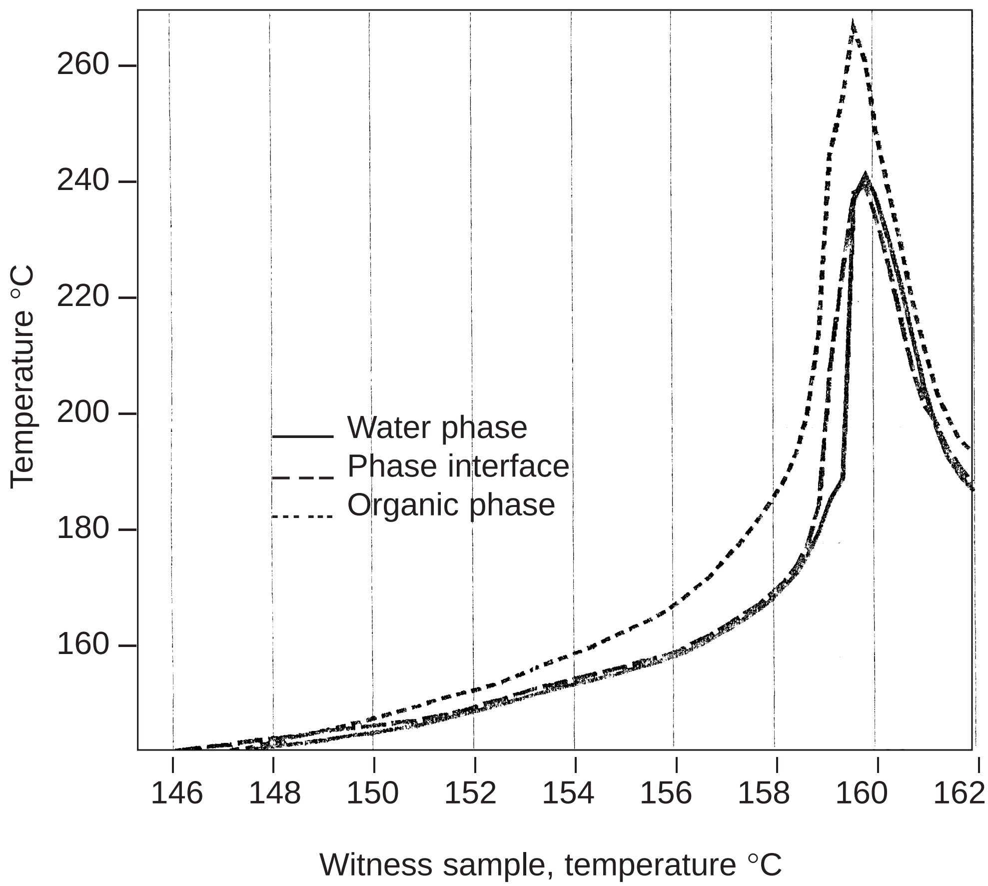
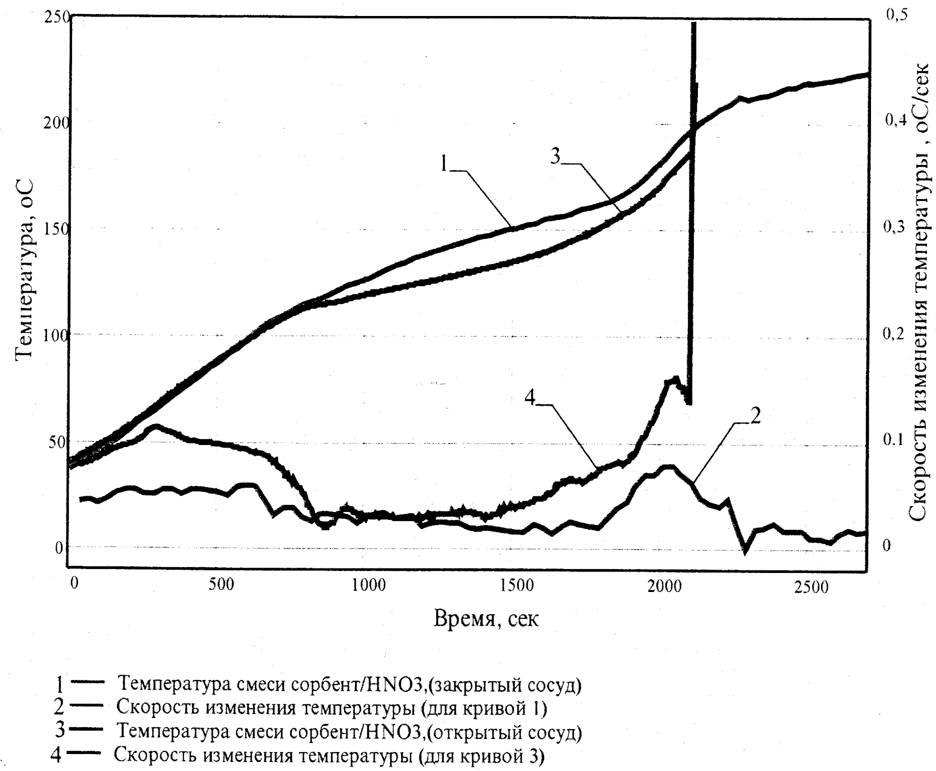


FIGURE 5. Variations in temperature in different zones of a two-phase mixture of 30% TBP in RZh-3 + 7 mol/L HNO_3.

Even for the source extraction mixtures, these temperatures are not so far from the maximum operating temperatures of the extraction and sorption operations, and are comparable to the temperatures at which nitrate solutions are inspissated. There is reason to believe that the "starting" temperatures of exothermic oxidation processes may decrease for irradiated and heat-treated mixtures, and also when the volumes are large.

Overall, if one takes into account the possibility of heating of mixtures by internal heat release (the heat of radioactive decay of the radionuclides), the attainment, in extraction and sorption mixtures, of the "starting" temperatures for a thermal explosion under production conditions does not appear to be such an unlikely event.

The information obtained to date makes it possible to determine the most important directions for future research on potentially hazardous oxidation processes in industrial mixtures of organic substances with nitrate oxidizers, especially in extraction and sorption mixtures.



KEY: (1) Temperature of sorbent/HNO_3 mixture (closed vessel); (2) Rate of change of temperature (for curve 1); (3) Temperature of sorbent/HNO_3 mixture (open vessel); (4) Rate of change of temperature (for curve 3); (a) Temperature (°C); (b) Rate of change of temperature (°C/sec); (c) Time (sec).

FIGURE 6. Typical behavior of temperature when mixtures of sorbent with nitric acid are heated in closed and open vessels. The relations are referred to a unified heating rate.

Reliable estimation of the temperatures of the onset of exothermic processes in mixtures is a priority task. To solve this problem, research needs to be conducted under the most stringent conditions: with slight heat exchange between reaction vessels and the environment, and with preirradiated and heat-treated mixtures containing nitrates of metals. It is especially important here to evaluate the heat release in the first stage of oxidation processes and the possibility of the resulting heating of mixtures to the "starting" temperatures of exothermic processes (thermal explosion).

A more remote but no less important task is to obtain raw data for making a computational estimate of heat release in specific industrial apparatus with specific mixtures. It includes a determination of the kinetic characteristics of the "preexplosion" period of occurrence of exothermic processes and an estimate of the amounts of heat released as a result of the reaction.

The fulfillment of these tasks will make it possible to draw up better-substantiated recommendations for the safe performance of radiochemical production operations and to obtain raw data for analysis of specific production operations, from the standpoint of the occurrence of accident situations, and for assessing their possible effects.

SAFETY ISSUES ASSOCIATED WITH SAFE SHUTDOWN AND OPERATION OF PLUTONIUM PROCESSING PLANTS

C. J. THOMPSON
British Nuclear Fuels plc
Sellafield, Seascale, Cumbria CA20 1PG, UK

1. Overview of Reprocessing at Sellafield

British Nuclear Fuels plc (BNFL) provide a complete nuclear fuel cycle service with its sites at Springfields (AGR/Magnox Fuel Fabrication) near Preston and Sellafield (Reprocessing and MOX Fuel Fabrication) in Cumbria. BNFL also generates electricity using Magnox Reactors at Sellafield (Calder Hall) and Chaplecross in Scotland. This paper provides an overview of Reprocessing Plant Safety Issues at Sellafield from receipt of irradiated fuel through to product re-cycling/waste immobilization and a review of safety issues associated with the operation and safe shutdown of Pu Processing and MOX fabrication plants.

Two reprocessing plants are in operation at Sellafield. The Magnox reprocessing plant, which commenced operation in 1964, processes irradiated uranium (U) metal fuel from UK and overseas Magnox reactors. The THORP (Thermal Oxide Reprocessing Plant), which commenced operation in 1994, reprocesses enriched uranium oxide fuel principally from Light Water Reactors (LWR) in Europe and Japan and Advanced Gas Cooled Reactors (AGR) in the UK.

The overall process is similar for the two plants and involves the use of solvent extraction methods to separate the uranium, the plutonium, and the fission products. The main elements of the reprocessing cycle are as follows:

- *Fuel Receipt*. In heavily shielded transport flasks via a rail link to UK stations/European stations (or the BNFL marine terminal at Barrow for sea transport from continental Europe and, for example, Japan);
- *Fuel Storage*. In dedicated storage ponds in either skips (Magnox/AGR) or Multi-Element Bottles (MEBs–oxide fuels);
- *Fuel Preparation.* Magnox fuel cladding is removed and the bare uranium bars fed into a magazine; this is then transferred to the reprocessing plant by rail. The fuel cladding is transferred to an encapsulation plant where it is immobilized in grout in stainless-steel drums. Oxide fuel assemblies are fed directly to the reprocessing plant;
- *Fuel Dissolution.* Within the Magnox reprocessing plant, fuel is fed from the magazine into a dissolver. In THORP, fuel assemblies are guillotined by a shearing mechanism and the sheared fuel fed to a dissolver in a basket.

Following dissolution of the fuel, the remains of the fuel cladding are transferred to an encapsulation plant;

- *Chemical Separation.* Both the Magnox and THORP plants employ solvent extraction methods using TBP/OK (tributyl phosphate/odorless kerosene) to separate out the fission products and uranium and plutonium components of the dissolver liquor. In the case of the Magnox plant, this is done entirely using mixer settlers; in THORP, a combination of pulsed columns and mixer settlers are employed. In both plants the overall process produces separate purified aqueous streams of uranyl nitrate and plutonium nitrate, which are subsequently treated to produce the respective oxide forms;
- *Uranium Finishing.* In both the THORP and Magnox plants, the uranyl nitrate stream is converted to uranium trioxide powder through concentration by evaporation and thermal denitration. The material is then stored prior to further processing into new uranium oxide or mixed oxide fuel;
- *Plutonium Finishing.* For both THORP and Magnox plants the plutonium nitrate is chemically conditioned and mixed with oxalic acid to precipitate the oxalate. After filtering and drying the resulting powder is calcined to convert it to dioxide powder. The material is then sent to a purpose built store prior to recycling via the MOX (mixed oxide) fuel route;
- *Liquid Effluents.* Four categories of liquid effluent arise from both the Thorp and Magnox processes; these are highly active, medium active, low active, and salt bearing. The types of liquid effluent are treated by various plants across the site and the resulting material either placed in long-term storage (e.g., vitrified HLW), treated to concentrate the waste material (e.g., by ultrafiltration) with the resulting low active liquors discharged to sea, or discharged directly to sea in the case of low active liquors;
- *Gaseous Effluents.* These are discharged from a number of points on the site via a variety of filtration systems.

2. Major Hazards Associated with Reprocessing Operations

The principal hazards associated with the operation of Reprocessing Plants at Sellafield are well recognized and may be categorized as criticality, fires/explosions, loss of containment, impact damage, and abnormal discharges. Examples of the potential causes of these include:

- *Criticality.* Accumulation, over-concentration, loss of poisoning/ favorable geometry;
- *Fire/Explosion.* Radiolytic hydrogen, red oil reactions, solvent fires, zircaloy fines;
- *Loss of Containment.* Leakage from process vessels, failure of cell cladding;
- *Impact Damage.* Impacts on equipment or plant containment dropped active loads (e.g., flasks);
- *Abnormal Discharges.* Failure in dissolver off-gas systems, contamination of effluents streams, misrouting of effluent streams.

3. Operation and Safe Shutdown of Plutonium Processing Plants

BNFL have safely operated Pu Processing and Storage Plants at Sellafield for more than 40 years. In recent years the Pu processing capability has been extended to include the recycling of Pu into MOX (mixed UO_2/PuO_2) fuel via the BNFL MOX Demonstration Facility plant, and is being developed on a larger scale using the Sellafield MOX Plant.

3.1 OPERATIONAL SAFETY IN THE PLUTONIUM FINISHING PROCESS

3.1.1 Process

Following the purification of the plutonium nitrate solution arising from reprocessing operations, the liquor enters the Pu finishing cycle. An identical process is used in both the Magnox and THORP reprocessing systems, which is as follows:

- Conditioning of the plutonium nitrate to convert all the plutonium into the 4-valent state by treating with hydrogen peroxide at 60°C;
- Precipitation of plutonium oxalate by adding oxalic acid;
- Filtration and washing of filter cake on a flat disc rotary vacuum filter;
- Drying and oxalate destruction by heating to 350°C in a controlled atmosphere furnace;
- Calcination by heating to 600°C in an argon atmosphere;
- Cooling in argon and sealing in storage containers.

3.1.2 Process safety (Magnox Plant)

External Radiation to Personnel. The plant is designed to be remotely operated from a central control room with minimum routine requirement for access to plant cell areas. All process plant is installed in cells, and shielding provided by the concrete walls ensures that radiation levels in operating areas remains well below 5 Svh^{-1}. Steel shielding on the equipment and glove boxes restricts radiation levels within the process cells generally to less than 50 Svh^{-1}. Operational control is provided to ensure that annual whole body and extremity doses to operators and maintenance personnel are maintained below the 10 mSv and 150 mSv plant targets wherever possible.

Internal Radiation to Personnel. The main protection is provided by design features which control spread of surface and airborne activity and minimize the need for routine access to process cell areas. Primary containment of the plutonium is generally provided by process plant with secondary containment by gloveboxes where appropriate. The cabinet extract ventilation system keeps the glove boxes under depression with respect to the cells and, in the event of a breach of containment, the emergency extract system maintains sufficient depression to help prevent the escape of activity. In the event of loss of glove box depression, all gas supplies to that box will be automatically isolated and appropriate process stages shutdown by interlock action. Cell and operating area ventilation is arranged to ensure that at all times airflow is in the correct direction for control of the potential spread of activity in the event of a release.

Continuous Pu-in-air monitoring is provided for all areas within the building, with alarms both local and repeated in the Health Physics control room. The system is available to initiate automatic area evacuation if widespread air activity is detected.

Criticality. The process plant is designed on the principle of safe geometry under all conditions (e.g., the use of HARP tanks). Continuous monitoring is provided by an approved criticality detection and alarm system. Where additional operational control is required to maintain safe conditions, this is specified in the appropriate nuclear safety assessments and Criticality Clearance Certificate. The latter specifies limits and conditions that need to be complied with during operations; for example, in respect of limited tap density, moisture content, isotopic inventory and mass. It identifies any systems or instrumentation that demonstrate that compliance is maintained during operation. In addition key points of the plant are monitored by neutron monitors to give early warning of the unanticipated build-up of solid plutonium which could lead to the development of unsafe conditions.

4. Safe Operation and Shutdown of the Sellafield MOX Plant

BNFL are in the final stages of commissioning a Mixed Oxide Fuel Fabrication Plant at Sellafield (Sellafield MOX Plant or SMP). This plant has a design throughput of 120 te fuel per year and will commence operations in 1998.

4.1 SAFETY UNDER NORMAL OPERATIONAL CONDITIONS

The safety assessment of SMP has identified routine operational dose uptake, criticality, and loss of containment as the most significant hazards influencing the design of SMP.

4.1.1 Operational dose uptake

The design of SMP has been developed largely around the concept of remote operation with appropriate local or bulk shielding provided around the process stages with relatively high dose rates. For example, the fuel assembly line operations are remotely operated and are enclosed in a concrete shielded room of appropriate thickness.

One of the major components of the operational dose uptake assessment process has been the development of a detailed plant operational model broken down to an individual task level. Preparation of the model commenced early in the project design and drew on the extensive experience gained by BNFL over many years of designing, operating, and decommissioning plutonium plants and more recent experience of operating MOX fuel fabrication plants. This experience provided valuable data on both the manpower requirements and task durations for both process and maintenance operations and on the main potential short- and long-term sources of operational exposure and how these exposures could be adequately controlled.

The SMP dose uptake assessments are therefore based on very robust base data. A detailed estimate of the dose uptake for SMP indicates that the annual average dose uptake for the SMP work force including maintenance operations will be less than 5 $mSvy^{-1}$ at full capacity and using high burn-up, long-aged Pu.

4.1.2 Criticality

Considerable effort has been concentrated on the development of a plant design which has an acceptable criticality safety case, but which is not constrained in terms of its flexibility to process a wide variety of feed material and manufacture a wide range of fuel assemblies. The principal objective of the design was to make the plant inherently safe where practicable for normal operations and potential fault conditions. Only where this objective is not practicable are additional safety protection systems provided. This enhances the robustness of the safety of the plant and reduces the requirement for potentially complex safety protection systems.

One of the major constraints in developing a criticality safety case for SMP has been the limited reliability data which is accepted by UK regulators for software-based control systems, as used on SMP. The hazard identification exercises carried out on the SMP design identified that failure of multiple control system functions could give rise to potential criticality incidents in certain process vessels which are not practicable to remove by equipment design. Therefore, although the design of SMP has minimized the requirement, a small number of safety protection systems have been incorporated into the SMP design to prevent potential criticality incidents initiated by multiple failure of control system functions. These systems are diverse and engineered to be fully independent of the control system. The potential for criticality incidents in the SMP plant has been calculated as less than $10^{-5}y^{-1}$.

4.1.3 Loss of containment

Most of the SMP processing operations, up to the loading of filled fuel rods into fuel magazines and assemblies are carried out in glovebox containment. A number of design features have been incorporated into the design to prevent potential loss of containment. These include:

- Prevention of glovebox pressurization by tripping glovebox gas feeds on detection of loss of glovebox depression;
- Utilizing vortex amplifiers to provide glovebox emergency breach protection;
- Prevention of potential impact damage to gloveboxes due to mechanical failure of in-box components e.g., by physically restricting movement of equipment.

Active material handled outside gloveboxes is contained in containment barriers of appropriate integrity (e.g., PuO_2 cans, fuel cans). The handling systems for these materials take due account of the integrity of the containment barriers by limiting the forces applied to the containment and limiting potential drop heights.

4.2 SAFETY UNDER PLANT SHUTDOWN CONDITIONS

The design of SMP and the relatively benign nature of the process has ensured that there is no requirement for services to maintain the plant in a safe state, under normal and emergency plant shutdown conditions. This has been achieved by the following design measures:

- All protection systems are designed to fail safe on loss of service or signal. Therefore, the plant will achieve a safe state under these circumstances;

- The design of the PuO_2 store ensures that a fully loaded store maintains the packages at a safe temperature indefinitely upon loss of the engineered cooling airflow;
- The design of the magazine and assembly stores incorporates a passive ventilation system that ensures that the fuel rods within the magazines and assemblies are maintained indefinitely at a safe temperature upon loss of the engineered cooling airflow.

5. Decommissioning of a Plutonium Finishing Plant

BNFL is currently undertaking a significant number of decommissioning projects at Sellafield, ranging in scope from the Pile 1&2 facilities to small plants associated with early phases of reprocessing operations.

As part of this ongoing program, a redundant Plutonium Finishing Line (FL3) that consists of 21 stainless-steel gloveboxes, and which was operated between 1962 and 1985 for the conversion of plutonium nitrate to plutonium oxide and plutonium metal, is now being decommissioned and dismantled.

A Post-Operational Clean Out (POCO) of the accessible parts of the line was completed by mid-1988. However, significant quantities of plutonium residues were anticipated in certain areas of the plant along with abnormally high radiation levels in these areas (up to 8 $mSvh^{-1}$ gamma and 2 $mSvh^{-1}$ neutron at contact) which are largely due to americium in-growth.

The gloveboxes are being removed one at a time to a purpose-built conditioning area. In the conditioning area, each glovebox is opened to enable collection of any plutonium residues left after POCO. The gloveboxes are then size-reduced, monitored, and placed in 200-liter drums prior to transfer to a designated storage area at Sellafield.

Size reduction operations take place within a tented area, within the conditioning area, which is provided with a ‘Tedak’ ventilation/filtration system. Where feasible the more efficient plasma arc size reduction methods are used rather than mechanical techniques to minimize operator dose. Glovebox removal, size reduction, and waste disposal operations are performed by operators wearing air-fed, pressurized, completely isolated suits (Windscale Suits). Entry into the Windscale Suit areas of FL3 is via a Windscale Suit recirculatory decontamination shower facility.

Originally it was intended to use manual methods for isolation and removal of items to the conditioning area and recovery of plutonium residues, followed by remote size reduction, waste monitoring, and disposal. Subsequent studies have indicated that there is little benefit, in terms of minimizing operator dose, from the use of remote techniques when the additional installation and maintenance activities associated with the more complex remote equipment is considered. Thus manual techniques are being employed.

The ‘Tedak’ fume extraction unit is a proprietary system that has been developed for decommissioning operations where plasma arc cutting devices are to be used. Plasma arc cutting ejects small particles of molten metal and generates a fine particulate fume. The ‘Tedak’ unit extracts these fine particulate fumes and filters them via a cyclone/sock filter system. This removes the majority of the particulate and avoids blinding of the HEPA filtration units downstream. The clean air then exhausts back into the size reduction room.

Radiological and Criticality assessments of the risks associated with the operations have identified a number of operating rules and safety mechanisms. An indication of the areas that these cover are provided below.

5.1 OPERATING RULES

Examples where operating rules guard against potential criticality scenarios include:

- Defining the sequence in which cabinets are moved out of the facility;
- Specifying how individual cabinets are to be dismantled;
- Specifying how individual components are to be handled dependent on the Pu content determined by monitoring.

5.2 SAFETY MECHANISMS

Examples of safety mechanisms that guard against potential criticality and radiological scenarios include:

- Differential pressure instrumentation and alarm on 'Tedak' HEPA filter monitoring loss of filtration/ventilation;
- Differential pressure instrumentation and alarm on building primary and secondary filtration systems;
- The monitor used to estimate the fissile content of items of plutonium contaminated material.

A number of requirements and pieces of equipment of lesser importance are embodied in the operating instructions and are designated as safety-related equipment.

5.3 ROUTINE DOSES

The upper bound estimate of collective dose (external + internal) for this project is 644 man-mSv over a 7-year duration, with the highest individual dose being <15 mSv (external + internal) over any 12-month period.

6. Conclusions

- BNFL has significant experience in the safe operation of nuclear fuel reprocessing plants at Sellafield.
- The Sellafield MOX Plant has been designed around the concept of remote operation to minimize operational dose uptake.
- The Sellafield MOX Plant has been designed such that it remains safe under plant shutdown conditions without the need for the provision of any services to maintain safety-significant equipment.
- Novel techniques employed in the decommissioning of redundant Pu finishing facilities have allowed operations to progress safely with minimal dose uptake and releases to the environment.

U.S. DOE SAFETY KNOWLEDGE BASE: ITS INTEGRATION AND UTILIZATION

FRED E. WITMER
U.S. DOE, DP-21, 19901 Germantown Road
Germantown, MD 20874, U.S.A.

1. Summary

The evolution of the U.S. Department of Energy (DOE) Environmental, Safety and Health (ES&H) information database has tracked the development and growth of personal computers and intra/inter-net capabilities. The scope and applicability of this database, which is primarily electronic, includes DOE's nuclear materials operations and is extremely broad, ranging from ES&H policy/safety initiatives/requirements (Orders) /standards and guides to the recording and analysis of events/accidents and sharing of lessons learned. Much of this information is available to the world via the Internet. Where pertinent, special effort has been made to link to other U.S. government agency ES&H databases. Certain DOE databases (typically preliminary and /or classified) are retained on intranets for appropriate timely access by DOE employees, contractors and/or oversight organizations.

Select access to this ES&H database by Minatom could accelerate and enhance the mutual understanding of U.S. and Russian Federation safety cultures. The sharing of lessons learned through the exchange of ES&H databases could result in improvements to both safety cultures.

Representative DOE ES&H search engines, database, and build-in analysis capabilities have been surveyed to provide limited "examples" to establish an appreciation for the capabilities and limitations inherent in DOE's major ES&H databases: the Occurrence Reporting and Processing System (ORPS), along with Defense Program's derivative binned information trending tool (DP/ORBITT); the Computerized Accident Incident Reporting System (CAIRS); the Medical Surveillance Program; Performance Measures (PM); and the Lessons Learned (LL) Program.

These and other DOE ES&H databases (along with built-in analytical aids) can be overwhelming to the first time user. Each database is specially structured/indexed to facilitate tailored applications, i.e., searching and manipulation within the database proper. Such features present compatibility problems in the consolidation of databases and/or during cross-cut comparisons.

Some databases have a “life of their own” because of organizational “stove piping.” Typically, time-based trending analysis of types of adverse events/causative factors is too general to identify cause and effect relationships. Without the development of normalized measurements, changes in mission and/or work load affect safety performance far more than do changes associated with the quality of safety management.

The use of advanced search engines for unstructured databases to address these deficiencies and to provide the requisite flexibility in dealing with different and changing applications across DOE’s ES&H database is under development.

2. ES&H Safety Database Overview

A convenient starting point for an overview is the DOE ES&H informational Internet Home Page (http://tis.eh.doe.gov/) which is reproduced in Figure 1. A wide range of safety related information is available via the “Windows” format of “pointing and clicking” the cursor to transfer to lower-tiered screens, ultimately pulling-up the appropriate document and/or compilation of data (table and/or graph). “Data Reporting & Analysis Services” (Fig. 1) provides a traceable gateway to the operational ES&H performance/operational databases.

2.1 OCCURRENCE REPORTING AND PROCESSING SYSTEM (ORPS)

A highly structured incident reporting system exists across the DOE complex as specified in DOE Order 232.1. The occurrence reporting process requires that the nature of occurrence and causative factors are coded; however, the body of the report relies on a narrative description of what occurred, its significance, corrective actions, and lessons learned, which are generally provided by personnel closest to the accident. There is a reporting-response time line, as the nature of the incident and contributing and root causes become better understood. The quality of the occurrence reports is generally uneven and there is residual concern if all reportable events are reported by the contractors. There is a disproportionately larger number of occurrences incurred by subcontractors, who are generally used for the higher risk activities, are in and out at the site, and are not fully indoctrinated in DOE safety procedures. Occurrence reports are searchable through a graphical users interface (GUI) that facilitates variable search criteria to be specified and can automatically produce bar charts, indicative of the number of occurrence reports for a given type of incident at a given facility. For example, in Figure 2, radioactive contamination incidences at the TA-55 Facility at the Las Alamos National Laboratory (LANL) have been plotted using the GUI. A breakdown on the nature of contamination (ingestion, skin, clothing, facility), insight into the dose/consequences or nature of the risk imposed by a specific task or mission is not available via this GUI plotting capability. In Figure 2, the higher incidence of reporting during the 1995-1996 time frame coincides with the production of the Pu-240 thermoelectric generators for the Cassini space mission at TA-55.

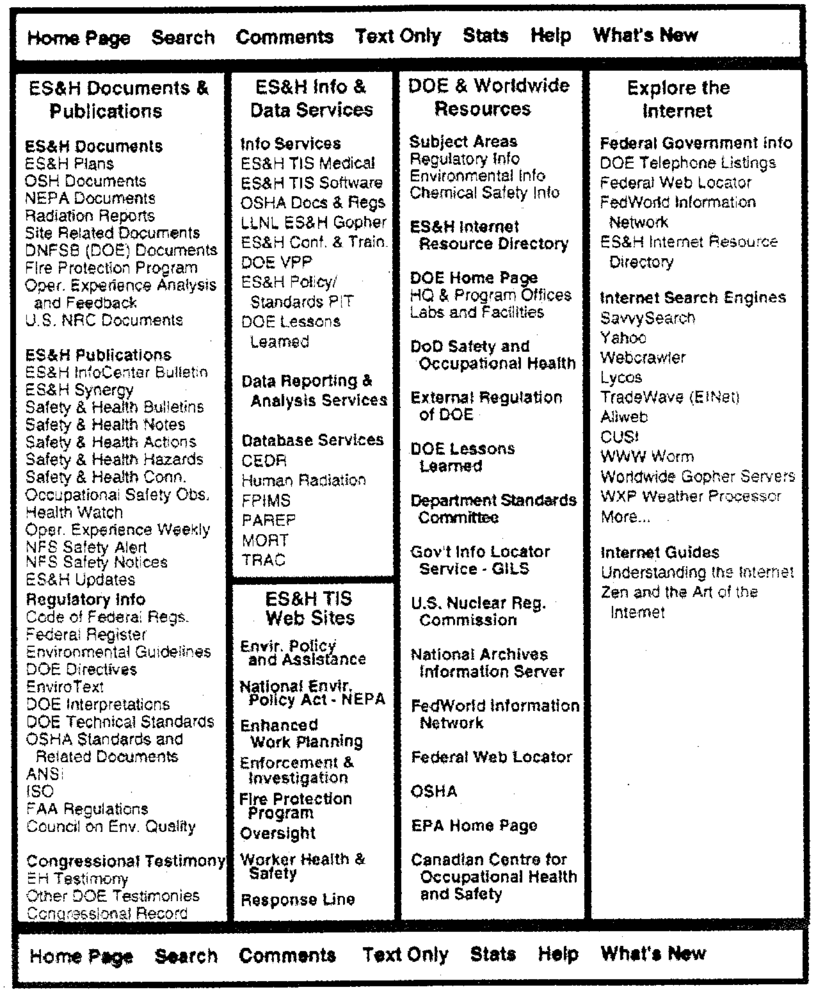


FIGURE 1. ES&H Safety Database Overview (http://tis.eh.doe.gov/)

2.2 DEFENSE PROGRAMS OCCURRENCE REPORTING BINNED INFORMATION TRENDING TOOL (DP/ORBITT)

Because the nature of event categorization processes inherent in the ORPS reporting processes are not conducive to trending operational attributes, along with determining the ES&H significance of the accident/incident, Defense Programs initiated a review and binning process to facilitate evaluations and trending for specific areas of operational weakness and/or strength. This process is a seamless derivative of ORPS and is performed without requiring any changes by DOE contractors in their execution of ORPS Order requirements. During the Defense Programs review, a significance weighting factor is placed on each incident consistent with established criteria. Because the review and classification process is performed by a standing technical panel, the process helps to establish uniformity and consistency in the DP/ORBITT database, thereby enhancing its value as a safety

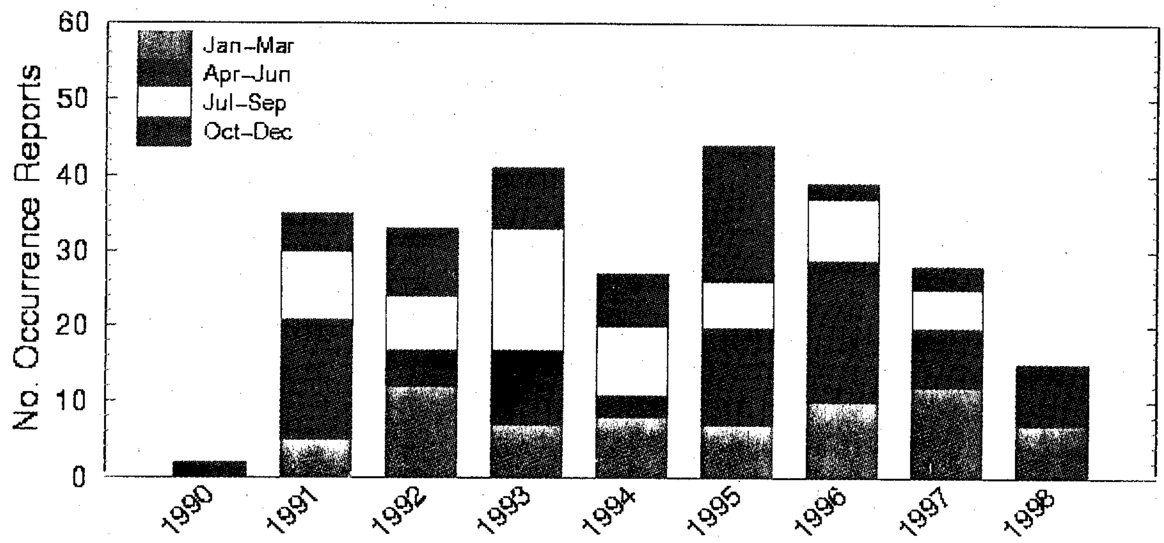


FIGURE 2. Graphical Users Interface (GUI) contamination incidence count vs. time at TA-55.

management tool to identify differences between sites/contractors and assist in understanding these differences.

Tables 1 and 2, respectively, list the operational bins and significance criteria used in the DP/ORBITT process. It should be noted that only 1 in 50 of the reported incidences has been classified as highly significant, while about 1 in 6 has been designated to be of medium significance. Of these, the majority are attributed to minor deviations from normal operations, symptomatic of potential break-down in operations, but not representing a specific precursor to a particular accident. About 34% of the incidences concerned some aspect of Conduct of Operations, while Radiological and OSHA/Industrial Hygiene Attributes each were involved in approximately 10% of the incidences. The remaining operational (major) attributes each were involved in 4-6% to the total incidences. Nearly half of the incidences binned under the OSHA/ Industrial Hygiene attribute were determined to be of medium significance.

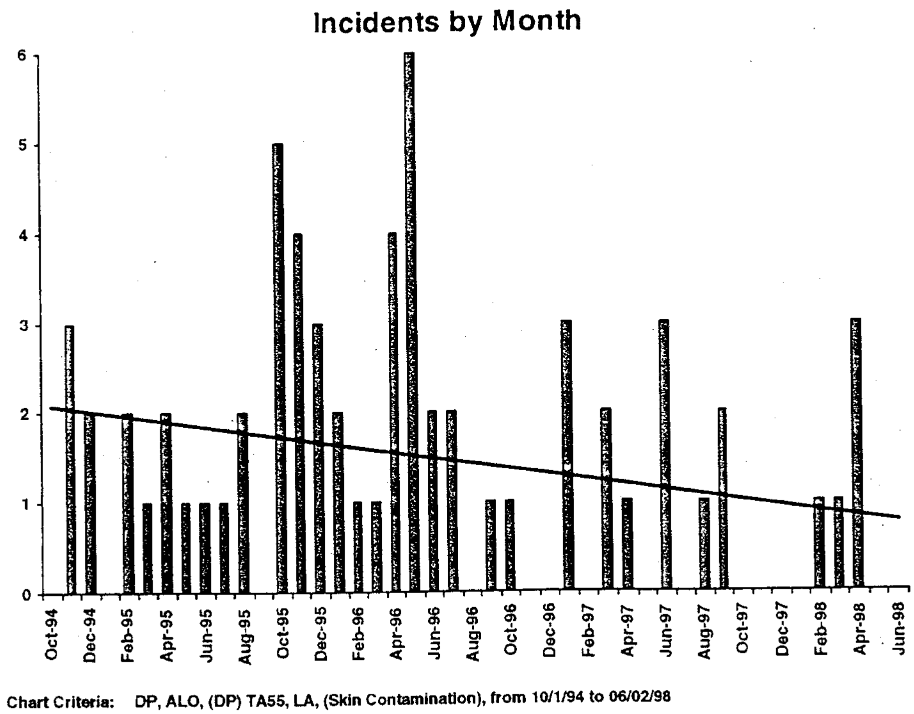
The DP/ORBITT database and plotting engine is available to Defense Programs staff via the DP HQ Limited Assess Network. Examples of the plotting capability are presented in Figure 3 that plot radioactive contamination at the LANL TA-55 Facility in greater detail than then available through the ORPS GUI. The elevated skin contamination during October 1995-May 1996 was due to Cassini start-up activities, latter coupled with the increased failure rate associated with the use of aged gloves in the glove-boxes.

TABLE 1. DP/ORBITT operational bins.

Conduct of Operations	**Instrumentation & Control**	**OSHA/Industrial Hygiene**
Conduct of Operations Misc.	Air Monitors	Indoor Air Quality
Configuration Management	Gas Analyzers	Industrial Hygiene Exposure
Criticality Issues	I &C Systems	Safety/Protective Equipment
Entry w/out PPE	I & C Components	Electrical Shock
Entry/ Exit Controls	Miscellaneous Monitors	Hoisting & Rigging
Improper Posting of RCA	Computer Components/Systems	Injury
Lockout/Tagout Issues		Fatality
Surveillences	**Mechanical /Structural**	Near Miss
Personnel Error	Mechanical Systems	
Procedure Compliance	Valves	**Security & Safeguards**
RadCon Compliance	Pump/Compressors	Loss of Mtls Accountability
Training Issues	Ventilation Systems	Classified Material
Inadequate Job Planning	Freeze Protection	Theft or Sabotage
	Misc. Structure/Mechanical	Fitness for Duty
Environmental	Seismic Events	Misc. Security Issues
Compliance Issues		
HAZMAT Release	**Radiological**	**Transportation**
Miscellaneous Release	Airborne Radiation	Transport/Shipping Issues
Oil/Gas Release	Exposure	Vehicle Accidents
Radioactive Release	Skin Contamination	
Underground Tanks	Clothing Contamination	**Miscellaneous**
	Facility/Equipment Contamination	Suspect Counterfeit Parts
Fire Protection		Corrosion/Mtls Degradation
Fire Suppression Actuation	**Electrical**	Emergency Mgmt & Comm
FPS Equipment Degradation	Emergency Diesel Generators	Chemical Reactions
Fire/Explosion	Circuit Breakers and Fuses	Safety Analysis/USQ Issues
	Power Outages	Natural Phenomena
	Batteries & Battery Chargers	Gloveboxes
	Uninterruptible Power Supplies	HEPA Filters
	Electrical Distribution	Nuclear Weapons
	Transformers	

TABLE 2. DP/ORBITT occurrence significance weighting factors.

Weighting factor	**Significance criteria**
1	Common occurrences. Minor deviations from normal operations
	Efforts to eliminate would not be cost effective
3	One remaining barrier to significant injury, electric shock (>50 volts AC), radiation exposure (one event >1000 mr uptake, intake or Committed Effective Dose Equivalent [CEDE]) or Industrial Hygience Exposure (>3 times OSHA limits), exceed criticality limit or double contingency is not maintained
	Hospitalization for Observation/Treatment
	Programmatic Breakdowns/Facility Standowns
	Intentional violation of procedures/requirements
	Cost basis >$10K
	Backup Safety Systems not available when needed
5	Life Threatening/serious injury: hospitalization >2 weeks
	Politically sensitive
	Major release affecting the public/environment
	Uncontrolled situation (major fire/explosion, unplanned criticality)
	Cost basis >$200K



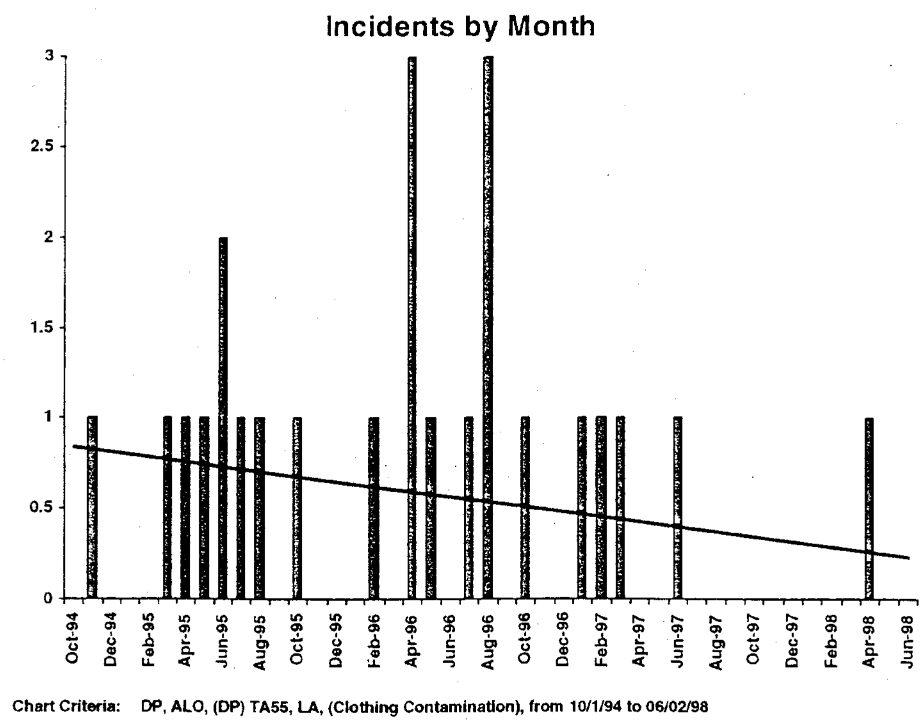


FIGURE 3. DP/ORBITT—contamination-type trending at two specific facilities.

2.3 COMPUTERIZED ACCIDENT INCIDENT REPORTING SYSTEM (CAIRS)

This database focuses on work-related personal injury and illness and property damage (including that due to the operation of motor vehicles). There is a general tabular breakdown relative to individual contractors; however, there is no detail as to the specific types of work or the assumed risk inherent in these particular types of work. This is an old database which was established to collect statistics for the U.S. Department of Labor. An important aspect is that the data is reported on a normalized per unit of work (200,000 person hours) basis which facilitates gross comparisons between and within industries. To emphasize the general nature of the CAIRS database, Total Injury and Illness Incidences Rates and Lost Workday Incidence Rates are depicted in Figures 4 and 5, respectively.

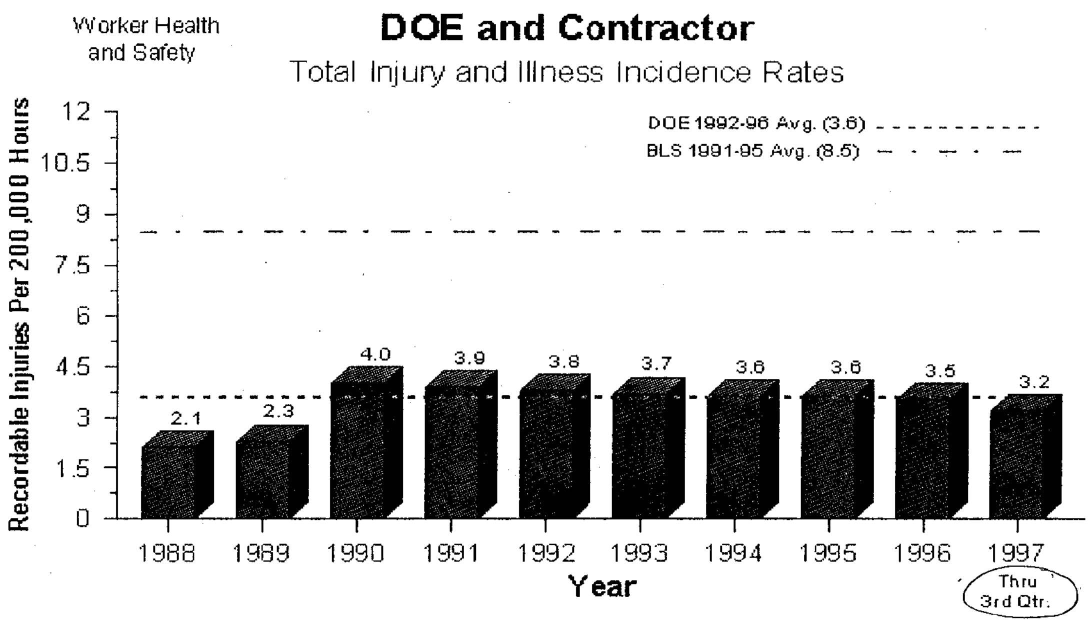


FIGURE 4. DOE and contractor total injury and illness incidence rates (CAIRS).

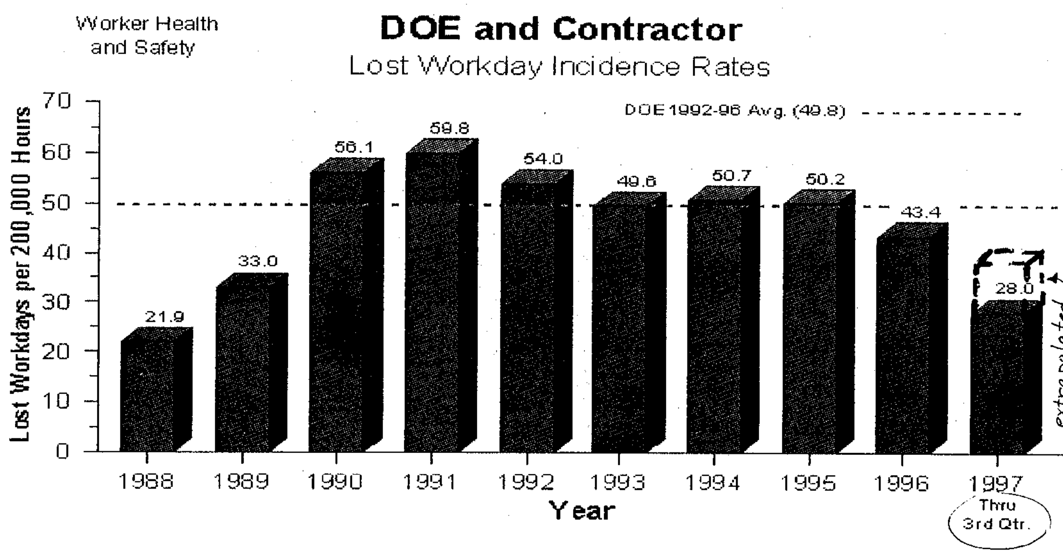


FIGURE 5. DOE and contractor lost workday incidence rates (CAIRS).

2.4 MEDICAL SURVEILLANCE PROGRAM

The methodology for this database, which is intended to track the heath histories for radiation and/or chronic insults over an individual's lifetime, is still under development. Concerns over retaining the individual's privacy and mitigating potential liabilities with regard to cause and adverse health affect have presented major challenges to the development of this (ambitious) database.

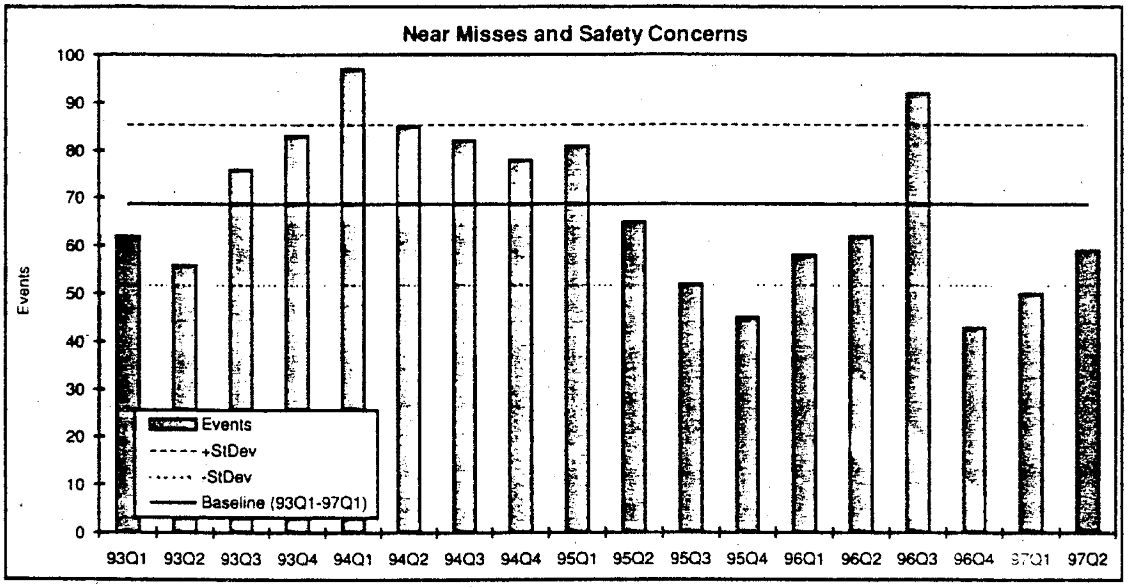
2.5 PERFORMANCE MEASURES (PM)

The identification of useful high-level time-trended ES&H performance measures and/or "indexes" (a mathematical combination of individual measures) to establish a performance baseline and indicate an impending process change has been difficult. For those performance measures selected, the lack of "normalization" and timely trending has severely limited their value to senior management to signal "actable" responses and establish comparability of performance expectations between different contractor organizations. Select examples of high-level performance measures prepared for DOE senior management are shown in Figures 6 and 7.

The use of time constant baselines to assess variance, depicted in Figure 6, are typically used for steady-state production operations to flag corrective actions for deteriorating performance (excessive variance); however, in the case of most DOE organizations, where a series of mission changes have been directed of their major elements, the application of a constant baseline is not realistic or meaningful. Generally, the "rollup" of the ORPS data used to derive the performance measures in Figure 6 is 6 to 12 months after the fact and it not timely enough to be "actable" by management. The measures shown are based on the number of events per quarter (time period) and approximate the rate of occurrence. In the fourth quarter of 1995, a revised occurrence reporting Order (232.1) with higher reporting thresholds was promulgated across the complex. One would expect a (damped) lowering in the rate of reported incidences due to the large organizational inertia involved in the actual implementation of the requirements of the revised Order and not the stepwise change implicit in the lower graph..

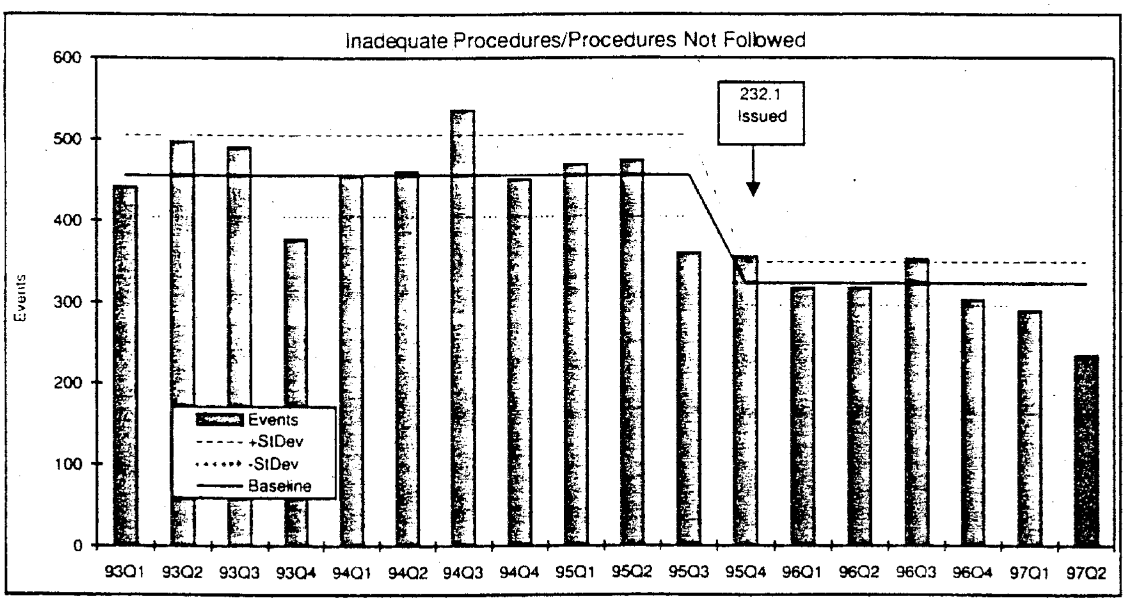
The rate at which suspect counterfeit items/parts (S/CI) have been identified across the complex, as shown in Figure 7, represents a better use of high-level performance indicators. In 1993-94, there was an extensive, complex-wide effort to identify counterfeit parts in safety related applications following the issuance of a quality alert bulletin. Fasteners accounted for 95% of the S/CI found. The finding and reporting of S/CI has since petered out—in part due to improved procurement and inspection practice and in part due to a lessening emphasis on finding S/CI because historically, when one looks for them, they turn up.

The best performance measures applications have been those that have been developed and deployed at the local level where the measures directly correlate to day-to-day operations and the affected parties have ownership. Figure 8 is an example of just such a practice. At DOE's Savannah River Site, the contractor has trended lock-out/tag-out failures that appear as an improving trend. However, if the ratio (%) of the lock-outs performed incorrectly to lock-outs attempted is plotted, one gets a different and correct picture that the quality of this particular operation is remaining constant. This is why it is essential for performance measures to be normalized to risk and/or opportunities to succeed.



Source: Review of Occurrence Reports by Department analysts.

(a)



Source: Review of Occurrence Reports by Department analysts.

(b)

FIGURE 6. Example of high level performance measures with constant baseline assumed and poor reporting timeliness (poor); (a) near misses and safety concerns; (b) inadequate procedures/procedures not followed.

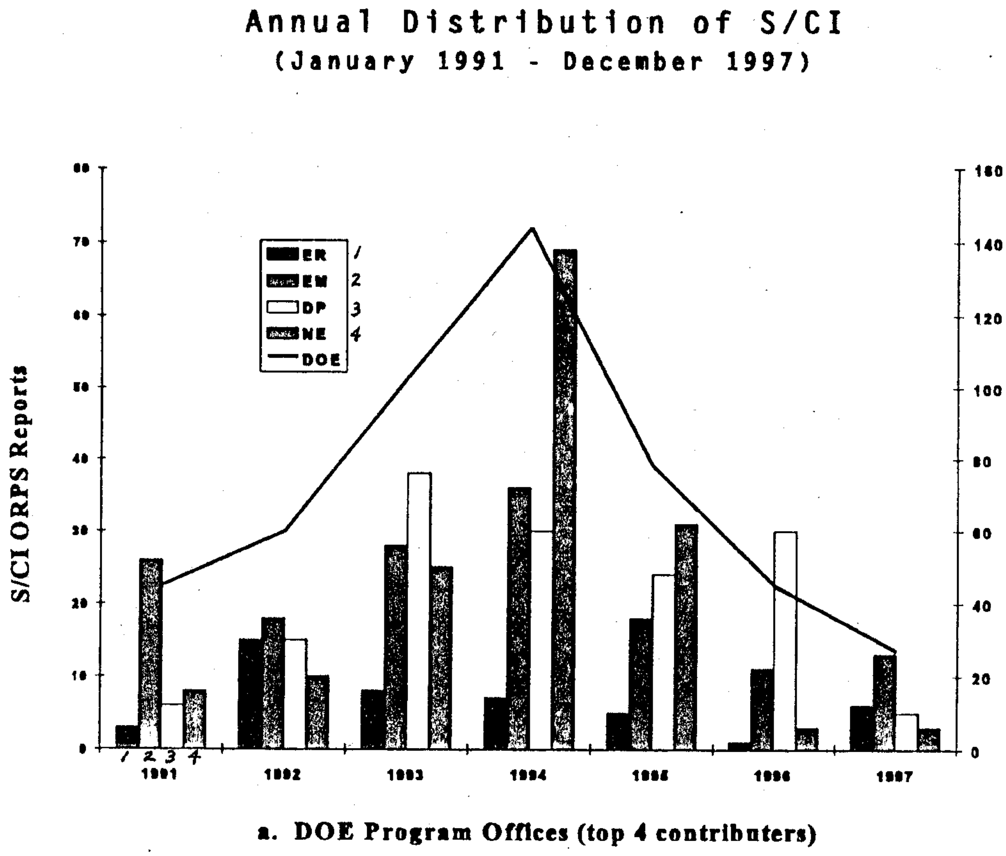


FIGURE 7. Example of high level measures used to track discovery of suspect counterfeit items (S/CI); annual distribution, Jan. 1991-Dec. 1997 (DOE program offices top 4 contributors).

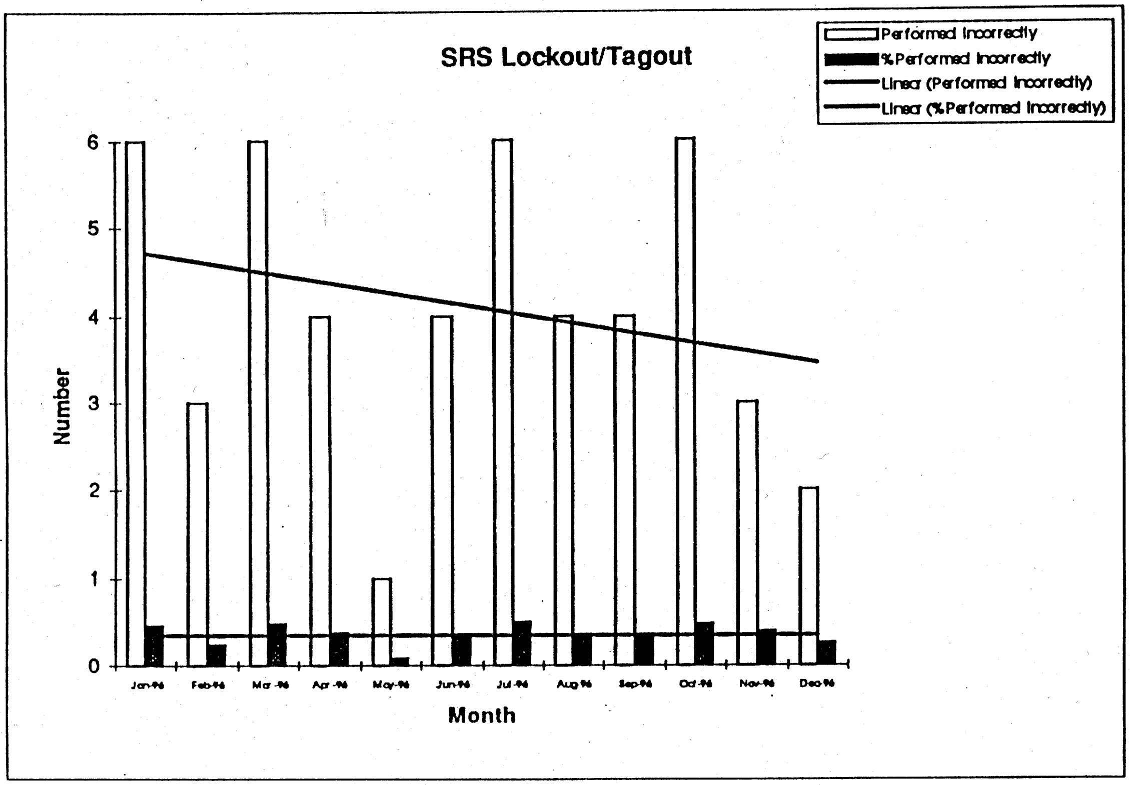


FIGURE 8. Example of lower level (Savannah River site-specific) normalized measure to track efficacy of Lockout/Tagout operations (good).

2.6 LESSONS LEARNED (LL)

A preponderance of DOE Orders that set the requirements for doing business within the Department call for the sharing of lessons learned that represent special knowledge gained from incidences, good practices, and success stories. One goal is to have an LL database across the complex that is searchable in a tailored manner for operations, trainers, procedure writers, appraisers, etc.

Many times the lessons learned activity is considered to be of parochial interest by an organization and not communicated to the complex. There now is a Department-wide e-mail system that distributes current lessons learned with an appropriate coding for urgency. Major DOE sites have a computer-based lessons learned database that is accessible from off-site; however, these databases must be searched independently with a large variability in the capability of the respective search engines, if one is present at all. Both the DOE ES&H and Defense Programs line organizations have review and analysis processes that contribute to the lessons learned database. These relationships are depicted in Figure 9.

At present, the major shortfalls of the LL activity are (1) there is a communications gap—customers in need are not targeted, (2) the LL database is incomplete and fragmented, and (3) parochial site and/or subject matter information is not shared. The use of an advanced commercially available search engine for an unstructured computer database, such as deployed by the intelligence community, has been proposed to address these short comings. The problem then becomes one of developing the front-end logic of the search engine to meet the needs of the user, opposed to forcing a preconceived indexing and binning structure in anticipation of how the user will use the system. Such a system is under consideration to enhance the applicability of LL. Such a system also possesses potential for broad application in the "archiving" of critical operational and safety information as nuclear weapon activities are curtailed, the work force ages and infrastructure is lost.

The general relationship between these respective databases and associated activities is depicted in Figure 10. The "key" designation on specific links indicate password control to access either via the intranet or web through appropriate controls or "fire walls."

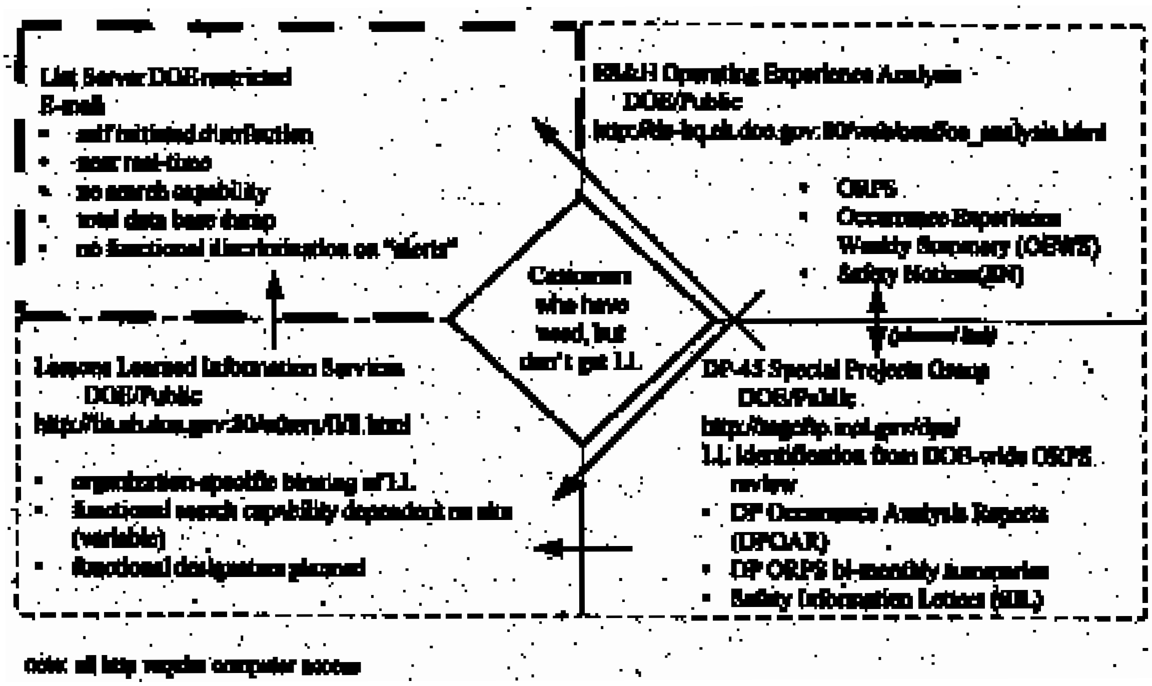


FIGURE 9. Current SELLS Lessons Learned database.

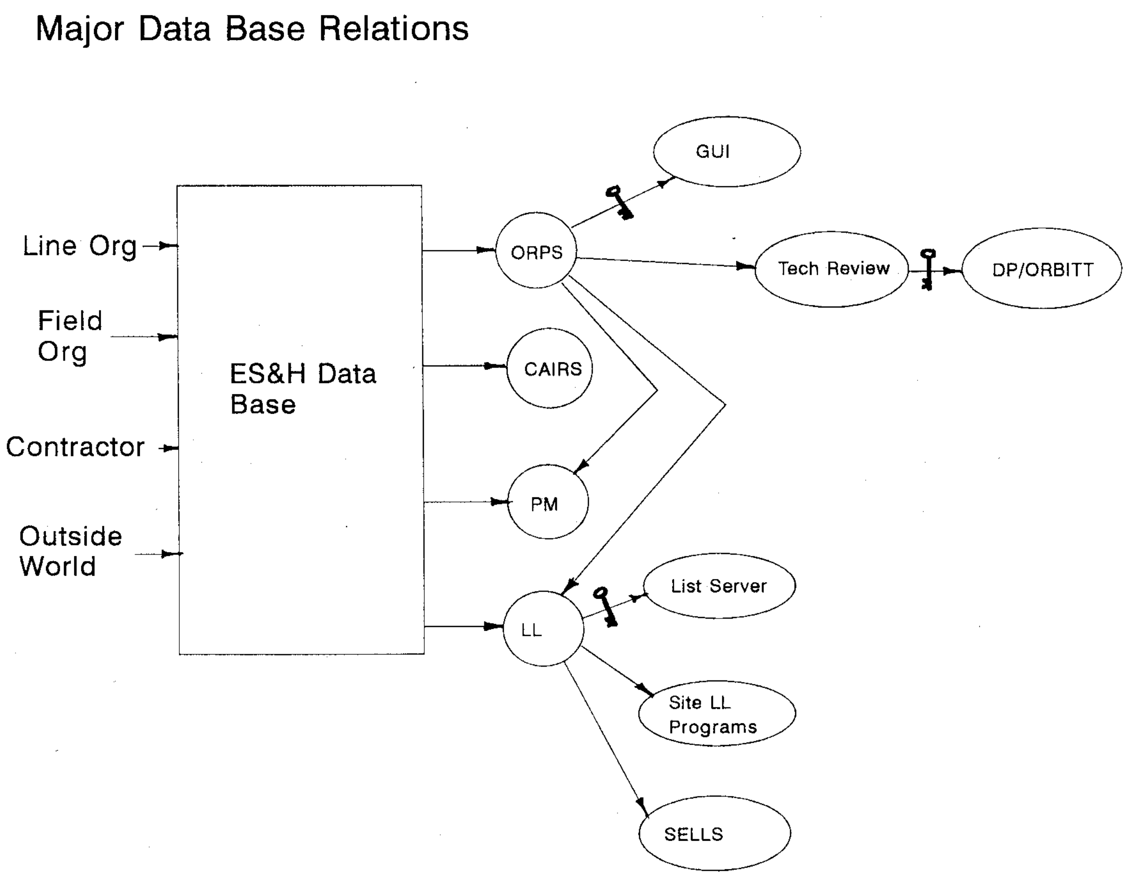


FIGURE 10. Major database relations.

3. Conclusion

While DOE possess an extensive ES&H database, with tailored applications and analytical aids, it can be overwhelming to potential users with special needs such as the worker or first-level supervisor interested in specific detail affecting their immediate operation or the senior manager who only wants significant, real ES&H vulnerabilities identified.

Many of the analytical and trending results (ORPS-GUI, and to some extent DP/ORBITT, CAIRS, and PM) are too general to establish cause-and-effect relationships. When monitoring the rate at which specific types of events occur as an indicator of performance, without the application of normalization techniques, changes in mission, work load, etc., far outweigh the effect of changes in safety management practice.

A number of DOE's ES&H databases have developed a life of their own and are fragmented because of differing customer requirements, need for stand-alone applications, organizational interests "stove piping," and historic isolation and parochialism at the site/contractor level.

Structured databases (e.g., ORPS, DP/ORBITT, CAIRS, PM and LL) require continued evolution and upgrading to meet changing customer needs to preserve continuity. Consideration is being given to the application of advanced search engines to address these deficiencies and enhance search/applications tailoring for ever changing customer lessons learned needs.

Select/controlled access to this evolving ES&H database by Minatom could contribute to an enhanced mutual understanding of the United States and Russian Federation safety cultures and would represent a significant step in the mutual sharing of lessons learned.

ESTABLISHING A BASIS FOR A UNITED STATES–RUSSIAN FEDERATION MULTI-YEAR PROGRAM IN NUCLEAR MATERIALS SAFETY

PAUL F. KRUMPE
U.S.DOE/DP
19901 Germantown Road, Germantown, MD 20874, U.S.A.

1. Introduction

The comprehensive nature of joint technical collaboration between the United States and the Russian Federation (RF) since 1994 in the field of radiochemical processing safety and the safe management of nuclear materials (as depicted within the nuclear fuels cycle program boundary in Figure 1), is well known and appreciated among the hundreds of U.S. and RF participants who have directly contributed to this expanding knowledge base. The future development and implementation of this joint initiative will depend on how well both countries access, share, adapt and utilize the extensive open information concerning each other's approaches to chemical and nuclear fuels/ materials safety and safety management systems. Sustainability of this activity in the foreseeable future will depend on securing the multi-year funding necessary to support continued bilateral (and possibly multilateral) technical exchanges, education and training, research and operational implementation of an improved safety management culture at key nuclear fuels facilities.

2. Program Significance

The potential of this program for increasing transparency must be understood as fundamental to U.S. non-proliferation objectives, especially in the safe reduction of danger at non-reactor nuclear fuels treatment facilities. This is an area not yet fully addressed under existing U.S.-RF non-proliferation or other activities. As both Russia and the U.S. move toward the disposition of excess fissile materials from various sources, including disassembled nuclear weapons, it is imperative that these initiatives incorporate systematic safety management functions in all phases of the mission. The U.S. National Academy of Sciences has recently identified surplus weapons plutonium as a "clear and present danger" to international peace and stability. We must acknowledge that another major accident in either country dealing with these special nuclear materials would seriously jeopardize the disposition efforts in both countries.

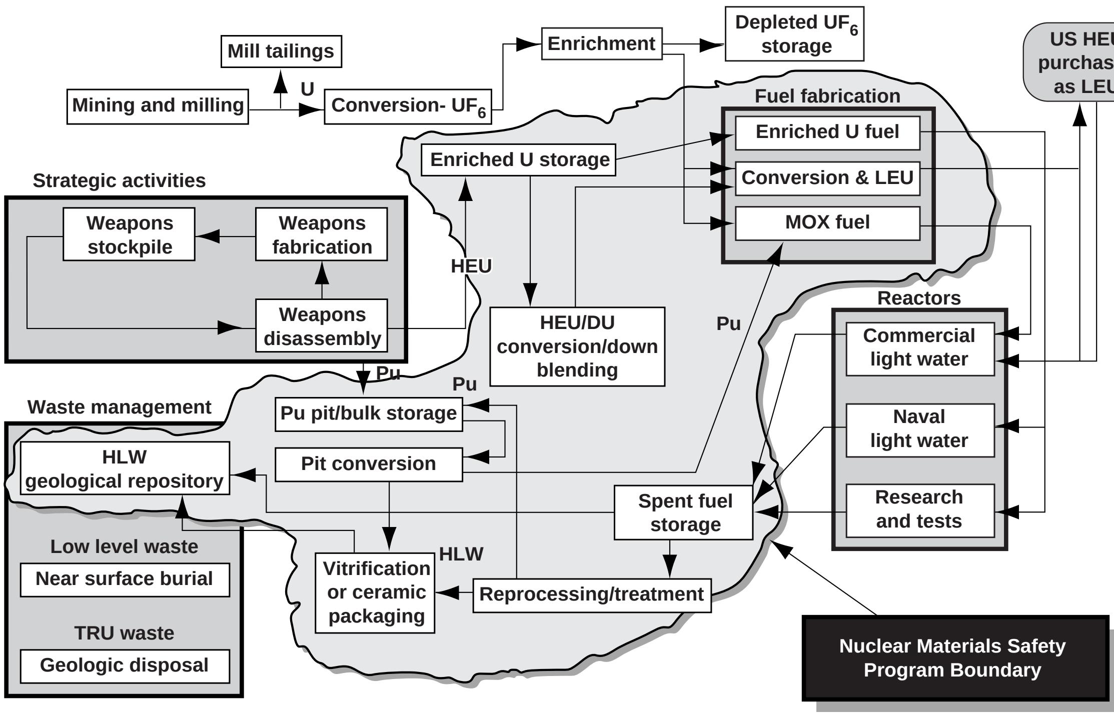


FIGURE 1. Flowsheet shows nuclear materials safety activities of interest to this ARW.

This could also cause substantial destabilizing financial conditions as well as other damaging economic, political, and social impact, especially in Russia. This program will focus attention on the prevention of future fuel-cycle and materials processing accidents. It will contribute significantly, at a modest annual cost, to keeping the weapons plutonium disposition effort on track by enhancing safety in the storage, handling, treatment and transportation of plutonium. Non-proliferation efforts can be accelerated by increasing transparency of the Russian nuclear fuel cycle by opening up facilities not covered by current U.S. Department of Energy (DOE) and Russian Federation Ministry of Atomic Energy (Minatom) interactions. A major contribution of this program to the future of international nuclear non-proliferation could be the eventual safe and secure shutdown of aging Russian nuclear fuels and fissile materials handling facilities. Establishing bilateral dialogue in this important area could significantly augment U.S.-RF non-proliferation goals as well as enhance national security interests.

3. Program Context

This program should be viewed within the overall context of existing key U.S.-Russian cooperative programs designed to reduce the nuclear danger. Given the current and near-future budgeting environment, the proposed Nuclear Materials Safety Management program would not likely exceed $8 million per annum over the five-year duration. This $40 million life-of-project level is relatively small when compared to the very large U.S. Department of Defense and DOE Cooperative Threat Reduction Program (CTR) which focuses on nuclear weapons dismantlement. Another very large program, the Materials Protection, Control and Accountability (MPC&A) program, complements the CTR and other DOE non-proliferation and arms control activities. The DOE Fissile Materials Disposition Program and the International Nuclear Safety and Cooperation Program are more modest in size and scope and could provide an umbrella for the proposed initiative. As of this writing, the most logical fit for this new activity is the International Nuclear Safety and Cooperation Program, primarily because of the program emphasis on safety management, operational safety procedures, education and training.

4. Mission Statement

The proposed nuclear materials safety program and previous technical exchanges and research activities originating with the Russian radiochemical accident at Tomsk in 1994 has evolved over the last several years. Emphasis today focuses more on the need for improving the safety culture at non-reactor nuclear facilities as well as preparing for and mitigating the potential causes of future accidents. As part of DOE's strategic planning initiative, especially given the increased importance of developing and maintaining strong ties with our Russian counterparts in the areas of nuclear weapons management, materials disposition, facility safety, and non-proliferation, the following program mission statement has been proposed: "To reduce the nuclear danger through improving nuclear materials safety management utilizing the existing joint experience base and mutual capabilities of the United States and the Russian Federation." This statement can provide the basis for establishing a bilateral Memorandum of Cooperation to formalize the program once multi-year funding has been identified.

5. Program Goal and Purpose

The proposed goal of this program, though possibly stated in different ways depending on the perspective of the program manager(s), is to reduce the risks (probability and consequences of potential accidents) at nuclear facilities while contributing to the clear understanding and demonstration of U.S. and RF nuclear facility safety practices and procedures in realizing nuclear materials non-proliferation. The program purpose is to ultimately ensure the safety of operations associated with the treatment, storage, and disposition of excess HEU and plutonium derived from various origins, including dismantled nuclear weapons. It is understood that both the program goal and purpose should be agreed upon by both the U.S. and Russian sides and that the legislative language authorizing and appropriating funds to support the program should also clearly state the bilateral program goal and purpose.

6. Key Program Elements

The two principal program elements under which work will be conducted during the five-year implementation period are:

1. Operational and infrastructure safety improvements at select nuclear fuel cycle defense facilities;
2. Academic exchanges, curriculum and degree program development, and joint research focusing on enhanced safety management.

Program direction will include the study of cross-cutting efforts in both countries dealing with reactor safety, safety issues involving transportation, plutonium vitrification, treatment options, and mixed uranium-plutonium oxide fuel fabrication. The program will also focus attention on safety in the storage, handling, treatment, and disposition of fissile weapons materials.

7. Participating Organizations

The U.S DOE Office of International Nuclear Safety and Cooperation (NN-30) within the Office of Nonproliferation and National Security in cooperation with DOE's Defense Programs and Materials Disposition Program will collaborate with Minatom, Department of Safety, Ecology and Emergency Situations, to implement the proposed program beginning in FY2000, subject to the availability of funds. The RF Khlopin Radium Institute (KRI), Moscow Engineering Physics Institute (MEPhI), the RF Ministry of Public Health, Gosatomnadzor (GAN) and Rosenergoatom, and the RF Minatom nuclear fuel cycle facilities/ sites will participate in the program as appropriate. The U.S. National Laboratories, including Lawrence Livermore (LLNL), Los Alamos (LANL), Sandia (SNL), Pacific Northwest (PNNL), and Argonne West (ANLW) are likely strong collaborators with counterpart Russian institutions to conduct lab-to-lab projects in nuclear materials safety. The Amarillo National Resource Center for Plutonium (ANRCP) in cooperation with Texas A&M University will conduct the nuclear materials safety masters degree program working closely with both Minatom and MEPhI.

8. Program Direction and First Year Focus

The program is expected to begin in early October 1999 (FY2000 Budget) provided that the Office of Management and Budget approves and the U.S. Congress authorizes and appropriates the necessary first-year funds for the program. The FY2000 effort will need to address radiochemical safety at the Russian facilities involved in treatment, handling, storage, and disposition of excess nuclear weapons materials as directed by the FY99 National Defense Authorization Act. Funds are expected to be allocated for planning and initial coordination as well as to address immediate safety concerns at key facilities. Additional funds will be needed for full implementation of technical training and the masters degree program design, technical exchange workshops, and site assessments. The first-year focus will also result in establishing joint working groups in nuclear materials safety topical areas, completion of radioactive aerosol and stack monitoring training, completion of accident process modeling, and radiochemical safety analyses of anion exchange in nuclear processing. Establishing a nuclear materials safety website to facilitate joint technical exchange and provide program updates will also be an important priority.

A comprehensive program plan and draft Memorandum of Cooperation in nuclear materials safety research and education is also an expected outcome during the first year of the program. A number of potential working group topical areas in nuclear materials safety management have been jointly proposed during the past few years at our technical exchanges. These include but are not limited to:

- Utilization and disposition of materials from disassembled weapons;
- Criticality safety in nuclear materials handling;
- Safety during the storage and transportation of nuclear materials;
- Safety considerations relating to the design and management of MOX fuel;
- Safety during transportation and storage of spent fuel;
- Ecological assessments in storage and disposal of nuclear materials in geologic formations;
- Preparedness for emergency situations and accident response planning;
- Radiation safety of workers, the public and facilities, and issues of public concern.

9. Future Proposed Activities

In addition to establishing joint working groups to address integrated safety management issue, we should begin to benchmark the safety authorization basis at selected Russian facilities.

It will also be necessary, to the extent possible, to determine the organizational structures and lines of authority at key facilities to ensure operational safety and accountability. Where possible, we will develop comparative analyses of U.S. and Russian safety management systems and document site-specific workplace prevention-based strategies. We will share our lessons learned systems specific to key processes and activities in an effort to help our counterparts avoid possible future accidents. With respect to on-site education and training, the working groups will prioritize joint technical training needs, qualification and certification requirements. Our national laboratories can provide proven methods for implementing risk-based management at selected facilities as well as methods for conducting Environmental Safety and Health (ES&H) assessments and audits. During the five-year program implementation phase, we will provide the means to help document major site-specific hazards and the tools necessary to maintain an integrated safety management capability beyond the life of this program.

10. Measuring Program Performance

Measuring performance of the joint U.S.-Russian Federation Nuclear Materials Safety program will focus on documenting safety improvements at key Russian industrial fuel cycle defense facilities and sites within mutually agreed upon time frames. The program goal is to reduce the probability of potential accidents while contributing to the clear understanding and demonstration of U.S. and RF nuclear facility safety practices and procedures in realizing nuclear non-proliferation. Performance will be quantitatively measured, as appropriate, in the program's lab-to-site technical exchanges and advanced research workshops to ascertain the degree to which Russian counterparts (both individuals and institutions) have embraced changes in their safety culture and have begun to implement Russian adaptations of standardized Western safety management practices, procedures, and regulations where applicable. Performance of the program's university nuclear materials safety management Masters of Science degree development activity will be measured by:

1. Number and quality of Russian students competing for entry into the program;
2. Number of graduates completing the degree program vs. the number entering;
3. Number of technical and safety management positions accepted by the program's graduates;
4. Recognition achieved by M.S. graduates among Russian authorities and ministries involved in nuclear materials management.

The ultimate program purpose is the assurance of safe operation at Russian nuclear materials industrial enterprises and sites associated with the treatment, storage, and disposition of excess HEU and Pu derived from various origins, including dismantled nuclear weapons. Efforts to benchmark safety management at select Russian defense fuel cycle facilities will be incorporated into all aspects of the program. During the five-year effort, improvements in the Russian safety culture, practices, and procedures will be documented and an increase in safety awareness will be evaluated. Key annual program milestones will be established and agreed upon jointly by both U.S. and Russian program managers. The ultimate success of this program can be measured by the sustainability of the transferred technology and safety culture (and educational programs) by the collaborating Russian institutions and ministries in the years and decades following conclusion of the program.

11. Program Mandate and Financial Support

This program represents a bold new initiative to address key non-proliferation issues relating to the safety of nuclear fuel cycle facilities in Russia. If and when specific facilities and processes in Russia are shut down and terminated with clean-up in progress (as has occurred in the DOE weapons complex), then it is imperative that this action be managed safely by highly trained technical individuals who understand not only the relevant radiochemical processes but also safety management principles applied to the worker, the public, and the environment. The U.S. House of Representatives National Security Committee has recognized the need for operational safety management at key Russian fuel cycle facilities and has recommended that DOE pursue the development of this program to address future nuclear materials safety critical needs in Russia. Toward that end, efforts are ongoing within DOE to develop a strategic plan to implement the program beginning in FY2000. While that effort progresses, we will continue to plan for a third U.S.-Russian technical workshop in nuclear materials safety management that will likely be held in the United States in the summer of 1999. This meeting will enable both the U.S. and Russian government technical representatives to finalize the program plan, implementation schedule, technical working groups, and priority projects for FY2000 and beyond. This entire effort will be subject to the availability of funds provided by Congress and in-kind contributions provided by our Russian counterparts.

THE UNIVERSITY-TO-UNIVERSITY COMPONENTS

K. L. PEDDICORD
PAUL NELSON
MARVIN L. ADAMS
DAVID R. BOYLE
ANDREW MCFARLAND
JOHN POSTON
EMILE SCHWEIKERT
Texas A&M University
College Station, Texas 77843-3133 U.S.A.

V.V. KHROMOV
V. KLIMANOV
V. BOLYATKO
E. KRYUCHKOV
Moscow Engineering Physics Institute (MEPhI)
31, Kashirskoe shosse, Moscow,115409, Russia

MAX ROUNDHILL
Texas Tech University
Lubbock, Texas 79409-1061, U.S.A.

Y. KAZANSKY
I. VOROBYOVA
Institute of Nuclear Power Engineering
Studgorodok 1, Obninsk Kaluga,,249020 Russia

CARL BEARD
Amarillo National Resource Center for Plutonium
Amarillo, Texas 97101 U.S.A.

1. Introduction

The Nuclear Materials Safety activity seeks to enhance safety and contribute to the development of a safety culture. This will be achieved through interaction among research laboratories, institutes and industrial sites to address current practical problems in nuclear materials safety and safety management. In addition, a second area of emphasis will be collaboration among universities to establish new academic curricula and degree programs which incorporate safety into the education of young engineers and scientists. In addition, faculty and students can contribute to research projects as well.

In the NATO Advanced Research Workshop on Nuclear Materials Safety Management [1], a number of initiatives were identified that would contribute to enhanced safety in the handling, transportation, storage, and treatment of materials in the various components of the nuclear fuel cycle. Particular emphasis was given to excess fissile materials from the weapons program. It was recognized that an accident in either country involving fissile weapons materials might seriously hamper the disposition efforts in both countries. The Nuclear Materials Safety activity represents a collaboration between the U.S. Department of Energy (DOE) and the Ministry of the Russian Federation for Atomic Energy (Minatom) to enhance nuclear materials safety and contribute to mutual national goals.

2. The Nuclear Materials Safety Activity

To increase the level of safety in the handling, treatment, transportation, and storage of nuclear materials, the concept of the Nuclear Materials Safety activity has been developed. This has grown out of exchanges of information between DOE and Minatom following the Tomsk tank accident. The interaction grew to encompass broader aspects of the nuclear fuel cycle. This concept of the Nuclear Materials Safety activity arose in conjunction with the Amarillo NATO Advanced Research Workshop in March, 1997. The Nuclear Materials Safety Program boundary was defined to incorporate the aspects of the nuclear fuel cycle—excluding reactors—that result in the disposition of excess weapons fissile materials. The safety activity does not include other components in which joint DOE/Minatom interaction programs are in place.

A principal feature of the Nuclear Materials Safety activity is a set of lab-to-lab projects between the national laboratories of DOE and the research institutes and industrial sites of Minatom. Projects that have been carried out to date include consequence assessments due to the accidental release of radionuclides, safety in the processing of nuclear materials in nitric acid, and storage and disposal of radioactive materials in geologic repositories. However, it is recognized that in addition to lab-to-lab projects, important contributions to nuclear materials safety can be made by including university-to-university interaction.

3. The University-to-University Component

A fundamental goal of the Nuclear Materials Safety activity is to provide scientists and engineers with a fundamental understanding of safety. This leads to the idea of working with students who are studying in these fields. It will be these individuals who during their professional careers will become the group leaders, section heads, directors and agency heads and will determine the procedures, policies, and approaches used in the nuclear enterprise. By incorporating safety into the curriculum and the educational process, the graduates will be better prepared in these disciplines and will contribute to the development of a safety culture. Furthermore, the participation of students will give them a greater understanding and appreciation of the fundamental importance of safety.

4. Curriculum Initiatives

The university-to-university component of the Nuclear Materials Safety activity will pursue these goals through a partnership of Russian and U.S. universities. The academic component contains several facets. New courses and curricula on the topics of nuclear materials safety will be developed and incorporated into current courses. By utilizing existing programs, this holds the possibility of reaching a large number of students.

In another promising initiative, described in the following paper by Khromov, et al., the Moscow Engineering Physics Institute (MEPhI) has developed new and innovative approaches to graduate education leading to the Master of Science degree in which the central theme will be safety. Two degree programs are envisioned: an M.S. degree in Nuclear Materials Safety and a second Master of Science degree in Radiation Safety of Man and the Environment, which will incorporate elements of health physics and radiation protection engineering. The total number of credit hours to obtain these degrees is similar to other M.S. degree programs. However, the format of these programs is especially unique. MEPhI intends to offer the instruction in English. This will have two benefits. The participants in the program will be able to readily access the international literature in these technical fields and remain up-to-date with the latest advances. In addition, by offering the curriculum in English, MEPhI could become the global educational center in nuclear materials and radiation safety and attract students from all over the world.

In another non-traditional approach, the courses will be offered in intensive two- to three-week segments. This will allow the participation of visiting experts in the various fields to serve as instructors in the courses. This will expose the students to the leading individuals from around the world who teach classes in their fields of specialization. The students will benefit from meeting and learning from the acknowledged leaders. In addition, electronic teaching methods and distance learning technologies may make it possible for lectures to be delivered to the students from virtually any point on the globe.

A third important part of the new degree programs will be practical experience for the students in which they would be exposed to current industrial practices in nuclear materials safety. This may provide an excellent opportunity to link with companies and organizations in Western Europe which are handling and processing nuclear materials on a commercial basis.

The new MEPhI graduate programs are a valuable contribution to raising the educational level of nuclear materials safety and contributing to building a wide spread safety culture.

5. University Contributions to Research in Safety

In addition to the curricular initiatives, a further important feature of the university-to-university component will be to have the faculty participate in joint research projects. This interaction is envisioned in several forms. Joint projects involving nuclear materials safety may be identified involving faculty from U.S. and Russian universities. Even more effective will be those projects in which students at the graduate level and possibly the undergraduate level are involved as well. In addition, faculty and students at both Russian and U.S. universities might also be a part of the lab-to-lab projects. This would have the advantage of bringing the students into contact with the specialists in the Minatom and DOE institutes and laboratories. Several areas in which faculty and students might contribute to research activities in nuclear materials safety are discussed in the paper entitled "University Contributions to Research in Nuclear Materials Safety" in this ARW.

6. Conclusions

The university-to-university component is a vital part of the Nuclear Materials Safety activity. By incorporating safety into the curriculum and involving faculty and students as part of the research program, long term but instrumental advances will be made in incorporating safety into all facets of the nuclear fuel cycle.

Acknowledgments

This paper was prepared with the support of the U.S. Department of Energy (DOE), Cooperative Agreement No. DE-FC04-95AL85832. However, any opinions, findings, conclusions, or recommendations expressed herein are those of the author(s) and do not necessarily reflect the views of DOE. This work was conducted through the Amarillo National Resource Center for Plutonium.

References

1. Peddicord, K. L., Lazarev, Leonard N., and Jardine, Leslie J., *Nuclear Materials Safety Management*, NATO ASI Series, Kluwer Publishers, Dordrecht, Holland (1998).

THE MASTER OF SCIENCE GRADUATE PROGRAM IN NUCLEAR MATERIAL SAFE MANAGEMENT

V.V. KHROMOV
E.F. KYUCHKOV
V.I. SAVANDER
N.I. GERASKIN
A.N. CHMELEV
Moscow State Engineering Physics Institute (MEPhI)
31, Kashirskoe shosse, Moscow, 115409, Russia

K.L. PEDDICORD
DAVID R. BOYLE
PAUL NELSON
Texas A&M University
College Station, Texas 77843-3133, U.S.A.

CARL BEARD
Amarillo National Resource Center for Plutonium
Amarillo, Texas 79101, U.S.A.

1. Introduction

The Master of Science Graduate Program in Nuclear Material Safe Management is presented here. This new education program being conducted at MEPhI was developed in close collaboration with the U.S. Amarillo National Resource Center for Plutonium (ANRCP) and Texas A&M University (TAMU). The professional staff of the following MEPhI departments participated in preparing education courses: "Theoretical and Experimental Physics of Nuclear Reactors," "Radiation Physics and Ecology," "Physical Basis of Material Sciences," as well as leading specialists from the RRC Kurchatov Institute, State Committee "Gosatomnadzor," Bochvar Institute of Inorganic Materials (VNIINM), and the Institute of Chemical Technology (VNIIChT).

Solution of the problems that arise in providing comprehensive safety of nuclear materials (NM) forms a basis for wide-scale and effective utilization of nuclear power technologies. The broad knowledge and great practical experience gained so far through NM applications in various branches of human activity form a solid ground for the provision of safety. The need to preserve and develop this experience as well as its use in practical works with NM under permanent rigorous safety requirements dictates the necessity to train highly skilled specialists. Therefore, for the safe development of the Russian nuclear power industry, it is quite necessary to organize the systematic academic training of specialists who will be certified based on a special education program.

The program for certified specialist training in NM safe management must be developed at the university level and contain, as necessary components, a common methodological knowledge base and an education program, in which the experience of safe NM management gained in leading Russian and foreign organizations must be involved. Such a training program must reflect real current demands and be easily adaptable to the demands of the near future.

The Moscow State Engineering Physics Institute (MEPhI, Technical University) is a head Russian higher-education institute for specialists' training in nuclear sciences and technologies. Therefore, the education program is being developed just in MEPhI, which has already launched the specialists' training in NM Safe Management at the Master of Science level. The specialists' training is accomplished on basis of a deepened understanding of the technologies used for NM utilization (including facilities of nuclear fuel cycle), guidelines and rules of NM Safe Management, accounting for the international experience of NM protection against unauthorized proliferation.

2. Main Objectives and Concepts of the Graduate Program

The main purpose of the Graduate Program is to train a new generation of highly skilled specialists who are able in practice to develop and to upgrade the technologies for NM safe management. The Master of Science Graduate Program in NM Safe Management is based upon the concept of an integrated systematic approach to the analysis of existing technologies and the development of new technologies of NM safe management. This includes a comprehensive analysis of the full nuclear fuel cycle as a single whole from the viewpoint of safety and efficiency. This concept is realized by inclusion in an education program covering a wide spectrum of special disciplines. They cover the following branches:

- Human and environmental safety;
- Nuclear radiation and chemical safety at back-end part of nuclear power fuel cycle;
- Scientific and technical aspects of the nuclear fuel cycles based upon MOX-fuel utilization:
- Problems of radioactive wastes management, including their prolonged storage and ultimate disposal.

Special attention is paid in the Graduate Program to the problems of safe and effective utilization of the weapons-grade NM released in process of nuclear disarmament on basis of existing reactor technologies as well as for development of the closed nuclear fuel cycle. Great importance is given to getting an acquaintance with and mastery of modern computer codes and technologies used in accounting for NM safe management. The specialized nuclear fuel cycles are addressed that possess the protection elements against unauthorized application of fissile NM. The technologies for the treatment of long-lived radioactive wastes and making them harmless are analyzed. The following problems were solved in the Graduate Program development:

1. Integration in a common education program of
   —Deepened scientific and engineering knowledge of university level;
   —Systematic approach to analysis of existing and to development of new technologies for safe NM management;
   —Legal and political aspects of NM management;
   —Computerized information technologies of NM accounting for provision of safe and effective NM management.
2. Development of a common methodological basis in form of lecture courses, laboratory practical works, and computer codes.
3. Organization of the specialists' graduation, dissemination of the experience gained, accommodation and modernization of the Graduate Program.

3. The Master of Science Graduate Program

The Master of Science Graduate Program in NM Safe Management is intended for students who have an education at the Bachelor's degree level in technical physics. Such a level is provided for by MEPhI and other Russian technical universities by a traditional four-year (eight-semester) physical, mathematical, and engineering training in nuclear power technologies. This is the basic education level required to enter the Master of Science program, and it is used as the approved Bachelor Graduate Program in Physical and Technical Problems of Nuclear Power according to direction #553100-Technical Physics, which meets requirements of the RF Ministry on Common and Professional Education.

The education plan assumes training for the Master of Science in three education semesters followed by a period for performing and writing the Master's thesis. The Master of Science Graduate Program consists of two blocks of education courses. The first block includes the courses delivered in the 9th semester. These courses are common to two Master of Science Graduate Programs, namely: Graduate Program in Nuclear Material Physical Protection Control and Accounting (MPC&A) and Graduate Program in Nuclear Material Safe Management. These courses provide a basic education at the Master of Science degree level in natural science and mathematics; technologies applied at all the stages of nuclear fuel cycle from mining of uranium ore to ultimate disposal of radioactive waste; legal aspects; ecology; NM control and accounting; and fundamentals of NM physical protection system design.

The knowledge gained in learning the first block of courses is a prerequisite for the profiling courses of the 10th semester (the second block of education courses). These courses are directly related to specific safety problems of NM management at all the stages of the nuclear fuel cycle (NFC); environmental impact of the NFC in case of weapons-grade NM involvement into NFC, and in case of regenerated fuel recycling with economical assessment of such a fuel utilization efficiency. During both semesters, significant attention is paid to learning a foreign language.

The final, 11th semester is devoted to reviewing the main concepts of NFC safety, to performing the practical studies on the topic chosen during fulfillment of the training research work, for completion and presentation of the Master thesis carried out under scientific supervision of the Department's professional staff and leading specialists of the nuclear power branch.

TABLE 1. Education load for the students of the MS Graduate Program.

Full scope of education load	4280 academic hours
Scientific practical works	20 weeks
Preparation of Master thesis	16 weeks

The Graduate Program in Nuclear Material Safe Management is realized within the framework of the MS program "Physical and Technical Problems of Nuclear Power" in direction #553100-Technical Physics. The MS Graduate Program structure and scope of education load completely corresponds to requirements of the RF Ministry on Common and Professional Education.

The range of the knowledge obtained in pursuing the MS Graduate Program covers detailed comprehension of the safe technologies used for NM management, NM control and accounting, and control over NM non-proliferation. This knowledge will enable the graduates to work in scientific research, development, design and utility entities of MINATOM, Government regulatory bodies, as well as to become a teacher after obtaining enough practical experience.

4. Conclusions

A significant scope of the work to develop and implement the MS Graduate Program was performed, in collaboration with specialists from leading scientific centers, during the period 1996 to early 1998. Existing education courses and laboratories are being upgraded while new ones are being developed. MEPhI professors participated in a number of the training courses, workshops, and conferences held by Russian and foreign scientific centers. The Graduate Program in Nuclear Material Safe Management was developed and now the program is being successfully implemented. From September 1, 1997, in accordance with the Graduate Program, based on the planned graduation of the bachelor's degree candidates, the process of student training was launched for the future Master of Science candidates who will graduate from MEPhI in 1999.

The authors express their gratitude to all the specialists who participated in the development and implementation of this Master of Science Graduate Program.

UNIVERSITY CONTRIBUTIONS TO RESEARCH IN NUCLEAR MATERIALS SAFETY

MARVIN L. ADAMS
IGOR CARRON
ANDREW R. MCFARLAND
PAUL NELSON
EMILE A. SCHWEIKER
Texas A&M University
College Station, Texas 77843-3123, U.S.A.

HANI MAHMASSANI
The University of Texas at Austin
Austin, Texas 78712, U.S.A.

CARL A. BEARD
Amarillo National Resource Center for Plutonium
Amarillo, Texas 79101, U.S.A.

JOHN C. RODGERS
Los Alamos National Laboratory ESH-4
Los Alamos, New Mexico 87545, U.S.A.

VICTOR V. BOLYATKO
EDWARD F. KRYUCHKOV
Moscow Engineering Physics Institute, 115409
Moscow, Russian Federation

D. MAX ROUNDHILL
Texas Tech University
Lubbock, Texas 79409-1061, U.S.A.

1. Introduction

Extant and planned collaborative research, related to nuclear materials safety and between the universities of the consortium operating the Amarillo National Resource Center for Plutonium and various Russian institutions of higher education, is outlined. The research includes activities in the following fields: sampling of airborne radionuclides, transportation of hazardous materials, safe removal and processing of nuclear materials, quality assurance practices, and disposition of weapons-grade plutonium in reactors.

Collaboration and cooperation between the Moscow Engineering Physics Institute (MEPhI) and the institutions of the consortium of universities (Texas A&M University System, Texas Tech University, and the University of Texas System.) that operate the Amarillo National Resource Center for Plutonium (ANRCP) extend back to 1990. At that time Professor Paul Nelson, of Texas A&M University (TAMU) and then serving as Chair of the Mathematics and Computation Division of the American Nuclear Society, accompanied Mrs. Betty F. Maskewitz, who was recently retired as Director of the well-known Radiation Shielding Information Center at the Oak Ridge National Laboratory (ORNL), on a visit to MEPhI. This visit occurred at the initiative of Dr. Victor V. Bolyatko, of what was then the

Department of Radiation Shielding at MEPhI. This was one of the first occasions for westerners from a nonsocialist country to visit the campus of MEPhI.

The visit to the Soviet Union by Mrs. Maskewitz and Professor Nelson had been arranged by the well-known transport theorist Professor Tatiana A. Germogenova of the Keldysh Institute of Applied Mathematics (KIAM) in Moscow. It was thus natural for many of the early cooperative efforts to focus upon the subject of computational transport theory. Some of the more notable activities within this framework were:

- Participation of a sizeable Soviet delegation in the International Topical Meeting on Advances in Mathematics, Computations and Reactor Physics, held in Pittsburgh, April 28–May 2, 1991. This participation was made possible through the financial support of the Academy of Sciences of the USSR, the American Nuclear Society (both headquarters and the host Pittsburgh Section) and TAMU. The proceedings of this biennial topical conference of the Mathematics and Computation Division of the American Nuclear Society were published by the American Nuclear Society [1].
- The Joint Workshop on Numerical Transport Theory, November 1–6, 1991, on the campus of TAMU in College Station, Texas [2]. This workshop was jointly sponsored by the Academy of Sciences of the USSR and the US National Science Foundation.
- The presence of a sizeable American (and other international) delegation at the International Symposium on Numerical Transport Theory, held in Moscow on May 26–28, 1992. The American participants principally were from academic institutions, but at least two Department of Energy (DOE) laboratories (Los Alamos and Oak Ridge) also were represented.
- Preparation of a proposal for the project *Partnership for Basic Research and Education in Nuclear Reactor Safety and Novel Applications of Transport Theory.* This proposal was submitted to, and ultimately accepted by, the International Science and Technology Center. KIAM was the lead organization for this project, with Professor Mikhail V. Maslennikov as Project Manager, and Dr. A. V. Voronkov as the Project Leader at KIAM, and Professor Nelson as Principal Consultant. Other participating Russian organizations were the Institute of Mathematical Modelling (Professor V. Ya. Gol'din, Project Leader) and MEPhI (with Dr. Bolyatko as Project Leader).
- Participation of a sizeable Russian delegation in the International Conference on Mathematics and Computations, Reactor Physics and Environmental Analyses, held in Portland (Oregon), April 30–May 4, 1995. This participation was made possible through the generous financial support of the International Science Foundation. The proceedings of this biennial topical conference of the Mathematics and Computation Division of the American Nuclear Society were published by the American Nuclear Society [3].

The ANRCP was founded in 1994, under a Cooperative Agreement between the DOE and the State of Texas. The mission of the ANRCP, under the terms of this initial Cooperative Agreement, is to "serve as a scientific and technical information resource on issues relating to the storage, disposition, potential utilization, and transportation of plutonium, high explosives, and other materials generated from weapons assembly and disassembly," in order to permit "concerned citizens and policy-makers in the state of Texas and elsewhere" to "be able to address issues with independent and reliable technical and scientific information." In furtherance of this mission, the Governing Board of the ANRCP under the leadership of Professor Dale E. Klein, in his capacity as Executive Director of the ANRCP, identified the importance of developing and maintaining cooperation and collaboration with Russian institutions of higher education having similar interests and capabilities with respect to the management of weapons-related nuclear materials. As of June 1, 1998, the members of the Governing Board are Dr. Kathleen E. Harris (representing Texas Tech University), Professor Dale E. Klein (Chair, and representative of the University of Texas System), Mr. Wales Madden (representing the Panhandle Region of Texas), and Professor Kenneth L. Peddicord (representing the Texas A&M University System).

Among institutions of higher education within the former Soviet Union, MEPhI clearly played a pre-eminent role within the Soviet nuclear weapons complex. Given that, it was natural for the ANRCP to seek to achieve this cooperation and collaboration by building upon the already existing relation, as outlined above, between MEPhI and Texas institutions of higher education. This has been accomplished primarily through a contract to MEPHI, from the ANRCP, for *Cooperative Studies in the Utilization and Storage of Excess Weapons-grade Plutonium.* The objective of this paper is to outline the present status and future plans for these and other cooperative efforts, especially as they are germane to the planned Nuclear Materials Safety Activity (NMSA). A smaller cooperative effort between the ANRCP and the Institute of Nuclear Power Engineering (INPE) at Obninsk also will be briefly described.

It is only appropriate to conclude this introduction by mentioning that NATO support played a key role in permitting the development of collaborations, on topics related to disarmament technologies, between the ANRCP universities and their various Russian counterparts. Some of the specific activities permitted by such support include the following:

- The award for "Neutronics of VVER Reactors," which permitted Professor V. V. Naumov (MEPhI) to visit the ANRCP universities and ORNL in November 1995, for the purpose of lecturing and consulting on this topic;
- The linkage grant for "Utilization and Protection of Excess Weapons-Grade Fissile Material," and the collateral grant for upgrading computing facilities at MEPhI;
- The NATO ARW on "Nuclear Materials Safety Management," the predecessor to this meeting, which was held in Amarillo in March 1997;
- The NATO ARW on "Problems in Nuclear Reactor Physics," held in Moscow in September 1997.

2. Transportation

As part of the contract for *Cooperative Studies in the Utilization and Storage of Excess Weapons-grade Plutonium,* between the ANRCP and MEPHI, there is a task directed toward "Issues of Safe Transportation in the Storage or Utilization of Excess Weapons-grade Plutonium." The senior individuals responsible for this task at MEPhI are Dr. Bolyatko and Dr. V. V. Kosterev (Principal Investigator). This work has taken the form of development of a computerised model for the assessment of risks from accidents and terrorist attacks during rail shipment of sensitive materials (e.g., weapons-grade plutonium). The model is based on an inference tree approach, with allowance for inference rules that treat either crisp or fuzzy descriptors of the probabilities of various types of individual events, and corresponding inference rules tailored to these types of descriptors. For the fuzzy analysis a computer code processes inference rules by implementing the Extension Principle of Zadeh [4] through the "α-cut" algorithm. Geographic information system technology and electronic maps of geographic area are involved as an instrument for handling cartographic data. Illustrative examples of the application of this methodology are given in [5,6].

This task on transportation issues is one of six tasks carried out by MEPhI, under the contract described above. The interested reader is referred to [7] for a detailed progress report for the performance period January 16, 1996, through January 15, 1997, on all six tasks. A similar progress report is in preparation for the more recent performance period, January 16, 1996, through January 15, 1997.

Within the universities of the ANRCP Consortium, there have been two distinct but coordinated efforts related to issues arising from transportation of sensitive materials. The group at the University of Texas at Austin (UTA), under the leadership of Professor Hani Mahmassani, has focused upon the development of improved algorithms for the routing of shipments of hazardous materials via road. This effort was initiated with a review and survey of criteria and models for the routing of radioactive materials [8]. Bowler and Mahmassani [9] subsequently adapted a time-dependent least-cost path algorithm to the problem of shipment of hazardous materials with curfews and waiting times, and applied this algorithm to illustrative examples of shipments of radioactive materials. Miller-Hooks and Mahmassani [10] applied recent advances in network analysis methodologies to the problem of selecting routes for shipment of hazardous materials. Several different procedures were considered, including both efficient heuristics and computationally intensive exact algorithms. All procedures considered both travel time and costs, but several different scenarios were considered in terms of the cost assigned to risks. This work received the Charlie Wootan Best Dissertation Award for 1997 from the Council of University Transportation Centers.

The group at TAMU, under the leadership of Professor Nelson, has focused upon developing and demonstrating the capability to use standard DOE methodology for the assessment of risks associated with shipment of radioactive materials. After an initial survey of available methodologies, the decision was made to concentrate upon the RADTRAN code, developed by the Sandia National Laboratories. Not the least of the reasons underlying this choice is the ready availability of TRANSNET by Internet access; see Section II of [11]. However, the documentation [12, 13] available at the time this work was done (1996 and early 1997) did not reflect the menu-driven version of RADTRAN that actually was on-line at that time. Therefore a supplemental user guide [11] that did reflect the features of the on-line version was developed.

The capability to use RADTRAN was subsequently demonstrated through a comparative risk assessment of two hypothetical versions of a campaign to transform the surplus weapons-grade plutonium currently stored at the DOE's Pantex site (near Amarillo) into mixed-oxide fuel, and ship it to a reactor site for utilization. In one of these hypothetical campaigns, the conversion and fabrication were performed on-site at Pantex, whereas in the second version the plutonium pits were shipped from Pantex to an intermediate site for conversion into oxide form and fabrication into fuel. The interested reader is referred to [14] for the results of this assessment.

The following are some potential elements of the NMSA that relate to transportation of radioactive materials:

- Development of a version of RADTRAN, or other code for analysis of the risk associated with transportation of radioactive materials, that is suited to the rail transportation system used in Russia;
- Development of materials for education of the public regarding the true magnitude of the risks stemming from accidents and incidental radioactive exposure occurring in the transportation of radioactive materials. (This is a problem that is common to the United States and Russia. Intuitive assessments by the general public, and even by nuclear professionals who have not had specific occasion to study the issue, tend to *substantially* overestimate the risks associated with the transportation of radioactive materials);
- Development of capability for assessing risks stemming from deliberate acts of sabotage or terrorism during transportation of radioactive materials, and education of the public regarding the magnitude of these risks. Again, these are problems that are common to the U.S. and Russia. The latter is a singularly difficult point, because on the one hand information regarding techniques used to enhance security seems to be an essential element of public confidence, while on the other hand this type of information could be used by potential malefactors to defeat the security;
- Development of specialised network algorithms for radioactive material routing and scheduling strategies that explicitly recognise the multi-objective and fuzzy nature of the associated risks, and that may be applied on-line in conjunction with real-time information on shipment location as well as up-to-the-minute updated risk assessments;
- Development of computer software to achieve optimum strategy for safe transportation of radioactive materials using fuzzy set theory methods.

3. Sampling of Airborne Radionuclides

3.1 INTRODUCTION

There have been a number of advancements in methodologies for characterizing the air environment associated with nuclear facilities. Included are:

- New methods for sampling stacks and ducts, including software for design of sampling systems;
- New apparatus for monitoring the nuclear workplace;
- New equipment for monitoring outdoor air in the proximity of a location where there is a nuclear emergency.

Documentation on the first two elements has been prepared and the materials are being translated into Russian; textual materials for the third topic are currently being prepared.

3.2 STACK SAMPLING

TAMU and Los Alamos National Laboratory (LANL) developed new methodologies for sampling stacks and ducts of the nuclear industry. These new methodologies serve as the foundation for revisions of the existing stack sampling standards of the American National Standards Institute (ANSI N13.1-1969) and the International Standards Organization (ISO 2885:1975). The ANSI and ISO standards had sought to obtain representative samples by defining the designs of the apparatus and by specifying that sampling must be done isokinetically if particles larger than 2- to 5-µm diameter could be present. Multi-point rakes of small diameter nozzles were required in larger ducts to try to average any non-uniformities in the contaminant concentration profile. However, the effect produced by use of the prescribed protocol was just the opposite if its intent; namely, the samples were generally not representative, but rather quite biased against larger particles. The reason is that larger particles could be lost to the internal walls of the sampling nozzle and transport line through the mechanisms of turbulent deposition and gravitational settling. The turbulent depositional losses were exacerbated by the use of small diameter transport lines (e.g., the lines were often less than 10-µm diameter).

The fundamental concept of the new methodologies developed by TAMU and LANL is that a representative sample can be obtained from a single point in a flow field provided the fluid momentum and any contaminants are both well mixed across the cross section of the flow. Criteria are placed on the performance of the sampling system; however, the system design is at the discretion of the user. For aerosol sampling, this requirement is fulfilled if it can be demonstrated that the velocity, tracer gas, and 10-mm aerodynamic diameter (AD) aerosol particle profiles are relatively flat at the sampling location. It must also be shown that the sampling probe will acquire a representative sample from the undisturbed flow stream in the stack or duct, and that at least 50% of 10-mm AD aerosol particles will be transported from the undisturbed free stream to a collector or analyser.

The manual that has been prepared and translated into Russian provides information on the methodology that can be used to demonstrate the uniformity of velocity, tracer gas, and aerosol profiles. Also, an English language version of PC-based software is provided that can be used to evaluate the aerosol penetration through sampling systems.

3.3 MONITORING AEROSOLS IN THE NUCLEAR WORKPLACE

Two types of transuranic aerosol monitoring devices are used in nuclear laboratories: filter air samplers (FASs) and continuous air monitors (CAMs). The purpose of a FAS is to provide a basis for estimating the dose to workers. This is achieved by operating the FAS for a given period of time (e.g., one week), and then measuring the radioactivity collected by the filter. The analysis is performed remotely and is done after sufficient time has elapsed for background radiation from radon and thoron progeny to decay. If there is an inadvertent aerosol release during the sampling period, the filter will generally be removed as soon as possible after the release and immediately analysed. The function of a CAM is to provide a near-real-time alarm for releases that are either intense and of short duration or low level and prolonged. Both FASs and CAMs should be capable of efficiently collecting samples and they should be strategically placed. Placement of CAMs is a particularly important factor because it may provide the only warning to a worker, yet because CAMs are expensive, their number is necessarily limited.

Collaborative work at LANL and TAMU has led to the development of new FASs and CAMs. The new FAS is fitted with a filter cartridge that accommodates both easy replacement in the field and achievement of quality assurance aspects of sample tracking. The device has a critical flow venturi that provides a constant flow rate through the filter independent of the level of applied vacuum, provided the vacuum exceeds a threshold of about 100 mm Hg. The particle sampling characteristics of the FAS have been determined through testing of the device in an aerosol wind tunnel. The new CAM features advanced background radiation suppression through use of mechanical means for stripping unattached radon/thoron progeny and numerical means for subtracting the radon/thoron progeny that are collected on the sampling filter.

The manual on FASs and CAMs, which is currently being translated into Russian, contains information on their design and operation, and on placement of CAMs. It is intended to serve as a text for either a short course on workplace sampling or a part of a college-level course on air sampling.

3.4 EMERGENCY RESPONSE MONITORING

In case of an accidental release, where the environment outside of a nuclear facility could be affected, it is highly desirable to obtain near-real-time data on the radionuclide concentration and projected atmospheric dispersion. Similarly, when contaminated soil is being removed, handled or processed, it is important to know if excessive amounts of radionuclides are being aerosolised. LANL has developed a transportable sampling system that allows near-real-time acquisition of radionuclide concentrations and meteorological parameters. The system includes a modified Canberra CAM, wind speed and direction indicators, and telemetry equipment. Lawrence Livermore National Laboratory (LLNL) has developed a computer code, HOTSPOT, that predicts the dispersion of contaminants in the atmosphere following an accident such as an explosion. The use of HOTSPOT and LANL environmental CAMs would facilitate estimation of the source strength, and subsequent near-real-time estimates of the impact of the accident. At the present time, the materials needed for coursework on the emergency response sampler is only partially complete. However, HOTSPOT documentation is currently available in an English language PC software code.

4. Safe Removal and Processing of Nuclear Materials

This work has been carried out by a group at Texas Tech University, within the Department of Chemistry and Biochemistry and the Strategic Metals Recovery Research Center, under the leadership of Professor D. Max Roundhill. The effort there has focused upon developing both computational and experimental strategies related to the safe removal and processing of nuclear materials. The group has collaborative interactions with both LANL, in conjunction with the ANRCP, and the Pacific Northwest Laboratory (PNL).

The ANRCP effort involves working with LANL to devise effective ways of removing gallium from weapons-grade plutonium, and to design experiments that will allow for such a process to be carried out on a batch production scale. The Texas Tech group is primarily involved in addressing the analytical problems that are involved in such an effort.

The collaborative program with PNL includes a team of theorists and experimentalists who are focused on identifying structural features that need to be incorporated into complexants in order for them to be selective for specific metal radionuclides. The project goal is to develop sufficient understanding of the problem so that it will be possible to design a complexant that will be specific for any particular metal radionuclide. Such knowledge would allow for strategies to be developed for the safe separation and handling of nuclear materials in order for them to be subsequently processed.

5. Quality Assurance Practices

A key aspect of the planned NMSA is to promote internationally recognised quality assurance (QA) practices. There are multiple reasons for introducing quality management standards in laboratories and processing facilities dealing with nuclear materials, including: the need for a system of documentation, prevention of repetition of work, traceability of experimental/operating conditions, and trustworthiness of the laboratory or processing facility.

It is the function of management to determine and implement a quality policy. A quality system consists of two parts: quality control for operation techniques and activities and internal QA to ensure that the intended quality is achieved. Irrespective of whether the managers of an organization believe they are operating under a quality system, it must be demonstrably visible to outsiders that the organization is indeed operating under such a system.

A growing number of industrial and commercial organizations, particularly those operating on an international scale, have set up or are implementing quality management systems. Curiously, universities and research institutes have so far largely ignored the needs and opportunities for education in this field. This lack of interest is remarkable, given that they train the future leaders for R&D, industrial, and managerial activities. The rationale for institutions of higher education to integrate quality management principles and practices in their science and engineering curricula is readily apparent:

- Minimise failure and repetition of experiments due to improper planning and/or execution;
- Ensure traceability of experiments and procedures;
- Prepare students for the needs of a technologically competitive society.

One might cite a couple of reasons for the lack of interest in and practice of quality management by universities. One argument could be that university graduates have the training to carry out their work without written procedures. However, a scientist or engineer can then become the “single point of failure”: the individual has the experimental/operational details in his/her head. When such a professional leaves, a tremendous amount of experience is lost. University and research laboratories also tend to assume implicitly that their procedures are of high quality. This assumption is often not verified. For instance, intercomparisons of analytical laboratory results have repeatedly demonstrated that laboratories that consider themselves “highly experienced” cannot produce the quality of data one would expect from this classification.

The NMSA provides an outstanding opportunity to foster demonstrable QA practices in the participating university programs and research institutes. The appropriate guides for implementing and evaluating QA practices are the ISO 9000 standards developed by the International Organization for Standardization [15-17]. The NMSA should spearhead the integration of quality management principles and practices into science and engineering curricula. The benefits of inculcating ISO 9000 standards in present and future generations of scientists and engineers will go far beyond the nuclear industry.

6. Plutonium Disposition in Reactors

There has been a great deal of collaborative research among universities in Texas and Russia on the issues surrounding the disposition of weapons plutonium in existing reactors. This research has studied questions ranging across the entire fuel cycle, from the creation of PuO_2 powder all the way to characterization of spent fuel. Most of the research has focused on questions related to safety. For example, the main goal of the reactor design studies has been to determine the safety margins of reactors that have various distributions of Pu in the fuel.

In any process or facility (excluding a reactor) that involves fissile material, it is obviously necessary to maintain material configurations that are subcritical. In the design of such processes and facilities, computer codes are used extensively to calculate multiplication factors of various configurations. Before they can be truly useful, these codes (with their cross section libraries) must be carefully benchmarked against experiments and their biases must be carefully quantified. The goal is a statement something like “If this code says that this type of configuration has a multiplication factor of 0.96 or less, then there is less than one chance in a million that the actual configuration will be critical.” There is a significant collaborative effort underway to attain this goal for several criticality codes in the U.S. and in Russia. This collaboration includes TAMU, UTA, ORNL, INPE at Obninsk, the Kurchatov Institute, and the Obninsk Institute for Physics and Power Engineering. See [18] for a recent progress report.

Given fuel pellets made from weapons Pu, there is a potential for corrosive interactions between fuel pellets and their “cladding,” which is material that separates the pellets from the coolant water in a reactor. This is because the corrosive element gallium is present in some weapons plutonium. TAMU and UTA have been collaborating with ORNL and LANL on several studies of this important question. One study is investigating dry chemical methods for removing the gallium; see [19] for details. Another task is studying the microscopic interaction between gallium and cladding under simulated irradiation conditions; see [20] for details. Yet another task studies the migration of gallium induced by temperature gradients in fuel pellets, as well as its subsequent interaction with cladding at elevated temperatures.

Because neutrons interact differently with plutonium than with uranium, one cannot simply replace uranium fuel pins with plutonium pins in a reactor. [A Pu-bearing fuel pin contains a mixture of plutonium dioxide and uranium dioxide, with the uranium usually depleted in U-235. This is usually called a “mixed-oxide” (MOX) fuel pin.] The challenge is to design and place MOX assemblies in a reactor in such a way that: (1) the power distribution is acceptable, with no “hot spots” that could lead to fuel failures; and (2) control and safety systems still function effectively. The first requirement can be difficult to meet if only a portion (say 1/3 or 1/2) of the reactor's pins are MOX, for power tends to peak in MOX pins that are adjacent to uranium pins. The second requirement can be difficult to meet if most or all of the reactor's pins are MOX, for a reactor's control systems tend to be less effective in the presence of Pu. Universities have participated in several collaborative efforts that have addressed these challenges. A collaboration between TAMU and Westinghouse designed fuel assemblies and loading patterns for the transition from a full uranium core to a partial MOX core such that safety margins were met [21]. A collaboration between Texas A&M University and ABB-Combustion Engineering designed fuel assemblies and loading patterns for an equilibrium cycle containing over 50% MOX, such that all calculated safety margins were met [22]. MEPhI has studied the utilization of weapons plutonium in Russian VVER-1000 reactors; see [7] for details. Finally, there is an ongoing collaboration between TAMU, North Carolina State University, and ORNL to address possible interruptions in MOX fuel supply during the disposition effort. This research aims to determine the best way to maintain reactor safety margins in the event that an interruption of the MOX fuel supply forces a fuel reload with only uranium fuel.

Researchers at TAMU and UTA have engaged in a small collaborative effort with ORNL to quantify important characteristics of spent MOX fuel from a reactor-based disposition effort. This is important for verification that the "spent fuel standard," which requires that weapons Pu be rendered as unattractive for future weapons use as is the Pu in existing commercial spent fuel, will be met by the reactor-based option. It is also important in the planning of storage, shipping, and ultimate disposal of the spent fuel. In a somewhat related but different effort, Savander and Glebov of MEPhI have devised a procedure for sensing whether a given "burned" fuel assembly was MOX or uranium. This procedure can be used to avoid the accidental misplacement of fuel assemblies during reload, which is very important for maintaining reactor safety margins [7].

In summary, there have been broad and significant contributions made by universities in the U.S. and Russia toward addressing important safety questions that surround the handling and disposition of weapons material. These contributions are continuing as the universities continue to play an active role in ongoing work.

7. Conclusions

The universities of the ANRCP have a strong history of collaboration with both counterpart Russian institutions of higher education, and with other institutes that supported the nuclear weapons complex within the former Soviet Union. Those collaborations are currently vital and active, and will become even more so under the planned Nuclear Materials Safety Activity.

These university research programs have a dual role: they both generate new information that will be beneficial to safety practice, and help to inculcate a safety culture into the nuclear practitioners and leaders of the next generation. As the NMSA unfolds, one can expect synergistic interactions to emerge between its laboratory-to-laboratory and university-to-university components that will even further enhance both of these objectives.

As an example of the types of synergism that might emerge, the Melcor Accident Consequence Code System (MACCS) is central to a currently ongoing laboratory-to-laboratory project between the V. G. Khlopin Institute and the Sandia National Laboratories.

One can envision the possibility of this important tool becoming integrated into existing course work on nuclear safety that exists at a number of the institutions of higher education mentioned above. As students become familiar with MACCS through these courses, it is almost inevitable that some will seek, in the projects they subsequently undertake for their theses or dissertations, to demonstrate additional capabilities of this important tool, and to add such capabilities as the need is indicated

References

1. Procs. of the International Topical Meeting *Advances in Mathematics, Computations, and Reactor Physics*, Vols. 1-5, American Nuclear Society (La Grange Park, Ill.), 1991 (ISBN # 0-89448-161-4).
2. The proceedings of this workshop were published as *Transport Theory and Statistical Physics,* **22**, Nos. 2 & 3 (1993).
3. Procs. of the International Conference *Mathematics and Computations, Reactor Physics*, *and Environmental Analyses*, Vols. 1 & 2, American Nuclear Society (La Grange Park, Ill.), 1995 (ISBN # 0-89448-198-3).
4. Zadeh, L.A. (1973) Outline of a new approach to the analysis of complex systems and decision process, *IEEE Trans. Sys. Man. Cyb.*, SMC-3, No.
5. Kosterev, V.V., Bolyatko, V.V., et al. (1998) Methods of fuzzy set theory applied to railway transportation risk assessment. International conference on soft computing and measurements *(SCM '98)*, St. Petersburg, June.
6. Kosterev, V. V., et al. (1998) Computer modelling for risk assessment of emergency situations and terrorist attacks during transportation using methods of fuzzy set theory, Procs. of the 12th International Conference on the Packaging and Transportation of Radioactive Materials *(PATRAM 98)*.
7. Bolyatko, V. V. (1998) "Cooperative Studies in the Utilization and Storage of Excess Weapons-grade Plutonium," Report No. ANRCP-NG-TWD-98-01, Amarillo National Resource Center for Plutonium, January 29.
8. Bowler, L. A. and Mahmassani, H. (1997) "Routing Criteria and Models for Radioactive Materials: A Review," Report No. ANRCP-NG-ITWD-97-08, Amarillo National Resource Center for Plutonium, September 9.
9. Bowler, L. A. (1997) "Routing of Radioactive Shipments in Networks with Time-varying Costs and Curfews," M. S. Thesis, Civil Engineering, University of Texas at Austin, December. (In preparation as an ANRCP report.)
10. Miller-Hooks, E. (1997) "Optimal Routing of Spent Plutonium and Other Hazardous Substances in Time-varying, Stochastic Transportation Networks," Ph.D. Dissertation, Civil Engineering, University of Texas at Austin, August. (In preparation as an ANRCP report.)
11. Caldwell, A. B. (1996) "RADTRAN User Guide Supplement," Report No. ANRCP-NG-ITWD-96-03, Amarillo National Resource Center for Plutonium, October 8.
12. Neuhauser, K. S. and Kanipe, F. L. (1992) "RADTRAN 4: Volume 2: Technical Manual," Report No. SAND 89-2370, prepared under Contract DE-AC04-76P00789 by Risk Assessment and Transportation System Analysis Division, Sandia National Laboratories, Albuquerque, NM and Livermore, CA for the U.S. Department of Energy, January.
13. Neuhauser, K. S. and Kanipe, F. L. (1992) "RADTRAN 4: Volume 3: User Guide," Report No. SAND 89-2370, prepared under Contract DE-AC04-76P00789 by Risk Assessment and Transportation System Analysis Division, Sandia National Laboratories, Albuquerque, NM and Livermore, CA for the U.S. Department of Energy, January.
14. Caldwell, A. B. (1997) "Risk Analysis of Shipping Plutonium Pits and Mixed-oxide Fuel," Report No. ANRCP-NG-ITWD-97-07, Amarillo National Resource Center for Plutonium, July 3.
15. "ISO-9000 Standard," International Organization for Standardization, Geneva, Switzerland, 1987.

16. J. Cottman, R. J. (1992) "Guidebook to ISO 9000 and ANSI/ASQC Q90," ASQC Quality Press, Milwaukee, WI.
17. Hutchins, G. B. (1993) "ISO 9000: a comprehensive guide, Oliver Wright Co. Press, Essex Junction, VT, 1993.
18. "Neutronics Benchmarks for the Utilization of Mixed-Oxide Fuel: Joint U.S./Russian Progress Report for Fiscal Year 1997" ORNL/TM-13603, Oak Ridge National Laboratory: Vol. 1 (Executive Summary) expected 1999; Vol. 2 (Calculations Performed in the United States) anticipated October 1998; Vol. 3 (Calculations Performed in the Russian Federation) has appeared; Vol. 4 (Additional Mixed Oxide Experiments) anticipated September 1998.
19. Philip, C. V., Pitt, W. W. Jr., Chung, K. and Anthony, R. G. (1997) Solid/Gas Catalytic Reduction for the clean Separation of gallium from Gallium Oxide/Plutonium Oxide, Procs. Plutonium Futures - The Science: Topical Conf. on Plutonium and Actinides, pp. 243-244, Santa Fe, NM, August 25-27, Los Alamos National Laboratory.
20. Hart, R. R., Rennie, J., Ünlü, K. and Ríos-Martinez, C. (1997) Gallium Interactions with Zircaloy Cladding, Procs. Plutonium Futures - The Science: Topical Conf. on Plutonium and Actinides, pp.105-106, Santa Fe, NM, August 25-27, Los Alamos National Laboratory.
21. Ankney, R. D., Alsaed, A. A. and Adams, M. L. (1997) Transition Cycle Fuel Management Using Weapons-grade Mixed-oxide Fuel, Procs. Conf. on Advances in Nuclear Fuel Management II, pp.18-17 through 18-29, Myrtle Beach, SC, March 23-26, American Nuclear Society.
22. Kantrowitz, M. L., Rosenstein, R. G., Alsaed, A. A. and Ramone, G. L. (1997) Core Designs for Disposition of Weapons-grade Plutonium in System-80 Reactors, Procs. Conf. on Advances in Nuclear Fuel Management II, pp. 18-9 through 18-16, Myrtle Beach, SC, March 23-26, American Nuclear Society.

THE LAB-TO-INSTITUTE COMPONENTS—RF VIEW

LEONARD N. LAZAREV
V.G. Khlopin Radium Institute
28, 2nd Murinsky Ave., 194021, St. Petersburg, Russia

1. Background

The present workshop on the Safety of Nuclear Materials Management is the fifth meeting of experts on this problem. A large body of important information has been submitted to these workshops. The exchange of experience is certainly useful for a reduction of the probability that incidents of different kinds may occur. However, along with the workshop form of cooperation, specific joint investigations are necessary for the most urgent problems. Although such programs are underway, they are not numerous.

Short reviews of these investigations were contributed earlier to the Conference SPECTRUM'96 [1] and the Fourth Workshop on Safety of Nuclear Materials Management in Amarillo [2].

This report is devoted to projects now underway and to the prospects of joint research development.

Recently, beginning in autumn 1997, two investigations were started on the incorporation of excess plutonium into glasses or ceramics for disposition of nuclear material. The customer for these works is Lawrence Livermore National Laboratory (LLNL). Experimental studies are carried out in Russia at V.G. Khlopin Radium Institute (KRI) and A.A. Bochvar All-Russian Institute of Inorganic Materials (VNIINM).

A working meeting of American and Russian specialists held in Livermore in May 1997 was an important step in stating the tasks for studies and choosing promising matrices for plutonium immobilization. The American party presented reports describing the results of laboratory studies of compositions based on glass and ceramics. It was also announced that, in the United States, the preference is given to ceramics. Russian specialists had a chance to visit an LLNL plutonium facility where immobilization studies are being conducted.

The task of studies carried out at KRI and VNIINM consists of selecting an optimum composition for incorporation of plutonium. Results of this work will be used to design pilot facilities, if a corresponding strategic decision is made. It should be remembered that in Russia the use of excess plutonium in mixed oxide (MOX) fuel is preferred.

In 1997, the investigation on the subject “Safety of anion-exchange processes in nitric acid media” was completed at KRI. The Military Academy of Chemical Protection and the Production Association “Mayak” participated in this project. The work was performed on order of Sandia National Laboratories (SNL) and the Savannah River Plant. Studies were carried out using Russian anionite VP-1AP. Recommendations were made on prevention of emergency situations. The results of this investigation were reported at the 10th Symposium on Separation Science and Technology for Energy Applications [3]. It is expected that the studies will be continued with the use of American vinyl pyridine anionites.

It should be noted that, after completing this work, preliminary data have been obtained showing the necessity of redetermination of permissible process parameters for ion-exchange purification in the case when the resin phase is saturated with nitric acid complexes of actinides. In particular, due to increase of nitrate ion concentration, the temperature of possible thermal explosion development decreases sharply. Therefore, it is urgent that additional studies be performed that take this factor into account.

In 1995, a joint project was started between SNL and KRI concerning the analysis of radiological accident consequences. The content and results of this work were reported at the preceding Workshop [4]. At present, KRI under the contract with SNL performs the verification of MACCS-2 Code by comparing calculated results with real field observations of radioactive contamination formed in 1957 during the explosion of the high-level tank in Chelyabinsk-65.

The enumeration of joint investigations shows that this form of collaboration has not yet got a systematic basis. Therefore, the information about an integrated Program of joint research distributed before the Workshop is of particular interest. I requested that Russian specialists look into the Draft Statement of Work (SOW) before defining the topics and content of their reports. Those giving reports were asked to give proposals on the participation of their institutions in tasks of joint research defining the priorities and the distribution of work between participating organizations. The discussion of these questions in Working Groups will allow us to define more concretely the Statement of Work.

2. Program Milestones

Concerning the milestones of the Lab-to-Lab component of the Program, the following proposals may be put forward:

2.1.1 Tasks L-1 and L-2

From the Russian party all Program elements should be executed under the direction of the Minatom Department of Safety, Ecology, and Emergency Situations. The V.G. Khlopin Radium Institute is able to be a center of scientific coordination in cooperation with Lawrence Livermore National Laboratory. In fact, KRI has played this role for some years.

2.1.2 Task L-3

This suggests the development of organizational structures for collaboration between U.S. National Laboratories and Research Institutes and Industrial Sites of the Russian Federation (RF). The foundations of these structures should be laid during this Workshop. Further improvement of organizational structures and their functioning will be achieved by lead organizations (LLNL and KRI).

2.1.3 Task L-4

This task suggests the creation of Joint Working Groups (JWGs) for each program category. It seems reasonable to limit not only the membership for each JWG but also the number of JWGs.

2.1.4 Task L-5

The development and maintenance of a website homepage will be performed by LLNL. The participation of KRI will be also necessary. The proposal on the development of a living glossary of specialized English and Russian terminology applicable to nuclear materials safety management was put forward at the Third Workshop on Non-Reactor Nuclear Safety held in Los Alamos in 1995. The specialists of the Lead Department of scientific and technical information of KRI can take part in the compilation of a bilingual English-Russian glossary or of a Russian equivalent of the English version. This staff is experienced in development of such documents under order of the International Atomic Energy Association.

2.1.5 Tasks L-6–L-9

These suggest the comparative assessment of methods for safety analysis in the RF and U.S., including the system of Nuclear Materials Safety Management. These are, certainly, multilevel problems. It is advisable to consider each task as applied to specific enterprise, branch of industry, and on federal level. On top level the system is founded on the development of a legislative basis and on implementation of specific functions by a number of government bodies including, for example, the State Atomic Supervision, Ministry of Health, regional authorities etc. And an important part is played by Minatom. This ministry performs also the direction of branch safety system. Therefore, it seems necessary that a representative of Minatom be the head of corresponding JWG. It is also advisable to charge this JWG with organizing of cooperation on tasks L-7 and L-9. The former task involves the preparation of hysterical review on serious accidents and lessons learned. On this subject there is a great deal of data requiring systematization and analysis. As it was mentioned in the SOW, planning joint technical training (Task L-9) will be based on the analysis performed according to Tasks L-6 and L-8. Thus, the first JWG should organize the cooperation for Tasks L-6, L-7, L-8, and L-9.

2.1.6 Task L-10

This covers a number of joint technical projects, the content and priorities of which may be discussed during this Workshop. Research institutes, designing organizations and industrial plants should take part in corresponding investigations. It is necessary to create a JWG for coordination of works. V.G. Khlopin Radium Institute may be charged with organizing of this group from Russian party.

2.2 FINANCIAL SUPPORT

It is necessary to keep in mind the requirements of financial support. As a minimum version, we can propose the following:

- Permanent working unit in the staff of KRI for the execution of Tasks L-1, L-2, and for participation in Tasks L-4 and L-5;
- JWG for assessment and comparison of safety systems in the USA and RF, responsible of execution of Tasks L-6, L-7, L-8, L-9;
- JWG for Joint RF–U.S. Projects (Task L-10).

It is desirable to get initial proposals on the personnel of groups in the course of this Workshop for subsequent approval by Minatom.

3. Conclusions

This report does not include the review of the university component of the SOW. But it is necessary to say that, unfortunately, the universities of St. Petersburg up to now have not been involved in bilateral cooperation in the field of Nuclear Materials Safety Management. This situation should be corrected, since many universities of St. Petersburg prepare specialists for nuclear science and industry.

The integration of the Lab-to-Lab and university components of the Program seems to be an important problem. There is no need to prove this fact. To carry out such an integration, one representative from the universities may be included in each JWG of Lab-to-Lab cooperation.

In conclusion, it should be noted that the discussed Statement of Work formulates clearly the backgrounds and tasks. This is a good foundation for development of a new phase of cooperation.

References

1. L. Lazarev, F. Witmer, E. Kudryavtsev, D. Carlson, P. Krumpe, M. Moshkov, "Russian-American Cooperation in Radiochemical Safety," *International Topical Meeting on Nuclear and Hazardous Waste Management*, Seattle, Washington, August 1996, v. 2, p. 943.
2. L.N. Lazarev, "Review of current Russian-American joint projects on safe management of nuclear materials," *Nuclear Materials Safety Management*. Kluwer Academic Publishers, 1998, p. 341.
3. M.L. Hyder, V.N. Romanovsky, S.A. Bartenev, L.N. Lazarev, S.A. Strelkov, G.M. Zachinyaev, E.R. Nazin, A.I. Maliych, "Russian Studies of the Safety of Ion Exchange in Nitric Acid," *Tenth Symposium on Separation Science and Technology for Energy Applications*, Gatlinburg, Tennessee, October, 1997.
4. D. Carlson, M. Young, L. Lazarev, B. Petrov, V. Romanovsky, "Overview of Sandia National Laboratories and Khlopin Radium Institute Collaborative Radiological Accident Consequence Analysis Efforts," *Nuclear Materials Safety Management*. Kluwer Academic Publishers, 1998, p. 333.

THE LAB-TO-INSTITUTE COMPONENTS—U.S. VIEW

LESLIE J. JARDINE
Lawrence Livermore National Laboratory
Livermore, California 94551 U.S.A.

1. Introduction: The Nuclear Materials Safety Initiative

The U.S.-Russian Nuclear Materials Safety (NMS) initiative is a proposed bilateral collaborative program to reduce the nuclear dangers associated with the treatment, storage, and disposition of excess highly enriched uranium (HEU) and plutonium from various origins, including dismantled warheads. The proposed program is based on the recognition that an accident in a nuclear facility in either country would have a significant detrimental impact in both countries including introducing major delays to the joint U.S.-Russian efforts for fissile materials disposition. The program would reduce the probability for potential accidents but also contribute to the transparency of U.S. and Russian nuclear facility safety practices and procedures and contribute to the global interests in national security and nuclear materials non-proliferation of both countries.

The proposed Nuclear Materials Safety program defines and incorporates integrated interactions between the U.S. Department of Energy (DOE), the Ministry of the Russian Federation for Atomic Energy (Minatom), associated national laboratories, research institutes, and industrial sites, as well as prominent universities with nuclear programs in both countries. The scope of the program avoids duplication of and truly complements efforts with other ongoing bilateral fissile materials and transparencies activities. The focus is on fissile materials associated with defense nuclear fuel cycle facilities in Russia and the United States that will be required for the initial storage and metal conversion process operations as depicted in the "program boundary" in Figure 1.

Because the initiative complements rather than duplicates existing DOE-Minatom collaborations, the NMS activity provides interactions of experts and access to facilities and activities not within the domain of current joint activities such as MPC&A, International Reactor Safety Program, U.S. HEU purchase agreement activities, and monitored reciprocal inspection activities. Finally, the NMS program will serve to inculcate a safety culture by focusing on and incorporating safety in the academic curriculum of students who will become the future group leaders, division heads, managers, institute directors, and ministers in Minatom and Russian nuclear sites.

The proposed program has two major components: (1) nuclear facility (Lab-to-Lab or site-to-site) technical projects and training initiatives; and (2) academic exchanges, curriculum and degree program development, and joint research to provide an enhanced safety focus. The objective of the NMS program is to improve the operational safety in the

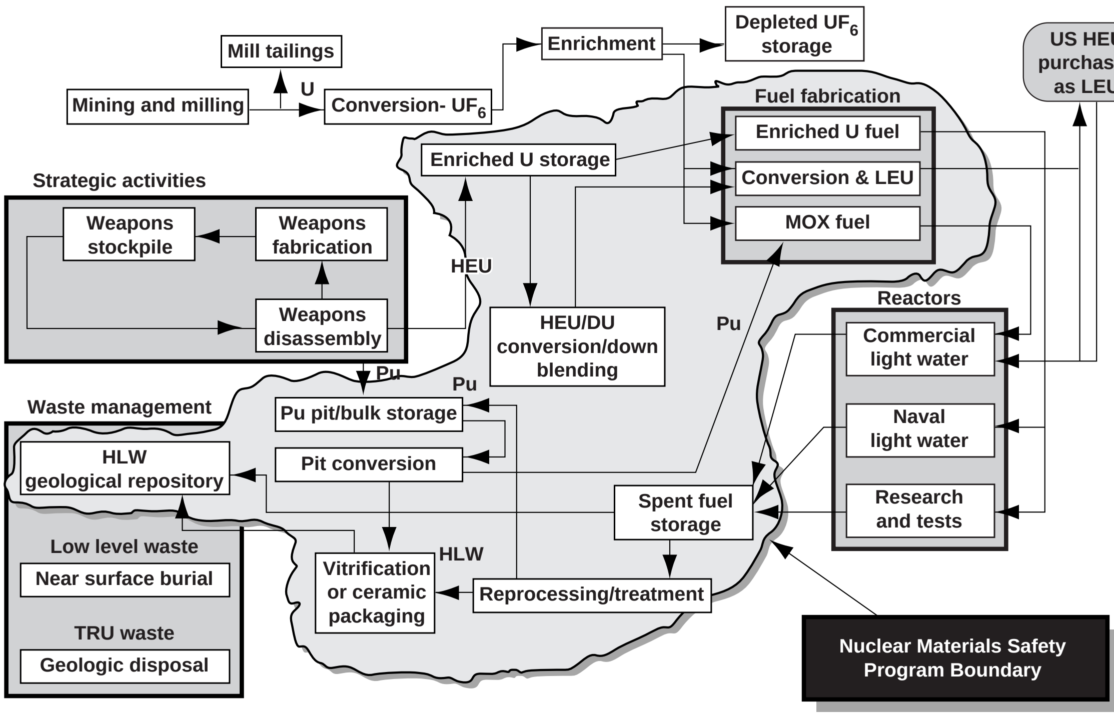


FIGURE 1. Flowsheet shows nuclear materials safety activities of interest to this ARW.

nuclear materials facilities and operations used for the storage and disposition of excess HEU and plutonium (Pu) from dismantled nuclear warheads and thereby enhance public and worker safety.

The Lab-to-Lab (or site-to-site) component involves projects between U.S. DOE national laboratories and Minatom research institutes and industrial sites in Russia. The corresponding proposed lead laboratories on this effort are the Lawrence Livermore National Laboratory (LLNL) and the V.G. Khlopin Radium Institute (KRI) in St. Petersburg, Russia. The Lab-to-Lab program will promote improved safety methodologies and their application to nuclear facilities for excess HEU and plutonium storage and disposition and improve training for facility operators.

Similarly, the academic component involves research and curriculum initiatives between U.S. and Russian universities. The proposed lead organizations for this component are the Amarillo National Resource Center for Plutonium (ANRCP) and the Moscow Engineering Physics Institute (MEPhI). The purpose of the academic component is to encourage the next generations of Minatom engineers, scientists, and managers to focus first on safety. Dr. Lee Peddicord addresses this academic component in detail in “The University-to-University Components” earlier in this volume.

2. Lab-to-Lab Nuclear Materials Program Components

The proposed Lab-to-Lab program component focuses on nuclear materials safety of excess HEU and plutonium storage and disposition. The strategic defense activities required to generate the excess fissile materials and all nuclear reactors used to burn the fissile materials are both excluded from the proposed Program as they are covered to the degree allowed by existing U.S.-Russian Federation agreements and other ongoing programs. This paper addresses the general approach and framework for developing a formal bilateral program. The following paper by Dr. Chin Ma and Dr. Leslie Jardine is based on this general approach; it contains the more specific recommendations needed to implement such a program.

First, joint technical projects in both countries will be identified, and classified into one of several different program categories for the purposes of organizing and managing a program. Candidate projects will be planned with appropriate milestones, performance measures, deliverables, and funding and personnel requirements. Based on pre-established criteria, the proposed projects will be evaluated for support. Those selected shall be funded and performed with the required DOE and Minatom oversight. Selected projects must result in exchanges of information or techniques that improve nuclear materials safety and strengthen the training of operators or result in a better understanding of potential accidents: meeting these objectives can to lead to better accident response planning and mitigating measures. Possible programmatic categories for organizing and managing a program could be:

- Worker safety;
- Public safety;
- Facility operational safety;
- Source terms;
- Consequences;
- Assessments;
- Emergency preparedness.

Joint projects are expected to fall into one of several technical organizing areas. For example, three areas could be defined as:

- Training and procedures (e.g., computer codes, audits, process simulators, operational practices, readiness reviews);
- Assessments (methodologies, safety analysis reports, consequences modeling, initiating events identification);
- Experimental (source terms, radiation monitoring, failure modes).

Technical exchange visits and specialized workshops of experts from Russia and the United States will also continue to be organized and held on a regular basis to promote improved safety within nuclear materials safety facilities and enhanced training of facility operators.

The lead U.S. laboratory, proposed as Lawrence Livermore National Laboratory, will integrate and implement the nuclear materials safety program together with up to six additional DOE laboratories, based on their unique capabilities and expertise in nuclear materials safety. The U.S. lead laboratory, in conjunction with the lead Russian Institute, will establish and maintain a website home page on nuclear materials safety. The V.G. Khlopin Radium Institute is proposed as the lead Institute in Russia, and already has been designated by Minatom with this responsibility. KRI will interface with the Minatom Department of

Safety, Ecology, and Emergency Situations. This Department has the overall management responsibility within the Minatom for nuclear material safety issues in Russia. In addition, the KRI will co-ordinate the institutes and industrial sites and identify for the Program those individuals and organizations in Russia with appropriate expertise. Joint Working Groups (JWG) of U.S. and Russian experts will be established for the programmatic categories in the first year, with the number of categories and members modified to respond to needs and opportunities identified as the Program progresses. The JWGs will develop, prioritize, and recommend projects and individuals for funding to the lead U.S. laboratory. LLNL will work with DOE, Minatom, and KRI to establish funding and obtain authorizations. With DOE and Minatom approvals, as appropriate, the responsible U.S. laboratories will then issue direct contracts to KRI and other institutes and industrial sites in Russia to achieve cost and schedule effectiveness based on the approved JWGs specific statements of work (SOWs). The projects will be managed by appropriate laboratories and institutes conducting and monitoring the funded SOWs.

The first fiscal year after funding of a bilateral program will focus on finding a common joint understanding of the existing nuclear materials safety management systems in Russia and the U.S. and how these might be improved by sharing experiences from each country. An initial program plan will be developed jointly and will be updated annually as needed. Organizational structures, specific JWGs, and operating procedures will be defined and implementation started consistent with approved funding. Funding at KRI will be established to allow the technical activities in Russia to be undertaken. Special attention will also be given to the provision of appropriate electronic mail equipment and personal computers to facilitate ready communication between principals. The two initial specific projects, a comparative assessment of safety management systems and a joint compendium of significant accidents and lessons learned, will be started. Also, the website homepage with supporting glossary will be completed.

The second fiscal year after funding will start nuclear materials safety projects for the initial excess HEU and plutonium storage and disposition operations. This may include storage,. plutonium metal conversion, MOX fabrication, Pu immobilization, and HEU conversion based on specific JWG recommendations and DOE and Minatom authorized SOWs. The third fiscal year after funding will focus on initiating projects beyond the initial excess storage and disposition activities, such as spent fuel storage, geologic repositories, and reprocessing

3. Conclusions

The proposed Nuclear Materials Safety program builds on a foundation of successful interactions on nuclear materials safety topics between DOE and Minatom staff and contractors starting with the April 1993 Tomsk-7 reprocessing plant accident. This ARW in St. Petersburg, in conjunction with the successful March 1997 Amarillo Advanced Research Workshop on Nuclear Materials Safety Management, completes a series of presentations by experts on nuclear materials safety in all of the key functional areas required for the safe and secure storage and disposition of excess HEU and plutonium (Fig. 1). The basis for developing a formal bilateral nuclear materials safety program has been established by these two ARWs. The Nuclear Materials Safety program capitalizes on technical exchanges to improve nuclear facility operational safety and develop mutual understanding and trust in key areas of the nuclear fuel cycle and nuclear materials operations. The proposed program fills a critical need to collaborate with Russia experts on safety and to maintain and enhance the momentum now underway in both the United States and Russia on nuclear materials non-proliferation, weapons dismantlement, and excess fissile materials disposition.

The experts have met, interacted, presented, and discussed their visions and opinions with each other, and are now ready to implement a much needed and commonly desired bilateral program on nuclear materials safety. We, the experts in the U.S. and Russia, await the decisions of those with funding authority to direct us to initiate this program.

POSSIBLE APPROACHES FOR A LAB-TO-LAB PROGRAM PLAN

CHIN W. MA
LESLIE J. JARDINE
Lawrence Livermore National Laboratory
Livermore, California 94551 U.S.A.

1. Introduction

The many presentations and discussions during this NATO Advanced Research Workshop showed strong interest and enthusiastic support for the U.S. and Russian Nuclear Materials Safety program. The goal of this program is to enhance the safety culture and practices of the storage and disposition activities for excess weapons plutonium (Pu) and highly enriched uranium (HEU). It is the consensus of the experts attending this meeting that this program will benefit both the United States and Russia, and also the Western European countries.

It is hopeful that the U.S. Department of Energy (DOE) will provide some funding for this Nuclear Materials Safety program in the year 2000 or even earlier. The program consists of two components, i.e. the laboratory-to-laboratory (Lab-to-Lab) program and the university-to-university program. The Lab-to-Lab program involves projects between the U.S. DOE national laboratories and Minatom research institutes and industrial sites in Russia. This paper provides a preliminary discussion on a possible plan for the Lab-to-Lab program. The discussion covers the program strategy and guidelines, the elements of safety culture and practices, the program organization, and the project approach. We would like to emphasize that this is merely a preliminary outline and we welcome any comments and suggestions.

2. Program Strategy and Guidelines

It is suggested that the following strategy and guidelines be incorporated in developing a program plan:

First, this program should have a focusing scope. The Joint U.S. and Russia safety technical exchanges started in 1993 after the Tomsk accident. This meeting is actually the sixth joint U.S. and Russia workshop. The scope of the previous safety technical exchanges, however, was not always very focused on a long-term objective. Therefore, it was somewhat difficult to find a sponsor for funding and the progress was somewhat slow. This time, the nuclear materials safety program is a new initiative, and it should focus entirely on the excess Pu and HEU storage and disposition activities. Because these activities have a very high visibility, it is expected that developing a program around these activities will be more successful.

Secondly, this program should select and work on mutually beneficial projects. The term “mutually beneficial” should be understood in a broad sense. Since the common goal and interest are to enhance the safety of excess Pu and HEU storage and disposition activities; one should recognize that any joint project that will enhance safety even in one country, either the U.S. or Russia, will actually benefit both countries. This is because an accident in either country will have a very negative impact on both countries and may delay the whole effort in both countries. We strongly believe that the U.S. and Russia will learn from each other through this program.

The third guideline is to develop a tailor-made approach for each country. It is recognized that there are large differences in nuclear practices between the U.S. and Russia. A successful approach should be tailor-made for each country. The U.S. approach should not be applied to Russia without change, and likewise, the Russian approach should not be applied to U.S. without change. There are, however, many common factors in safety culture and practices that will contribute to the success of excess Pu and HEU storage and disposition activities. One should try to identify these common factors and incorporate them in the practices of each country.

In order for this program to have a real impact on the enhancement of safety; it must interface closely with the ongoing excess Pu and HEU storage and disposition activities and must include the Russian industrial sites.

This program should obtain a long-term commitment of key personnel. It is recognized that this is a joint effort of two countries. Therefore, the direct personal relationship is an important factor for the success of this program. We are glad to see that through the safety technical exchanges of the last six years, a set of specific experts on nuclear materials safety from Russia, Europe, and the U.S. have been identified and many of them have already developed good personal relationships.

This program should allocate a significant portion of the budget to Russia. It is expected that more Russian colleagues and fewer U.S. colleagues will participate in this program.

Finally, because this is a new program and the participants do not have much experience, it is suggested that this program be adjusted based on the progress and feedback as appropriate during implementation.

3. Elements of Safety Culture and Practices

The goal of this nuclear materials safety program is to enhance the safety culture and practices of the excess Pu and HEU storage and disposition activities. What do we mean by safety culture and practices? A lengthy discussion on this subject has been given in a report published in 1991 by The International Atomic Energy Agency. This section will provide only a very brief discussion. Basically, the safety culture and practices consist of three elements:

- The first element is safety regulation. It sets forth the safety requirements by the government for the construction and operation of a nuclear material facility. There are many kinds of regulations, such as law, standards, and orders.

- The second element is the safety analysis and technology. It determines the conditions to meet the safety requirements, such as evaluation of source terms and consequences, engineering design to prevent and mitigate an accident.
- The third element is the safety management. It implements and maintains the conditions, such as operator training and job planning.

The safety culture also concerns people's attitudes towards safety. There are three important basic attitudes:

- Clear responsibility and expectations;
- Self assessment;
- Continuous improvement.

These three attitudes mean that organizations and all individuals must fully commit to safety. Safety concerns should be placed before production concerns. People should not just mechanically follow procedures and comply with regulations—that is a passive attitude. Rather, everyone should clearly know their responsibility and the impact of their work on safety, frequently make a self-assessment of their performance, and continuously try to improve their work regarding safety.

All the above factors are closely related; they should all be considered for the enhancement of the safety culture and practices of excess Pu and HEU storage and disposition activities.

4. Program Organization

It is suggested that the program could be organized as follows:

1. First, the U.S. DOE and the Russian Minatom will oversee the program. They have the final authority.
2. Second, the lead laboratories will be the Lawrence Livermore National Laboratory on the U.S. side and the Khlopin Radium Institute on the Russian side. They will lead and coordinate the Lab-to Lab program.
3. Third, other U.S. laboratory and Russian institutes and industrial sites will also participate in the program.
4. Fourth, a minimum number of joint working groups will be formed. Each joint working group will have a U.S. co-chair and a Russian co-chair. The functions of the joint working groups are to:
   —Develop, prioritize, and recommend projects for funding;
   —Review, and if necessary develop, Statements of Work for projects.

4.1 WORKING GROUPS

At this time, it is suggested that five working groups, each with a different scope may be appropriate to handle the program activities. They are:

- Accident Phenomenology and Accident Analysis Group. It will cover topics such as hazards, accident events, source terms, consequences, probability of occurrences, methodology, and computer codes.
- Safety Technology and Experiments Group. It will cover topics such as engineering designs for prevention, mitigation, and detection of an accident,

aerosol monitoring, fire protection, dry cask storage, and source term experiments.

- Operation and System Safety Group. It will cover topics such as criticality control, administrative controls to prevent accidents, radiation protection, normal emission, system failure mode, and human error.
- Safety Management and Regulation Group. It will cover topics such as worker training, job planning, quality assurance, audit, inspection, operation readiness review, and regulations.
- Emergency Planning Group. It will cover topics such as fire department and evacuations.

As the program develops, some working groups may be eliminated or combined, or new working groups may be added.

5. Project Approach

Once the program is developed, it is suggested that detailed works will be performed by various projects. Each project will be defined by a Statement of Work, which includes the work scope, the budget, the schedule, the deliverables, and the personnel.

A project team will be formed for each project. In general, each project will be limited to one year duration and less than $100,000 budget for Russia and an appropriate amount for the United States. In general, we believe that many small projects is a better approach than a few large projects.

Some examples of the project approach are given below.

5.1 EXAMPLES OF PROPOSAL PROCESS

Each participating U.S. national laboratory and Russian institute and industrial site could submit a proposal for a project. The proposal should include a clearly defined Statement of Work. The proposal will be submitted to the appropriate joint working group for review. Since it is expected that there will be many proposals and the resources are finite, the joint working group will prioritize which projects to fund.

5.2 EXAMPLES OF POSSIBLE INITIAL PROJECTS

- Compile, evaluate, and compare the U.S. DOE safety management system and the equivalent Russian Minatom approach to management control, with emphasis on training, job planning, and management responsibility.
- Assess and compare the safety regulations in both countries applicable to the excess Pu and HEU storage and disposition activities, with emphasis on developing the contents and formats of safety analysis reports.
- Assess and compare the methodologies of hazard and accident analysis in both countries applicable to the excess Pu and HEU storage and disposition activities.

- Compile a compendium of significant accidents and lessons learned in the U.S. and Russia relevant to the excess Pu and HEU storage and disposition activities.
- Develop a website homepage for nuclear materials safety at LLNL and KRI.

5.3 EXAMPLES OF POSSIBLE PRODUCTS AND DELIVERABLES

- Organize and conduct workshops and publish proceedings;
- Organize technical exchange visits;
- Conduct training;
- Perform analyses, studies, experiments, and publish topical reports.

6. Conclusion

The disposition of large quantities of excess weapon plutonium and highly enriched uranium remains a crucial and urgent issue for world peace. The experts participating in this Advanced Research Workshop can play a significant role in developing and implementing a nuclear materials safety program to enhance the safety culture and practices of the disposition activities. The program will reduce the probability for potential accidents while contributing to the transparency of U.S. and Russian nuclear facilities safety practices and procedures and contributing to the global interests in national security and nuclear materials non-proliferation of both countries. A strong base of cooperation between U.S., Russian, and European experts on this safety issue now exists due to the dialogue and technical exchanges of the last six years. The timing is right and the challenge is great.

APPROACHES TO PRIORITIZING THE LAB-TO-LAB PROGRAM

MARY YOUNG
DAVE CARLSON
The Sandia National Laboratories, Albuquerque
Albuauerque, New Mexico 87110 U.S.A.

1. Summary

The U.S. and Russian weapons dismantlement process is producing hundreds of tons of excess plutonium (Pu) and highly enriched uranium (HEU) fissile materials. The nuclear operations associated with the final disposition of these materials will be occurring in both countries for decades. A significant accident during these operations could delay the disposition process. Russia-U.S. collaborative efforts to address safety issues associated with disposition processes have been ongoing since 1993. The experience of these collaborative efforts have demonstrated the need for a systematic and formalized approach to identifying and prioritizing collaborative projects. A systematic approach to the successful implementation of a formal program will require the definition of year-by-year program objectives, specific technical program areas, a process for the prioritization and selection of projects, and identification of performance measures to evaluate the success of projects. Specialized working groups established for each technical area are needed to define research priorities, review research proposals, and recommend proposals for funding. A systematic approach to the establishment of a formal U.S.-Russia cooperative program will serve to ensure the safety and continuity of disposition processes and reduce the nuclear proliferation risks presented by this material.

The U.S. and Russian weapons dismantlement process is producing hundreds of tons of excess Pu and HEU fissile materials. The U.S. and Russia are both converting and blending HEU into low enriched uranium (LEU) for use in existing reactors. Russia also plans to fuel reactors with excess Pu. The United States is on a dual-path approach for the disposition of excess Pu: (1) use of Pu in existing reactors and/or (2) immobilization of the Pu in glass or ceramics followed by geologic disposal. The fissile nuclear materials storage, handling, processing, and transportation processes associated with the disposition process will be occurring in both countries for decades. A significant accident at any point in the process could significantly delay the disposition process.

Russia-U.S. collaborative efforts to address safety issues associated with nuclear processes required for the disposition of excess weapons grade nuclear materials were initiated in response to the 1993 Tomsk-7 accident. A joint Russia-U.S. team evaluated the causes of an explosion in a nuclear fuel reprocessing tank at the Tomsk-7 reprocessing facility. The success of the joint Russia-U.S. Tomsk-7 accident assessment team resulted in a continuation of the collaborative efforts. Table 1 summarizes the major events comprising Russia-U.S. joint efforts to address common safety concerns relating to the disposition of excess weapons grade nuclear material.

In addition to the activities listed in Table 1, a number of joint U.S.-Russia pilot projects have been completed. The pilot projects were initiated to evaluate the effectiveness of joint U.S.-Russia research efforts. The technical areas studied in the pilot projects have included investigations into the safety of anion exchange, accident consequence assessment methods, and the application of principles of geological similarity to study alternative methods for the safe disposal of radioactive wastes. The experience obtained from the pilot projects and workshops listed in Table 1 indicates that there would be significant value in establishing a formal U.S.-Russia collaborative program to enhance and ensure the safety of nuclear processes required for the safe disposition of excess weapons grade nuclear material. These initial efforts provide significant groundwork for the establishment of a formal program.

A number of types of nuclear facilities required for the disposition process are not included under existing U.S.-Russia nuclear safety programs and agreements. The types of nuclear facilities and activities not currently covered under existing formal agreements are listed Table 2.

During the Krasnoyarsk-26 and Amarillo workshops, subgroups of Russian and U.S. specialists met to discuss specific technical areas of concern. Table 3 lists the topics discussed in the subgroups. The subgroups identified areas of research that could potentially benefit from a collaborative approach. Potential research projects were identified and prioritized in terms of their perceived need. Russian and U.S. institutes best

TABLE 1. The history of the joint U.S.-Russian effort to promote the safe management of nuclear materials.

Date	**Location**	**Activity**
June 1993	TOMSK-7 Russia	U.S. DOE Technical Team on-site review of TOMSK-7 incident
September 1993	Hanford, Washington U.S.A.	First joint U.S.-Russia meeting on Radiochemical processing safety
November 1994	St. Petersburg and Krasnoyarsk-26, Russia	Second U.S.-Russia Joint Workshop on Radiochemical Operational Safety
August 1995	Los Alamos, New Mexico, U.S.A.	Third U.S.-Russia Workshop on Non-Reactor Nuclear Safety
August 1996	Seattle, Washington U.S.A.	Program Review and Planning Meeting for Future Technical Exchanges
March 1997	Amarillo, Texas, U.S.A.	NATO Advanced Research Workshop: 4th U.S.-Russia Workshop: Nuclear Materials Safety Management Initiative

TABLE 2. Nuclear material operations not currently within the scope of existing U.S.-Russia Safety Programs.

UF_6 production	Depleted UF_6 storage
Enrichment	Enriched uranium storage
Fuel fabrication Enriched U fuel Conversion and LEU MOX fuel	Spent fuel storage
	Pu Pit/bulk storage
	Vitrification or ceramic packaging
	Pit conversion
HEU/DU conversion/down blending	Radiochemical processing/treatment
HLW geological repository	

TABLE 3. Program areas identified and discussed at Krasnoyarsk-26 and Amarillo.

Krasnoyarsk-26 Workshop	**Amarillo Workshop**
Radiochemical modeling and experiments	Nuclear materials (Pu) storage, transportation, and handling
Radiochemical facilities operational safety	MOX production, transportation and handling
Safety and risk assessment	Spent fuel storage, transportation and handling
Radioactive waste management safety	Geologic disposal, wastes, and environmental issues
Nuclear materials storage safety	

suited for each project were identified. The discussions during the subgroup meetings demonstrated the need for a systematic and formalized approach to identifying and prioritizing collaborative projects. A systematic approach to the successful implementation of a formal program will require the definition of year by year program objectives, specific technical areas within the program scope, a process for the prioritization and selection of projects, and identification of performance measures to evaluate the success of projects.

Program areas and annual program objectives could be defined by The U.S. Department of Energy (DOE) and Ministry of the Russian Federation for Atomic Energy (Minatom) program managers with input from specialized working groups. A sample set of possible program areas is listed in Table 4. Specialized working groups established for each program area could define research priorities, review research proposals, and recommend proposals for funding.

TABLE 4. Possible set of program technical areas.

Accident phenomenology and analysis
Safety technology and experiments
Operation and system safety
Safety management and regulation
Emergency planning

The working groups could consist of primarily U.S. and Russian subject matter experts but also include specialists from other countries. The Amarillo workshop included participants from Japan, France, Great Britain, and Germany. The international participation demonstrated the valuable expertise relating to material disposition processes that exists in a number of countries. One Russian and one U.S. chairperson could be appointed to coordinate the efforts of the working group.

The objective for the first year of a formal program could be the development of a common joint understanding of the existing nuclear material safety management systems in the Russian Federation and United States. The next logical step would be to develop an understanding of the safety related vulnerabilities in each system and to prioritize these vulnerabilities.

Collaborative projects must provide benefit to both the Russian and U.S. disposition process. Projects should be selected based on criteria such as project cost, potential reduction of operational risks, and the likelihood of project success.

The synergistic application of U.S. and Russian expertise and resources to maximize the safety of nuclear material processes will serve to ensure the successful disposition of excess weapons grade nuclear material in both countries. Collaborative efforts to date have established productive professional relationships between U.S. and Russian nuclear process safety specialists. Valuable experience has been gained that can be applied in the formulation of a successful program structure. A formal U.S.-Russia cooperative program will serve to ensure the safety and continuity of disposition processes and reduce the nuclear proliferation risks presented by this material.

WORKSHOP SUMMARY AND WRAP-UP: PANEL AND PARTICIPANT DISCUSSIONS

LESLIE J. JARDINE
Lawrence Livermore National Laboratory
Livermore, CA 94551, U.S.A.

L. R. PEDDICORD
Department of Nuclear Engineering
Texas A&M University
College Station, TX 77843-3133, USA

M. MOSHKOV
LEONARD N. LAZAREV
V.G. Khlopin Radium Institute
28, 2nd Murinsky Ave., 194021, St. Petersburg, Russia

1. Summary

The Russian Institute of Public Health (IPH) commented they have a wealth of occupational safety data and accident data that could be made available to a formal nuclear materials safety bilateral program. The IPH is currently working with Ministry of the Russian Federation for Atomic Energy (Minatom) in developing revised regulatory documents and expressed a strong desire to see how the equivalent organizations in the United States work together. The IPH would like to participate in future nuclear materials safety activities.

Several participants stated a need for a special Minatom-U.S. Department of Energy (DOE) umbrella agreement for this bilateral nuclear materials safety initiative. Such an agreement was believed necessary by several Russian participants before groups could be formed and any work started on developing and prioritizing joint nuclear materials safety projects for possible implementation. It was also stated that the Advanced Research Workshop (ARW) attendees do not prepare such an agreement and at best can only facilitate and assist the development of such an agreement, probably by getting higher level DOE and Minatom officials to discuss the need and start the drafting. One Russian suggested a clause for nuclear materials safety interactions that could be incorporated into a revised Material

Protection, Control, and Accountability (MPC&A) umbrella agreement being prepared for renewal. However, this was not viewed as a favorable approach by many attendees.

The use of groups of experts and joint working groups (JWGs) to prioritize acceptable projects, as suggested by Dr. Chin Ma, Lawrence Livermore National Laboratory, was widely endorsed. There was no consensus on the number, size, and scope of the JWGs; some stated that large groups were needed and others suggested only a small group of people with broad experience. The idea of using a senior Joint Steering Committee and lower level JWG groups was suggested by one Russian attendee. The importance of linking and integrating the university activities and the laboratory/institute projects was recognized. It was suggested that the two different types of groups (i.e., lab/institute and academic groups) include representatives on each other's groups. The suggested number of five JWGs seemed to be viewed as reasonable.

The European participants were questioned as to how they could be involved in a bilateral nuclear materials safety program. They had stated the proposed nuclear materials safety interactions would be beneficial to them. The most obvious contributions were that the Europeans could host site visits and tours in their operating plutonium facilities and discuss the safety systems and methods of actual operating plants. They could also support the attendance of their experts at future meetings on nuclear materials safety. It was suggested that the French-German-Russian trilateral mixed oxide fuel (MOX) project could be used as a possible way to initiate and involve the United States and remaining European parties.

There was general agreement with the draft plan and objectives as presented by Dr. Ma for a nuclear material safety bilateral program. These were viewed as reasonable and only minor revisions may be needed. It was stated that the proposed program still needs marketing to gain wider support.

Gosatomnadzor (GAN) raised the issue of Russian Institutes that have plutonium and highly enriched uranium (HEU) within the city limits of both Moscow and St. Petersburg. It was suggested that a joint working group be formed to look into this issue and recommend a long-term approach for proceeding with minimizing the risks to the public in these cities. For example, common criteria could be developed by the JWG and used to assess and track the potential situations in large cities with lots of people. Other European countries could also assist Russia in addressing this problem systematically in the future.

It was agreed that there are many common safety issues that could be shared among the United States, Europe, and Russia. All could learn from each other if a formal program were established and implemented on nuclear materials safety.

One Russian Institute proposed some activities on the nuclear safety of the processes of plutonium-containing materials management. These included:

- Assess issues of nuclear and radiation safety during the immobilization of Pu-containing materials into glass-type forms;
- Evaluate the environmental impacts of MOX fuel fabrication that uses a process of aqueous precipitation and granulation;
- Assess the nuclear, radiation, and environmental safety of pyrochemical processes of MOX fuel fabrication compared to aqueous processes;
- Analyze the nuclear and radiation safety of MOX fuel fabrication for existing light water and fast reactors;
- Determine the technological and environmental safe management of operations during reprocessing of nuclear materials, including the organic radioactive wastes and means of protection of workers and the public;
- Assess and substantiate the technological processes of Pu conversion, MOX fuel fabrication, and immobilization from the viewpoint of nuclear and radiation safety.

INDEX OF AUTHORS AND TITLES

GLOSSARY

AGR	Advanced gas cooled reactor
ALb	admissible level in water, group b population
ANDRA	French National Radioactive Waste Management Agency
ANLW	Argonne National Laboratory West
ANRCP	Amarillo National Resource Center for Plutonium
ARW	Advanced Research Workshop
AVM	Marcoule Vitrification Facility (France)
Balakovo	RF nuclear power plant
Bilibino and Beloyarsk	Russian nuclear power plants
BN-800	reactor type
BNFL	British Nuclear Fuels plc
BWR	water reactor
CAIRS	Computerized Accident Incident Reporting System
CAM	continuous air monitors
CEA	French Atomic Energy Commission
CFCa	Cadarache plant, Belgonucleaire
CIC	Can-in-cannister
CIS	Central or Communist Independent States
Cogema	French Nuclear Power company
CSIS	Independent Scientific Committee
CTR	Cooperative Threat Reduction Program
DOE	Department of Energy
DP/ORBITT	derivative binned information trending tool
DSIN	French nuclear safety authority
DWPF	Defense Waste Processing Facility (US)
EDF	Electricitee d' France
EOIs	Emergency Operating Instructions
ES&H	Environmental Safety and Health (ES&H)
FAS	filter air samplers
FB	fuel bundle
FP	fission products
GAN	Gosatomnadzor
Gosgortekhnadzor	RF State Committee for Oversight of the Safe Conduct of Mining Operations
GSPI	Specialized State Design Institute
GUI	graphical users interface
HAL	Highly Active Liquor
HEPA	high-efficiency particulate air filter
HEU	high(ly) enriched uranium
HEWC	High Enriched Waste Concentrates

HLLW	High-level liquid waste
HLW	high-level waste
HM	heavy metals
HSE	Home Secretary for Energy
IAEA	International Atomic Energy Agency
IBRAE	Electrogorsk Research and Engineering Center for Nuclear Power Plant Safety
ICRP	International Commission on Radiological Protection
INPE	Institute of Nuclear Power Engineering, Obninsk
INRSP	International Nuclear Reactor Safety Program
IPH	Russian Institute of Public Health
ISO	International Standards Organization
KfK	Kernsforschungszentrum, Karlsruhe, Germany
KIAM	Keldysh Institute of Applied Mathematics
KRI	V.G. Khlopin Radium Institute, St. Petersburg
LANL	Las Alamos National Laboratory
LEU	low enriched uranium
LEWC	Low Enriched Waste Concentrates
LL	Lessons Learned
LLNL	Lawrence Livermore National Laboratory
LRW	liquid radioactive waste
LWR	light water reactor
MACCS	Melcor Accident Consequence Code System
Mayak	Russian power plant
MCIW	Mining Chemical Integrated Works, Krasnoyarsk
MEB	Multi-Element Bottle
MEPhI	Moscow Engineering Physics Institute
Minatom	Ministry of the Russian Federation for Atomic Energy
MOX	mixed oxide, i.e., mixed $U0_2/Pu0_2$
MPC&A	Material Protection, Control, and Accounting
NATO	North Atlantic Treaty Organization
NFC	Nuclear Fuel Cycle
NM	nuclear materials
NMS	Nuclear Materials Safety
NMSA	Nuclear Materials Safety Activity
NO_x	nitrogen oxides (i.e., NO or NO_2 or mixtures)
NPP	Nuclear Power Plant
NSD	Nuclear Safety Division
ORNL	Oak Ridge National Laboratory
PAD	Personal Alarmed Dosemeters
PM	Performance Measures
PNNL	Pacific Northwest National Laboratory
POCO	Post-Operational Clean Out
PRA	Probabilistic Risk Assessment
PSA	permissible specific activity
Pu	plutonium
PVA	permissible values of average annual volume activity

PWR	Pressurized water reactor
Pzero (P0)	Belgian industrial fab plant
RF	Russian Federation
RMBK, VVER	basic reactor designs
SCI	counterfeit items/parts
SCIW	Siberian Chemical Integrated Works, Mayak Plant
SCR	spontaneous chain reaction
SEC	State Education Center
SEEID	Safety, Ecology and Extreme Incident Department
SFB	Spent fuel bundle
SICR	seismic impacts of critical intensity
SNF	spent nuclear fuel
SNL	The Sandia National Laboratories
SOW	statements of work
SSC	State Scientific Center, NSD/IPPE
TAMU	Texas A&M University
TBP	tributyl phosphate
TBP/OK	Tributyl Phosphate/Odourless Kerosene
tHM	tons, heavy metal
THORP	Thermal Oxide Reprocessing Plant
TRACTEBEL	Belgium industrial group
U.S.	United States
USD	U.S. dollars
VNIIChT	Institute of Chemical Technology
VNIINM	A.A. Bochvar All-Russian Institute of Inorganic Materials
VNIPIPT	All-Russian Design and Research Institute of Production Engineering of MINATOM of Russia
VPS	vitrified product store
VSM	vertical super melter
VVER-1000	reactor type
UK	United Kingdom
WG-Pu	Weapons-grade plutonium
WHO	World Health Organization
WVP	Windscale Vitrification Plant (also waste vitrification plant or process)
Zaporizhzhya	RF nuclear power plant

Technical Information Department · Lawrence Livermore National Laboratory
University of California · Livermore, California 94551