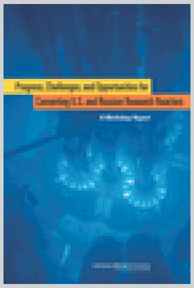

Progress, Challenges, and Opportunities for Converting U.S. and Russian Research Reactors: A Workshop

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Progress, Challenges, and Opportunities for Converting U.S. and Russian Research Reactors

A Workshop Report

U.S. Committee on Progress, Challenges, and Opportunities for
Converting U.S. and Russian Research Reactors from
Highly Enriched to Low Enriched Uranium Fuel

Nuclear and Radiation Studies Board
Division on Earth and Life Studies

NATIONAL RESEARCH COUNCIL
OF THE NATIONAL ACADEMIES

Russian Committee on Progress, Challenges, and Opportunities for
Converting U.S. and Russian Research Reactors from
Highly Enriched to Low Enriched Uranium Fuel

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A number of individuals and organizations contributed to the successful completion of this report. The committee wishes to thank the National Nuclear Security Administration (NNSA) and Dr. Jeffrey Chamberlin, NNSA's liaison to the committee, for supporting this project. Of course, the success of this project is due in large part to the high-quality presentations provided by the symposium speakers, who are listed in Appendix A.

The U.S. committee also wishes to thank the Russian Academy of Sciences (RAS) for hosting the symposium at its facilities in Moscow. Dr. Yuri Shiyan, RAS liaison to the committee, served as the primary link between the U.S. and Russian committees and provided effective and tireless support for both committees. Dr. Nikolay Arkhangelsky (Rosatom), Dr. Yuri Cherepnin (Dollezhal Scientific Research and Design Institute of Energy Technologies [NIKIET]), and Dr. Evegeny Ryazantsev (Kurchatov Institute) provided helpful reviews and fact checking of the committee's final report.

The committee extends special thanks to the staff of the National Research Council for supporting this study. Study director Dr. Sarah Case took the lead for organizing the symposium and was primarily responsible for shaping the committee's final report. Dr. Kevin Crowley, director of the Nuclear and Radiation Studies Board (NRSB), assisted with report preparation and ably handled report review and publication. Ms. Erin Wingo skillfully managed the logistics for the committee's U.S. meetings, the Moscow symposium (in close consultation with Dr. Yuri Shiyan), and report preparation, review, and publication. Dr. Rita Guenther, staff for the Committee on International Security and Arms Control, and Ms. Toni Greenleaf of the NRSB also provided valuable advice on symposium logistics.

This report has been reviewed in draft form by individuals chosen for their diverse perspectives and technical expertise, in accordance with procedures approved by the Report Review Committee of the National Research Council (NRC). The purpose of this independent review is to provide candid and critical comments that will assist the NRC in making its published report as sound as possible and will ensure that this report meets institutional standards for objectivity, evidence, and responsiveness to the study charge. The review comments and draft manuscript remain confidential to protect the integrity of the deliberative process. We thank the following individuals for their participation in the review of this report:

- Pablo Adelfang, International Atomic Energy Agency
- Thomas Newton, Massachusetts Institute of Technology
- Jordi Roglans, Argonne National Laboratory
- Jasmina Vujic, University of California at Berkeley

Although the reviewers listed above provided many constructive comments and suggestions, they were not asked to endorse the contents of this report, nor did they see the final draft of the report before its release. The review of this report was overseen by Rodney C. Ewing, University of Michigan. Appointed by the National Research Council, he was responsible for making certain that an independent examination of this report was carried out in accordance with institutional procedures and that all review comments were considered carefully. Responsibility for the final content of this report rests entirely with the authoring committees and the institution.

Contents

Overview

Highly enriched uranium (HEU) is used for two major civilian purposes: as fuel for research reactors and as targets for medical isotope production. This material can be dangerous in the wrong hands. Stolen or diverted HEU can be used—in conjunction with some knowledge of physics—to build nuclear explosive devices. Thus, the continued civilian use of HEU is of concern particularly because this material may not be uniformly well-protected.

This report focuses on the civilian use of HEU for research reactor fuel. It summarizes the proceedings of a joint symposium organized by the National Research Council of the U.S. National Academies and Russian Academy of Sciences (RAS) to address progress, challenges, and opportunities for converting U.S. and Russian research reactors from HEU to low enriched uranium (LEU) fuel. This symposium—held in Moscow on June 8-10, 2011—was sponsored by the U.S. Department of Energy–National Nuclear Security Administration's (NNSA) Office of Defense Nuclear Nonproliferation.

This report provides a summary of the symposium presentations and discussions; it does not represent a consensus of the symposium participants or the authoring committees.[1] Many important points were made by individual participants during the symposium,[2] particularly regarding possible future actions for reducing or managing the proliferation risks posed by

[1] This report was authored by committees of the National Academy of Sciences (NAS) and RAS. These committees are responsible for the report's quality and accuracy.

[2] No effort was made in this report to attribute statements of fact to individual participants.

HEU-fueled U.S. and Russian research reactors. These points include but are not limited to the following:

- **Many symposium participants from both the United States and Russia emphasized the importance of reducing and, where possible, eliminating the use of HEU in research reactor fuel.** Participants noted that conversion of research reactors to LEU fuel provides for permanent threat reduction and may reduce the requirements for (and potentially the costs of) facility security.
- **Research reactors currently serve important purposes for research and industry, and they will continue to serve important purposes into the future.** Prominent examples include medical isotope production and research associated with the design of next-generation nuclear plants.
- **The United States and other nations have been able to convert research reactors to LEU fuel while maintaining their performance for key missions.** In fact conversions of research reactors in the United States have resulted in improved understanding of their operating characteristics and, in some cases, improved performance. In the United States, all reactors that can be readily converted with existing LEU fuels have been converted. Many symposium participants observed that conversion studies of research reactors in Russia has started but conversion is lagging behind the United States.
- **The economic and performance challenges associated with conversion are likely to be surmountable in many cases, particularly with government assistance and the involvement of reactor operators and customers.** The development of higher-uranium-density LEU fuels could reduce fears of loss of performance by reactor customers.
- **Collaboration between the United States and Russia on conversion of research reactors has been and is likely to continue to be valuable.** Several participants noted that collaborative U.S.-Russian work on fuel development has provided opportunities to advance conversion of both countries' reactors. Additionally, the United States has already confronted regulatory challenges associated with conversion. This experience could be useful to Russia.
- **Some facilities may not be easily convertible to LEU fuel, including fast spectrum reactors, fast critical assemblies, reactors with small core volumes, and reactors with high specific power per unit volume of active core.** The feasibility of conversion depends to some extent on policy choices by host nation governments. Several workshop participants suggested that one way of minimizing the use of HEU for essential or unique missions would be to create major international nuclear centers to house these reactors and to ensure that those facilities have strong security and safeguards protections.

1

Introduction and Background

This report is a summary of a joint symposium held on June 8-10, 2011, by the National Research Council (NRC) of the U.S. National Academies and the Russian Academy of Sciences (RAS) on progress, challenges, and opportunities for converting United States and Russian Federation (R.F.) research reactors[1] from highly enriched uranium (HEU) to low enriched uranium (LEU) fuel.[2,3] This symposium was organized in response to a 2010 request from the U.S. Department of Energy (DOE), National Nuclear Security Administration's (NNSA) Office of Defense Nuclear Nonproliferation.

NNSA requested that a symposium be organized and a subsequent summary document be produced to address:

- Recent progress on conversion of research reactors, with a focus on U.S.- and R.F.-origin[4] reactors;

[1] In this report, the term "research reactors" is defined to include research, test, and training reactors, including critical and subcritical assemblies.

[2] By international agreement, HEU is defined as uranium enriched to a concentration of 20 percent uranium-235 or greater, whereas LEU is defined to be uranium enriched to a concentration of less than 20 percent uranium-235.

[3] This symposium focused on HEU-fueled reactors; however, some research reactors are also fueled with plutonium. The challenges of managing plutonium-fueled reactors—which will need to be accomplished through materials protection, control, and accounting measures—are mentioned in this report but were not the focus of this symposium.

[4] The terms "origin," "supplied," and "designed" are used interchangeably in this report to describe reactors that were developed by the United States and Russia for both domestic and third-country use.

- Lessons learned for overcoming conversion challenges, increasing the effectiveness of research reactor use, and enabling new reactor missions;
- Future research reactor conversion plans, challenges, and opportunities; and
- Actions that could be taken by U.S. and Russian organizations to promote conversion.

The statement of task for the project is included as Appendix C.

The preparation of the symposium agenda and the production of this summary report were carried out by a committee of U.S. experts appointed by the National Academies and a committee of Russian experts appointed by the Russian Academy of Sciences. Biographical sketches of the committee members are provided in Appendix B. These organizing committees met jointly three times over the course of the project: First, in November 2010 to plan the symposium; second, in June 2011 to hold the symposium; and third, in September 2011 to finalize the symposium report. The agenda for the symposium is provided in Appendix A, along with a list of briefings presented at the November 2010 meeting.

NNSA and the NRC agreed that the symposium would not produce consensus findings or conclusions but would instead be used to encourage discussion among U.S. and Russian participants. For this reason, this symposium summary does not contain findings, conclusions, or recommendations and does not represent a consensus of symposium participants.[5] This report represents a summary record of the briefings and discussions that occurred during the symposium. Although the U.S. and Russian organizing committees are responsible for the content of this report, any views contained in the report are not necessarily those of these committees, the National Academies, or the Russian Academy of Sciences.

The remainder of the chapter provides background information on proliferation risks associated with civilian use of HEU; basic operating principles and terminology associated with research reactors; and potential impacts of reducing HEU use in research reactors. Much of the content of this discussion is drawn from symposium briefings (Adelfang, 2011; Arkhangelsky, 2011; D'Agostino, 2011; Dragunov, 2011; Matos, 2011; Roglans, 2011a). Additionally, some basic concepts and definitions were added for the benefit of non-expert readers.

[5] Important statements of opinion are attributed to individual workshop participants where appropriate, but no attempt has been made to attribute statements of fact.

PROLIFERATION AND CIVILIAN TRADE IN HEU

The availability of HEU—particularly in the civilian sector—is a significant proliferation and security concern. In 2001, the U.S. National Research Council stated in its report, *Making the Nation Safer*, that "(t)he primary impediment that prevents countries or technically competent terrorist groups from developing nuclear weapons is the [lack of] availability of special nuclear material (SNM),[6] especially HEU" (NRC, 2001). The availability of HEU in the civilian sector—as opposed to the military sector—is of particular concern, because resources may not be available or used to protect the material adequately during storage or transport.

If HEU is available, either stolen or purchased, it is plausible that a nuclear weapon could be built by either a state or a non-state actor.[7] The technical barriers to constructing such a weapon are not impassably high. As Pablo Adelfang of the International Atomic Energy Agency (IAEA) noted during the symposium (Adelfang, 2011), individuals with a basic knowledge of physics and machining could build a functioning bomb from stolen HEU. This is largely because HEU is only weakly radioactive—making it relatively easy to handle—and because such a device would not require explosive testing to be assured of some yield.

In the civilian sector, HEU is primarily used to fuel research reactors and produce radioisotopes for use in medical procedures. The stockpiles of HEU held for these purposes and others are significant. At the end of 2003, the estimated global stockpile of HEU (both civilian and military) was around 1,900 metric tons. Although the vast majority of this HEU is under military control, about 175 metric tons is civilian HEU (ISIS, 2005). This quantity of HEU is sufficient to fabricate about 3,500 nuclear weapons.[8] The vast majority of this civilian HEU is located in the United States (124 metric tons) and in Russia (15-30 metric tons) (ISIS, 2005).

The potential proliferation risk associated with the use of HEU-fueled research reactors—the focus of the symposium and this summary report—arises from the need to transport and store both unirradiated and irradi-

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[6] "The term 'special nuclear material' means plutonium, uranium enriched in the isotope 233 or in the isotope 235, and any other material that the [Nuclear Regulatory] Commission … determines to be special nuclear material." (42 U.S.C. § 2014)

[7] Although LEU could, in principle, be enriched and converted into HEU for use in building a nuclear weapon, this process would require a significant technical infrastructure, and the mass of LEU required would be very large. The international community could track an effort to enrich LEU more effectively than one involving the theft of HEU.

[8] Assuming 50 kilograms of HEU per explosive device. This may be a conservative assumption. The IAEA defines the *siqnificant quantitity* of HEU to be 25 kilograms. Significant quantity is defined as "the approximate amount of nuclear material for which the possibility of manufacturing a nuclear explosive device cannot be excluded" (IAEA, 2001).

ated[9] HEU fuel. This fuel must be protected at all times and is potentially vulnerable to theft while in transit, including across national borders. Proliferation risk exists even in nuclear weapons states.

It is possible to replace HEU in many civilian applications with LEU, which is considered to have a lower proliferation risk because it is not suitable for use in a nuclear device. Such replacements are possible using current technologies or technologies that are under development. For example, in 2009, the NRC found that the HEU targets used for the large-scale production of the medical isotope molybdenum-99 could be replaced by LEU targets (NRC, 2009). Similarly, many existing research reactors can operate using LEU fuel rather than HEU fuel (see Chapters 2 and 3 of this report). In fact, as discussed elsewhere in this report, many reactors have been successfully converted from HEU to LEU fuel, and many other conversions are under way. The continuation of this trend could significantly reduce the proliferation risk associated with the civilian trade in HEU.

As will be discussed in the next section, 40 percent of the world's operating research reactors are located in the United States and Russia, and nearly all of the world's research reactors are fueled with either U.S.- or R.F.-origin fuel. For these reasons among others, the United States and Russia combined have significant influence on the nature and extent of the worldwide trade in civilian HEU.

RESEARCH REACTORS

Following U.S. President Dwight Eisenhower's 1953 Atoms for Peace speech to the United Nations (Eisenhower, 1953), the U.S. and Russia exported research reactors to about 40 countries. At present, the IAEA lists 254 operational research reactors in 55 countries (Adelfang, 2011; see Figure 1-1). According to the IAEA, 75 civilian research reactors (excluding defense and icebreaker reactors) are currently operating using HEU fuel (see Figure 1-2). Nearly all HEU-fueled research reactors are supplied with HEU of U.S. or Russian origin, with the exception of a very few that are supplied with Chinese-origin HEU. About 25 percent of all research reactors are located in developing countries, including Bangladesh, Algeria, Colombia, Ghana, Jamaica, Libya, Thailand, and Vietnam.[10]

Civilian research reactors are used for a wide variety of missions, for example, to perform research in a broad range of scientific and engineer-

[9] Much research reactor used fuel is not considered to be "self-protecting" (formally defined as producing a dose rate greater than 100 rad per hour at 1 meter in air) because of its low radioactivity. However, irradiated fuel from virtually all of the high-performance reactors mentioned in this report would be considered to be self-protecting, as would irradiated fuel from commercial power reactors, for a period following removal from the reactor.

[10] www-naweb.iaea.org/napc/physics/ACTIVITIES/Research_Reactors_Worldwide.htm.

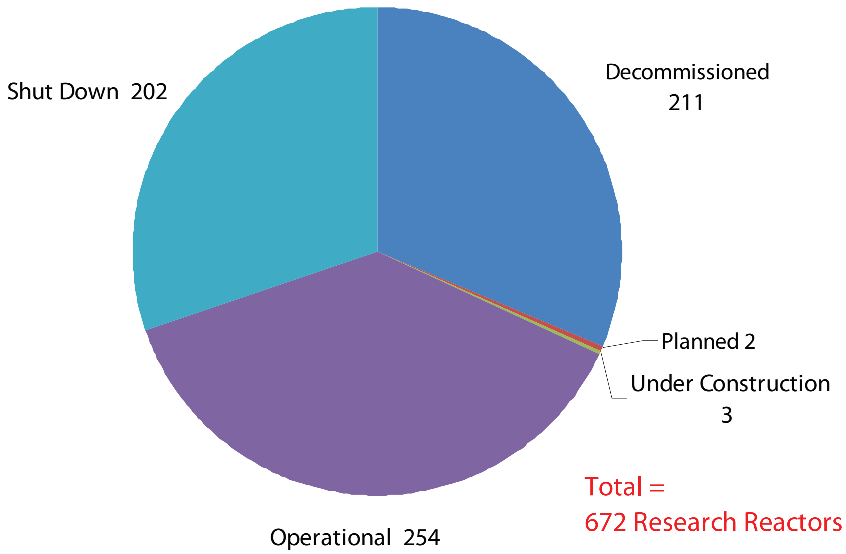

FIGURE 1-1 Research reactors of the world. More than 670 research reactors have been constructed. At present, fewer than half (254 reactors) are operational. SOURCE: Adelfang (2011).

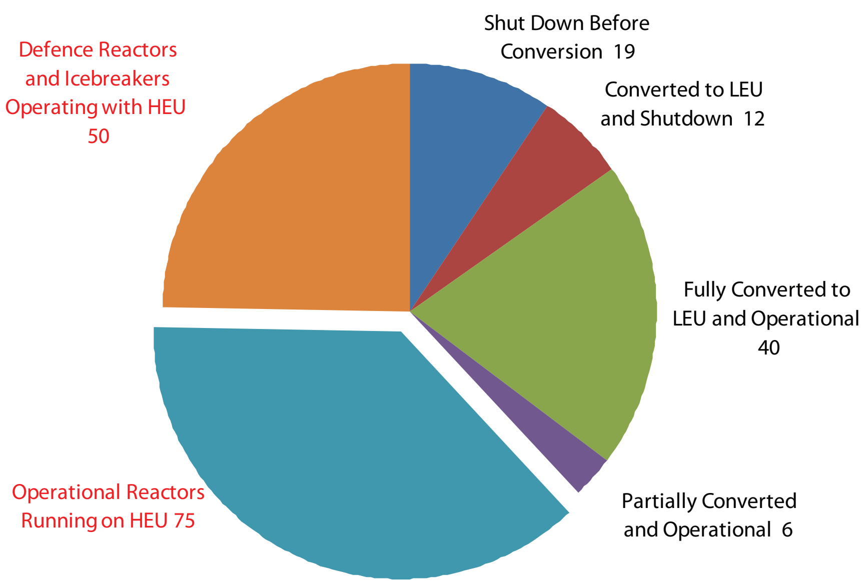

FIGURE 1-2 HEU-fueled research reactors of the world. At present, 75 civilian research reactors are operated using HEU fuel; the remainder have been converted to LEU fuel and/or shut down. SOURCE: Adelfang (2011); data as of 2009.

ing disciplines, including research related to nuclear engineering, nuclear physics and chemistry, materials science, and biology. In addition, research reactors have become indispensible for the production of medical isotopes for diagnostic and therapeutic procedures and are also used for industrial purposes such as silicon doping.

Research reactors' key missions require them to be designed differently from commercial power reactors. Most notably, research reactors are typically designed to produce higher thermal neutron fluxes at much lower thermal outputs than power reactors. Most research reactors are also physically much smaller than power reactors (typically having core volumes of less than a cubic meter versus tens of cubic meters) and require far less fuel (typically a few kilograms versus thousands of kilograms).

Research reactors have a broad range of designs in terms of power levels, moderators,[11] fuel types, and cooling systems, among other design features. In many cases, these reactors are one-of-a-kind or few-of-a-kind, complicating efforts to convert them to LEU fuel. For illustrative purposes, one common broad category of research reactor—the pool- or tank-type water-moderated reactor—is described in the following paragraphs. A broad range of other designs exist, including fast research reactors, which require no moderator and use plutonium as fuel, and "homogeneous reactors," in which the reactor core is a solution of dissolved uranium salts contained in a tank.

Pool-type or tank-type research reactors (see Figure 1-3) comprise a cluster of fuel assemblies and control rods[12] in a pool or tank of water, which serves as both a moderator and a coolant.[13] The core is often surrounded by graphite, beryllium, or heavy water (the "reflector") that is used to slow down (moderate) neutrons and reflect them into the core to maximize the neutron flux. The core and reflector typically contain empty channels for irradiation of targets and test materials, and some reactors are designed with apertures in their pool or tank walls through which neutron beams can be accessed. Figures showing the core configurations for a number of different research reactors can be found in Chapters 2 and 3.

Fuel assemblies (also referred to as "fuel elements") contain the uranium fuel that powers the reactor. A fuel assembly is comprised of individual fuel plates, tubes, or rods, the latter of which is also referred to as

[11] A moderator is a material used to slow down neutrons (i.e., reduce their kinetic energies), which increases the probability of fission when the neutrons are captured by uranium nuclei. Light materials such as water and graphite are commonly used as moderators.

[12] Control rods contain materials (e.g., boron) that absorb neutrons; they are used to control fission rates in the reactor fuel and hence the power levels in the reactor.

[13] Tank-type research reactors are similar to pool-type reactors in overall design, but they typically operate at higher power densities, requiring higher coolant flows and pressures, making it necessary to separate the coolant from the remainder of the pool contents.

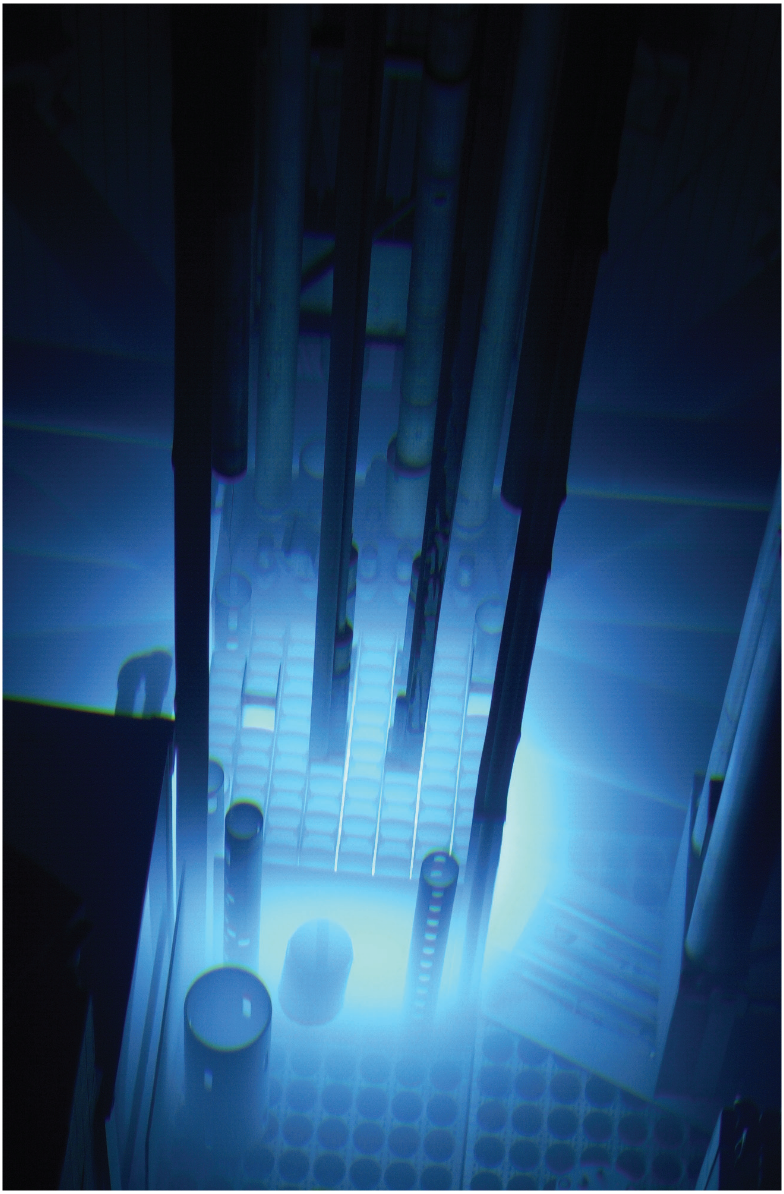

FIGURE 1-3 Pool-type research reactor. This photo shows the core of the Ford Nuclear Reactor at the University of Michigan, the first reactor converted to use LEU fuel under the U.S. Reduced Enrichment for Research and Test Reactors program. The conversion was completed in 1984. The reactor was shut down in 2003 and subsequently decommissioned. SOURCE: Michigan Memorial Phoenix Project.

"pins." Each fuel plate or tube consists of the uranium fuel itself (the "fuel meat") sealed in a "cladding" most typically constructed of aluminum. The number of fuel plates or tubes in an individual fuel assembly can vary widely. For example, a Russian MIR.M1 fuel assembly contains four tubes, whereas the outer fuel assembly of the U.S. High Flux Isotope Reactor contains 369 plates. An illustration of a Russian IRT-4M fuel assembly is shown in Figure 1-4.

Plate-type and TRIGA pin-type fuel is most commonly used in pool- and tank-type research reactors of U.S. origin, whereas tubular or pin-type fuel is used in Russian-origin reactors. Different fuel production methods—rolling in the United States and extrusion in Russia—are used as well.

RESEARCH REACTOR CONVERSION

The United States and the Russian Federation have had active efforts to convert research reactors from HEU fuel to LEU fuel for more than 30 years. The history of these conversion efforts is outlined in the following section, followed by a brief discussion of the current state of research reactor conversion efforts in both countries.

History of Research Reactor Conversion Efforts

The first U.S.- and Soviet-supplied research reactors, which were constructed beginning in the 1950s, were designed to operate on LEU fuel. During the 1960s and 1970s, power upgrades[14] in U.S.-supplied reactors required increased uranium-235 element loadings to reduce fuel consumption and contain fuel fabrication costs. HEU fuel enriched to 93 percent uranium-235 became standard in these reactors. During the same time period, power upgrades in Soviet-supplied research reactors also required increased uranium-235 element loadings; HEU fuel enriched to 80 to 90 percent uranium-235 became standard in these reactors (Arkhangelsky, 2011).

However, in the 1970s, concerns in both the United States and Soviet Union about potential links between the civilian trade in HEU and nuclear proliferation began to increase following a nuclear weapons test in India, unsafeguarded nuclear activities in other countries, and growing terrorist activities around the world. In 1978, the U.S. Department of Energy (DOE) established the Reduced Enrichment for Research and Test Reactors (RERTR) program to develop technologies to minimize and eventually

[14] Power upgrades of U.S.- and Soviet-supplied research reactors were undertaken to increase neutron fluxes in experimental positions.

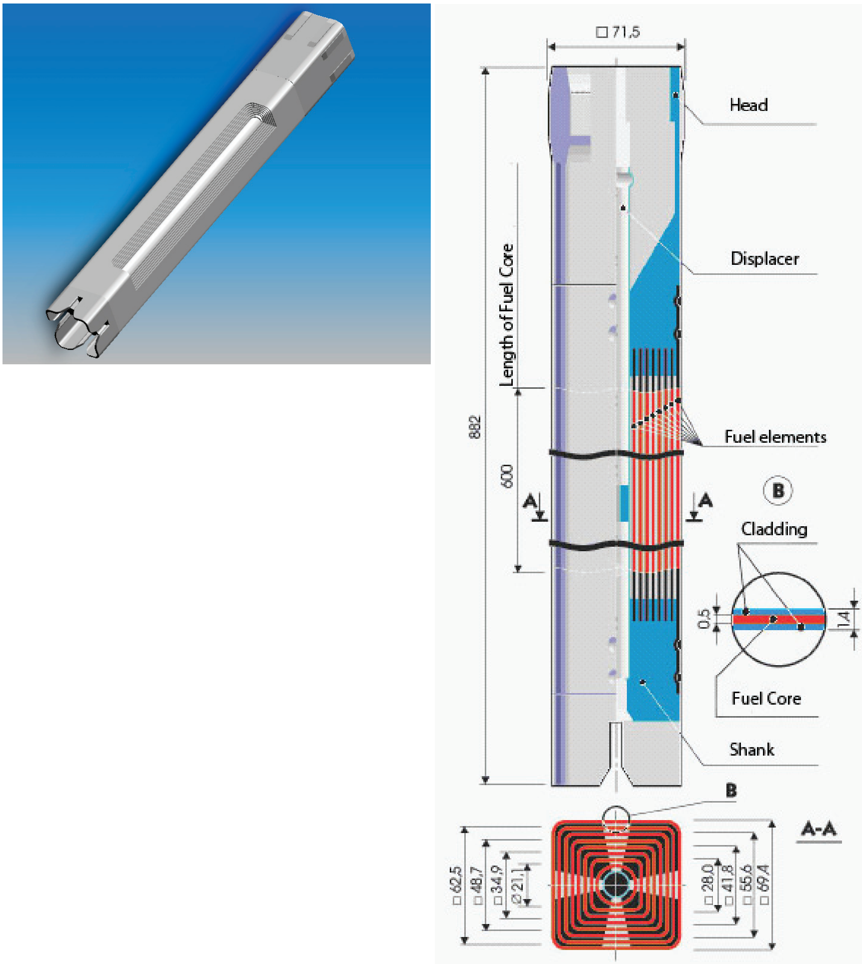

FIGURE 1-4 Illustrations of the Russian IRT-4M fuel assembly. A partial cutaway of a complete fuel assembly is shown on the left. A cutaway view of the fuel assembly (right top) reveals the individual fuel tubes; a cross-section of the fuel assembly (bottom right) shows the nested tubes. SOURCE: Cherepnin (2011).

eliminate the civilian use of highly enriched uranium.[15] At present, all of DOE's HEU elimination efforts for civilian research and test reactors[16] are

[15] More information about this program can be found at www.rertr.anl.gov.
[16] Research, test, and training reactors that have military or national security missions are outside the scope of DOE's conversion program.

currently being carried out under the Global Threat Reduction Initiative (GTRI), into which RERTR was absorbed in May 2004.[17]

Also around 1978, the U.S.S.R. Ministry of Atomic Energy initiated a similar program, the Russian Program of Reducing of Enrichment in Research Reactors (RPRERR), to reduce the enrichment of fuel for research reactors in its client states from 80-90 percent enriched uranium to 36 percent enriched uranium. At this time, the U.S.S.R. began work on high-density LEU research reactor fuels for use in foreign research reactors operating with Soviet fuel (Arkhangelsky, 2011). However, there was no contact or collaboration between these U.S. and Soviet conversion programs until 1993.

The first formal contact to discuss collaboration on research reactor conversions took place in Moscow in March 1993. At that meeting it was decided to initiate a contract between Argonne National Laboratory (ANL) and the Dollezhal Scientific Research and Design Institute of Energy Technologies (NIKIET) on conversion studies and fuel development. Following these interactions, the Russian program began to develop fuel with a less than 20 percent enrichment based on uranium dioxide fuel for the conversion of foreign research reactors.[18]

Significant progress has been made to convert HEU-fueled research and test reactors around the world. As of June 2011, a total of 74 research reactors have been converted from HEU fuel to LEU fuel or shut down since 1978. Of these, 35 have been converted or shut down since 2004, including seven U.S. domestic conversions; 18 foreign conversions; and 10 domestic and foreign shutdowns prior to conversion (Chamberlin, 2010; Roglans, 2011b).

At present, the United States and Russia are cooperating on the conversion of U.S.- and Russian-designed reactors in other countries. The February 2005 Joint Statement by President George W. Bush and President Vladimir V. Putin on nuclear security cooperation affirmed this cooperation:

> The United States and Russia will continue to work jointly to develop low-enriched uranium fuel for use in any U.S.- and Russian-design research reactors in third countries now using high-enriched uranium fuel, and to return fresh and spent high-enriched uranium from U.S.- and Russian-design research reactors in third countries. (Bush-Putin, 2005)

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[17] DOE and GTRI assist reactor operators to perform feasibility studies and safety analyses required for regulatory approval to convert and procure LEU replacement fuels. GTRI also funds work to develop and qualify higher-density uranium-molybdenum (UMo) LEU fuel to convert high-performance research reactors (see Chapter 2).

[18] In 1996 the Bochvar All-Russian Research Institute of Inorganic Materials (VNIINM) became the lead Russian institute under the contract with ANL.

This cooperation was reaffirmed and expanded by U.S. President Barack Obama and Russian President Dmitry Medvedev in a July 2009 joint statement (Obama-Medvedev, 2009). To implement the Obama-Medvedev Joint Statement, Rosatom Director General Sergey Kiriyenko and DOE Deputy Secretary Daniel Poneman signed an agreement during their December 6-7, 2010, meeting to begin studies to determine the technical feasibility and economic impact of converting six HEU-fueled research reactors in Russia (Arkhangelsky, 2011; D'Agostino, 2011).

Current Conversion Status of U.S. and Russian Research Reactors

There were 34 civilian research reactors in operation in the United States in 2011 (Table 1-1)[19] As of June 2011, all but 8 of these reactors had been converted to LEU fuel. Two of these 8 reactors[20] appear to be convertible using current-type LEU fuels. DOE is completing studies to confirm the feasibility of converting these reactors using current-type LEU fuels. Additional research will be required to more fully develop the capability to fabricate these LEU fuels.

However, the following six reactors (including one critical assembly[21]) comprise what DOE refers to as "high-performance" reactors that pose many challenges for conversion, as discussed in more detail in Chapters 2 and 3:

- Advanced Test Reactor (ATR) at the Idaho National Laboratory
- The ATRC critical assembly associated with the ATR
- High-Flux Isotope Reactor (HFIR) at the Oak Ridge National Laboratory in Oak Ridge, Tennessee
- Massachusetts Institute of Technology Reactor (MITR) in Cambridge
- Missouri University Research Reactor (MURR) in Columbia
- National Bureau of Standards Reactor (NBSR) at the National Institute of Standards and Technology in Germantown, Maryland

New high-density LEU fuels are now under development to convert these reactors (Roglans, 2011a). These fuel development efforts are described in Chapter 2.

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[19] These reactors are regulated by the U.S. Nuclear Regulatory Commission or the U.S. Department of Energy.

[20] The NTR General Electric Reactor in California and the Idaho National Laboratory's TREAT reactor (Roglans, 2011b).

[21] A critical assembly contains sufficient fissionable and moderator material to sustain a fission chain reaction at a low (close to zero) level. It is designed so that fissionable and moderator materials can be easily rearranged in various geometries to mock up different reactor designs.

TABLE 1-1 Civilian Research Reactors in Operation in the United States in 2011

Reactor	Institution, Location	Thermal Power (kW)	Peak Steady-State Thermal Flux (n/cm^2-s)	Date of Commission
AFRI TRIGA*	AFRRI,[a] Bethesda, MD	1,000	1.0×10^{13}	1/1/1962
AGN-201*	Idaho State Univ., Pocatello	0.005	2.5×10^{8}	1/1/1967
AGN-201*	Univ. of New Mexico, Albuquerque	0.005	2.5×10^{8}	10/1/1966
AGN-201*	Texas A&M Univ., College Station	0.005	2.0×10^{8}	1/1/1957
ARRR*	Aerotest, San Ramon, CA	250	3.0×10^{13}	7/9/1964
ATR	Idaho National Laboratory, Idaho Falls	250,000	8.5×10^{14}	7/2/1967
DOW TRIGA*	Dow Chemical, Midland, MI	300	5.0×10^{12}	7/6/1967
GSTR*	USGS,[b] Denver, CO	1,000	3.0×10^{13}	2/26/1969
HFIR	ORNL,[c] Oak Ridge, TN	85,000	2.5×10^{15}	8/1/1965
KSU TRIGA MK II*	Kansas State Univ., Manhattan	250	1.0×10^{13}	10/16/1962
MITR-II	Mass. Inst. of Technology, Cambridge, MA	6,000	7.0×10^{13}	7/21/1958
MURR	Univ. of Missouri, Columbia	10,000	6.0×10^{14}	10/13/1966
MUTR*	Univ. of Maryland, College Park	250	3.0×10^{12}	12/1/1960

TABLE 1-1 Continued

Reactor	Institution, Location	Thermal Power (kW)	Peak Steady-State Thermal Flux (n/cm^2-s)	Date of Commission
NBSR	NIST,[d] Gaithersburg, MD	20,000	4.0×10^{14}	12/7/1967
NSCR*	Texas A&M Univ., College Station	1,000	2.0×10^{13}	1/1/1962
NTR General Electric	GE, Sunol, CA	100	2.5×10^{12}	11/15/1957
OSURR*	Ohio State Univ., Columbus	500	1.5×10^{13}	3/16/1961
OSTR*	Oregon State Univ., Covallis	1,100	1.0×10^{13}	3/8/1967
PSBR*	Penn State, University Park	1,000	3.3×10^{13}	8/15/1955
PULSTAR*	North Carolina State Univ., Raleigh	1,000	1.1×10^{13}	1/1/1972
PUR-1*	Purdue Univ., West Lafayette, IN	1	2.1×10^{10}	1/1/1962
RINSC*	Rhode Island Atomic Energy Commission, Narrangansett	2,000	2.0×10^{13}	7/28/1964
RRF*	Reed College, Portland, OR	250	1.0×10^{13}	7/2/1968
TREAT	Idaho National Laboratory, Idaho Falls	250	5.0×10^{12}	10/12/1977
TRIGA Univ. of AZ*	Univ. of Arizona, Tucson	100	2.0×10^{12}	12/6/1958
TRIGA Univ. UT*	University of Utah, Salt Lake City	100	4.5×10^{12}	10/25/1975

continued

TABLE 1-1 Continued

Reactor	Institution, Location	Thermal Power (kW)	Peak Steady-State Thermal Flux (n/cm^2-s)	Date of Commission
TRIGA II*	Univ. of Texas, Austin	1,100	2.7×10^{13}	3/12/1992
UC Davis*	Univ. of California, Davis	2,000	3.0×10^{13}	1/20/1990
UCI*	Univ. of California, Irvine	250	5.0×10^{12}	11/25/1969
UFTR*	Univ. of Florida, Gainesville	100	2.0×10^{12}	5/28/1959
UMLR*	Univ. of Mass., Lowell	1,000	1.4×10^{13}	1/2/1975
UMRR*	Univ. of Missouri, Rolla	200	2.0×10^{12}	12/11/1961
UWNR*	Univ. of Wisconsin, Madison	1,000	3.2×10^{13}	3/26/1961
WSUR*	Washington State Univ., Pullman	1,000	7.0×10^{12}	3/13/1961

NOTES:
*Currently operating with LEU fuel.
[a] Armed Forces Radiobiology Research Institute.
[b] U.S. Geological Survey.
[c] Oak Ridge National Laboratory.
[d] National Institute of Standards and Technology.

There were 24 operating research reactors, 30 critical assemblies, and 12 subcritical assemblies in the Russian Federation in 2011 (Bezzubtsev, 2011; see Figure 2-10 in Chapter 2).[22] Basic information on currently operating Russian research reactors is given in Table 1-2. Several civilian reactors pose substantive technical challenges to conversion, such as reactors using fuel pins consisting of UO_2 dispersed in a copper-beryllium matrix with stainless steel cladding designed to operate at significantly higher fuel temperatures than most research reactors.

[22] Not including naval or other defense-related reactors.

TABLE 1-2 Russian Research Reactors in Operation in 2011

Reactor	Institution, Location	Thermal Power (kW)	Peak Steady-State Thermal Flux (n/cm^2-s)	Date of Commission
Argus	Kurchatov, Moscow	20	4.0×10^{11}	12/1/1981
BOR-60	RIAR,[a] Dmitrovgrad	60,000	2.0×10^{14}	12/1/1969
F-1	Kurchatov, Moscow	24	6.0×10^{9}	12/25/1946
Gamma	Kurchatov, Moscow	125	9.0×10^{11}	1/4/1982
Hydra	Kurchatov, Moscow	10	2.2×10^{10}	1/1/1972
IBR-2M Pulsed R	JINR,[b] Dunba	20,000	1.0×10^{13}	11/30/1977
IGRIK	Minatom, Chelyabinsk	30	2.5×10^{10}	12/15/1975
IR-8	Kurchatov, Moscow	8,000	1.5×10^{14}	8/12/1981
IR-50	NIKIET,[c] Moscow	50	1.7×10^{12}	2/20/1961
IRT	MEPhI,[d] Moscow	2,500	4.8×10^{13}	5/26/1967
IRT-T	Tomsk Polytechnic Institute	6,000	1.1×10^{14}	7/22/1967
IRV-2M	Res. Inst. of Scientific Instruments, Lytkarino	4,000	8.0×10^{13}	1/1/1974
IVV-2M	Inst. of Nuclear Mat., Zarechny	15,000	5.0×10^{14}	4/22/1966
MIR.M1	RIAR, Dmitrovgrad	100,000	5.0×10^{14}	12/26/1966
OP-M	Kurchatov, Moscow	300	8.4×10^{12}	12/1/1989
PIK	Petersburg Nuclear Physics Institute	100,000	4.0×10^{15}	Under construction

continued

TABLE 1-2 Continued

Reactor	Institution, Location	Thermal Power (kW)	Peak Steady-State Thermal Flux (n/cm^2-s)	Date of Commission
RBT-6	RIAR, Dmitrovgrad	6,000	2.2×10^{14}	1/10/1975
RBT-10/2	RIAR, Dmitrovgrad	7,000	1.6×10^{13}	11/24/1983
SM-3	RIAR, Dmitrovgrad	100,000	5.0×10^{15}	1/10/1961
U-3	Krylov Shipbuilding Research Institute, St. Petersburg	50		12/13/1964
YAGUAR	Minatom, Chelyabinsk	10		6/29/1990
WWR-M	Petersburg Nuclear Physics Institute	18,000	1.5×10^{14}	12/29/1959
WWR-TS	Karpov, Obninsk	15,000	1.0×10^{14}	11/4/1964

Note: This table does not include critical assemblies.
[a] Research Institute for Atomic Reactors.
[b] Joint Institute for Nuclear Research.
[c] Dollezhal Scientific Research and Design Institute of Energy Technologies.
[d] Moscow Engineering Physics Institute.

SOURCE: IAEA (2011).

REPORT ROADMAP

The symposium featured a range of briefings from R.F., U.S., and international experts on policy, science, and engineering issues relevant to the conversion of research reactors from HEU fuel to LEU fuel. These briefings were organized into several sessions, reflected in the four chapters of this report:

- Chapter 1 (this chapter) provides the context for this study and introductory material from the symposium;
- Chapter 2 addresses challenges associated with conversion as well as potential solutions;

- Chapter 3 addresses the challenges and successes associated with converting eight specific U.S. and Russian reactors; and
- Chapter 4 addresses future research directions and opportunities, including opportunities for further interaction between the U.S. and Russia on research reactor conversion.

REFERENCES

Adelfang, P. 2011. Reduction of Commercial Traffic in HEU. Presentation to the Research Reactor Conversion Symposium. June 8.

Arkhangelsky, N.V. 2011. Problems of the Research Reactors Conversion from HEU to LEU: History and Perspectives. Presentation to the Research Reactor Conversion Symposium. June 8.

Bezzubtsev, V. 2011. Regulating Safe Operation of Russian Research Reactors. Presentation to the Research Reactor Conversion Symposium. June 9.

Bush-Putin. 2005. Joint Statement on Nuclear Security. October. Available at moscow.usembassy.gov/transcripts_photo_bio/joint-statement-by-president-george-w.-bush-and-president-vladimir-v.-putin-on-nuclear-security-cooperation-bratislava-february-24-2005.

Chamberlin, J. 2010. Global Threat Reduction Initiative: Reactor Conversion Program. Presentation to the Research Reactor Conversion Committee. November 29.

Cherepnin, Yu. 2011. Experience of Resolving the Problems Rising in Conversion of Research Reactors. Presentation to the Research Reactor Conversion Symposium. June 9.

D'Agostino, T. 2011. Remarks on HEU Minimization. Presentation to the Research Reactor Conversion Symposium. June 8.

Dragunov, Y.G. 2011. Types, Problems and Conversion Potential of Reactors Produced in Russia. Presentation to the Research Reactor Conversion Symposium. June 8.

Eisenhower, D. 1953. Atoms for Peace Speech to United Nations. Available at www.iaea.org/About/history_speech.html

IAEA (International Atomic Energy Agency). 2001. IAEA Safeguards Glossary, 2001 Edition. International Nuclear Verification Series No. 3, Vienna, Austria, International Atomic Energy Agency. Available at www-pub.iaea.org/MTCD/publications/PDF/nvs-3-cd/PDF/NVS3_prn.pdf.

IAEA. 2011. Database of World Research Reactors. nucleus.iaea.org/RRDB.

ISIS (Institute for Science and International Security). 2005. Global Stocks of Nuclear Explosive Material. Available at isis-online.org/isis-reports/detail/global-stocks-of-nuclear-explosive-materials/17.

Matos, J. 2011. Experience with Solutions to Conversion Challenges for U.S.-Supplied Research Reactors. Presentation to the Research Reactor Conversion Symposium. June 8.

NRC (National Research Council). 2001. Making the Nation Safer. Washington, DC: National Academy Press.

NRC. 2009. Medical Isotope Production without Highly Enriched Uranium. Washington, DC: The National Academies Press.

Obama-Medvedev. 2009. Joint Statement on Nuclear Cooperation. July 6. Available at www.whitehouse.gov/the_press_office/Joint-Statement-by-President-Barack-Obama-of-the-United-States-of-America-and-President-Dmitry-Medvedev-of-the-Russian-Federation-on-Nuclear-Cooperation/.

Roglans, J. 2011a. Maintaining Performance and Missions. Presentation to the Research Reactor Conversion Symposium. June 9.

Roglans, J. 2011b. Private communication with committee members on research reactor inventories, September 2.

2

Challenges and Opportunities Associated with Conversion

Session 2 of the symposium (see Appendix A) focused on technical challenges associated with conversion and potential solutions for overcoming those challenges. Three panels of Russian Federation (R.F.) and U.S. speakers were organized to address these topics:

- Panel 2.1: *Technical challenges associated with conversion and potential solutions* featured Russian and U.S. presentations on low enriched uranium (LEU) fuel design, core modifications, and approaches for maintaining reactor performance and missions after conversion.
- Panel 2.2: *Other technical challenges associated with conversion* featured presentations on ageing and obsolescence, regulatory challenges, and challenges posed by research reactors that cannot be converted.
- Panel 2.3: *How challenges associated with previously converted reactors were overcome* featured presentations on approaches for overcoming the conversion challenges identified by the other panels in this session.

These panel presentations are summarized in this chapter along with key thoughts from the participant discussions.

FUEL DESIGN FOR CONVERSION

Two presentations on fuel design for conversion were given by Panel 2.1 speakers: Daniel Wachs (Idaho National Laboratory) reported on efforts to develop LEU fuels for converting U.S.-origin reactors (Wachs, 2011), and Yu.S. Cherepnin (Dollezhal Scientific Research and Design Institute of Energy Technologies [NIKIET]) described progress and prospects for reduction of fuel enrichment in Russian-origin reactors (Cherepnin, 2011).

Fuel Design for U.S.-Origin Reactors

Daniel Wachs

Highly enriched uranium (HEU) fuel elements in U.S.-origin research and test reactors consist of aluminum-clad plates (see Chapter 1) that contain a UAl_x or U_3O_8-aluminum dispersion fuel meat clad in aluminum or a uranium-zirconium hydride ($UZrH_x$) fuel meat clad in stainless steel (TRIGA fuel). Work carried out by Argonne National Laboratory and the Idaho National Laboratory, in cooperation with other American, European, and Korean organizations, has resulted in the development of three LEU dispersion fuel systems[1] for conversion of plate-type reactors:

- UAl_x (density = 2.3 grams of uranium per cubic centimeter [gU/cm^3])
- U_3O_8 (3.2 gU/cm^3)
- U_3Si_2 (4.8 gU/cm^3)

These fuel systems are adequate for converting all but "high performance" research and test reactors.[2] There are six HEU-fueled high-performance research reactors in the United States[3] as well as four HEU-fueled high-performance research reactors in Europe that cannot be

[1] The Reduced Enrichment for Research and Test Reactors (RERTR) program (see Chapter 1) also participated in the qualification of a fourth LEU fuel system: a uranium-zirconium hydride with an erbium burnable poison ($UZrH_x$-Er) fuel system that is used for the conversion of TRIGA (Test, Research, Isotope production—General Atomics) reactors. General Atomics began developing a higher-density fuel (up to 3.7 gU/cm^3) before the RERTR program was started in 1978. The RERTR program performed irradiation tests on 20/20 (i.e., 20 weight percent uranium, 20 percent enriched), 30/20, and 45/20 fuels. The 30/20 fuel was used to convert the Oregon State TRIGA Mark II reactor, discussed later in this chapter, and the University of Wisconsin Nuclear Reactor, discussed in Chapter 3, as well as a number of other TRIGA reactors in the United States and abroad.

[2] These high-performance reactors have high-power-density (i.e., high-flux-density) cores. Fuels having higher uranium densities than are available with existing LEU fuels are required to convert these reactors.

[3] As noted in Chapter 1, there are two additional HEU-fueled research reactors in the United States (NTR General Electric and TREAT; see Footnote 20 in Chapter 1) that appear to be convertible using current-type LEU fuels. The Department of Energy (DOE) is completing studies to confirm the feasibility of converting these reactors using current-type LEU fuels. Additional research will be required to more fully develop the capability to fabricate these LEU fuels.

converted with these existing LEU fuel systems. The U.S. reactors are shown in Table 1-1 in Chapter 1; the European reactors are the following:

- Belgian Reactor 2 (BR2) at the Belgian Nuclear Research Centre in Mol, Belgium
- Forschungsreaktor München II (FRM-II) at the Technical University of Munich, Germany
- Jules Horowitz Reactor (JHR), under construction at the CEA Cadarache Research Centre in Cadarache, France (discussed in Chapter 4)
- Réacteur à Haut Flux (RHF) at the Institut Max von Laue-Paul Langevin (ILL) in Grenoble, France

Higher-density LEU fuel systems based on uranium-molybdenum (UMo) alloys are now under development for use in converting these U.S. and European reactors. Test irradiations have been carried out on several UMo alloys to assess their suitability for use as fuel for these reactors. Testing revealed that alloy phases with high U/Mo ratios (e.g., U-10Mo[4]) were most stable under irradiation because they suppressed the formation of fission gas bubbles.[5]

Two LEU fuel systems based on this alloy are now under development by Idaho National Laboratory and partners:

- UMo dispersion fuel: A UMo alloy dispersed in an aluminum matrix with uranium densities up to 8.5 gU/cm^3. An LEU fuel system based on this material is being developed for conversion of BR2, RHF, and JHR.[6]
- Monolithic UMo fuel: Metallic UMo foils with a uranium density of 15.5 gU/cm^3. An LEU fuel system based on this material is being developed for conversion of ATR, HFIR, NBSR, MITR, and MURR (Figure 2-1).

Test irradiations of fuel elements containing both of these materials are now being carried out to develop and qualify these fuel systems.

UMo Dispersion LEU Fuel

Initial irradiations of fuel elements containing UMo dispersions resulted in the formation of interaction layers between the UMo and Al particles and the development of porosity and distortion (pillowing). The addition of small amounts (~2 percent) silicon to the aluminum phase was

[4] That is, alloys consisting of 9 parts uranium to 1 part molybdenum by weight.

[5] Fission gas bubbles are formed in the fuel phase as a result of the production of gaseous fission products.

[6] At present, no LEU replacement fuel has been identified for the FRM II reactor.

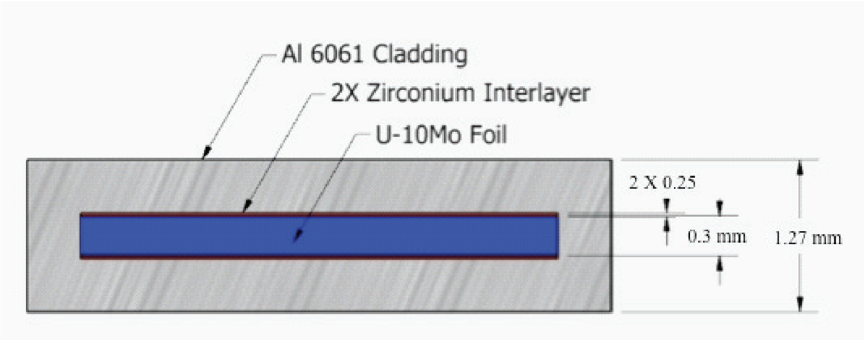

FIGURE 2-1 Schematic cross-section of a research reactor fuel element containing monolithic UMo. SOURCE: Wachs (2011).

found to suppress the development of this interaction layer at burnups of up to 70 percent. However, test irradiations of this fuel material at high power (~ 500 watts per square centimeter [W/cm^2]), high uranium loadings (> 8 gU/cm^3), and high burnup (> 70 percent) resulted in the formation of small blisters on the fuel plates. Follow-up experiments are planned for the fall of 2011 to determine why such blistering occurs and how the fuel element can be modified to eliminate it. A bounding-case irradiation of this fuel material in BR2 is planned for 2011-2012.

UMo Monolithic LEU Fuel

Fuel plates under development for high-performance U.S. reactors consist of a UMo alloy foil ("U-10Mo Foil" in Figure 2-1) surrounded by a zirconium fission recoil barrier ("2X Zirconium Interlayer" in Figure 2-1) in an aluminum cladding ("Al 6061 Cladding" in Figure 2-1). The barrier is intended to prevent interactions at the interface of the fuel meat and cladding. A key issue for this fuel is the stability of this interface. Although the interface is mechanically stable, swelling of the fuel meat during irradiation could lead to the development of porosity at the interface and eventual delamination of the foil from the cladding. Such swelling and delamination could prove to be a life-limiting factor for this fuel system.

Qualification testing of this fuel for three high-performance research reactors (MITR, MURR, and NBSR) is currently under way. A partial fuel assembly[7] is currently being irradiated in ATR at the Idaho National Laboratory (Figure 2-2), and irradiation of ATR fuel elements is planned

[7] As the name suggests, a partial fuel assembly contains only portions of a full fuel assembly. For example, a partial assembly might contain fewer fuel plates than a full assembly.

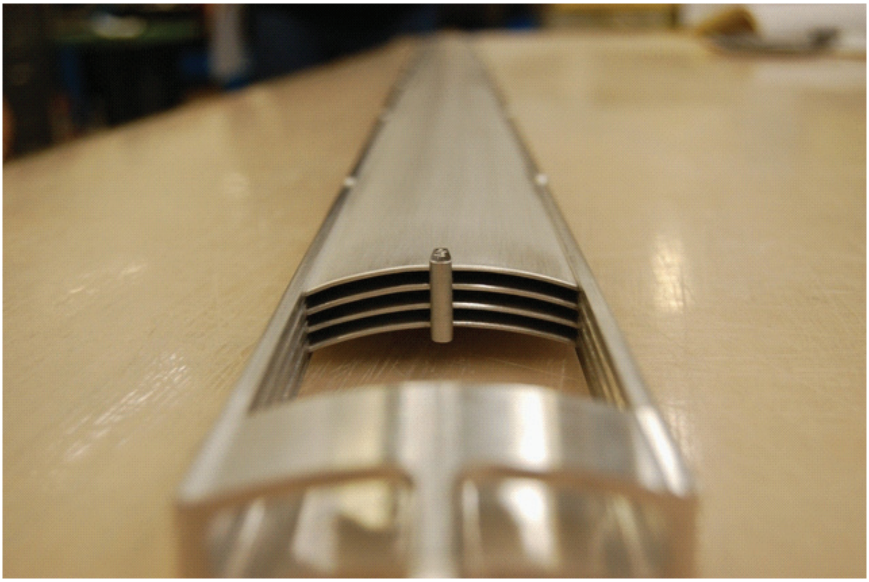

FIGURE 2.2 End view of a partial fuel assembly (AIFP-7) containing monolithic UMo fuel that is currently undergoing test irradiations in the ATR. SOURCE: Wachs (2011).

to begin in 2012. Lead test assembly irradiations are planned once these irradiations are completed.

Testing of this fuel system for use in the highest-performance U.S. reactors (i.e., ATR, HFIR) is planned to begin in late 2011. Bounding-condition irradiation tests (greater than 500 W/cm^2 and greater than 60 percent burnup) on a full-size fuel plate will be carried out at the ATR in late 2011. Fuel qualification testing will be initiated after these irradiation tests are completed.

Fuel Design for Russian-Origin Reactors

Yu.S. Cherepnin

Most Russian research and test reactors use HEU fuels consisting of UO_2-aluminum dispersions fabricated as thin-walled tubular elements of various enrichments and configurations. A Russian program was started in the 1990s to further reduce the enrichment of fuel used in Russian-origin research reactors that are located outside of the Russian Federation. This work has been led by three Russian organizations (NIKIET, Bochvar All-

Russian Scientific Research Institute for Inorganic Materials [VNIINM], and Novosibirsk Chemical Concentrates Plant [NZKhK]) with the collaboration of several other organizations and customers (i.e., research reactor operators) and has resulted in the development of LEU fuels.

The initial phase of this program created UO_2-Al LEU fuel assemblies for conversion of all existing Russian-origin research reactors that are located outside of the Russian Federation. The aim was to reduce the enrichment of uranium in the fuel elements without changing fuel element geometry. LEU fuel assemblies of several designs have been developed (Figure 2-3):

- VVR-M2 fuel assembly. This assembly has a tubular geometry and contains a UO_2-aluminum dispersion fuel meat with a density of 2.5 gU/cm^3. These fuel assemblies have undergone a full cycle of design, testing, and licensing and are currently being manufactured at the fuel production facility at NZKhK in Novosibirsk. This fuel is being supplied to Russian-origin research reactors in Hungary, Vietnam, and Romania.
- IRT-4M fuel assembly. This assembly has a square geometry and contains a UO_2-aluminum dispersion fuel meat with a density of 3.0 gU/cm^3. This fuel, which is fully licensed, is the highest-demand fuel for Russian-origin research reactors located outside of the Russian Federation. This fuel is being supplied to Russian-origin research reactors in the Czech Republic, Uzbekistan, and Libya.
- VVR-KN fuel assembly. This assembly has a hexagonal geometry and is being developed for use in a Russian-origin research reactor in Kazakhstan. It will replace a 36 percent enriched assembly that is now in use. Three assemblies have been manufactured and are now being irradiated in the reactor. Conversion studies and fuel qualification activities for this reactor are proceeding in close cooperation with the reactor operator, producing good results.
- MR fuel assembly. Design work is about to begin to develop a UO_2-aluminum dispersion fuel for this tubular fuel assembly. The fuel meat (which currently has an enrichment of 36 percent) is expected to have an enrichment of 19.5 percent with a density no less than 3.5 $g\ U/cm^3$. It is expected to take about a year to complete this design work and manufacture fuel assemblies for testing. The 19.5 percent enriched fuel will be used in the Russian-origin MARIA research reactor in Poland.

The transition to these LEU fuel assemblies has proceeded using the same fabrication technologies and equipment for producing HEU fuel. However, the use of LEU fuels can reduce reactor "performance" (i.e., reduce neutron flux densities in the core and reflector regions) by up to about 15 percent and shorten fuel replacement cycles. Consequently, the develop-

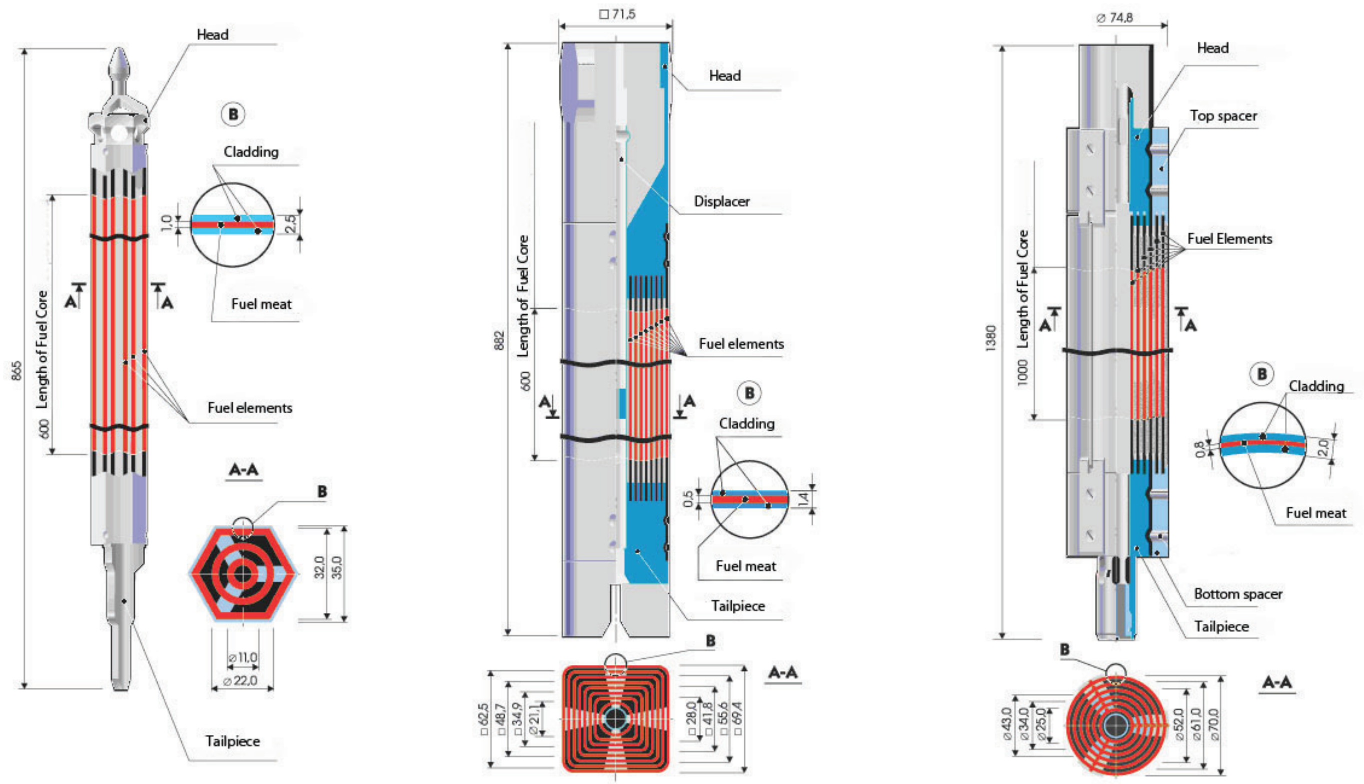

FIGURE 2-3 Schematic illustrations of (left) VVR-M2 tubular fuel assembly, (middle) IRT-4M square fuel assembly, and (right) MR tubular fuel assembly. The VVR-KN hexagonal fuel assembly is not shown. SOURCE: Cherepnin (2011).

ment of higher-density LEU fuels is needed to maintain reactor performance and fuel cycle length and also to increase fuel robustness by allowing an increase in cladding thickness.

The development of higher-density fuels is being carried out in a second phase of the Russian program to reduce fuel enrichments. Work is proceeding on a UMo dispersion LEU fuel with a density of about 5 gU/cm^3.[8] Test irradiations of this material have been carried out to burnups of 40-60 percent. Design efforts are under way for two fuel assembly types: IRT-3M (which has a tubular geometry) and IRT-U (which has a pin geometry).

The third phase of the reduced enrichment program is envisaged to involve the development of completely new fuel designs for future reactors. These new designs should be safe, reliable, easy to fabricate, and economically efficient compared to current designs. UMo monolithic LEU fuels manufactured in the form of pins appear to be a promising future design concept. These could be arranged in geometries to mimic the tubular, square, and hexagonal geometries of current-generation fuel assemblies that are used in Russian-origin research reactors.

CORE MODIFICATIONS FOR CONVERSION

Two presentations on modifications of research reactor cores to address the technical challenges of conversion were given by Panel 2.1 speakers: John Stevens (Argonne National Laboratory) provided a U.S. viewpoint on core modifications (Stevens, 2011), and I.T. Tetiyakov (NIKIET) provided a Russian viewpoint (Tetiyakov, 2011).

U.S. Viewpoint on Core Modifications

John Stevens

The conversion of a research reactor from HEU to LEU fuel can result in performance penalties in the reactor, primarily arising from the reduced density of uranium-235 and absorption of neutrons by uranium-238. Modifications to a reactor core may be required to overcome these penalties. Several core modification strategies have been used to overcome the penalties associated with the conversion of U.S.-origin research reactors; these include modifications to the following:

[8] Extrusion processes are used to manufacture research reactor fuel in Russia, whereas rolling processes are used to produce research reactor fuels in the United States and Europe. Both processes produce suitable fuels, but fuel produced by extrusion generally has a lower density than fuel produced by rolling.

- fuel plate thickness and reflector locations;
- fuel meat thickness;
- uranium and burnable absorber loading; and
- fueled height of the core.

When making modifications to a reactor core one should strive to change as little as possible. Two particularly successful strategies for overcoming performance penalties that entail minimal changes are (1) tuning the burnable absorber to match the fuel composition; and (2) if cost is acceptable, modifying reflector materials and/or geometries.

Of course, the fuel will, by definition, change from HEU to LEU during the conversion process, and the LEU fuel must be "acceptable" for conversion. An LEU fuel is considered to be acceptable for conversion when it meets the following criteria:

- Qualified: the fuel assembly has been successfully irradiation tested and is licensable.
- Commercially available: The fuel assembly is available from a commercial manufacturer.
- Suitable: The fuel assembly satisfies the criteria for LEU conversion of a specific reactor; safety criteria are satisfied; fuel service lifetime is comparable to current HEU fuel; and the performance of experiments is not significantly lower than for HEU fuel.
- The reactor operator and regulator agree to accept fuel assembly for conversion.

Successful conversion requires the involvement of reactor operators to understand their needs and constraints.

The following examples were presented to illustrate some of the core modification options that are available to overcome conversion penalties. Some of the reactors described in these examples have already been converted, whereas others have not yet been converted.

Oregon State TRIGA Mark II Reactor

The Oregon State TRIGA reactor is licensed to operate at a steady state power of 1.1 megawatts (MW) and can pulse to 2,500 MW with a peak steady-state thermal flux of about 10^{13} neutrons per square centimeter per second (n/cm^2-s) in the B1 position. The reactor was originally fueled with a 70 percent enriched $UZrH_x$ fuel with a 1.6 weight percent erbium burnable absorber. The reactor was converted to a 19.75 percent enriched $UZrH_x$ fuel with a 1.1 weight percent erbium burnable absorber.

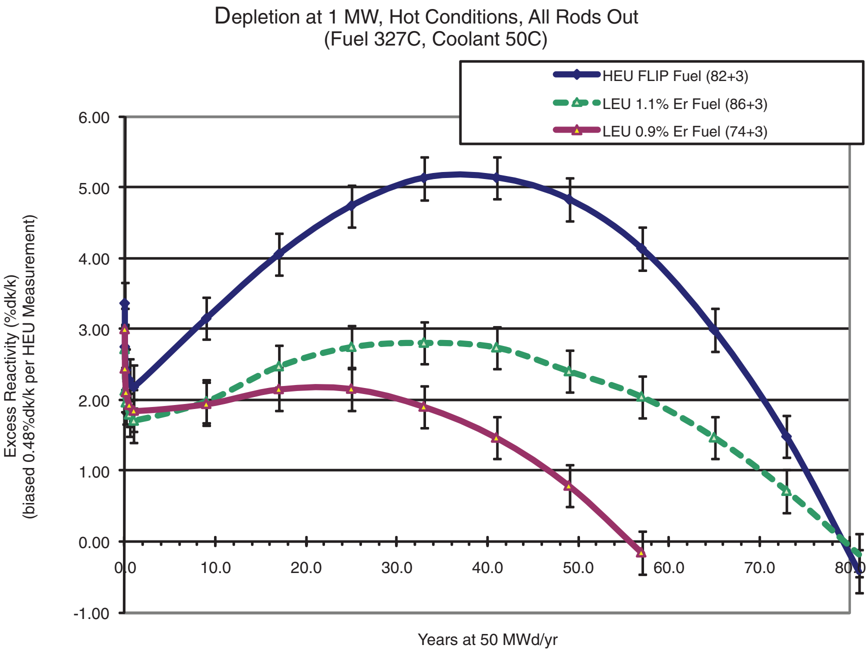

FIGURE 2-4 Plot of excess reactivity versus time at constant burnup rate for the Oregon State TRIGA Reactor. Adjusting the burnable poison to 1.1 percent in the LEU core provided an acceptable shutdown margin and maintained the longevity of the core (middle curve in the figure). SOURCE: Stevens (2011).

This reactor has a lifetime core, and it was important to the reactor operator to maintain a full grid of fuel assemblies in the converted core to maintain flexibility for conducting irradiation experiments. However, maintaining a full core reduced the shutdown margin (i.e., raised the excess reactivity) at the beginning of life of the new reactor core. Adjusting the erbium burnable poison to 1.1 percent in the converted core restored the shutdown margin and maintained the longevity of the core (Figure 2-4).

RPI Research Reactor

The RPI research reactor is licensed to operate at 1 MW power and has a peak flux of about 3.1×10^{13} n/cm^2-s. The core was converted from a 93 percent enriched UAl_x-aluminum dispersion fuel to 19.75 percent enriched uranium silicide (U_3Si_2)-aluminum dispersion fuel in 2007. The LEU fuel contains slightly more uranium-235 than the HEU fuel it replaced to account for the increased neutron absorption by uranium-238.

The conversion goal for this reactor was to allow for 10 years of operation at acceptable neutron flux density levels using the same number or fewer fuel assemblies. A silicide fuel with the same fuel meat thickness as the original HEU fuel met this goal when the core contained 17 fuel assemblies. However, by increasing the thickness of the fuel meat by 0.1 millimeters, the conversion goal could be met using only 13 fuel assemblies, a savings of 4 assemblies. Additionally, by changing the locations of some of the beryllium reflector blocks, designers were able to increase neutron flux densities in key locations in the reactor core to better suit experimental needs.

MURR

MURR is a high-performance research reactor with a very compact core (core volume of only 33 liters with 4.3 liters of fuel meat) with a peak thermal flux of about 6.0×10^{14} n/cm^2-s (Figure 2-5). The reactor is

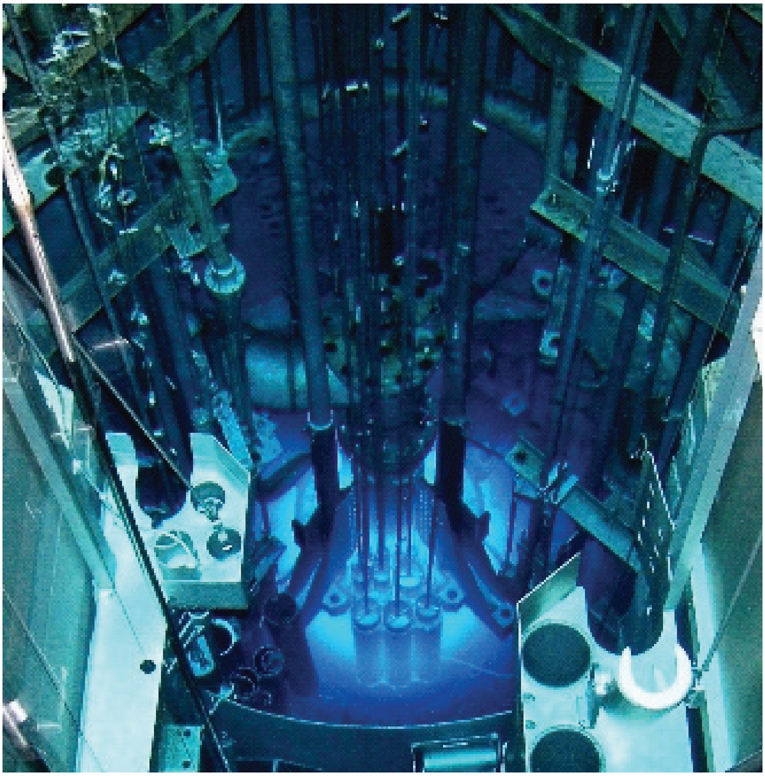

FIGURE 2-5 Photo of the MURR reactor core. SOURCE: Roglans (2011).

refueled weekly to maintain a greater than 90 percent capacity factor for efficient production of medical isotopes.

Conversion studies for this reactor showed that if the fuel geometry was unchanged, conversion using UMo monolithic LEU fuel would result in a harder neutron spectrum and, thus, increased power in some regions of the reactor. A means to control this higher power density needed to be identified for conversion to become possible.

The reactor fuel plates are curved, and there is no flexibility to rearrange them to reduce power peaking. However, it was determined that by using four distinct thicknesses of fuel meat in the assemblies (ranging from 0.23-0.43 millimeters), peaking factors could be reduced to acceptable levels.

Belgian BR2

The Belgian BR2 reactor typically operates at 50-80 MW with a peak thermal flux of about $0.8\text{-}1.1 \times 10^{15}$ n/cm^2-s. The fuel consists of curved plates that are swaged together at stiffener joints to form six concentric tubes. The fuel meat is 93 percent enriched uranium containing integrated boron and samarium burnable poisons.

The reactor is planned to be converted using a 19.75 percent enriched UMo dispersion LEU fuel. However, integrating a burnable poison into these fuel plates will be difficult owing to the high-volume fraction of UMo in the dispersion. Consequently, the reactor operator plans to install cadmium wires in the swage joints between the curved fuel plates to control reactivity, a technique that has been used successfully in some other conversions to silicide fuel.

RHF

RHF has a maximum power of 58 MW and a peak thermal neutron flux of about 1.5×10^{15} n/cm^2-s. The core consists of a one-time-use assembly consisting of 280 curved plates arranged between two concentric cylindrical “sideplates.” The reactor is currently fueled with a 93 percent enriched UAl_x-aluminum dispersion fuel with boron-10 burnable poisons at the tops and bottoms of the fuel plates.

This reactor is used as a neutron beam source, and a key requirement for conversion is the preservation of “brightness” (i.e., intensity) of these beams and the reactor cycle length. To meet these objectives, the fueled height of the reactor core will be increased by eliminating the burnable poison zones at the tops and bottoms of the fuel plates. (These poisons will be moved to another location in the reactor.) However, even with this change there will still be a 5-10 percent loss in brightness at key experimental positions. This loss of brightness can be compensated for by increasing the beam times for some experiments, but it will not affect overall throughput of experiments in the reactor.

Russian Viewpoint on Core Modifications

I.T. Tetiyakov

When converting a research reactor from HEU to LEU fuel it is important to avoid degradation of the following:

- Consumer characteristics: neutron flux level, thermal power, neutron spectrum, and adequacy of safety systems.
- Safety characteristics: reactivity margins, effectiveness of control rods, and peak power density.
- Performance characteristics: fuel cycle duration, number of planned reactor shutdowns, and reactor serviceability.
- Technical and economic indices: mass of uranium loading, volume of spent fuel to be reprocessed, and financial expenditure for fuel purchase and reprocessing of spent fuel.

There are two potential paths for converting a research reactor while maintaining these characteristics. One path is to design a new core that can fit into the existing reactor. The other path is to maintain the geometric configuration of the current core but change the design and arrangement of fuel and/or reflector elements.

Conversion to LEU fuel may result in decreased uranium-235 content and will result in increased uranium-238 content in the reactor core. This can change the neutronic characteristics of the core, which in turn can change its reactivity, the effectiveness of control rods, and the dynamics of fuel burnup. All of these changes can affect reactor safety. Consequently, safety analyses must be carried out to demonstrate that conversion will preserve reactor safety at required levels, including neutron-physical analysis, thermal-hydraulic analysis, and an analysis of transient and emergency operations.

As illustrated by the following three examples, for some Russian research reactors there are no developed LEU fuel elements that would enable conversion with acceptable consumer characteristics. Moreover, some Russian research reactors are approaching the ends of their operating lives, and there is a need to consider whether to shut down these reactors or modernize them.

IRT (Moscow Engineering and Physics Institute)

The IRT is a medium-flux, 2.5 MW pool-type reactor with a square core containing 16 IRT-3M fuel elements enriched in uranium-235 to 90 percent. Initial studies have been carried out to examine the feasibility of converting this reactor to 19.75 percent enriched fuel elements of an IRT-4M design containing a UO_2-aluminum dispersion fuel meat.

These studies indicate that conversion would result in some consumer and economic penalties compared to HEU fuel: neutron flux densities in the fuel and reflector regions would decrease by 20-30 percent and 10-20 percent, respectively, and the number of fuel elements in the core would increase by 2-4 elements.[9] The economics of conversion will depend on the cost of LEU fuel elements and their reprocessing compared to the costs for HEU fuel elements.[10] However, there would be no unacceptable changes in safety characteristics, and fuel burnups would not change.

IVV-2M (Institute of Nuclear Materials, Zarechny)

The IVV-2M is a high-flux, 15 MW pool-type reactor with a hexagonal core containing 42 hexagonal fuel elements enriched in uranium-235 to 90 percent. Initial studies have been carried out to examine the feasibility of converting this reactor to 19.75 percent enriched fuel containing a UO_2-aluminum dispersion fuel meat.

This reactor is being very effectively operated at present and has a high level of utilization, so any significant loss of consumer characteristics following conversion would be problematic. Initial conversion studies have focused on identifying a fuel type that would meet consumer needs. Analytical studies have examined the reactor characteristics that would result from conversion to dispersion fuels having uranium densities of 3.5 and 6.5 gU/cm^3 as well as a UMo-aluminum dispersion fuel.

Conversion to a 3.5 gU/cm^3 fuel that was manufactured using existing (extrusion-based) fuel fabrication technologies would result in insufficient reactivity reserve and the deterioration of other consumer characteristics such as burnup. Conversion using 6.0-6.5 gU/cm^3 fuel would improve the feasibility of conversion if fuel elements with such material were able to be manufactured economically.

[9] This conclusion was reached by studying the use of existing IRT-4M LEU fuel. A feasibility study with the IRT-3M UMo fuel of higher density, currently under development, is under way at MEPhI (see Chapter 3) and may reach different conclusions when completed.

[10] In Russia, reprocessing costs are based on fuel mass.

WWR-M (Petersburg Nuclear Physics Institute, Gatchina)

The WWR-M is an 18 MW pool-type reactor with a hexagonal core containing 145 fuel elements of WWR-5M design that are enriched in uranium-235 to 90 percent (Figure 2-6). Although this reactor entered service in 1959, it is still operating very effectively and has achieved several increases in power and flux densities since it was commissioned. It is now the highest power reactor of its type in existence.

Initial studies have been carried out to assess the feasibility of converting this reactor to 36 percent enriched and 19.75 percent enriched fuel. It was observed that as enrichments decrease, burnups, thermal neutron flux densities, and fast neutron flux densities also decrease. These studies indicate that fuel having a uranium density of 8.25 gU/cm^3 would be required to convert this reactor without sacrificing needed consumer characteristics. However, fuels with this density are not available at present.

MAINTAINING PERFORMANCE AND MISSIONS

Two presentations discussing performance and missions of reactors after conversion were given by Panel 2.1 speakers: Jordi Roglans (Argonne National Laboratory) provided a U.S. viewpoint on maintaining performance and missions (Roglans, 2011), and A.L. Petelin (Research Institute of Atomic Reactors [RIAR]) provided a description of several Russian research reactors at RIAR and their missions (Svyatkin et al., 2011).

U.S. Viewpoint on Maintaining Performance and Missions

Jordi Roglans

The Global Threat Reduction Initiative (GTRI) strives to achieve several goals when converting research reactors:

- Develop or identify an LEU fuel assembly that is acceptable for conversion.
- Ensure that the ability of the reactor to perform its scientific mission is not significantly diminished.
- Ensure that conversion can be achieved without requiring major changes in reactor structures or equipment.
- Demonstrate that the LEU fuel meets all safety requirements and that conversion and subsequent operations can be accomplished safely.
- Ensure that annual operating costs do not increase significantly as the result of conversion.
- Develop a conversion schedule that is based on operational requirements, capabilities, and regulatory processes.

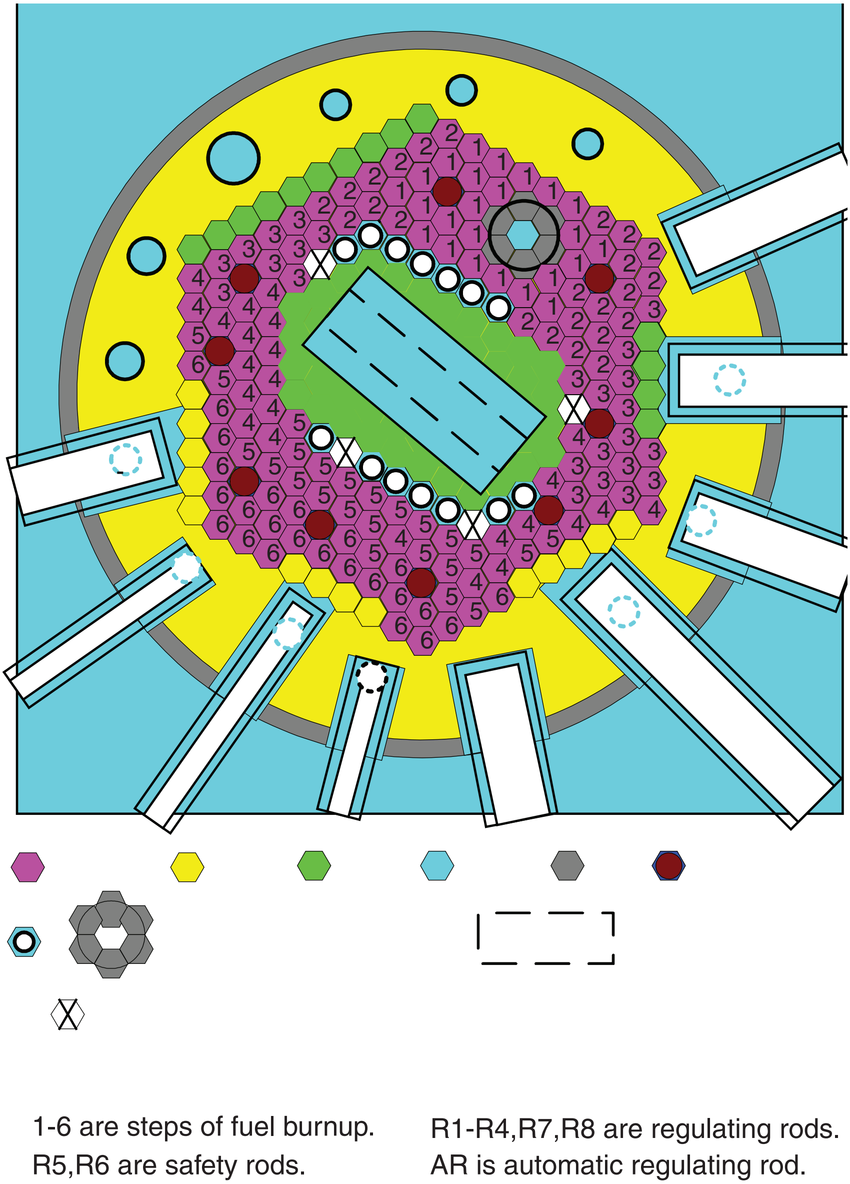

FIGURE 2-6 Schematic illustration of the WWR-M reactor core. SOURCE: Tetiyakov (2011).

As noted in John Stevens' presentation (summarized elsewhere in this chapter), a fuel assembly is considered to be acceptable for use in a conversion project when it meets the following criteria and the reactor operator and regulator agree to accept fuel assembly for conversion:

- Qualified: The fuel assembly has been successfully irradiation-tested and is licensable.
- Commercially available: The fuel assembly is available from a commercial manufacturer.
- Suitable: The fuel assembly satisfies the criteria for LEU conversion of a specific reactor; safety criteria are satisfied; fuel service lifetime is comparable to current HEU fuel; and the performance of experiments is not significantly lower than for HEU fuel.

When converting from an HEU to LEU fuel, one should strive to make as few changes as possible in the fuel assembly and core geometries. Conversion should also be carried out in a way that has the least possible effect on scientific operations in the facility.

The annual operating costs of a reactor will be affected by the costs of the LEU fuel assemblies compared to the HEU fuel assemblies they are replacing. The new very-high-density UMo fuels will likely cost more to fabricate because there are more manufacturing steps. However, work is under way to minimize those cost differences with the goal of maintaining or even reducing when feasible the number of LEU fuel assemblies that are consumed in a reactor each year compared to HEU fuel assemblies.[11] The number of fuel assemblies consumed per year dominates costs when LEU and HEU fuel assemblies are of similar cost.

Analytical studies are typically needed to determine whether conversion can be accomplished without a significant impact on reactor performance and missions. However, such formal studies may not be required for HEU-fueled reactors that are of a similar type and performance to reactors that have already been converted to LEU.

The analytical studies needed to assess the potential for conversion include:

- Feasibility studies that identify suitable LEU fuel assemblies (either existing qualified fuels or new fuels under development), compare reactor performance with HEU and LEU fuels, and calculate key safety parameters.
- Operational and safety analyses to demonstrate that the transition from HEU to LEU fuel can be done safely and without interrupting normal

[11] Service lifetimes of LEU fuel assemblies can be increased if the uranium-235 loadings are higher than comparable loadings in the HEU fuel.

reactor operations, and also that the converted reactor satisfies all safety requirements.

One also needs to formulate safety requirements and resolve any issues raised by regulators regarding the reactor's safety documentation. Additionally, economic impact studies may be required to determine the overall impact and acceptability of conversion.

A feasibility study entails many activities. Initially, fuel requirements and experimental performance indicators must be defined. With respect to the latter, it is important to determine what the most important experimental positions are in the reactor and what performance characteristics (e.g., flux densities and neutron energy distributions) are required in those positions. Iterative modeling studies are used to determine these characteristics as well as other operating criteria such as shutdown margins. Fuel assembly and reactor core designs are adjusted, and the models are rerun until acceptable performance and other important reactor characteristics are achieved. The final LEU fuel assembly design can be selected once these studies are completed.

Some high-performance reactors may require fuel-design optimization and possibly facility-specific mitigation measures to address any performance penalties arising from conversion. For U.S. high-performance reactors, the anticipated unmitigated decreases in performance resulting from conversion do not preclude any current applications but could affect application throughputs. The high demand for these reactors is already limiting scientific output and isotope production. Consequently, several mitigation strategies are being pursued to avoid throughput penalties.

For the U.S. high-performance reactors, the following mitigation strategies are being pursued:

- HFIR: The anticipated performance penalty of 10-15 percent will be mitigated by increasing reactor power from 85 MW to 100 MW. This could result in small gains in performance.
- MITR: The anticipated performance penalty of 5-10 percent will be mitigated by increasing reactor power from 6 MW to 7 MW.
- MURR: The anticipated performance penalty of 15 percent will be mitigated by changing LEU plate thickness (see the presentation by John Stevens elsewhere in this chapter) and by increasing reactor power from 10 MW to 12 MW.
- NBSR: The anticipated performance penalty of 10 percent will be mitigated by upgrading the cold neutron source.

Power increases in HFIR, MITR, and MURR are possible because their existing cooling systems are adequate to handle the increased heat loads. As a result of these mitigation strategies, no current applications are expected to be precluded by conversion. In ATR, preliminary studies indicate that there could be a 5-10 percent performance penalty after conversion. A strategy to mitigate this penalty has not yet been identified.

The key to successful conversion is collaboration. In the case of high-performance reactors or reactors with unique designs, iterative collaborations among facility operators, fuel designers, and conversion analysts are essential to optimize fuel and core design and minimize performance impacts.

Descriptions of Russian Research Reactors

A.L. Petelin

The Russian presentation focused on current characteristics and missions of the research reactors at RIAR in Dimitrovgrad. RIAR is Russia's largest complex for examinations of full-scale components of nuclear reactors and irradiated materials. It also has equipment and facilities for fuel cycle research and a radiochemical complex for investigation and production of transuranic elements and radioisotopes.

RIAR currently operates five research reactors. A sixth reactor is being decommissioned. The characteristics and missions of the operating reactors are described briefly in the following sections.

SM-3

SM-3 is a 100 MW pressurized water flux trap-type reactor containing 32 fuel elements enriched in uranium-235 to 90 percent. The reactor has a compact square core (420 mm in plan dimension and 350 mm in height) with a central trap. Up to 41 positions are available for irradiation experiments in the central trap, core, and reflector. The maximum thermal neutron flux density in the central trap is 5×10^{15} n/cm^2-s. Thermal neutron flux densities of 1.5×10^{13} to 1.5×10^{14} n/cm^2-s can be obtained in the reflector.

The reactor has two low-temperature coolant water loops and a high-temperature loop that can be used for fuel testing, examination of fission-product releases from leaky fuel rods and their removal from primary cooling circuits, and the irradiation of structural and absorbing materials. The spectral characteristics and neutron-flux-density variability in this reactor also make it useful for producing a range of isotopes, including transplutonium elements and industrial isotopes such as cobalt-60.

This reactor is potentially useful for other high-dose irradiation applications, for example, testing of fuel and structural materials for high-temperature reactors, fast-boiling reactors, and supercritical reactors, as well as new designs for research reactors. In particular, new LEU fuel compositions can be examined for applications in high-flux reactors. The reactor can also be used for training.

MIR.M1[12]

MIR.M1 is a 100 MW loop-type reactor that uses 48-58 fuel elements enriched in uranium-235 to 90 percent (see Figure 3-8 in Chapter 3). It has seven loop facilities: Two with water coolant (PV-1, PV-2), two with water/boiling-water coolant (PVK-1, PVK-2), two with water/boiling water and steam coolant (PVP-1, PVP-2), and one with nitrogen and helium coolant (PG). The facility also contains hot cells and cooling pools. The maximum thermal neutron flux in the loop channel is 5-7 × 10^{14} n/cm^2-s.

A variety of experimental activities are currently performed in this reactor. These include the examination of advanced VVER-1000 fuel, testing of VVER-1000 fuel with high burnup (greater than or equal to 60 megawatt days per kilogram of uranium [MWd/KgU]), testing of new VVER cladding materials, and examination of fission-product releases from VVER-1000 fuel rods containing artificial defects. The reactor is also used to test LEU fuel and produce the industrial isotope iridium-192.

This reactor is potentially useful for other types of experimental applications, including high-temperature and high-pressure testing of reactor materials, simulation of severe reactor accidents, testing of innovative fuel and cladding materials, and expanded production of isotopes. Realizing some of these activities would require upgrades to some of the reactor loops.

RBT-6 and RBT-10/2

The RBT-6 and RBT-10/2 reactors are pool-type reactors of similar design. The RBT-6 operates with 56 fuel elements at a power of 6 MW, whereas RBT-10/2 operates with 78 fuel elements at a power of 10 MW. Both reactors have neutron flux densities of about 1 × 10^{14} n/cm^2-s. The fuel for both reactors is UO_2 dispersed in a copper-beryllium matrix enriched to 90 percent.

Although these reactors operate at full power most of the time, their experimental channels (up to 8 for RBT-6, slightly more channels for RBT-10/2) only have about 50 percent utilization. There is interest in increasing the usage of these reactors. Possible additional experimental activities

[12] This reactor is also discussed in another symposium presentation that is summarized in Chapter 3.

include silicon doping, isotope production (including molybdenum-99 production), testing of industrial materials, and neutron capture therapy. Some of these activities would require redesign of the experimental channels.

BOR-60

BOR-60 is a 60 MW sodium-cooled fast reactor that can produce up to 12 MW of electricity. It is fueled with UO_2 or UO_2-PuO_2 fuel with uranium-235 enrichments of 45-90 percent and plutonium content of 70 percent. It has a maximum neutron flux density of 3.7×10^{15} n/cm^2-s.

This reactor is currently used for test irradiations of reactor fuels and materials, including new fuels, cladding, and structural materials for fast reactors, water cooled reactors, and fusion reactors. It is also being used for transmutation research, other fuel cycle research, and isotope production. The experimental applications could be expanded to include advanced reactor and fuel cycle research.

AGEING AND OBSOLESCENCE OF RESEARCH REACTORS

Two presentations on understanding and addressing the ageing and obsolescence of research reactors were given by Panel 2.2 speakers: H.-J. Roegler (an independent consultant from Germany, formerly with Siemens[13]) described an International Atomic Energy Agency (IAEA) initiative on research reactor ageing and ageing management (Roegler, 2011). E.P. Ryazantsev (Kurchatov Institute) provided a historical description of the research and test reactors at the Kurchatov Institute (Ryazantsev, 2011).

IAEA Initiative on Research Reactor Ageing and Ageing Management

H.-J. Roegler

The IAEA's activities in ageing and ageing management for research reactors began in the mid 1990s. In March 1995, the IAEA issued a TECDOC report (IAEA, 1995) on how to manage ageing in research reactors. Two months following the release of this report, the IAEA sponsored a conference on research reactor ageing; the conference was held in Germany and involved more than 100 participants. In December 2008—more than a decade after publication of the TECDOC and sponsorship of the follow-up conference—the IAEA hosted an expert meeting at its Vienna headquarters to review the history of the agency's efforts on ageing, including the ade-

[13] And under a Contract Service Agreement with the IAEA.

quacy of existing documentation, and to consider whether an initiative to collect additional information was warranted.

As the result of this expert meeting, the IAEA initiated the development of a database on research reactor ageing. This database is intended to address ageing as a technical and safety issue and explicitly excludes reactor conversion to LEU fuel. Information for the database was collected from research reactor operators using a standard template that was developed by the IAEA. The template permitted the reporting of a maximum of 3 ageing problems, classified by 13 possible ageing mechanisms in 76 reactor systems arranged in 9 groups. The template provided space for descriptions of ageing problems and actions taken to mitigate or cure them. A contact address for the reporting reactor was also required.

The templates were distributed in February 2009 to 133 research reactor operators plus 28 other manufacturers and authorities. Responses from these organizations were incorporated into the database in October that same year. A total of 188 templates were initially submitted from 77 reactor facilities plus 6 other institutions (contributors were permitted to submit more than one template per facility). After review and revisions of the initial submissions, a total of 155 templates reporting on 367 ageing problems were included in the database.

There was a rather high-level of non-participation (43 percent) in this survey, which could have been caused by several factors, including language barriers, inexperience with completing these types of templates, or concern that the ageing problems might be publicly disclosed. One non-respondent justified the lack of participation as follows:

> We do not have an ageing management program, because we do not have the funding for such a thing. We fix things when they break. That is unfortunately the nature of our business here due to monetary constraints. For me to fill out your template with something that is irrelevant is not worth your time, or ours. ...We also do not necessarily wish to have this information be publicly available.

However, a convincing number of useful observations emerged from the template data that were submitted to the IAEA:

- The 77 reactors represented in the template responses range from less than 5 years to more than 50 years old. The average age was 37.8 years (Figure 2-7).
- The most frequently reported ageing problems were obsolescence and technology changes (92 out of 367 reports); corrosion (73 out of 367); and changes to regulatory requirements and standards (49 out of 367).

Other frequently reported ageing problems included mechanical fatigue and wear and radiation-induced ageing (Figure 2-8).

- There were more ageing problems reported for younger reactors than for older reactors. This suggests that ageing problems begin with the initiation of operation of a research reactor.

Taken together, these data demonstrate the need for the future management of ageing in research reactors.

Although the database intentionally excluded information related to conversion, as noted previously, the information in the database is still potentially useful for conversion planning, because conversion needs to consider past as well as future ageing. The information in the database could be used, for example, to identify:

- Ageing systems and mechanisms to investigate
- Issues to discuss with the authorities
- Contacts for advice on addressing every type of ageing problems

The IAEA is planning to undertake a first update of this ageing database in August 2011. This will involve the reconfirmation of research reactor operator contacts, updates to the content of templates, and fresh approaches to the research reactor operators who did not provide information in 2009.

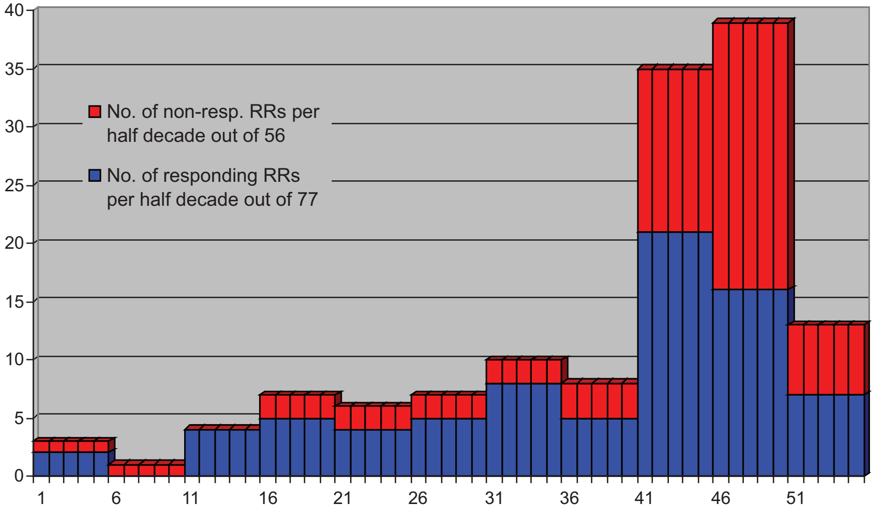

FIGURE 2-7 Age distribution of research reactors surveyed by the IAEA. SOURCE: Roegler (2011).

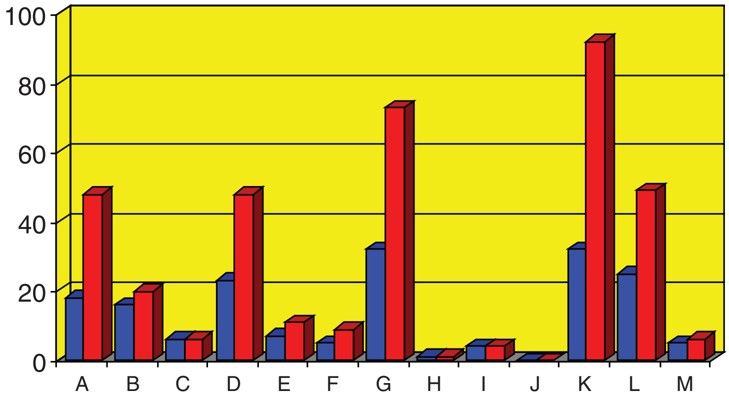

FIGURE 2-8 Reported ageing mechanisms at research reactors surveyed by the IAEA.
NOTE: A = Radiation induced; B = Temperature induced; C = Creep due to stress; D = Mechanical displacement/fatigue/wear; E = Material deposition; F = Erosion; G = Corrosion; H = Damage (power excursion); I = Flooding consequences; J= Fire consequences; K= Obsolescence/technology change; L = Required/standard changes; M = Other.
Blue = Different systems (out of 76) nominated per mechanism.
Red = Total nominated issues (out of 367) per mechanism.
SOURCE: Roegler (2011).

Reactors at the Kurchatov Institute[14]

E.P. Ryazantsev

The practical use of atomic energy for civilian and military purposes in the Soviet Union began with the launching of research reactor F-1 in December 1946. The reactor is graphite moderated and is fueled with 50 tonnes of natural uranium. Its operational range extends from 25 kW to 4 MW. This reactor is still operating today and is used as a reference source for neutron fluxes.

There have been a total of 80 research reactors constructed by the Soviet Union, including the following 15 reactors that were constructed in foreign countries:

- VVR-S (2-10 MW power): Constructed in East Germany, Czechoslovakia, Romania, Poland, Hungary, and Egypt between 1957 and 1961.

[14] Some of the Russian reactors described in this presentation are also discussed in another presentation that is summarized in Chapter 3.

- IRT-2000 (2-10 MW): Constructed in China, Bulgaria, North Korea, and Iraq between 1961 and 1967.
- TBP-C (10 MW): Constructed in China in 1959.
- RA (10 MW): Constructed in Yugoslavia in 1959.
- IRT-10000 (10 MW): Constructed in Libya in 1981.
- MARIA (30 MW): Constructed in Poland in 1974.
- IVV-9 (0.5 MW): Constructed in Vietnam in 1983.

Eleven research reactors besides F-1 have been constructed at the Kurchatov Institute:

- RFT: Channel graphite reactor; initial power 10 MW, later upgraded to 20 MW; began operations in 1957 and was partially demolished in 1962.
- VVR-2: Pool-type reactor; initial power 0.3 MW, later upgraded to 3 MW; began operations in 1954 and was dismantled in 1983.
- IRT: Pool-type reactor; initial power 2 MW, later upgraded to 5 MW; began operation in 1957 and was dismantled in 1979.
- MR: Channel-type reactor immersed in a pool; initial power of 20 MW, later upgraded to 50 MW; began operation in 1963 and was shut down in 1993.
- Chamomile: High-temperature neutron thermoionic converter; 0.1 MW; began operation in 1964 and was shut down in 1996.
- Hydra: Homogeneous pulse reactor; 0.01 MW (30 mega Joules per pulse); began operations in 1972 and is currently operational.
- Yenisei: High-temperature neutron thermoionic converter; 0.1 MW; began operation in 1973 and was dismantled in 1986.
- IR-8: Pool-type reactor; 8 MW; began operation in 1981 and is currently operational (Figure 2-9).
- Argus: Homogeneous reactor; 0.02 MW; began operations in 1981 and is currently operational.
- Gamma: Cabinet water-cooled reactor; 0.125 MW; began operation in 1982 and is currently operational.
- OR (referred to as OP-M in Table 1-2 in Chapter 1): Pool-type reactor; 0.3 MW; began operation in 1989 and is currently operational.

These reactors created an experimental base for nuclear and materials research at the Kurchatov Institute.

The remainder of this presentation focused on the characteristics of the MR and IR-8 reactors at the Kurchatov Institute and activities at a branch institute in Sosnony Bory (Leningrad region).

MR was equipped with 10 experimental loops, each of which functioned as a small prototype power reactor. Several coolants were used in these loops, including pressurized water, steam-water mixtures, helium, carbon dioxide, and liquid lead bismuth. The neutron flux density in the reflector was 5×10^{14} n/cm^2-s. This reactor was used to work out the structure of active zones of nuclear reactors and test 400 fuel assemblies and more than 8,000 fuel rods for VVER, RBMK, ACT, high-temperature, and naval reactors.

IR-8 has a compact core with an effective reflector that provides for large thermal neutron densities of 2.3×10^{14} n/cm^2-s. The core contains 12

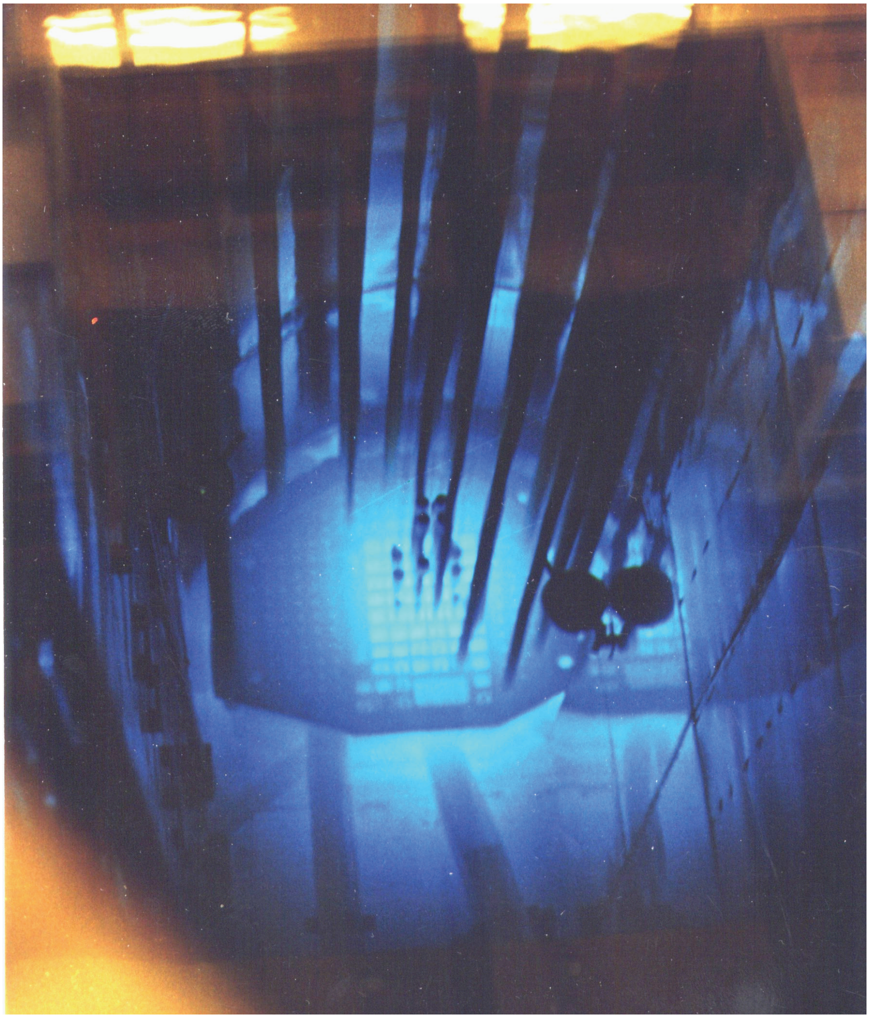

FIGURE 2-9. Photograph of the IR-8 reactor at the Kurchatov Institute. SOURCE: Ryazantsev (2011).

experimental channels in a horizontal orientation. This reactor is used to carry out fundamental research in nuclear physics, solid state physics and superconductivity, and other experiments.

The Scientific Research Technological Institute (NITI), a branch of the Kurchatov Institute, was created in Sosnovy Bor in 1964. It has a full-scale prototype submarine reactor.

REGULATORY CHALLENGES TO CONVERSION

Two presentations on the regulatory challenges of converting research reactors were given by Panel 2.2 speakers: Alexander Adams (U.S. Nuclear Regulatory Commission) provided a U.S. viewpoint (Adams, 2011), and V.S. Bezzubtsev provided a Russian viewpoint (Bezzubtsev, 2011).

U.S. Viewpoint on Regulatory Challenges

Alexander Adams

The mission of the U.S. Nuclear Regulatory Commission (USNRC) is to ensure that the commercial use of nuclear materials in the United States is conducted safely. The USNRC is responsible for regulating civilian research reactors, including research reactor fuels and conversions, but the agency does not regulate U.S. Department of Energy (DOE) reactors.

Regulation of Research Reactor Fuel

Research reactor fuel development is the responsibility of DOE under the GTRI program. The USNRC does not get involved directly in these fuel development activities, but it does have the responsibility for approving LEU fuels for use in USNRC-licensed reactors.

USNRC approval of new LEU fuels is based on information submitted by DOE, including:

- Results of LEU fuel development and testing.
- Information on LEU fuel fabrication.
- LEU fuel qualification reports.

The USNRC must conclude that an LEU fuel is suitable and acceptable for use before approving it for use in USNRC-licensed reactors. Once an LEU fuel is approved, licensees can reference the USNRC evaluation in their Safety Analysis Reports; licensees do not have to justify the generic aspects of an LEU fuel that has been approved by the USNRC. However, licensees are required to address any facility-specific issues related to use of that fuel.

To date, the USNRC has approved three LEU fuels for use in USNRC-licensed reactors:

- Uranium silicide (U_3Si_2) fuel;
- U-ZrH_x fuel for TRIGA reactors; and
- Special Power Excursion Reactor Test (SPERT) fuel elements.

Regulation of Research Reactor Conversions

When regulatory requirements for conversion became effective there were 26 HEU-fueled civilian research reactors in the United States under the regulatory authority of the USNRC. Most of these reactors were being operated by universities. The current conversion status of these reactors is shown below:

- Sixteen reactors were converted to LEU fuel, and five of those reactors were subsequently shut down after conversion.
- The licenses of four reactors were terminated before conversion.
- Decommissioning was approved for two reactors before conversion.
- No suitable fuel has been identified for one reactor (MITR).
- Unique purpose applications (described later) are pending for two reactors.
- Suitable fuel has been identified but no funding is available to convert one reactor (NTR General Electric).

The first group of reactor conversions (10 reactors) was completed in 2000. The second group of reactor conversions (6 reactors) began in 2006 and was completed in 2009. In 2007, the USNRC staff turned its attention to conversion of three of the four remaining HEU-fueled reactors that it licenses, which are high-performance reactors: MITR, MURR, and NBSR.[15]

The Commission issued a policy statement in 1982 that fully supported the Reduced Enrichment for Research and Test Reactors (RERTR) program. Initially, many research reactor licensees resisted the call for conversion, informing the USNRC that they preferred instead to implement additional security measures at their facilities. The Commission members and staff engaged licensees through a number of outreach activities, and a Commission-sponsored LEU study group comprising licensed technical experts prepared a report on the technical feasibility of conversion.

[15] The USNRC does not regulate the High Flux Isotope Reactor at the Oak Ridge National Laboratory or the Advanced Test Reactor and its critical assembly at the Idaho National Laboratory. These reactors are the responsibility of DOE.

The Commission also developed a conversion rule, which was promulgated in Title 10, Section 50.64 of the Code of Federal Regulations (10 CFR 50.64, Limitations on the Use of Highly Enriched Uranium [HEU] in Domestic Non-power Reactors) in 1986. At about the same time this rule was issued, the Commission initiated steps to reduce the amount of unirradiated HEU fuel that licensees were authorized to possess at their facilities. Licensees now minimize their onsite inventories.

The regulations in 10 CFR 50.64 prohibit new construction permits for HEU-fueled reactors unless those reactors have a "unique purpose." It also prohibits acquisition of additional HEU fuel for current reactors if LEU fuel acceptable to the Commission is available, again unless the reactor has a unique purpose. The regulations also require reactor licensees to replace HEU fuel with LEU fuel acceptable to the Commission in accordance with an approved schedule. To be acceptable to the Commission, LEU fuel must (1) meet the operating requirements of the existing license, or (2) based on a safety review and approval by the USNRC, be used in a manner that protects public health and safety and promotes the common defense and security, and (3) limit to the maximum extent possible the use of HEU fuel.

The USNRC defines "unique purpose" as a project, program, or commercial activity that cannot be reasonably accomplished without HEU. This includes specific projects, programs, or commercial activities that significantly serve the U.S. national interest; reactor physics or reactor development; research based on HEU flux levels or spectra; or reactor cores of special design.

The Commission initially received four unique purpose applications from U.S. licensees. Two of these (for the MITR and the Cintichem Reactor[16]) were withdrawn, and the other two (for MURR and NBSR) have been pending for about 20 years. The Commission staff decided to defer decisions on these applications shortly after they were submitted; these decisions will continue to be deferred until a fuel acceptable to the Commission is developed for use in these reactors.

The timing of conversion depends on several factors: The availability of government funding; the availability of LEU fuel acceptable to the Commission; the availability of shipping casks to remove HEU fuel from the facility after conversion[17]; and the level of reactor usage.

NUREG 1537 (Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors) contains guidance for licensees to submit conversion applications to the USNRC. The conversion application must include an update of the reactor's Safety Analysis Report relating to

[16] The Cintichem Reactor was located in Tuxedo, New York. It was shut down in 1990.

[17] Depending on its design, HEU fuel is shipped to either the Savannah River Site in South Carolina (for aluminum-based fuels) or the Idaho National Laboratory (for other fuel designs), where it is stored.

issues that are impacted by conversion to LEU. Specific areas of focus in the application include the following:

- Reactor neutronics and thermal hydraulics: Codes and calculations that have been benchmarked against the HEU reactor should be used to analyze the LEU reactor. The licensee should show that margins of safety are maintained in the LEU reactor.
- Reactor accidents: The licensee should reanalyze the HEU Safety Analysis Report accidents using LEU fuel to determine the impacts from conversion. Particular concerns include changes in power per fuel element, fission product inventory, and reactivity. The licensee must also perform a review to determine whether conversion to LEU fuel introduces new accident scenarios. Conversion should not have a significant impact on accident analysis results and normally should not introduce new accident scenarios.

The application also identifies all necessary changes to the license, facility, and operating procedures arising from conversion. The application must be limited to conversion and cannot include other changes or upgrades. Those are handled through the normal license amendment process.

Once the USNRC reviews and accepts an application, it issues an enforcement order directing the licensee to convert to LEU fuel and make any necessary changes to its license, facility, and procedures. By issuing enforcement orders, the USNRC assumes the burden for defending against any legal challenges that arise from conversion, thereby relieving the licensee from this responsibility.

Several lessons have been learned from the civilian research reactor conversions that have been carried out to date in the United States. First, updating the safety analyses and preparing the conversion application take time and effort and can result in the discovery of other technical issues. Second, the key to successful conversions is to develop an LEU reactor design that can be successfully analyzed and built. Finally, conversion has benefits beyond the elimination of HEU: Most notably, it can result in increased technical expertise among reactor staff and improved knowledge of reactor characteristics and operating behavior. Conversion also provides valuable training opportunities: At university reactor facilities, many students have been involved in the development of conversion analyses.

Russian Viewpoint on Regulatory Challenges

V.S. Bezzubtsev

The Russian Federation has been cooperating with the United States and the IAEA in several GTRI programs. These include the return of

Russian-origin HEU fuel to the Russian Federation from countries in Eastern Europe and Asia; reduction of fuel enrichment in research and test reactors; and enhancement of physical security for high-risk radioactive sources. Active international cooperation and collaboration are necessary for achieving the strategic objectives of GTRI.

ROSTEXNADZOR is the nuclear safety watchdog in the Russian Federation. It is responsible for regulating more than 6,000 facilities in the Russian Federation, including research and test reactors.[18] It has three primary functions: regulatory control, licensing, and supervision of atomic energy facilities.

The federal codes and standards developed by ROSTEXNADZOR are of two types: (1) general and (2) facility specific. The agency develops and promulgates federal codes and standards for atomic energy use, administrative regulations, guidelines, and safety guides. The federal codes and standards provide general safety provisions for each type of atomic energy facility, for example, nuclear power plants, research reactors, icebreaker reactors, and nuclear fuel cycle facilities. These codes and standards also provide specific provisions for activities at these facilities including siting, construction, operation, and decommissioning.

There are 10 separate codes and standards for research nuclear installations, which include research reactors. These include, for example:

- General Safety Assurance Provisions for Research Nuclear Installations (NP-033-01)
- Requirements for the Content of Research Nuclear Facility Safety Analysis Reports (NP-049-03)
- Rules of Nuclear Safety for Research Reactors (NP-009-04)
- Requirements for a Content of Action Plan for Protection of Personnel in Case of an Accident at a Research Nuclear Installation (NP-075-06)

Many of these codes and standards draw from IAEA documents, either in full or part, the latter being adapted to local conditions.

An effort is currently under way to enhance the regulatory framework for nuclear and radiation safety at research reactors in the Russian Federation. This includes the modification of current regulatory documents and the development of new regulations. The new regulations would require periodic safety reviews of research reactors, development of rules for withdrawing research reactors from state supervision, and development of procedures for modifying the design, engineering, and operating documentation of research reactors.

[18] This number includes radiation sources at hospitals.

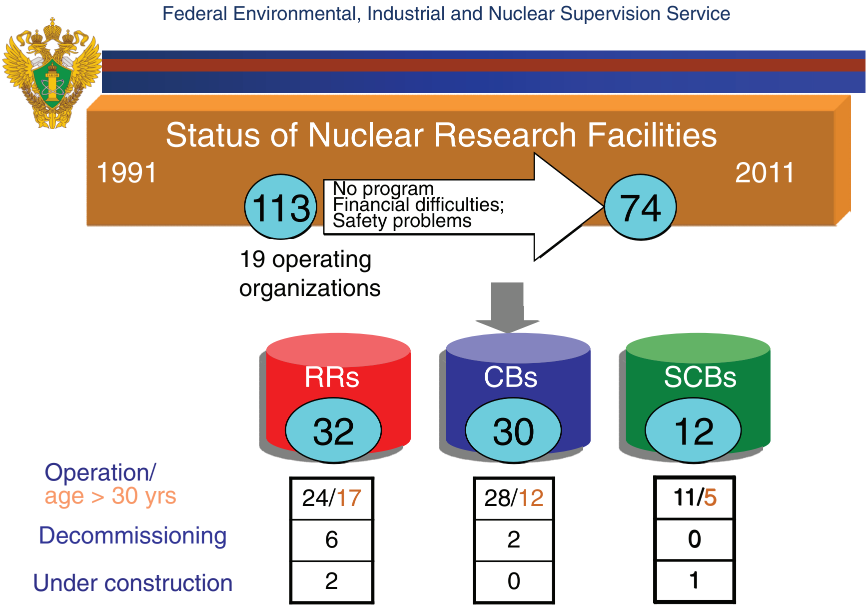

FIGURE 2-10 Status of research reactors in the Russian Federation. RR = research reactors; CBs = critical assemblies; SCBs = subcritical assemblies. SOURCE: Bezzubtsev (2011).

There were 74 licensed research reactors (including critical and subcritical assemblies) in the Russian Federation in 2011. These are being operated by 19 organizations, including Rosatom and the Russian Academy of Sciences. These reactors comprise (Figure 2-10):

- 32 research reactors (24 operating, 6 decommissioned, and 2 under construction)
- 30 critical assemblies
- 12 subcritical assemblies

The average operation age of the research reactors is 24 years, but 17 reactors have been operating for more than 30 years.

ROSTEXNADZOR is just beginning to develop regulations for the conversion of research reactors in the Russian Federation. The regulator does not see any serious barriers or obstacles that might prevent conversion-related licensing activities. The USNRC's rich experience with fuel development and conversion-related approval activities would be useful for ROSTEXNADZOR in organizing its work.

The specific issues that will need to be addressed by ROSTEXNADZOR in research reactor conversion in the Russian Federation are the following:

- R&D for design and fabrication of new LEU fuel, LEU fuel tests, and validation of LEU fuel characteristics and operating conditions.
- Safety demonstrations of fabrication, transportation, storage, and disposal of new LEU fuel.
- Analysis of flux kinetics and distribution in reactor cores with LEU fuel.
- Thermohydraulic analysis.
- Safety analysis, including certification of computer codes; justification of safe operation limits and conditions; accident initiators; and modification of Safety Analysis Reports, plans of personnel and public protection, quality assurance programs, and operational procedures.
- Modification of research nuclear installation designs.

CHALLENGES POSED BY REACTORS THAT CANNOT BE CONVERTED

Two presentations on the challenges posed by research reactors that cannot be converted were given by Panel 2.2 speakers: Jeffrey Chamberlin (U.S. Department of Energy, National Nuclear Security Administration) provided a U.S. viewpoint (Chamberlin, 2011), and G. Pshakin (Institute for Physics and Power Engineering in Obninsk) provided a Russian viewpoint (Zrodnikov et al., 2011).

U.S. Viewpoint on Challenges

Jeffrey Chamberlin

GTRI is the key program within the U.S. government for implementing the U.S. policy to minimize the civilian use of HEU. GTRI's mission is to reduce and protect vulnerable nuclear and radiological materials located at civilian sites worldwide. Its specific goals are to: (1) *convert* research reactors and isotope production facilities from HEU to LEU; (2) *remove* and dispose of excess nuclear and radiological materials; and (3) *protect* high-priority nuclear and radiological materials from theft and sabotage.

GTRI's Reactor Conversion Program is focused on converting civilian research reactors worldwide to operate on LEU fuel. Its goal is to convert or verify the shutdown of 200 civilian research reactors and HEU facilities by 2020.[19] However, GTRI does not specifically encourage the shutdown of

[19] This deadline slipped to 2022 while this report was being completed because of Fiscal Year 2011 federal budget reductions.

research reactors; such decisions are made by facility operators. A research reactor does not have to be considered to be vulnerable to be a candidate for conversion. GTRI is focused on converting civilian reactors and HEU facilities that use HEU fuel because it provides for permanent threat reduction.

Since the inception of GTRI in 2004, 23 HEU-fueled research reactors have been converted as part of the program, including 7 research reactors in the United States and 16 research reactors in other countries.[20] The most recent conversions were the Kyoto University Research Reactor in Japan (March 2010) and the Rez Reactor in the Czech Republic (April 2011).

As noted in Chapter 1, nearly all U.S. HEU-fueled reactors that can convert with existing LEU fuels have successfully been converted (see also Footnote 3 in this chapter). As noted in previous presentations, there are six HEU-fueled U.S. research reactors (ATR and its critical assembly, HFIR, MITR, MURR, and NBSR) that cannot be converted until a new LEU fuel is developed. Additionally, in December 2010, DOE and Rosatom signed an Implementing Agreement to perform feasibility studies for the possible conversion of six HEU-fueled research reactors in the Russian Federation.

The reduction of HEU use in civilian applications is supported at the highest levels in the U.S. and Russian governments. In a joint statement issued on July 6, 2009, Russian Federation President Dmitry Medvedev and U.S. President Barack Obama issued a joint statement expressing their strong support for HEU minimization:

> We declare an intent to broaden and deepen long-term cooperation to further increase the level of security of nuclear facilities around the world, including through minimization of the use of highly enriched uranium in civilian applications and through consolidation and conversion of nuclear materials.

This call for minimization was echoed in UN Security Council Joint Resolution 1887, which was issued in September 2009, and in the April 2010 Nuclear Security Summit.

GTRI works in cooperation with reactor owners/operators to convert reactors to LEU fuel. This cooperation involves:

- Performance of feasibility studies to determine if reactors can be converted and still achieve their missions without major changes in reactor structures or equipment.

[20] In Chapter 1, it was noted that 35 conversions or shutdowns of HEU-fueled reactors have occurred since 2004. This larger number includes 10 reactors that were shut down and 2 reactors that were converted to LEU under domestic programs rather than GTRI.

- Ensuring that required fuel assembly criteria for LEU conversion are satisfied; LEU fuel provides a similar service lifetime as the HEU fuel; there is no significant penalty in reactor performance; and safety criteria are satisfied.
- Development of a schedule for conversion based on operational requirements, capabilities, and regulatory processes.
- Demonstrating that conversion and subsequent reactor operations can be accomplished safely.
- Determining, to the extent possible, that overall costs associated with conversion do not significantly increase the annual operating expenditures for reactor owners/operators.
- Obtaining/verifying that agreements and authorities are in place to proceed with conversion.

GTRI's starting assumption for reactor conversions is that "anything is possible." The experience gained from previous conversions demonstrates that there are many ways to overcome technical barriers. Indeed, many of the recent successful conversions of U.S. reactors were not thought to be possible 20-30 years ago.

Although GTRI policy is to take all reasonable steps to convert facilities and reduce the use of HEU, there may be some facilities that are not feasible to convert. For example, a feasibility study for a particular reactor might indicate that conversion is not feasible because LEU fuel assembly criteria are not satisfied and a unique fuel development effort is not technically or economically feasible. This might be the case for fast reactors, fast critical assemblies, or HEU reactors with very small core volumes.

In such cases, there are four options for addressing HEU minimization at such facilities:

- Option 1: Assess the possibility of changing the facility mission such that it can be accomplished with LEU fuel. However, GTRI does not advocate a change of reactor mission for the sole purpose of converting.
- Option 2: Reduce HEU enrichments. This may be technically feasible in some cases where LEU conversion is not. Note, however, that reduced enrichments above 20 percent are not considered HEU minimization under international norms or GTRI policy.
- Option 3: Shut down the facility or consolidate it with similar facilities if it is underutilized.
- Option 4: If no other options exist for the facility other than to operate with HEU, remove all excess HEU and enhance physical protection measures to achieve threat reduction.

GTRI considers each of these options to be "last resort" and does not endorse them as a matter of policy. These options must be considered on a case-by-case basis by the facility and the host government.

Russian Viewpoint on Challenges

G. Pshakin

The BFS-1 and BFS-2 critical assemblies[21] at the Institute for Physics and Power Engineering in Obninsk (Figure 2-11) provide a good example of reactors that cannot be converted to LEU fuel. These reactors, which are fueled with HEU and plutonium, were constructed in the late 1950s and early 1960s as part of the Soviet Union's fast breeder program for nuclear energy development. Although these assemblies cannot be used for designing commercial-scale fast breeder reactors, they are useful for simulating fast breeder reactor cores, for fuel cycle research, and for studying the transmutation of minor actinides. This fuel used in these assemblies is not self-protecting[22] and therefore poses special security concerns.

Converting these facilities to LEU fuel cannot be accomplished without sacrificing the current mission. Moreover, even if the uranium enrichment of the fuel could be reduced, plutonium would still be required to simulate the cores of fast breeder reactors.

There are two options for addressing the security concerns associated with these facilities: (1) shut down the facility and remove all nuclear materials; or (2) organize a state-of-the-art materials protection, control and accounting (MPC&A) system and enhance the culture of personnel through proper training, motivation, and support. The second option is obviously preferable.

The facility has cooperated with the United States to develop an MPC&A system. It includes a non-destructive analytical system based on high-resolution germanium detectors for isotopic measurement of accounted items; neutron coincident counters for nuclear material mass measurements; and specially designed access and monitoring systems. This program has to protect more than 100,000 HEU and plutonium discs that are used to model the cores of fast breeder reactors.

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[21] As noted in Chapter 1, a critical assembly contains sufficient fissionable and moderator material to sustain a fission chain reaction at a low (close to zero) level. It is designed so that fissionable and moderator materials can be easily rearranged in various geometries to mock up different reactor designs.

[22] As noted in Chapter 1, a material is considered to be "self-protecting" if it produces a dose rate greater than 100 rad per hour at 1 meter in air. These high levels create substantial radiological barriers to illicit use.

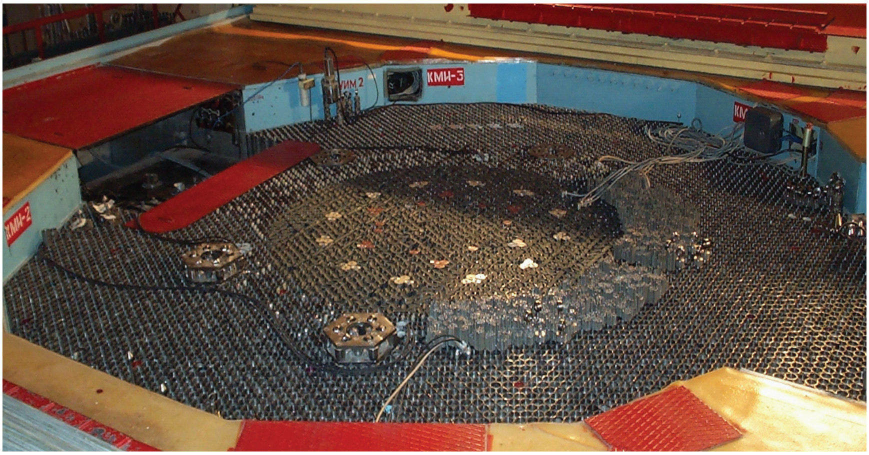


FIGURE 2-11 Photograph of a BFS critical assembly (BFS-1). SOURCE: Zrodnikov et al. (2011).

DISCUSSION

Time was set aside during this session for free discussion among symposium participants. Some of the key comments from that discussion are presented in this section.

- **Research reactors will continue to be an essential tool for many applications.** B. Myasoedov commented that he expected the role of research reactors to grow in the future to support the development of more complex reactor designs for nuclear power plants, including those based on fast reactor designs; for radiopharmaceutical production; and for analytical methods (such a neutron activation analysis) to support safety monitoring and control. He suggested that Russia and the United States should agree to work together and with third-party countries to design a standardized research reactor that could be produced on an industrial basis. This would eliminate the need to design individual, customized cores and fuel elements.
- **Past experience suggests that successful conversion solutions can be found for most reactors.** Jim Snelgrove commented that in view of the success that has occurred in converting reactors in the United States and some other countries, a key take-away message from this symposium should be that it is possible to find conversion solutions if one works hard enough to uncover them. Yu.S. Cherepnin added that some of the presentations in this session documented how enrichment levels could be reduced without degrading reactor performance. These examples should be publicized. H.-J. Roegler commented that conversion can result in improvements to reactors.

- **Current work under way in Russia on monolithic fuel development could pave the way for conversion of many Russian research reactors.** Jim Matos commented that the densities of the LEU dispersion fuels described in the Russian presentations are too low to be used in converting many Russian reactors. Jim Snelgrove noted that monolithic pin-type LEU fuel is also being tested in Russia. This fuel is a potential replacement for the tube-type fuel that is now being used in Russian research reactors. The recent agreement between DOE and Rosatom to assess the feasibility of converting six Russian research reactors could play an important role in assessing the potential utility of this LEU fuel.
- **There may be some research reactors that cannot be converted.** V. Ivanov noted that there may be some reactors with unique purposes that cannot be converted. For example, the multipurpose fast breeder reactor to be built in Dimitrovgrad will be fueled with HEU and plutonium. The concept of reducing risk by eliminating HEU does not make sense for this reactor because the HEU is used alongside plutonium. This is also true for critical assemblies. He also noted that the concept of "unique mission" has not yet been defined in Russia, and he suggested that there should be a limited list of parameters that could be applied to determine uniqueness. N.V. Arkhangelsky reminded symposium participants that it was recognized from the very beginning of the RERTR program that there are a number of research reactors that would not lend themselves to conversion, including fast breeders.
- **Reactor ageing is a potential complication for conversion, but it can be managed.** V. Ivanov noted that unless national regulatory requirements dictate conversion, the decision to convert, upgrade, or shut down a reactor will be made by the operator/owner. The owner/operator must determine whether it makes sense to convert the reactor if the remaining lifetime is negligible. H.-J. Roegler commented that, in his experience, research reactor ageing problems can be cured, although in some cases it can take time. A.N. Chebeskov commented that different reactor facilities may have access to different resources to manage ageing. Having a set of best practices to manage ageing could be a topic for international cooperation.
- **Reactor customers (users) are an important part of the conversion process.** V. Ivanov commented that conversion work needs to be transparent to customers, not just designers and research reactor specialists. He suggested that it would make sense for the international community, including the customers of research reactors, to cooperate more closely on conversion.
- **There may be economic advantages to conversion.** Richard Meserve noted that conversion may have economic advantages that were not discussed by any of the symposium presenters. In particular, LEU costs could be lower, depending on how that material is priced, and costs for securing

LEU fuel should be much lower than for HEU fuel. Jordi Roglans commented that transportation costs, especially international transportation costs, will be lower for LEU fuel because HEU is often transported by the military.

- **A worldwide ethic on conversion should be developed.** Yu.S. Cherepnin suggested that the world community should develop a new ethic against operating reactors with HEU. Strong signals should be sent to operators of HEU reactors that they need to convert, and funding should be demanded from governments to support conversion.
- **Working together, the Russian Federation and the United States have played and will continue to play important global roles in research reactor conversion.** N. Laverov noted that the Russian Federation has decommissioned 200 nuclear submarines and, working with the United States, has returned 100,000 tonnes of natural uranium and 500 tonnes of HEU from foreign countries. The recent agreement between DOE and Rosatom to assess the feasibility of converting six Russian research reactors is an important step for eliminating HEU use in Russian research reactors. It is important that the Russian Federation and the United States serve as an example to countries by reducing the enrichments of their research reactors to lower levels.

REFERENCES

Adams, A. 2011. Regulatory Challenges and Solutions: High-Enriched to Low-Enriched Uranium Fuel Conversion. Presentation to the Research Reactor Conversion Symposium. June 9.

Bezzubtsev, V. 2011. Regulating Safe Operation of Russian Research Reactors. Presentation to the Research Reactor Conversion Symposium. June 9.

Chamberlin, J. 2011. Challenges Posed by Research Reactors That Cannot be Converted (U.S. Viewpoint). Presentation to the Research Reactor Conversion Symposium. June 9.

Cherepnin, Yu. 2011. Experience of Resolving the Problems Arising in Conversion of Research Reactors. Presentation to the Research Reactor Conversion Symposium. June 9.

IAEA [International Atomic Energy Agency]. 1995. Management of Research Reactor Ageing. IAEA-TECDOC-792. Vienna: International Atomic Energy Agency.

Roegler, H. 2011. Obsolescence & Ageing: Findings from the IAEA Initiative on Research Reactor Ageing and Ageing Management. Presentation to the Research Reactor Conversion Symposium. June 9.

Roglans, J. 2011. Maintaining Performance and Missions (U.S. viewpoint). Presentation to the Research Reactor Conversion Symposium. June 8.

Ryazantsev, E. 2011. Ageing and Obsolescence of Research Reactors. Presentation to the Research Reactor Conversion Symposium. June 9.

Stevens, J. 2011. Core Modifications to Address Technical Challenges of Conversion. Presentation to the Research Reactor Conversion Symposium. June 8.

Svyatkin, M.N., Izhutov, A.L., and Petelin, A.L. 2011. Use of Research Reactors of Scientific Centre RIAR. Presentation to the Research Reactor Conversion Symposium. June 8.

Tetiyakov, I.T. 2011. Modification of the Reactor Cores. Presentation to the Research Reactor Conversion Symposium. June 8.

Wachs, D. 2011. Research and Test Reactor Fuel System Development. Presentation to the Research Reactor Conversion Symposium. June 8.

Zrodnikov, A., Pshakin, G., and Matveenko, I. 2011. Research Reactors That Cannot be Converted. Presentation to the Research Reactor Conversion Symposium. June 9.

3

Reactor Conversion Case Studies

Session 3 of the symposium focused on technical challenges associated with conversion of specific U.S. and Russian reactors. Eight case studies of individual research reactors' potential for conversion—three U.S. reactors and five Russian reactors—were presented in this session. These presentations and some key thoughts from the participant discussions are summarized in this chapter.

As was discussed in Chapter 2, there are several analyses that need to be performed to enable conversion of a research reactor from HEU fuel to LEU fuel:

1. Neutronics analysis[1] is performed to determine neutron fluxes in various regions of the new LEU core, reactivity effects, including burnup effects, and various reactor safety parameters.
2. Thermal and hydraulic analysis is performed to ensure that the new LEU core can be adequately cooled during normal and accident conditions.
3. Accident analysis is performed to analyze the potential for fission product release under hypothetical accident conditions.

Because of the uniqueness of many research reactors, conversion studies

[1] "Neutronics" refers to an analysis of the neutron flux throughout the core, which entails analysis of fission and neutron capture events caused by absorption of neutrons by the reactor core, scattering of the neutrons, and losses of the neutrons from the reactor.

need to be carried out for each individual reactor, and the challenges encountered can be very different for different reactors.

U.S. REACTOR CONVERSION CASE STUDIES

The following three case studies of U.S. research reactor conversions are summarized in this chapter:

- Paul Wilson (University of Wisconsin) reported on the successful conversion of the University of Wisconsin research reactor (UWNR) (Wilson, 2011).
- Thomas Newton (Massachusetts Institute of Technology) reported on the status of conversion plans for the Massachusetts Institute of Technology Reactor (MITR) (Newton, 2011).
- David Cook (Oak Ridge National Laboratory; ORNL) reported on the status of conversion plans for the High Flux Isotope Reactor (HFIR) (Cook, 2011).

These reactors are quite different: MITR is planned to be the first research reactor to convert to using high-density uranium-molybdenum (UMo) monolithic LEU fuel and is considered to be a relatively straightforward conversion for a high-performance reactor. In contrast, HFIR is planned to be the last U.S. domestic reactor to convert to LEU fuel and is likely to pose far greater conversion challenges. Current approaches and plans for converting these reactors are described in the following sections.

University of Wisconsin Nuclear Reactor

Paul Wilson

UWNR is a 1 megawatt (MW) TRIGA pool reactor (see Chapter 1) housed on the University of Wisconsin campus in Madison, Wisconsin. Its primary mission is the training of undergraduate and graduate nuclear engineering students; however, it is also used to perform research, including irradiation for neutron activation analysis.

The reactor first went critical as a 10 kilowatt (kW) LEU-fueled reactor in 1961 and, following several power upgrades, was converted to HEU fuel in 1979. It was converted back to LEU fuel 30 years later, successfully achieving criticality in 2009. At that time, UWNR was converted from using 70 percent enriched TRIGA-FLIP (Fuel Life Improvement Program) fuel to TRIGA LEU 30/20 (30 percent uranium by weight, 20 percent enriched) fuel. The new LEU fuel is, like the previous FLIP fuel, a standard TRIGA-type fuel element containing erbium-doped uranium-zirconium hydride ($UZrH_x$-Er) fuel (see Chapter 2).

Neutronics Analysis

A number of key neutronics analyses were performed for a range of reactor core states, including the beginning-of-life, middle-of-life, and end-of-life states. These studies included analyses of:

- Power distributions (for use in the thermal/hydraulic analyses), including (1) total fuel assembly power and core power distributions; and (2) axial and radial power distributions in the maximum power fuel assembly;
- Shutdown margins as a function of fuel burnup; and
- Key reactivity parameters, including (1) delayed neutron fraction[2]; (2) prompt neutron lifetime[3]; (3) control element worth[4]; and (4) prompt temperature coefficient.[5]

The neutronics of the reactor core were modeled using Los Alamos National Laboratory's Monte Carlo n-Particle code, version 5 (MCNP5) with the core nuclear reaction database ENDF/B-VII maintained by the National Nuclear Data Center. In addition, Argonne National Laboratory's REBUS codes for analysis of fuel cycles were used for the burnup analysis. Finally, some confirmatory analysis was performed using the HELIOS two-dimensional generalized-geometry lattice physics transport code.[6]

Several challenges were associated with performing these analyses at Wisconsin. First, sufficient information was not available on the operational history of the HEU core to be able to calculate fuel composition for use in benchmarking the model. As a substitute, analysts worked backwards to estimate the composition of the fuel using the current critical conditions for the core. This does not provide a benchmark but gives some confidence in the validity of the model. Second, large computing resources were required

[2] Delayed fission neutrons are neutrons emitted spontaneously from decay of a fission product from a prior fission event, whereas prompt neutrons are neutrons emitted from the fission process directly. The delayed neutron fraction is the ratio of the mean number of delayed fission neutrons per fission to the mean total number of neutrons per fission (prompt plus delayed).

[3] The prompt neutron lifetime is the average time between the emission of neutrons and either their absorption in or their escape from the system.

[4] The control element worth is the negative reactivity change caused by inserting a control element into the reactor. UWNR has five separate control elements.

[5] The prompt temperature coefficient is the change in reactivity per degree change in fuel temperature.

[6] This confirmatory analysis was a two-dimensional deterministic analysis coupled with a one-dimensional diffusion approximation.

for some of the analyses, beyond what was easily available at the university. Finally, the university had only a modest existing capacity for performing such reactor analyses. This capacity had to be built up for the analyses to be carried through successfully.

The major difficulty associated with conversion was related to system reactivity. The like-for-like replacement of HEU-FLIP fuel with LEU 30/20 fuel increased the reactivity of the core. The modeled core with LEU fuel could not be shut down even with all control elements fully inserted. To reduce system reactivity and meet shutdown margin requirements, the core design was changed from a 23 fuel assembly/10 reflector configuration (in which the assemblies are arranged in an "H" pattern) to a 21 fuel assembly/14 reflector configuration (in which the assemblies are arranged in an "X" pattern) (see Figure 3-1). However, the reduction in the number of fuel assemblies resulted in a slightly reduced core lifetime following conversion.[7]

Thermal/hydraulic Analysis

The thermal/hydraulic analysis focused on the behavior of the high-power channel at steady state, low-power pulse, and high-power pulse.[8] The analysis yielded estimates of:

- Coolant flow rate; and
- Temperatures at the fuel centerline, the axial/radial temperature profile, and the minimum departure from nucleate boiling ratio (DNBR).[9]

The U.S. Nuclear Regulatory Commission's (USNRC's) RELAP5/Mod3.3 code was used to perform the thermal/hydraulic analysis. A single channel analysis was performed with the highest-power channel, involving 20 axial nodes (15 in the fuel meat) and 27 radial nodes (21 in the fuel meat). To model the reactor pulsing mode, a two-channel model was used, with the two channels defined as (1) the hot channel and (2) the rest of the core. For the pulsing analysis, a RELAP point reactor kinetics model was used, with temperature coefficients obtained from the MCNP5 analysis that was described previously. Finally, a two-channel model was used to model

[7] When preparing for conversion, the University of Wisconsin was provided with two additional LEU fuel assemblies so that the reactivity could be boosted if required.

[8] This analysis assumed no cross-flow, i.e., no exchange of coolant with adjacent fuel or reflector assemblies.

[9] DNBR is the ratio of the heat flux needed to cause departure from nucleate boiling to the actual local heat flux of a fuel rod. Departure from nucleate boiling is the point at which the heat transfer from a fuel rod rapidly decreases because of the insulating effect of a steam blanket that forms on the rod surface when the temperature continues to increase (USNRC, 2011).

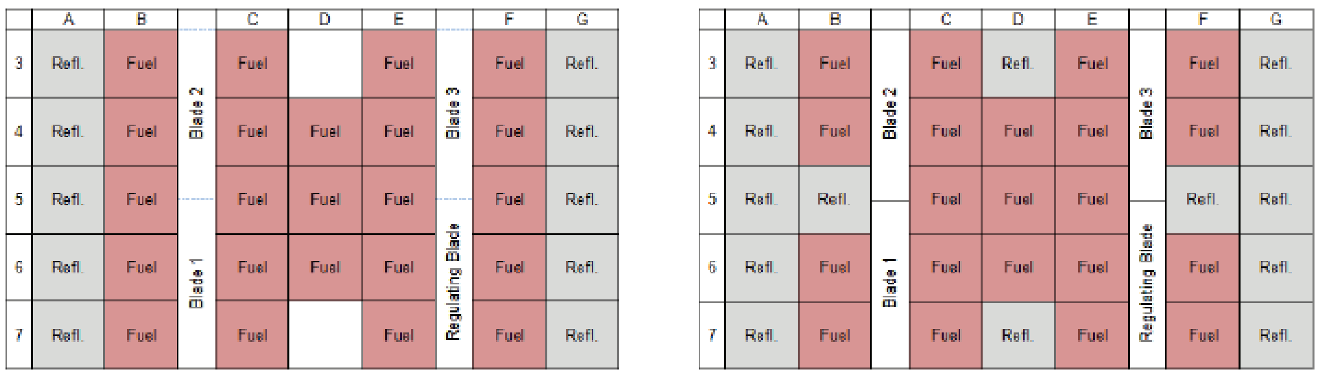


Figure 1, Core Map: HEU on Left, LEU on Right

FIGURE 3-1 Core map of the University of Wisconsin reactor before (left) and after (right) conversion from HEU to LEU fuel. Fuel elements are shown in red, and beryllium reflector elements are shown in grey. SOURCE: Austin (2010).

a loss-of-coolant accident using three phases to represent different water levels remaining in the pool and assuming axial conduction in the fuel.

The thermal/hydraulic analysis faced four major challenges. First, the analysis was very sensitive to gap thickness, so additional sensitivity analyses needed to be carried out. Second, a discrepancy was found between the two critical heat flux correlations used to analyze the natural circulation mode.[10] Third, there was some uncertainty in the natural convection heat transfer models. Finally, it was challenging to determine appropriate air-cooled temperature safety limits for the new LEU 30/20 fuel type.

The overall outcome of the thermal/hydraulic analysis was encouraging. The average fuel assembly power increase associated with the use of fewer assemblies caused small changes to appear in the models of the steady-state operation of the reactor following conversion. However, the definition of the fuel temperature-limiting safety setting[11] was updated, ensuring that the fuel temperatures remained below the set points. The temperatures were calculated to be within technical specifications for the reactor: The maximum fuel temperature under pulsing operation at 1 kW and at 1.3 MW was calculated to be within maximum allowable temperatures, and the loss-of-coolant fuel temperatures were less than 700 °C.

Accident Analysis

The potential for a fission product release under accident conditions was analyzed for a maximum hypothetical accident consisting of cladding failure in the high-power fuel assembly (25 kW) after continuous full-power operation. The accident analysis was carried out with and without an intact water pool and operating ventilation system.[12] Reactivity insertion was also analyzed. Additionally, a loss-of-cooling accident was analyzed to determine the fuel temperature and radiation dose from the exposed core.

The accident analysis used Oak Ridge National Laboratory's ORIGEN code to calculate the fission product inventory in case of accident. An analysis of release fractions used a Gaussian plume model, and radiation doses were calculated using MCNP5.

[10] This is the relationship between the conditions in a heated channel and the heat flux at which the heat transfer becomes impaired as a result of the transition from nucleate boiling to film boiling. These conditions may include the mass flow rate, channel geometry, and thermal properties of the fluid (e.g., the density of liquid and vapor, heat of vaporization, specific heat). The critical heat flux correlations used in this analysis were the Groeneveld 2006 look up tables and the Bernath correlation. More information on these correlations can be found in Vitiello (2008).

[11] The fuel temperature-limiting safety setting is the temperature below which the fuel is required to be maintained to prevent fuel element failure.

[12] A "near maximum hypothetical accident" maintains an intact pool and ventilation.

The accident analysis had an overall positive outcome. LEU conversion required no changes in response to any accident. The reactor remained within regulatory limits under all variations to the maximum hypothetical accident.

This analysis had another positive benefit: The university's capabilities to analyze core accidents increased significantly; previously, only simple methods and models had been used to analyze such accidents. As a result, a more detailed understanding of the potential radiation dose was gained, including the time-dependent behavior and the spatial distribution of dose.

Results of Conversion and Future Plans

Although the overall conversion experience was positive, the converted reactor core behaved somewhat differently than the calculated core. In particular, the converted reactor was substantially less reactive than was calculated. The reason for this difference is still not fully understood. In the near term, the UWNR staff is pursuing a plan to shuffle the core and reduce the number of reflectors. This will cause a slight reduction of the neutron flux in some positions; however, the reshuffling should increase the flux in other positions. This reshuffling would take the core from the "X" configuration of 21 fuel assemblies and 14 reflectors to a "+" configuration of 21 fuel assemblies and 6 reflectors (see Figure 3-2).

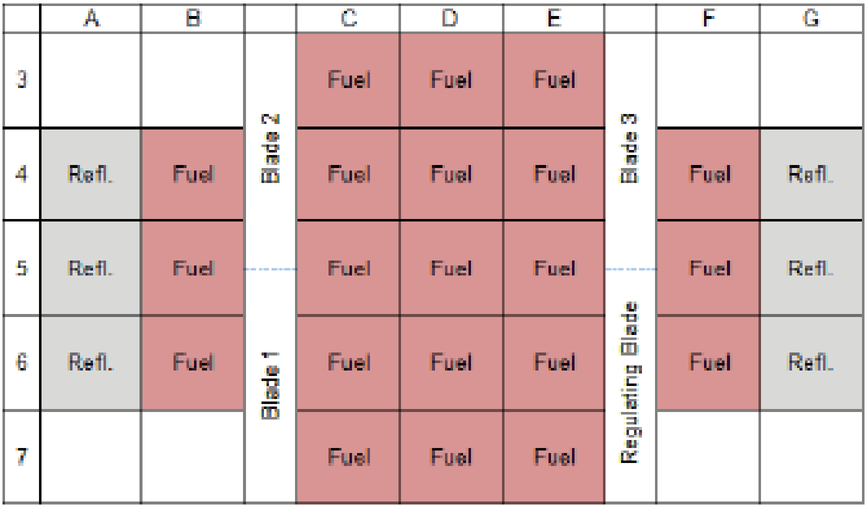
Overall, the conversion-related work enabled a widespread upgrade in UWNR staff's analysis capabilities, and it has also provided opportunities for further analysis. For example, experimental research is ongoing to better understand natural circulation heat transfer in TRIGA-relevant conditions, and the fresh LEU core provides a wide variety of benchmark data for continuing to improve analytical capabilities.

Massachusetts Institute of Technology Reactor

Thomas Newton

MITR is a 6 MW research reactor that is currently operating using aluminide (UAl_x) dispersion fuel that is 93 percent enriched in uranium-235. Its primary mission is research, although it is also used for student training, particularly for nuclear engineers. The research performed at MITR focuses primarily on fast neutron experiments, including irradiation testing of cladding for next-generation light-water reactors and advanced nuclear fuel experiments.

The reactor core is highly compact and has a hexagonal geometry with 27 fuel assembly positions. Twenty-four of these positions contain fuel; the remaining three positions are reserved for experiments (see Figure 3-3). The



d. N21-R6 (final design)

FIGURE 3-2 Planned future core map for the UWNR reactor. Fuel elements are shown in red, beryllium reflector elements are shown in grey, and white boxes are empty positions. SOURCE: Austin (2010).

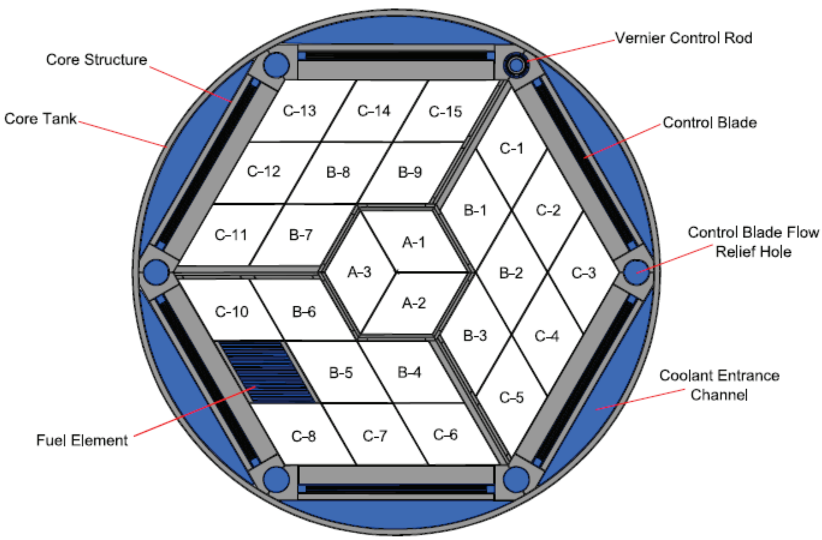


FIGURE 3-3 Overhead view of the MITR reactor core. The 27 fuel assembly positions are labeled A-1 through C-15. Twenty four of these positions hold fuel. A fuel element is shown in dark blue in position C-9. SOURCE: Newton (2011).

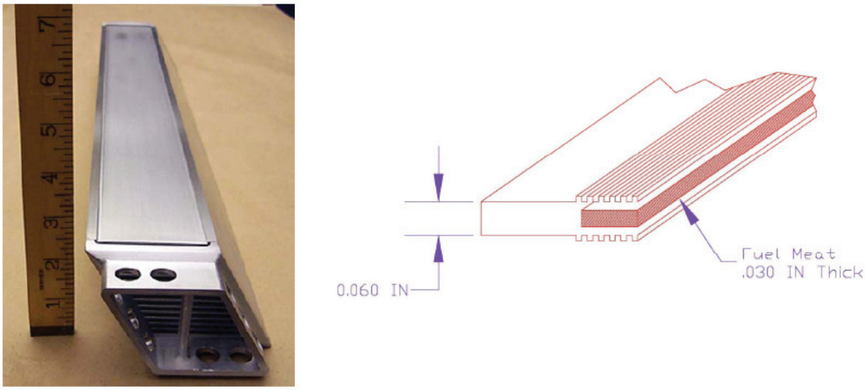


FIGURE 3-4 MITR's unique finned fuel elements. A complete fuel assembly consists of 15 stacked fuel plates in an aluminum shell. The fins can be seen on the individual fuel plate on the right. The fuel meat is 93 percent enriched UAl_x dispersed in aluminum. SOURCE: Newton (2011).

aluminum-clad fuel plates (15 per assembly) are designed with longitudinal fins to increase the heat transfer area (see Figure 3-4). The thermal and fast neutron fluxes in the core region are approximately 3×10^{13} and 1×10^{14} neutrons per square centimeter per second (n/cm^2-s), respectively. The core is light-water cooled and moderated, with six control blades located around the periphery.

MITR has not yet been converted to LEU fuel because an appropriate fuel has not yet completed development and qualification. In fact, MITR's unique fuel assembly design and highly compact core complicate conversion. Currently available LEU fuels were judged not to be appropriate for use in MITR because they would not allow criticality to be maintained and would also require a complete redesign of the core. However, the use of high-density UMo monolithic LEU fuel (discussed in Chapter 2) is likely to allow conversion of the reactor core to LEU. It is the reference fuel used in the conversion analyses. This fuel is 19.75 percent enriched in uranium-235 and has a density of 15.5 grams of uranium per cubic centimeter (gU/cm^3).

Neutronic and Thermal/hydraulic Analyses

A major challenge for MITR is to convert to LEU while still meeting the performance requirements for experiments in the reactor. Meeting these requirements will entail optimizing the fuel design to maximize heat transfer and neutron flux. In particular, the neutron flux optimization is focused on maintaining HEU-equivalent fast neutron fluxes in in-core materials experiments and thermal neutron fluxes in out-of-core experiments. To prepare for conversion, the existing neutronics and thermal/hydraulics models for the MITR core were improved in several ways.

Several major improvements were made to the neutronics codes. The primary change enabled more accurate burnup modeling and benchmarking. The improvements entailed an extensive review of the model's core structure and dimensions as well as an update of the cross-section libraries, homogenized volume fractions, and discrete structures. Two initial HEU cores were modeled, and the results compared favorably with measurements.

The neutronics codes were improved in other ways as well. A graphical user interface was added, as was the capability to model HEU, LEU, and mixed core geometries. In addition, it is now possible to model all fuel movements, including flipping, rotating, and fuel storage. The models are also now able to track and plot the mass of isotopes as well as the power distribution in the core.

The improved burnup modeling has shown good results. Twelve recent cores have been modeled; the results were in good agreement with measured beginning- and end-of-cycle control blade positions. There was also good agreement among different models.

In addition to the neutronics modeling, thermal/hydraulics models were also updated and modified. Specifically, the models were modified to include the fuel's longitudinal fins for the steady-state and loss-of-flow analyses. Initial results have shown that the LEU design core has a higher margin to onset of nucleate boiling and a lower peak cladding temperature with loss of flow.

The results of the neutronic and thermal-hydraulic analyses have been used to design an LEU fuel for this reactor. The LEU fuel assembly will contain more plates and use a thinner fuel meat (0.51 mm for UMo LEU fuel versus 0.76 mm for HEU fuel) and cladding (0.28 mm for UMo LEU fuel versus 0.35 mm for HEU fuel). Fuel developers have informed MITR staff that fuel and cladding of this thickness is feasible to manufacture.

As was noted in Chapter 2, the reactor's operational power will need to be increased from 6 MW to 7 MW to counter the expected loss of performance after conversion. This will result in an increase in the cycle length from 40-50 days for the HEU fuel to 60-70 days for the LEU fuel.

Safety Analysis

Prior to beginning the conversion analysis, some safety analysis parameters at MITR were not well known, particularly for the finned cladding.

To complete the safety analysis, further information is being gathered on the following three topics:

- Finned channel hydraulic pressure drop. A flow experiment has been built, measurements have been made, and a finned channel correlation[13] describing the relationship between the pressure drop and the flow rate has been developed.
- Adequacy of the onset of nucleate boiling (ONB) correlation for finned channels.[14] For this purpose, the MITR boiling flow loop is being constructed to measure ONB for the LEU channel geometry and validate the Bergles-Rohsenow ONB correlation[15] for finned channels. This facility will be operational later this year.
- Oxide distribution in the finned cladding. The current burnup limit of 1.7×10^{21} fissions per cubic centimeter is based on an even 50 micrometer-thick aluminum oxide distribution on the cladding. However, particularly within the finned region, the actual oxide distribution is unknown. MITR is currently using an eddy current probe for fin-tip measurements of oxide thickness. This thickness will then be compared with the operational history of the fuel element. Finally, a selected fuel element will be shipped to Idaho National Laboratory for evaluation of the oxide distribution in the areas between the fins.

Other Potential Challenges

Aside from the challenges discussed above, there are several other potential challenges that MITR will face during the transition to an LEU-fueled core. First, MITR is likely to be the first reactor to convert using UMo monolithic LEU fuel. MITR staff is presently working to better understand how best to introduce this first-of-a-kind fuel into the reactor. The current plan is to gradually introduce LEU fuel into the HEU core. To evaluate this plan, a mixed-core analysis will be carried out prior to conversion. Two challenges are foreseen in the mixed-core transition: Power peaking is generally higher in new LEU elements, and steady-state HEU and LEU margins to ONB decrease with an increasing number of LEU fuel elements in the mixed

[13] This is the relationship between the pressure drop and the conditions in the flow channel such as mass flow rate.

[14] This is the relationship between the conditions in the flow channel at the time when there is net vapor generation. It is a relationship between parameters such as mass flow rate, heat flux, and thermal properties of the liquid (e.g., thermal conductivity, specific heat, saturation temperature).

[15] The Bergles-Rohsenow correlation is commonly used for prediction of ONB in narrow rectangular coolant channels and relates cladding local heat flow at ONB to local heat flux and pressure.

core. MITR staff project that partially burned HEU elements may need to be kept in reserve as the reactor is transitioned to the full LEU core.

Second, the heating from gamma ray absorption outside the fuel is significantly less for LEU than for HEU, resulting in a lower heat load on the deuterium reflector and shield system. However, some in-core materials experiments rely on gamma heat for temperature control and may need to be redesigned.

Finally, it is also possible that mechanical stresses from the heavier loading of the denser LEU fuel will necessitate some redesigning of the facility. However, current analyses indicate that heavier loading is not likely to pose a problem.

Future Plans and Projected Results of Conversion

Once the UMo monolithic LEU fuel is qualified, MITR has a number of future plans to prepare for conversion that draw on the technical work described in the previous sections. First, a preliminary safety analysis report will be prepared and approved prior to conversion. Second, because the fuel fabrication requirements are unique to the MITR reactor, fuel manufacturing tolerances will need to be determined. Finally, all cores up to the last full HEU core will need to be analyzed using the newly upgraded neutronics model; this model will also be used for fuel management.

MITR staff is using the requirement to convert to LEU fuel to improve its analysis capabilities and obtain a greatly improved understanding of the reactor. These improved capabilities have resulted in an optimized LEU fuel design for the MITR reactor.

Oak Ridge: High Flux Isotope Reactor

David Cook

HFIR currently operates at 85 MW—following a derating from 100 MW in the early 1990s because of embrittlement of the reactor pressure vessel—using a U_3O_8-Al dispersion fuel that is 93 percent enriched in uranium-235. HFIR's original primary mission was the production of transuranic isotopes. With the addition of the cold neutron source in 2007, the facility began hosting world-class cold and thermal neutron scattering research. The facility also meets critical needs for materials irradiation and the performance of neutron activation studies.

The reactor core is cooled and moderated by pressurized light water and is very small (50.8 cm active fuel height and 43.5 cm diameter). The HFIR core contains one inner and one outer cylindrical fuel element (see Figure 3-5). At the center of the inner fuel element is a 13 cm-diameter

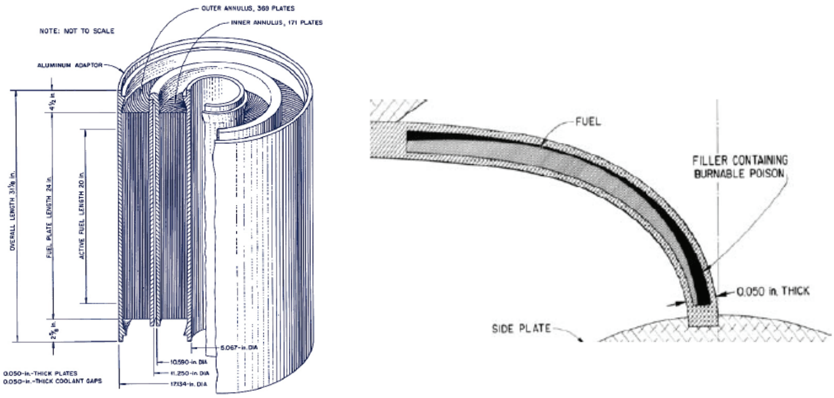


FIGURE 3-5 HFIR's core (left) and fuel plates (right). HFIR's core consists of two fuel elements, concentrically arranged into an inner annulus and an outer annulus, each comprised of many individual fuel plates, shown on the left. The involute shape of the fuel element can be seen, as can the thinning of the fuel meat at either end of the element (with the thickening of the burnable poison). SOURCE: Cook (2011). Image courtesy of Oak Ridge National Laboratory.

hole (the "flux trap") that contains vertical experimental targets for isotope production (californium). Outside the fuel elements is a beryllium reflector that contains additional experimental positions.

The inner fuel element contains 171 fuel plates, and the outer fuel element contains 369 fuel plates. The fuel plates, are involute-shaped, and the fuel meat is radially contoured along the involute—the fuel distribution is peaked in the center and thinner on the edges to suppress power peaking (see Figure 3-6). The inner element plates also contain a burnable poison (boron-10). These fuel plates are complex to manufacture because of the plate form and the welds at the sideplates.

HFIR has a number of unique design features that complicate conversion; consequently, it will be the most complex—and the last—of the U.S. research reactors to convert from HEU to LEU. Conversion will not occur until it is clear that the reactor's primary operating missions and safety will not be significantly impacted. This includes maintaining the very high fluxes that HFIR is capable of generating, particularly in the flux trap region.

HFIR cannot be converted until an appropriate LEU fuel has completed development and qualification. Like MITR, HFIR's unique fuel assemblies and highly compact core complicate conversion. Also like MITR, the use of high-density UMo monolithic LEU fuel along with additional changes in the reactor design is likely to allow for conversion. The new fuel will be 19.75 percent enriched in uranium-235 and have a density of 15.5 gU/cm^3.

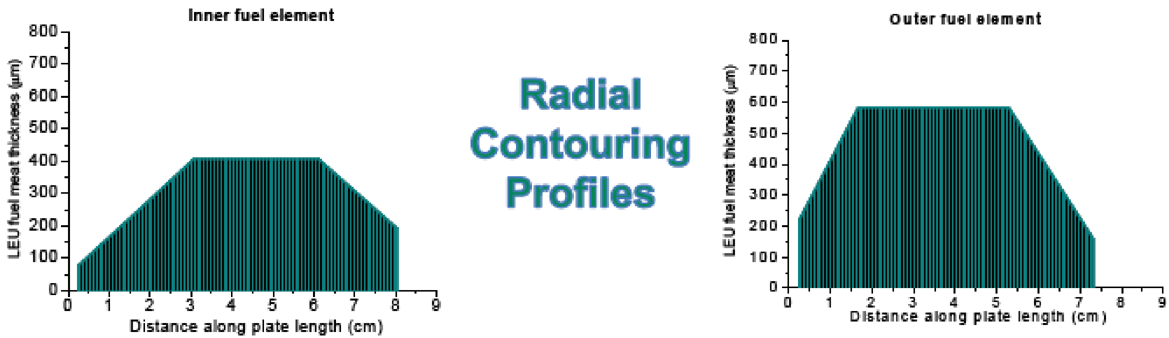


FIGURE 3-6 Radial contouring of fuel plates for reference LEU fuel design. The reference LEU fuel elements are shown in cross-section. The inner fuel element, shown on the left, will be more dramatically asymmetric than the outer fuel element, shown on the right, although both fuel elements are noticeably asymmetric. SOURCE: Cook (2011). Image courtesy of Oak Ridge National Laboratory.

Neutronics and Thermal/hydraulic Analyses

HFIR staff has developed a reference fuel design, but there is significant work remaining to evaluate its safety. HFIR's conversion plan requires maintaining the fuel plates' involute shapes, the overall core geometry, thermal/hydraulic system parameters, and key neutron fluxes, particularly in the flux trap region. To create a "proof of concept" reference LEU fuel design, state-of-the-art HEU-validated neutronics analyses were coupled to the original HFIR thermal design analysis. The current analysis uses the Nuclear Energy Agency's Monte Carlo depletion interface code VESTA (MCNP/ORIGEN) that accounts for the zirconium interlayer between the UMo fuel and cladding (see Figure 2-1 in Chapter 2).

Current calculations indicate that essential neutron fluxes as well as fuel-cycle length can be preserved using UMo monolithic LEU fuel if the reactor power is increased from 85 to 100 MW. The fuel plates will need to be radially contoured (see Figure 3-6) and axially contoured on the lower edge to avoid flux peaking at the edges of the fuel.

Increasing the reactor power to 100 MW will require changes in the thermal/hydraulics safety basis, and new safety limits and protective system setpoints[16] must be derived from a revised thermal analysis. Transient analyses must also be reevaluated as well as fission product release, transport, and consequence analyses.

The return to 100 MW operation will also increase the heat flux from the fuel plates. However, ORNL plans to maintain the current primary coolant inlet temperature, flow rate, and pressure (pressure is constrained to 475 pounds per square inch atmospheric [psia] because of the embrittled vessel). Consequently, the safety limits and associated protection system setpoints will need to be changed, which will require the identification of additional safety margin, either through analysis or changes in fuel design. There are several resources that could be used to find the additional required safety margin: (1) use a modern multidimensional physics analysis to evaluate the safety margin and demonstrate its adequacy; (2) revise the manufacturing uncertainties included in the safety analysis; and (3) revise the approach to the consideration of uncertainties (statistical versus multiplicative).

ORNL plans to begin the revised thermal/hydraulic analysis by performing a modern multidimensional analysis to analyze the safety margin at higher operating power. This analysis will use a COMSOL[17]-based, three-dimensional, detailed multiphysics LEU model to replace the existing HFIR steady-state heat transfer code. At present, ORNL staff is working

[16] At specified setpoints, the reactor will shut down by opening the circuit breakers that supply electrical power to control rods.

[17] COMSOL is a multiphysics engineering software package.

to develop the integrated multiphysics modeling tools and place them into production. The new COMSOL model will require validation against the old (HEU) data, new (LEU) data, and separate effects testing, plus acceptance by the regulator (the U.S. Department of Energy).

LEU Fuel Design and Testing

There are remaining fuel development and fabrication challenges associated with producing the UMo monolithic LEU fuel that will be required to convert HFIR:

- Fuel development is still under way, and the results will affect the final LEU fuel design. A fuel irradiation test series is currently ongoing at Idaho National Laboratory (INL); ORNL expects that this testing (which includes fuel failure testing) will guide safety and other calculations. The analysis of the fission product release from these tests will be most helpful.
- Fuel fabrication methods still need to be developed for producing variable radial and axial fuel thicknesses. In addition, the safety analyses rely on precise manufacturing tolerances, so for HFIR more than other reactors, fuel fabrication will need to be very precise.
- Criticality testing of the fuel will need to be completed. A facility will need to be identified for this purpose.
- Reactor startup testing will need to be conducted both at low power (to validate the analyses) and at full power (to demonstrate fuel performance and the preservation of key mission capabilities).
- Finally, ORNL will need to plan and assess the impacts of initial commissioning as well as the transition from HEU operation to full LEU operation.

Other Challenges

The identified changes in power, nuclear characteristics, and fuel weight will affect the HFIR facility infrastructure. At present, it appears that ORNL will need to: (1) increase the capacity of the cooling tower and the cold source helium refrigerator; (2) modify the reactor instrumentation and control systems to manage the increased heat load; (3) modify the fuel handling tools; and (4) reevaluate the core structural and seismic analyses. In addition, the spent fuel systems will need to be reevaluated, including the design analyses for the pool storage and fuel shipping containers.

RUSSIAN REACTOR CONVERSION CASE STUDIES

Although Russia has successfully converted many foreign research reactors from HEU fuel to LEU fuel, it has not historically had a domestic conversion program comparable to that of the United States. To date, no research reactors in the Russian Federation have been converted from HEU to LEU fuel. However, as noted in Chapter 1, in December 2010 the U.S. and Russian governments agreed to initiate feasibility studies to analyze the conversion potential of the following six Russian research reactors that are currently operating with HEU fuel:[18]

1. MIR.M1 (Research Institute of Atomic Reactors [RIAR], Dimitrovgrad);
2. IR-8 (Kurchatov Institute, Moscow);
3. OR (listed as OP-M in Table 1-2 in Chapter 1) (Kurchatov Institute, Moscow);
4. ARGUS (Kurchatov Institute, Moscow);
5. IRT (Moscow Engineering Physics Institute [MEPhI], Moscow); and
6. IRT-T (Tomsk Polytechnical Institute, Tomsk).

The feasibility studies for these reactors are planned to be completed at the end of 2011.

During the symposium, significant concern was expressed by many members of the Russian delegation regarding the possibility of performance degradation accompanying conversion from HEU to LEU cores. Many members of the U.S. delegation were significantly more optimistic that good design of the replacement LEU core could eliminate concerns about performance degradation. This difference in view may be attributable to the considerably greater U.S. experience with research reactor conversions (see Chapter 2).

The current missions and currently assessed conversion potentials of five of the six reactors listed above were described by Russian presenters during the symposium:

- V.A. Starkov (RIAR) discussed the conversion potential of MIR.M1 (Starkov, 2011).
- V.A. Pavshuk (Kurchatov Institute) discussed the conversion potential of Argus (Pavshuk, 2011).
- V.A. Nasonov (Kurchatov Institute) discussed the conversion potential of IR-8 (Nasonov, 2011).

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[18] Descriptions of some of these reactors were provided in the presentations that are summarized Chapter 2.

- Yu.A. Tzibulnikov (Tomsk Polytechnic Institute) discussed the conversion potential of IRT-T (Tzibulnikov, 2011).
- E.A. Kryuchkov (MEPhI) discussed the conversion potential of IRT (Kryuchkov, 2011).

Because the feasibility studies of these reactors were at an earlier stage of development than the U.S. studies when the symposium was held, less detail is provided in presentation summaries than was given for the U.S. reactor conversions.

MIR.M1

V.A. Starkov

The MIR.M1 reactor is a 100 MW pool-type research reactor located at RIAR in Dimitrovgrad. It has a maximum thermal neutron flux at the experimental positions of 5×10^{14} n/cm^2-s. Its primary mission is to test experimental fuel assemblies and fuel rods under normal, abnormal, and accident conditions.

The core and beryllium reflector blocks are stacked in a hexagonal grid comprising 127 hexagonal blocks 148.5 mm in size, installed at a pitch of 150 mm (see Figure 3-7). Four central rows of beryllium blocks operate as

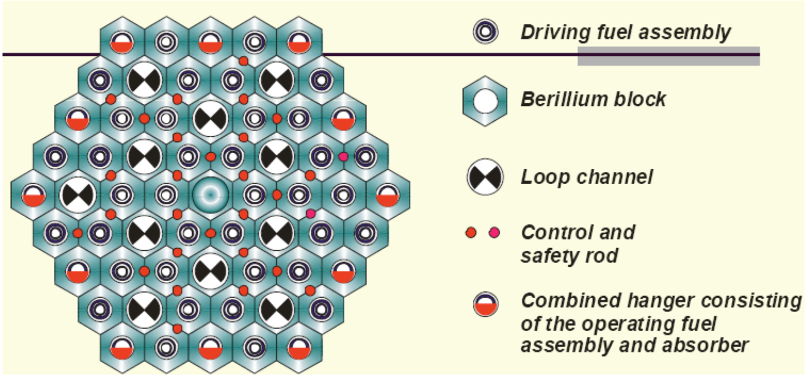

FIGURE 3-7 Diagram of the MIR.M1 reactor core. The core is composed of hexagonal beryllium blocks with channels cut through their centers. Individual fuel assemblies can be seen (silver circles), as can experimental positions (black and white). Each experimental position is surrounded by six fuel assembly channels. SOURCE: Starkov (2011).

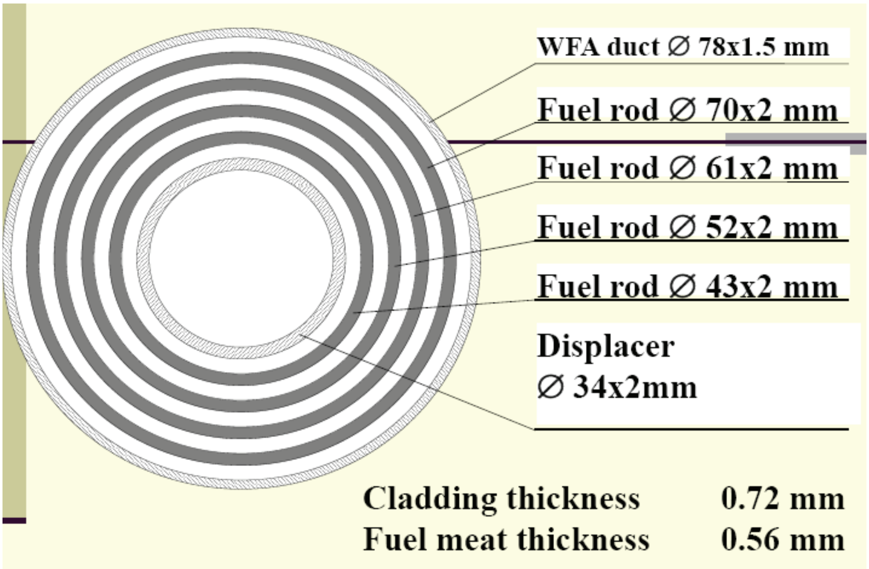

FIGURE 3-8 Diagram of a MIR.M1 fuel assembly. Each fuel plate is cylindrical and has a fuel meat thickness of 0.56 mm and a cladding thickness of 0.72 mm. SOURCE: Starkov (2011).

a moderator, and two external rows of beryllium blocks act as a neutron reflector. The core also contains 11 loop channels where experiments are placed. Each experimental channel is surrounded by six fuel assemblies to maximally isolate each experiment from neighboring experiments. Each fuel assembly consists of four cylindrical fuel tubes arranged concentrically (see Figure 3-8). Absorbing rods are located along the edges of the blocks. For every channel there are two to three such absorbers for a total of about 30. This core design is very flexible and allows for the simultaneous irradiation of multiple experiments in different power regimes.

Potential and Plans for Conversion

The MIR.M1 reactor has had a long-running research program focused on HEU minimization. In addition, further work is being undertaken as part of the contract (described previously) that was recently signed with the United States to study the feasibility of converting MIR.M1 from HEU to LEU.

If MIR.M1 is converted from HEU to LEU, several key performance characteristics will need to remain the same to allow the reactor to continue to fulfill its main missions. The thermal neutron flux to the experiments cannot be degraded, and the reactor power (100 MW) and campaign duration (30 days) will also need to remain constant.

Two fuel types were considered as candidates for converting the MIR.M1 reactor: (1) a UMo dispersion LEU fuel (described in Chapter 2), and (2) a uranium dioxide (UO_2) dispersion LEU fuel (the existing technology). A uranium silicide fuel type was considered at an earlier stage but was ruled out because the technology for producing UO_2 and UMo dispersion LEU fuels is better understood in Russia.[19]

UMo dispersion LEU fuel is the most likely candidate for conversion of the MIR.M1 reactor. Recent calculations have shown that to retain the required performance characteristics after conversion, the density of uranium in the core will need to be higher than is possible technologically for UO_2 LEU fuel but that is obtainable using UMo dispersion LEU fuel.

UMo dispersion LEU fuel has been tested extensively in Russia. Different material compositions (e.g., additions of silicon to the aluminum matrix) as well as different fuel fabrication technologies have been tested both with and without coatings. The results have been positive, particularly when the fuel is coated with titanium nitride. Four tests on full-scale assemblies have been performed so far—primarily to validate the conversion of the research reactor in Tashkent, Uzbekistan—and the findings have been reported by Russian scientists at conferences on enrichment reduction (Chernyshov et al., 2002). Post-irradiation materials science studies have been performed and are still ongoing.

MIR.M1 staff has found that changes in the thermal loading will require the fuel assemblies to be changed slightly from the original HEU design. Preliminary analysis has shown that using UMo dispersion LEU fuel is feasible if the fuel meat thickness is increased from 0.56 mm to 0.94 mm. Under this scenario, the annual fuel consumption for LEU would be four times higher than for HEU, but the number of fuel assemblies used would decrease by a factor of approximately 1.75.

Overall, it appears that the quality of the core can be improved by using UMo dispersion LEU fuel and changing the fuel meat thickness. The next stage of the feasibility analysis will involve verification using precision programs. However, some outstanding problems remain to be solved for the UMo dispersion LEU fuel before adopting it for use in MIR. RIAR (working in collaboration with Argonne National Laboratory) expects to complete the feasibility study for MIR.M1 by the end of 2011.

[19] In addition, Dr. Starkov stated during the symposium that he believed there have been some problems in reprocessing silicide fuels. As was noted previously, Russia reprocesses its research reactor fuel unlike in the United States.

Argus

V.A. Pavshuk

The Argus reactor at the Kurchatov Institute in Moscow is one of three HEU-fueled research reactors at the Institute to be included in the U.S.-Russia conversion feasibility study agreement.[20] The Argus reactor is a 20 kW light-water cooled and moderated solution reactor with a core volume of 22 liters of UO_2SO_4 solution containing 1.71 kg of 90 percent enriched uranium. The reactor is used for neutron radiography, neutron activation analysis, and production of isotopes and nuclear filters.

Potential and Plans for Conversion

Concrete plans are in place to convert the Argus reactor to LEU fuel. At present, work is underway to assess the feasibility of converting the reactor from HEU to 17.5 percent enriched LEU. The neutronics and thermal hydraulics calculations have been completed and will be sent to the customer by the end of 2011. Once this has been done, the documentation will need to be completed, the fuel will need to be qualified, and a license to operate with LEU fuel will need to be obtained. These activities are planned to occur in 2012.

Also throughout 2012, Argus staff will begin preparations for reloading the reactor with LEU fuel. Presently, reloading is planned to be completed by the end of 2012. After reloading, the reactor will restart and a safety validation will be performed through 2013. The conversion is planned to be completed in 2014.

IR-8

V.A. Nasonov

IR-8 is a pool-type reactor operating at 6 MW (but rated to 8 MW) with 90 percent enriched HEU fuel. The 60-cm-high IR-8 reactor core contains 16 IRT-3M fuel assemblies with a beryllium reflector. There are 12 horizontal experimental channels and a number of vertical experimental channels located in the core, the reflector, and the vessel. IR-8 is used to perform research in a broad range of fields, including nanotechnology, materials science, solid-state physics, nuclear physics, and medicine.

[20] The other reactors are IR-8, discussed in the next section, and OR (OP-M in Table 1-2), a 0.3 MW pool-type reactor. See Chapter 2.

Potential and Plans for Conversion

Maintaining the current fast and thermal neutron fluxes at IR-8 after conversion is essential for carrying out the facility's primary missions. Although the reactor is currently operating at 6 MW, there are plans in the near future to increase power to the rated 8 MW. If neutron fluxes are insufficient after this power increase, further uprating will not be possible; although the reactor can in principle operate to 30 MW, its power is restricted because of its location in the heart of Moscow. Maintaining the high fluxes with limited power will require a high-density LEU fuel.

Kurchatov has proposed to convert this reactor using IRT-3M fuel assemblies with UMo dispersion LEU fuel having 19.7 percent enrichment. Similar to the MIR.M1 reactor, two options were initially considered for transitioning to LEU: the IRT-3M assemblies with UMo dispersion LEU fuel, and IRT-4M fuel assemblies with UO_2 fuel. The IRT-4M assemblies were determined to be inadequate to maintain the needed neutron fluxes because the 3 gU/cm^3 uranium density in the UO_2 fuel is too low.

Although it is possible to increase the uranium density in the UO_2 fuel to high-enough levels to obtain the needed fluxes, fuel reliability is likely to decrease. Previous experiments were conducted with an enhanced UO_2 density (3.85 gU/cm^3) in the IRT-3M fuel; however, some fuel elements failed after being installed in the reactor for testing, and the satisfactory fuel elements achieved burnups of only 40 percent.

Kurchatov has developed a set of full-scale models to describe the reactor geometry in detail, including the reflector, core, fuel elements, and experimental channels. The staff at Kurchatov has also carried out a neutronics analysis to assess the feasibility of conversion. This analysis relies on recently developed Monte Carlo codes (MCU-PTR codes with the database MDBPTR50, along with ASTRA for thermal hydraulics calculations). These codes were benchmarked during reactor operation by measuring the isotopic composition of materials in the core and reflector.

The Kurchatov Institute (working in collaboration with Argonne National Laboratory) expects to complete the feasibility study for IR-8 by the end of 2011.

IRT-T

Yu.A. Tzibulnikov

The IRT-T reactor at Tomsk Polytechnic Institute (TPU) in Tomsk is a 6 MW[21] pool-type research reactor. The reactor is primarily used for training

[21] Tomsk Polytechnic Institute has recently requested that this reactor be licensed to operate at 11 MW.

engineers and managerial staff for nuclear power facilities as well as other specialists, but it also supports a significant amount of research, particularly industrial research. As of the time of the symposium (June 2011), up to a third of the experiments performed at IRT-T were for industrial purposes, which allowed the reactor to operate as a source of revenue for TPU. The facility plans to become financially self-sustaining between 2015 and 2017.

In 2011, TPU began implementing a growth and development program for nuclear physics research. The institute also maintains a silicon alloy laboratory (for silicon doping studies), a laboratory devoted to radioactive pharmaceuticals, and a laboratory devoted to instrumental neutron activation and analysis. A large part of the industrial production involves silicon doping, but the facility also produces some medical isotopes, including molybdenum-99.

Because of the importance of industrial revenue to the operation of the reactor, it is essential to understand how the conversion might affect the technological capability of IRT-T before proceeding—not simply the current capability, but also projected increases in technological capability. For example, current plans call for an improved instrumental and technical base to boost production to 8 tonnes per year of doped silicon from the current production of 2 tonnes (with a current capacity of 4 tonnes).

Potential and Plans for Conversion

TPU has an agreement with Argonne National Laboratory to perform computations to study the feasibility of converting IRT-T to LEU fuel. At the same time, TPU is performing independent calculations. In early 2012, TPU plans to complete an analysis comparing the performance of IRT-T using both HEU and UMo dispersion LEU fuel. At present, only initial computations and analytical work on the potential impacts of reactor conversion have been performed.

The results of these initial computations have been alarming. The use of UMo dispersion LEU fuel results in a harder neutron spectrum compared to HEU fuel, which could create problems for silicon doping applications as well as for the production of radiopharmaceuticals. Another issue of concern is the potential for higher fuel costs for LEU relative to HEU. TPU purchases its own fuel, so a significant increase in fuel costs could negatively impact revenues.

IRT

E.A. Kryuchkov

The IRT reactor at MEPhI in Moscow is a 2.5 MW pool-type research reactor. The IRT facility was designed primarily as a student training facility and secondarily to conduct a wide range of research activities. For example, MEPhI performs scientific experiments for producing short-lived isotopes, tests sensors for power stations, and hosts medical physics research, particularly the development of equipment for neutron therapy. Because of the relatively low neutron flux densities (described below), materials testing and industrial-scale isotope production are not performed at this facility. Beyond the training and research missions, MEPhI also hosts visits to the facility by members of the public. These visits are intended to improve public relations and demonstrate the safety and reliability of the reactor.

The IRT reactor uses IRT-3M type fuel enriched to 90 percent uranium-235. The reactor has a maximum fast neutron flux in the core of 4.3×10^{13} n/cm^2-s and a maximum thermal neutron flux in the core of 4.8×10^{13} n/cm^2-s. The reactor uses a beryllium reflector.[22] IRT has 48 vertical experimental locations, with 6 of these locations occupied by fuel. IRT also has 10 horizontal experimental channels, allowing for a range of training and scientific work to be conducted (see Figure 3-9).

Two horizontal experimental channels are currently being used for neutron therapy. The first is used to irradiate animals, and the second is being reconfigured for human testing. These channels require specific parameters that would not be changeable if the reactor were to be converted to LEU fuel.

In particular, it is important to MEPhI to address the following issues in converting to LEU fuel:

- Ensure that safety parameters will continue to conform to existing regulations. The radiation safety parameters for IRT are set to be stricter than for many other research reactors because of the public tours. This will continue to be true after conversion.
- Retain current neutron fluxes. Although IRT is not used to produce isotopes—meaning that the neutron flux in the vertical channels is not the key parameter—it is important to maintain the capabilities required for neutron capture therapy and to enable research by maintaining current neutron flux densities in some locations.
- Estimate the annual operation costs when working with LEU. Operating funds for the IRT reactor are limited, so economic parameters will be essential when considering conversion.

[22] Aluminum is now being used rather than beryllium in a number of locations because of swelling of some of the beryllium blocks.

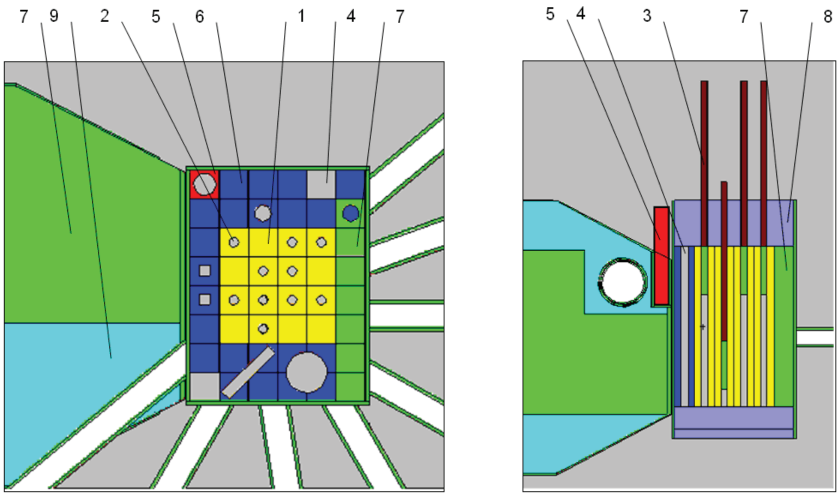

FIGURE 3-9 Core of the IRT research reactor. The diagram on the left shows the IRT core from overhead, and the diagram on the right shows the core from the side. The fuel assembly positions (1) are shown as yellow boxes; the fuel assemblies with channels for control rods (2) are shown as yellow boxes with gray circles (3) at the center, representing the control rods; the beryllium reflector positions (6) are shown as blue boxes; and the aluminum reflector positions (7) are shown as green boxes. The large green object on the left of each diagram is aluminum, and the aquamarine object to the left is graphite. SOURCE: Kryuchkov (2011).

Potential and Plans for Conversion

MEPhI is in the early stages of conversion analysis for the IRT facility. An initial neutronics analysis of the HEU core has been completed and further analysis on the neutronics and thermal-hydraulics of the core is currently under way. For these two tasks, MEPhI has used an application developed within its institute and qualified by the Russian nuclear regulator.

Using this application, the safety, experimental performance, and fuel assembly consumption parameters of an HEU core were determined for comparison with the proposed LEU core. The key parameter used for the analysis is the neutron flux density in two channels.

MEPhI staff has modeled the reactor's lifetime (with the HEU core) in great detail, including the isotopic composition of every unloaded bundle. MEPhI staff has determined that high-density UMo dispersion LEU fuel will be necessary to successfully convert the IRT reactor, as is the case for most of the other Russian reactors discussed in this chapter. However, staff remains concerned about the economic uncertainties associated with using this fuel. At this time, UMo dispersion LEU fuel has not yet been licensed in Russia, so there is no answer at this time as to what the allowed burnup will be.

The IRT reactor has been in operation for 44 years and is in need of some refurbishment. For example, the beryllium reflectors should be replaced and soon the control rods will also need to be replaced. Although the need for the reactor refurbishment is not directly connected with the conversion to LEU fuel, modifying the fuel enrichment without updating the reactor would be problematic; it would be best to combine these two tasks.

DISCUSSION

Following the individual case study briefings some time was set aside for free discussion among the workshop participants. The major points made by individuals (sometimes multiple individuals) over the course of this discussion are summarized in the paragraphs below.

- **Much work remains to be done to convert Russian reactors.** The feasibility of converting from HEU fuel to LEU fuel has been studied for a number of Russian reactors, but some participants noted that a significant amount of work still remains to be done to successfully convert them. On the other hand, it was also noted that although the United States has successfully converted a number of domestic reactors, challenges still lie ahead as the United States continues to research what will be needed to convert its high-performance research reactors.
- **Fuel development activities in both the United States and Russia are progressing quickly.** Throughout the first two days of the symposium, particularly during discussions of the case studies, one of the most frequently discussed issues involved fuel development. It was noted that the focus of U.S. efforts was on the development and qualification of UMo monolithic LEU fuel with densities of up to 15.5 gU/cm^3. Efforts in Russia are focused on the development and qualification of UMo dispersion LEU fuel with densities of more than 5 gU/cm^3. Argonne National Laboratory has been working closely with the Bochvar Institute to develop and qualify UMo dispersion LEU fuel for use in Russian research reactors. The opinion was expressed by several individuals on the U.S. side that Russian fuel development is progressing quickly.
- **The technical feasibility of reactor conversion appears to be high, but the economics of conversion in Russia still needs study.** It was observed that it may be possible in the near future to successfully convert many of the reactors discussed in this chapter without significant degradation in mission. However, particularly on the Russian side, it appears that the

economics of conversion have yet to be studied in detail. Several Russian participants noted that such an economic analysis will be essential in the coming years.

REFERENCES

Austin, K.T. 2010. LEU Core Design for the Conversion of University of Wisconsin Nuclear Reactor. Paper presented at International Meeting on Reduced Enrichment for Research and Test Reactors, October 10-14, 2010. Available at www.rertr.anl.gov/RERTR32/pdf/S2-P3_Austin.pdf.

Cook, D. 2011. Challenges Associated with Conversion of the High Flux Isotope Reactor. Presentation to the Research Reactor Conversion Symposium. June 9.

Chernyshov, V.M., Ryazantsev, E.P., Egorenkov, P.M., Nassonov, V.A., Yuldashev, B.S., Karabaev, Kh.Kh., Dosimbaev, A.A., Aden, V.G., Kartashev, E.F., Lukichev, V.A., Aleksandrov, A.B., and Yenin, A.A. 2002. Results of IRT-4M Type FA's Testing in the WWR-CM Reactor (Tashkent). Paper presented at International Meeting on Reduced Enrichment for Research and Test Reactors, 2002. Available at www.rertr.anl.gov/web2002/2003web/fullpapers-pdf/chernyshov.pdf.

Kryuchkov, E.F. 2011. Problems (peculiarities) of IRT MEPhI Research Reactor Conversion. Presentation to the Research Reactor Conversion Symposium. June 9.

Nasonov, V. 2011. Conversion of the IR-8 Reactor. Presentation to the Research Reactor Conversion Symposium. June 9.

Newton, T. 2011. Challenges with the Conversion of the MITR. Presentation to the Research Reactor Conversion Symposium. June 9.

Pavshuk, B.A. 2011. Classification of Reactors by Type of Task. Presentation to the Research Reactor Conversion Symposium. June 9.

Starkov, V.A. 2011. The Status of Testing LEU U-Mo Full-Size IRT Type Fuel Elements and Mini-Elements in the MIR Reactor. Presentation to the Research Reactor Conversion Symposium. June 9.

Tzibulnikov, Yu.A. 2011. Business Plans and Key Projects at TPU. Presentation to the Research Reactor Conversion Symposium. June 8.

USNRC (U.S. Nuclear Regulatory Commission). 2011. Nuclear Glossary. Available at: www.nrc.gov/reading-rm/basic-ref/glossary/.

Vitiello, B. 2008. Thermal Hydraulic and Safety Analysis of the University of Wisconsin Nuclear Reactor. Masters Thesis, University of Wisconsin-Madison.

Wilson, P. 2011. LEU Conversion of the University of Wisconsin Nuclear Reactor. Presentation to the Research Reactor Conversion Symposium. June 9.

4

Managing Proliferation Risks and Maintaining Missions

The final session (Session 4) of the symposium focused on possible futures for research reactors and how the proliferation risks associated with them can be managed. Five briefings presented at the symposium (Appendix A) on these topics are summarized in this chapter:

- A. Zrodnikov (Rosatom Institute for Physics and Power Engineering) discussed missions for future research reactors (Zrodnikov, 2011);
- R.P. Kuatbekov (Dollezhal Scientific Research and Design Institute of Energy Technologies [NIKIET]) and P. Lemoine (Commissariat à l'Énergie Atomique) provided Russian and French perspectives on future research reactor plans and designs (Kuatbekov, 2011; Lemoine, 2011); and
- A.N. Chebeskov (Rosatom Institute for Physics and Power Engineering) and Robert Bari (Brookhaven National Laboratory) provided Russian and U.S. perspectives on proliferation risks associated with highly enriched uranium- (HEU-) fueled research reactors (Bari, 2011; Chebeskov, 2011).

Following these briefings, symposium participants engaged in a discussion about future opportunities for the United States and Russia related to research reactor conversion. This discussion is summarized in the last section of this chapter.

FUTURE MISSIONS FOR RESEARCH REACTORS

A. Zrodnikov

Research reactors in the United States and Russia serve a variety of industrial and biomedical missions and enable research in fields such as physics and nuclear engineering. Missions mentioned during the course of the symposium that seem likely to continue include silicon doping, radioisotope production, notably including molybdenum-99, and neutron therapy. It is essential to maintain the capability to meet these research and industrial needs. Other means (e.g., particle accelerators) may be developed in the future for generating some radioisotopes and producing neutron beams, but research reactors will be far more difficult to replace for some other applications. In particular, future research related to nuclear energy and the nuclear fuel cycle will necessitate maintaining and improving current research reactor capabilities in the United States and Russia as well as in other countries. Research reactors are especially needed to conduct basic research for nuclear power development.

Nuclear power generation faces major challenges in the coming decades. Increasing quantities of commercial spent nuclear fuel are being accumulated around the world, and in the long-term, supplies of uranium-235 will begin to decrease. Fast neutron reactors ("fast reactors") are being studied in the United States and in Russia for their potential to help meet these challenges. Such reactors have the potential to "burn" long-lived actinides in spent fuel and also to produce and operate using plutonium, thereby extending current fuel supplies. However, more research remains to be done on these topics to effectively design the needed facilities and processes.

Beyond the design and testing of future fast reactors, further research could also help to extend the capability of nuclear power plants to meet new tasks. For example, research on heat- and radiation-resistant materials could lead to the deployment of high-temperature nuclear plants to meet the needs of heat-intensive industrial processes, including water desalination, production of synthetic fuels, and hydrogen production. If fossil resources that currently fuel these processes are exhausted, nuclear power will be needed to fill the gap.

Several research problems related to these topics will need to be investigated in the coming decades, including improving the scientific understanding of:

1. Nuclear physics of the interaction of radiation with matter.
2. Radiation damage of metallic and nonmetallic reactor materials.

3. Changes in macroscopic material properties caused by neutron and charged particle irradiation.

Research reactors will also be used in theoretical, computational, and experimental studies on thermo-physical, physical-chemical, corrosion, and physical-mechanical properties of advanced high-temperature coolants, fuel materials, and core structural materials. Moreover, data generated from such studies will help researchers to develop complete nuclear data libraries. This knowledge can be used to develop new nuclear technologies.

Much of the research work involving fast reactors may require capabilities that only a few current research reactors possess. A research reactor with a stationary steady-state fast neutron flux of about 10^{16} neutrons/cm^2-s will be required to support this research.

In the subsequent discussion, Thomas Newton (Massachusetts Institute of Technology [MIT]) agreed that this need for fast neutrons was also true at MIT, and observed that, after conversion, MIT plans to take advantage of the harder neutron spectrum that can be acquired with low enriched uranium (LEU) for fast neutron experiments.

FUTURE RESEARCH REACTOR PLANS AND DESIGNS

R.P. Kuatbekov

Current research reactors are unlikely to meet all needed missions over the next few decades. Many of the currently operating research reactors are ageing, and many missions are projected to grow in importance. Consequently, there is a need to design and build new research reactors. In many cases, particularly for industrial applications, new reactors can be designed from the beginning to use LEU rather than HEU fuel. In other cases, particularly if HEU or even plutonium fuel is required to retain essential performance characteristics, alternative solutions may need to be found to meet nonproliferation goals.

The customers of research reactors do not care whether the reactor is fueled by HEU or LEU—they simply need the results within a reasonable period of time and at reasonable cost. This is true whether the results are completed research, produced materials, or medical isotopes. Consequently, two key qualifications for any new research reactor will be: (1) its ability to meet customer needs; and (2) economic and technical feasibility. With respect to economics, both initial costs and refueling costs of the reactor should be considered to be reasonable by the operator.

In addition, not all countries can afford to perform experiments to optimize fuel for their research reactors, as the United States and Russia have done, and these countries are likely to be a major market for certain types of research reactors in the coming decades. For these reasons, NIKIET is using reliable and tested fuel types and design solutions in its new research reactor designs. At the same time, proliferation concerns will need to be accounted for.

NIKIET is in the process of designing several new types of LEU-fueled research reactors for industrial, biomedical, training, and research applications. The focus is on the development of pool-type reactors with integrated passive safety systems. Pool-type reactors are convenient for the end-user because they allow for flexibility in the core configuration and easy access to experimental positions. NIKIET uses standardized components in its reactor designs, which reduces costs and simplifies future repairs.

NIKIET is focusing on narrow-purpose reactor designs that optimize each reactor for the customer's primary end use. There are two end uses that are in highest demand, both mentioned elsewhere in this report: medical isotope production and silicon doping. NIKIET is focusing on optimizing designs of two reactor types for these applications: (1) a low-power (500 kW or less) reactor with natural circulation for silicon doping and (2) a 15 MW reactor for isotope production. Some preliminary computations have been carried out on these reactor designs, and NIKIET plans to improve these designs in the future with additional computations and design work.

It is feasible to meet most customer needs using LEU-based research reactors. Designing reactors to use LEU from the start will not be as much of a challenge as retrofitting some current HEU-fueled reactors. In fact, modifying research reactor cores that were originally designed to use HEU can be very expensive and technically challenging, as illustrated by the case studies in Chapter 3. If the core is optimized during the design stage, then it can simultaneously be optimized for its missions. For example, if the core is initially designed to use LEU fuel, then differences in neutron fluxes and spectra can be accounted for from the initial design stages.

In fact, NIKIET has found that several of its designs for new LEU reactors maintain high flux levels to meet customer requirements and achieve reliable operation with high fuel burnups. NIKIET has now proven computationally that these LEU reactors should operate as well as similar HEU reactors.

On the other hand, some cutting-edge research requires reactors with unique designs or higher fast or thermal neutron fluxes. This research may not be able to be carried out using the types of standardized designs described here. In many cases, unique LEU-fueled reactors can be designed from the start to meet these needs; however, it was suggested by some at the symposium that maintaining a small number of special-purpose reactors fueled by HEU or plutonium could have value, particularly for fast reactor research.

Discussion

N.V. Arkhangelsky noted that a provision formulated at the start of RERTR—backed both by Russia and the United States—acknowledges that there are a number of reactors that will not lend themselves to conversion, including fast breeders. This provision remains in effect. One example of such a reactor is the Russian BOR-60 with an HEU and plutonium core. Before the end of this decade, a multipurpose fast reactor of this type will have been built in Dmitrovgrad. V. Ivanov stated that this reactor must also be viewed as unique or qualifying for special treatment, because the work done there will be important to future advances in nuclear technology.

Another participant noted that very few reactors of such unique types are likely to be needed. Several delegates at the symposium suggested that large international centers could be used to house a small number of research reactors operating using HEU or plutonium. These high-performance reactors would be pursued on an international basis, making adequate capacity available for the international research community. This approach could eliminate the need for the United States and Russia to duplicate their capabilities in this area and allow top-quality personnel and the highest standards of physical protection and materials protection, control, and accounting to be focused at a small number of sites.

NEW RESEARCH REACTOR CASE STUDY: THE JULES HOROWITZ REACTOR

P. Lemoine

The Jules Horowitz Reactor (JHR) is a 100 MW multipurpose materials testing reactor that was commissioned to replace another reactor, OSIRIS, which was built in the 1960s. JHR was initially designed to operate with a new high-density LEU fuel; however, because of difficulties in the development and qualification of this fuel, the reactor will begin operation with HEU fuel instead as described in the paragraphs to follow.

The JHR fuel elements consist of eight circular rings of curved fuel plates, each 1.37 mm thick (see Figure 4-1). The fuel elements have a 98 mm external diameter and a 600 mm active height. The nominal hydraulic gap (“coolant gap” in Figure 4-1) between the fuel plates is 1.95 mm; light water, which streams upward through the gap at a speed of 15 meters per second, is used for both cooling and moderating the core.

The core can contain 34 to 37 fuel elements and has up to 10 experimental positions (see Figure 4-2). The designed neutron fluxes are 5.5×10^{14} fast neutrons per square centimeter per second (n/cm^2-s) in the core and up to 4.5×10^{14} thermal n/cm^2-s in the reflector.

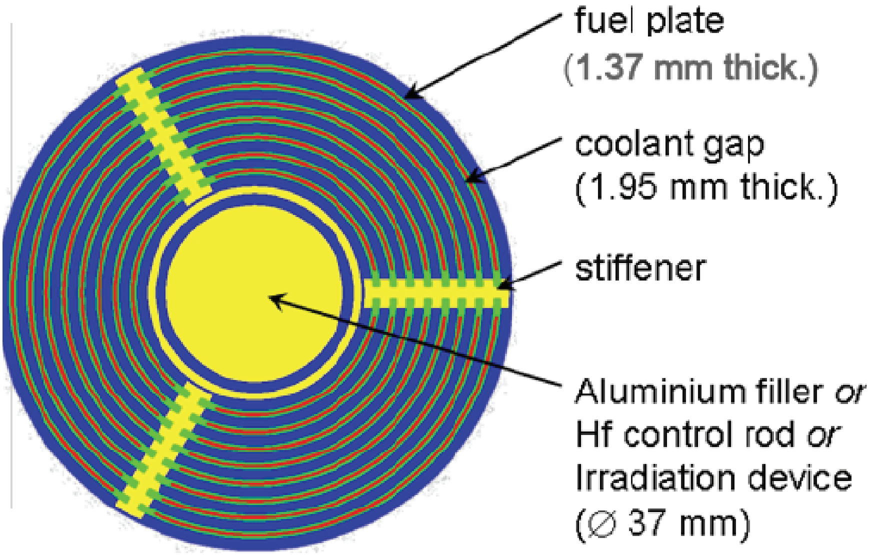

FIGURE 4-1 Schematic illustration of a JHR fuel element. The 1.37-mm-thick fuel plates form eight concentric rings, with coolant gaps of 1.95 mm between the plates. The center of the fuel element contains aluminum filler, a hafnium control rod, or an experimental position. SOURCE: Lemoine (2011).

The reactor was designed in 2002 using a reference fuel of high-density (8 grams uranium per cubic centimeter [gU/cm^3]) UMo dispersion LEU fuel. Original plans had called for this fuel—in development under the RERTR program—to be qualified in 2006. In 2004, however, problems with the fuel's irradiation behavior indicated that it would be unlikely to be available in time for JHR's completion. At the time of this symposium, UMo dispersion LEU fuel was still under development by the European initiative LEONIDAS, which is supported in part by the U.S. Department of Energy (DOE). Further optimization still needs to be done to qualify this fuel and demonstrate that it will be available at reasonable cost.

JHR still intends to use UMo dispersion LEU fuel when it becomes available. However, for the time being, JHR plans to use a neutronically equivalent uranium silicide (U_3Si_2) dispersion fuel enriched to 27 percent uranium-235. The higher enrichment of the silicide fuel is intended to balance its lower density (4.8 gU/cm^3) relative to UMo dispersion LEU fuel. The neutron-equivalent U_3Si_2 fuel is currently under qualification. Although this fuel has been used in other reactors, qualification for JHR is needed because its operating level is much higher than the operating levels of other reactors that use this fuel.

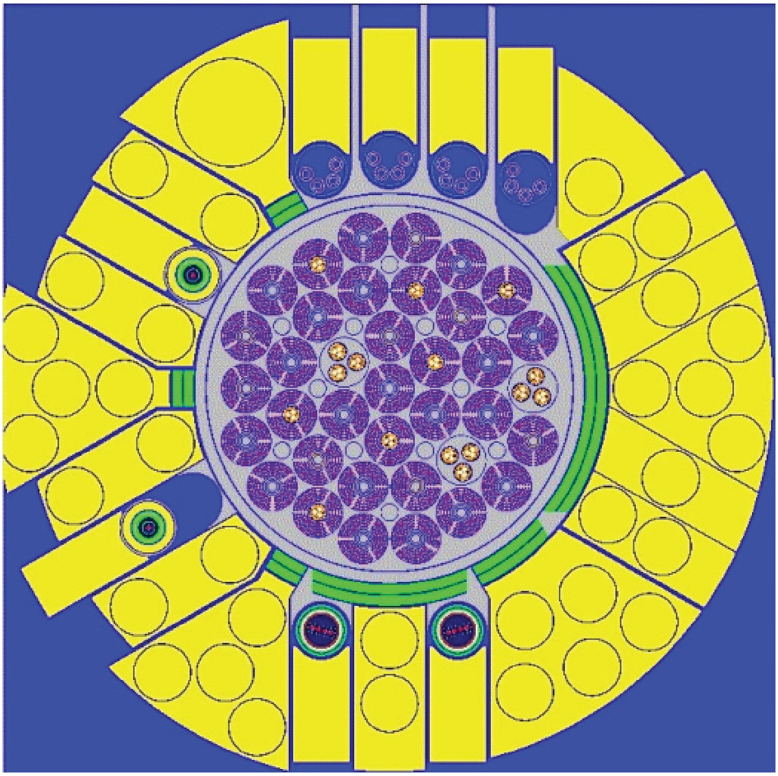

FIGURE 4-2 Schematic illustration of the JHR core. The fuel elements are shown in purple. Ten experimental positions are shown in yellow, with seven located in the center of individual fuel elements. Three “triple” experimental positions are available in fuel element positions. The core is surrounded by a beryllium reflector with additional fixed experimental positions and eight cross water channels for mobile devices. SOURCE: Lemoine (2011).

METHODS TO IMPROVE THE ASSESSMENT OF THE RISKS POSED BY HEU-FUELED RESEARCH REACTORS

If research reactors will continue to be needed in the foreseeable future it is important to understand as clearly as possible their risks. As noted previously, conversion of research reactors from HEU to LEU lowers risk. However, some reactors may not be able to be converted, so it is important to understand the risks associated with their continuing operation. This risk goes beyond the reactor itself to involve all facilities and associated infrastructures, including fuel manufacturing; transportation; fresh fuel storage; irradiated fuel storage; and reprocessing or final repository placement.

Robert Bari described two different types of risk associated with research reactor facilities and infrastructures (systems) as follows (Bari, 2011):

- Proliferation risk of an HEU-fueled research reactor's fuel cycle is associated with the diversion or undeclared production of nuclear material or misuse of technology by a host state seeking to acquire nuclear weapons or other nuclear explosive devices.
- Terrorism risk of an HEU-fueled research reactor's fuel cycle is associated with the theft of materials suitable for nuclear explosives or radiation dispersal devices and the sabotage of facilities and transportation by sub-national entities and/or non-host states.

The following sections describe two methodologies to structure and improve the understanding of proliferation and terrorism risk: First, assessing the relative attractiveness of various nuclear materials; and second, proliferation risk assessment methods.

Material Attractiveness

A.N. Chebeskov

As noted in Chapter 1, the lack of availability of special nuclear material (SNM) that can be used to build a nuclear weapon is widely agreed to be a major barrier to nuclear proliferation (see Chapter 1). Thus, an essential part of understanding the proliferation risk associated with a research reactor involves understanding how straightforward it would be for a host state or terrorist organization to successfully misuse the reactor's fuel material.

The attractiveness of a nuclear material from a proliferator's point of view is determined in large part by a material's ability to sustain a nuclear chain reaction. Material attractiveness is also influenced by whether it is necessary to process the material to make it usable in a nuclear weapon. To categorize fissile materials qualitatively, four categories (classes) might be used: very attractive, attractive, low attractive, and unattractive.

Several variables are relevant to the attractiveness of SNM. For example, for a given quantity of uranium, its attractiveness is proportional to both its enrichment and its mass. Higher-enriched materials are more attractive than lower-enriched materials; for example, HEU enriched to 90 percent uranium-235 is far more attractive than LEU, which is regarded to be unattractive. Similarly, higher masses are more attractive than lower masses for a given level of enrichment. In general, the higher the enrichment, the less mass is required to obtain an equivalent amount of uranium-235.

Of course, nuclear weapons can be constructed using plutonium as well, but it is difficult to compare the attractiveness of different materials. Different grades of plutonium can be rated relative to one another as reactor grade (less attractive) and weapons grade (more attractive). However, very highly enriched uranium is the most desirable material for a potential proliferator, because of the relative simplicity of constructing a nuclear explosive device using HEU as opposed to plutonium.

As an example, at the MEPhI reactor, the small size and mass of very highly enriched fuel assemblies represent a higher theft risk than heavier power reactor fuel assemblies, especially for fresh fuel assemblies. The uranium contained in the MEPhI fuel assemblies would not need further enrichment to be usable in a nuclear explosive device. For irradiated fuel assemblies this risk is smaller because of the presence of strong radiation.

Risk Assessment

Robert Bari

Quantitative risk assessment has been used successfully to estimate safety risks, for example, at nuclear power plants. However, more research is needed before proliferation and terrorism risks can be effectively estimated using such a methodology. Such risk assessment methods are easier to apply to safety, for several reasons:

- The likelihood of an accident is more easily estimated than the likelihood of a deliberate attack. A deliberate attack depends on the choices of an intelligent adversary, making likelihoods and methods of failure difficult to estimate.
- Inherent features and engineered systems with known characteristics provide safety, whereas both intrinsic (i.e., barriers intrinsic to the technologies themselves) and extrinsic (e.g., guns, guards, gates, safeguards) systems provide security. The effectiveness of some extrinsic measures, particularly those that involve human action, can be difficult to estimate.
- For safety, defense in depth and safety margins are universally embraced.

Workable proliferation risk models still need significant development.

The methodology summarized here is one of several possible approaches and is analogous to the approach developed for the Generation IV Forum: Proliferation Risk and Physical Protection (PR&PP).

To perform an effective risk assessment, it is important to gather a great deal of information about the research reactor facility as well as the country in which it is located. There are many countries with research reactors, each having its own national and geopolitical interests that could impact the potential for proliferation. In addition, a number of key assumptions need to be considered in the analysis. These include assumptions about potential threats, such as diversion, misuse, breakout, theft, and sabotage; extrinsic factors such as sources of fresh fuel supply, spent fuel disposition, and fuel transportation; and facility design and operational information that impact proliferation risk.[1]

The assessment itself involves building a range of scenarios by which proliferation could occur; analyzing specific scenarios to determine whether an attempted proliferation was successful and the barriers that were encountered along the way; then using the responses to construct a risk estimate.

The key elements of an effective proliferation risk assessment include:

- Gather information on facility design.
- Define country (or countries) context.
- Establish/define international safeguards design.
- Establish/define physical protection design.
- Define adversary mission success.
- Identify facility targets (for adversary).
- Perform pathway analysis to define potential scenarios for proliferation.
- Evaluate pathways for each threat and measure.
- Assess and interpret results.

Further research will be needed before this type of analysis can be carried out in a dependable way for research reactors. The range of possible scenarios has not been explored in much detail. In addition, combining the information produced by each stage of the analysis described above to produce an overall understanding of risk remains challenging. However, such a risk assessment process can still be worthwhile to perform. In particular, the process itself can provide useful insights, not just the final result.

Discussion

Many measures that can be taken to reduce the risk of proliferation from research reactors are already well known. Some measures mentioned by symposium attendees included avoiding the use of HEU fuel where possible in favor of LEU fuel; maintaining adequate nuclear materials

[1] For example, one facility might require only very few radiation protection measures to isolate nuclear materials, whereas another facility might require more sophisticated measures. These operational characteristics affect the proliferation risk of the facility.

protection control and accountability (MPC&A) and physical protection measures; and using appropriate insider prevention methods, as practical.

Many of the participants at the symposium observed that the principal means of reducing proliferation risk is conversion of research reactors to LEU; however, as noted previously, this may not be possible in all cases. Other participants noted that some risk also accompanies the use of LEU. Consequently, appropriate MPC&A and physical protection measures will continue to be needed, although to a lesser extent than with HEU fuel.

A symposium participant posed a question about the relative priorities between conversion to LEU and better physical protection. In particular, is it possible to compensate for HEU use through improved security? Robert Bari's reply was that one cannot separate conversion from physical protection. Clearly, maintaining HEU fuel poses a greater risk, but LEU use does not mean zero risk. Richard Meserve clarified that at a gross overview level, conversion lowers risks as well as the costs for physical protection.

FUTURE OPPORTUNITIES FOR THE UNITED STATES AND RUSSIA

Near the close of the symposium, participants were asked to summarize important ideas that had been mentioned over the preceding three days and to identify potential future opportunities for both the United States and Russia on the conversion of research reactors from HEU to LEU fuel. During this discussion, many key points were brought up by individuals in attendance at the symposium. These points include the following:

- **Many symposium participants from both the United States and Russia emphasized the importance of reducing and, where possible, eliminating the use of HEU in research reactor fuel.** Over the past few decades, the trend in research reactor development—as well as in civilian applications as a whole—has been to reduce the use of HEU.
- **Research reactors currently serve important purposes for research and industry, and they will to continue to serve important purposes into the future.** In some cases, accelerators or other sources of neutrons could be used to replace research reactors for medical isotope production and other applications. However, for scientific research, some types of irradiation phenomena, and advanced fuel cycle work, research reactors will continue to be invaluable into the foreseeable future. Several workshop participants stated that these reactors must continue to operate safely and cost effectively and fulfill their missions in ways that meet the needs of customers.
- **Collaboration between the United States and Russia on conversion of research reactors will continue to be essential and fruitful.** Daniel Wachs observed that past collaborative U.S.-Russian work on fuel development has provided opportunities to advance conversion of both countries'

reactors; he stated that the cross-fertilization of ideas, lessons learned, and technological advances has been helpful and should continue to be encouraged. In addition to technical collaboration, one participant observed that there is significant potential for collaboration on the regulatory aspects of conversion as well. Alexander Adams and V. Bezzubtev noted that Russia will face many challenges in regulating its to-be-converted reactors; the United States has previously faced many similar challenges and may have helpful advice for Russia on this issue.

- **The United States and other nations have been able to convert research reactors to LEU fuel while maintaining performance required for key missions, e.g., research as well as medical and industrial applications.** H.-J. Roegler observed that prior to conversion of many research reactors in Europe there were a number of concerns about maintaining needed functionality after conversion. However, in the end, the performance of many research reactors was improved as a result of the conversion process through design changes and better understanding of reactor behavior. P. Adelfang added that an analogy might be made to molybdenum-99 production. In 2001, Argentina's Cómision Nacional de Energía Atómica (CNEA) made the decision to convert its domestic production from HEU targets to LEU targets. At that time, it was considered to be infeasible to produce molybdenum-99 in significant quantities using LEU; however, CNEA showed that it could be done. After nine years it has become abundantly clear that high-quality molybdenum-99 production is possible with LEU targets.
- **The economic and performance challenges associated with conversion are likely to be surmountable, particularly with government assistance and the involvement of reactor operators and customers.** Research reactor conversions have been successfully completed in many countries, but many of these efforts would have been unlikely to occur without U.S. government support. B. Myasoedov and Jeffrey Chamberlin agreed that government involvement is critical to future conversion successes in Russia and the United States. Jordi Roglans noted that governments' decisions regarding future HEU use would likely be influenced by the potential for economic and other upheavals if a terrorist event involving HEU occurred related to research reactors or otherwise.
- **Some facilities may not be easily convertible to LEU fuel, including fast reactors, fast critical assemblies, reactors with small core volumes, and reactors with high specific power per unit volume of active core.** The feasibility of conversion depends to some extent on policy choices by the host nation's government. Several workshop participants suggested that one way of minimizing the use of HEU for essential or unique missions would be to create major international nuclear centers to house the few reactors needed for these missions and to ensure that those facilities have strong security and safeguards protection. A. Zrodnikov observed that international

centers would complement conversion, because a large international facility would allow research to be done that would be more challenging than at a smaller facility. In addition, he observed that at such facilities it would be easier to manage high-quality MPC&A as well as physical protection because of the international attention that such facilities would receive, especially if such facilities were placed in nations with well-developed nuclear infrastructures. The suggestion regarding major international centers received support from several Russian participants.

REFERENCES

Bari, R. 2011. Estimation of Risk for Research Reactors. Presentation to the Research Reactor Conversion Symposium. June 10.

Chebeskov, A. 2011. An Approach to Proliferation Risk Assessment for Research Nuclear Reactors. Presentation to the Research Reactor Conversion Symposium. June 9.

Kuatbekov, R.P. 2011. Types and Designs of Prospective Research Reactors with LEU Fuel. Presentation to the Research Reactor Conversion Symposium. June 10.

Lemoine, P. 2011. Fuel Design and LEU Development for the Jules Horowitz Reactor. Presentation to the Research Reactor Conversion Symposium. June 10.

Zrodnikov, A. 2011. Power Development and Missions of Prospective Research Reactors. Presentation to the Research Reactor Conversion Symposium. June 10.

Appendix A

Symposium Agenda

Russian-American Symposium on the Conversion of Research Reactors to Low Enriched Uranium Fuel

6–10 June, 2011
Moscow
RAS Presidium
Presidential Hall
(Leniniskii Road, 32a, 2nd Floor)

June 8, Wednesday

SESSION 1: WELCOME, PURPOSES, AND TASKS OF SYMPOSIUM

10:00 **Welcome and opening remarks**
Academician N. Laverov, co-chair of symposium (Russian Academy of Sciences)

10:10 **Welcome from Rosatom**
S.V. Kiriyenko (Rosatom)

10:25 **Welcome from Russian Federation Ministries of Formation and Science**
S.N. Mazurenko (Ministries of Formation and Science)

10:40 **Tasks and purpose of symposium**
R. Meserve, co-chair of symposium (Carnegie Institution for Science, USA)

COFFEE BREAK (11:00 – 11:20)

11:20 **Keynote briefing: Non-proliferation and the reduction of commercial traffic in HEU**
P. Adelfang (International Atomic Energy Agency)

12:00 **Types, purposes, and conversion potential of Russian origin research reactors**
Yu.G. Dragunov (NIKIET)

12:30 **Challenges associated with converting reactors to low enriched fuel: History and prospects**
N.V. Arkhangelsky (Rosatom)

LUNCH (1:00 – 2:30 pm)

2:30 **Welcome from U.S. Department of Energy**
T. D'Agostino (U.S. Department of Energy National Nuclear Security Administration)

2:40 **Types, purposes, and conversion potential of U.S. origin research reactors**
J. Roglans (Argonne National Laboratory)

SESSION 2: OVERVIEW OF TECHNICAL CHALLENGES ASSOCIATED WITH CONVERSION AND POTENTIAL SOLUTIONS

Panel 2.1: Technical challenges associated with conversion and potential solutions

3:00 **LEU fuel design for research reactors**
D. Wachs (Idaho National Laboratory)

3:20 **Reduced enrichment in research reactors: Current status and prospects**
Yu.S. Cherepnin (NIKIET)

3:40 **Core modifications (U.S. viewpoint)**
J. Stevens (Argonne National Laboratory)

4:00 **Core modifications (Russian viewpoint)**
I.T. Tetiyakov (NIKIET)

COFFEE BREAK (4:20 – 4:40)

4:40 **Maintaining performance and missions (U.S. viewpoint)**
J. Roglans (Argonne National Laboratory)

5:00 **Maintaining performance and missions (Russian viewpoint)**
A.L. Petelin (NIIAR)

5:20 **Discussion and question and answer period**
Led by J. Snelgrove, Committee member

6:00 Closing remarks

June 9, Thursday

SESSION 2: OVERVIEW OF TECHNICAL CHALLENGES ASSOCIATED WITH CONVERSION AND POTENTIAL SOLUTIONS, CONTINUED

Panel 2.2: Other technical challenges associated with conversion

10:00 **Ageing and obsolescence of research reactors**
H.-J. Roegler (Siemens, ret.)

10:15 **Ageing and obsolescence of research reactors**
E.P. Ryazantsev (Kurchatov Institute)

10:30 **Regulatory challenges (U.S. viewpoint)**
A. Adams (U.S. Nuclear Regulatory Commission)

10:45 **Regulatory challenges (Russian viewpoint)**
V.S. Bezzubtsev (ROSTEXNADZOR)

11:00 **Challenges posed by research reactors that cannot be converted (U.S. viewpoint)**
J. Chamberlin (U.S. Department of Energy National Nuclear Security Administration)

11:15 **Challenges posed by research reactors that cannot be converted (Russian viewpoint)**
A.V. Zrodnikov (Institute for Physics and Power Engineering)

COFFEE BREAK (11:30 – 11:50)

11:50 **Discussion and question and answer period**
Led by V. Ivanov, Committee Member

Panel 2.3: How challenges associated with previously converted reactors were overcome

12:20 **Experience with solutions to conversion challenges (U.S. viewpoint)**
J. Matos (Argonne National Laboratory)

12:40 **Experience with solutions to conversion challenges (Russian viewpoint)**
Yu.S. Cherepnin (NIKIET)

LUNCH (1:00 – 2:30)

2:30 **Discussion and question and answer period**
Led by R. Bari, (Brookhaven National Laboratory; consultant to the committee)

SESSION 3: TECHNICAL CHALLENGES ASSOCIATED WITH CONVERSION OF SPECIFIC U.S. AND RUSSIAN REACTORS (CASE STUDIES)

Panel 3.1: Converting two U.S. reactors

3:00 **Challenges associated with the conversion of American reactor MITR**
T. Newton (Massachusetts Institute of Technology)

3:20 **Challenges associated with the conversion of American reactor HFIR**
D. Cook (Oak Ridge National Laboratory)

3:40 **Discussion and question and answer period**
Led by R. Meserve, U.S. Chair

COFFEE BREAK (4:00 – 4:20)

Panel 3.2: Converting two Russian reactors of the six intended for conversion under the recent U.S.-Russian agreement

4:20 **Concrete challenges and solutions**
V.A. Starkov (NIIAR)

4:40 **Classification of reactors according to type of decided tasks**
V.A. Pavshuk (Kurchatov Institute)

5:00 **Discussion and question and answer period**
Led by N. Laverov, R.F. Chair

Panel 3.3: Converting training research reactors

5:20 **Conversion of the U.S. training research reactors**
P. Wilson (University of Wisconsin, Madison)

5:40 **Problems with conversion of reactor IR-8**
V.A. Nasonov (Kurchatov Institute)

6:00 **Challenges associated with converting training research reactor MIFI**
E.F. Kruchkov (MEPhI)

6:20 **Challenges associated with converting training research reactor TPU (Tomsk)**
Yu.A. Tzibulnikov (TPU)

6:40 **Discussion and question and answer period**
Led by A. Zrodnikov, Committee member

June 10, Friday

SESSION 4: FUTURE OPPORTUNITIES FOR CONVERSION

10:00 **Estimation of the risks of the propagation of fissionable elements with the operation of the research reactors**
A.N. Chebeskov (Institute for Physics and Power Engineering)

10:20 **Estimation of risk for research reactors**
B. Bari (Brookhaven National Laboratory)

10:40 **Desires of reactor users and the future tasks that cannot be solved today on the existing types of the research reactors**
A.V. Zrodnikov (Institute for Physics and Power Engineering)

11:00 **Design and engineering of future LEU fuel for research reactors: Application to Jules Horowitz Reactor project**
P. Lemoine (Commissariat à l'Énergie Atomique, France)

11:20 **Types and designs of future LEU research reactors**
R.P. Kuatbekov (NIKIET)

COFFEE BREAK (11:40 – 12:00)

12:00 **Discussion and question and answer period**
Led by B. Myasoedov, Committee member

12:20 **Discussion of the actions which Russian and U.S. organizations could undertake for the realization of the conversion of research reactors**

12:50 **Summary of symposium results**
R. Meserve, U.S. Chair
N. Laverov, R.F. Chair

1:10 **Adjourn symposium**

Appendix B

Committee and Staff Biographical Sketches

Richard A. Meserve (U.S. Chair) became the ninth president of the Carnegie Institution for Science in 2003. Dr. Meserve was the chairman of the U.S. Nuclear Regulatory Commission (USNRC) from October 1999 until March 2003. He is currently Senior of Counsel in the Washington, D.C. law firm of Covington & Burling, where he was a partner before joining the USNRC. He devoted his legal practice to technical issues arising in environmental and toxic tort litigation, counseling scientific societies and high-tech companies, and nuclear licensing. Dr. Meserve also served as an adviser to the President's Science and Technology Advisor from 1977-1981, and as a law clerk to Justice Harry A. Blackmun of the United States Supreme Court and Judge Benjamin Kaplan of the Massachusetts Supreme Judicial Court. Among other affiliations, he is a member of the American Philosophical Society and an elected fellow of the American Academy of Arts and Sciences, the American Association for the Advancement of Science (AAAS), and the American Physical Society. He has served as chairman or a member of numerous committees of the National Academies, including the Committee on Science, Technology and Law, the Board on Energy and Environmental Systems, the Board on Radioactive Waste Management, and the Nuclear and Radiation Studies Board. He also was chair of the Committee on Upgrading Russian Capabilities for Controlling Highly Enriched Uranium and Plutonium. He received his bachelor's degree from Tufts University in 1966, a law degree from Harvard in 1975, and his Ph.D. degree in applied physics from Stanford in 1976. He was elected to the National Academy of Engineering in 2003.

Nikolay P. Laverov (Russian Chair) is vice president of the Russian Academy of Sciences (RAS) and former director of the Institute of Geology of Ore Deposits, Petrology, Mineralogy, and Geochemistry. He has worked in and with the USSR and Russian governments on a range of ecological problems, particularly nuclear waste disposal, and has been a leader in radiogeological studies aimed at using the protective properties of the geological environment to prevent pollution of the ecosphere by radionuclides. In addition to his research activities, Dr. Laverov has held a variety of prominent positions in scientific administration and government, including chief of the Scientific Research Organizations Administration of the USSR Ministry of Geology (1972-1983), pro-rector of the Academy of the National Economy (1983-1987), president of the Kyrgyzstan Academy of Sciences (1987-1989), and USSR deputy prime minister and chairman of the USSR State Committee for Science and Technology (1989-1991). In 1989, Dr. Laverov was elected vice president of the USSR Academy of Sciences, a post to which he was subsequently re-elected in the RAS. In 1992, he was named co-chair of the Earth Science Joint Working Group, which is under the auspices of the U.S.-Russian Space Agreement. He is also a member of the Council on Science and Technology under the President of the Russian Federation. Dr. Laverov graduated from the M.I. Kalinin Nonferrous Metals and Gold Institute in Moscow in 1954 and earned a doctorate in geological-mineralogical sciences in 1958. A full member (academician) of the RAS since 1987, he has authored or co-authored more than 250 publications including 20 books and has served as editor-in-chief of the journal *Geology of Ore Deposits* since 1989.

Vladmir Asmolov is first deputy director general-director for scientific-technical policy of Energoatom Concern OJSC. Prior to this position, he served as deputy director general-director for science and engineering of FSUE Concern Rosenergoatom. He has also served as director of the Kurchatov Institute, and from 2003-2004, he served as deputy minister of atomic energy of the Russian Federation. In addition, Dr. Asmolov is currently serving as representative of the Russian Federation to the Organization for Economic Cooperation (OECD) Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI); as a professor at Moscow Power Engineering Institute (Technical University); as chairman of the Scientific and Technical Panel of Rosatom (Federal Agency of Atomic Energy, formerly Minatom); as chairman of the scientific and technical panel of Concern Rosenergoatom; and as a member of the International Nuclear Safety Group (INSAG). He has received a certificate of appreciation from the U.S. Nuclear Regulatory Commission, the Order of Courage from the President of the Russian Federation, and the Order of Honour from the President of the Russian Federation. He received a master's degree from the Moscow Power-Engineering Institute and a Ph.D. from the Kurchatov Institute.

David J. Diamond is chief scientist in the Nuclear Science and Technology Department at Brookhaven National Laboratory. He is also acting leader of the Nuclear Analysis Group. He has extensive experience in nuclear reactor safety, primarily through his work for the U.S. Nuclear Regulatory Commission (USNRC). He has also worked on safety issues with regulatory bodies in more than a half dozen countries as well as the International Atomic Energy Agency. His technical contributions are through the application of neutronics and thermal-hydraulics models, and the combining of deterministic and statistical analyses. The applications have been to problems in light and heavy water power and non-power reactors. For research and test reactors (RTRs) he has led a team providing support in reactor analysis and other disciplines for the research reactor at the National Institute of Standards and Technology (NIST) Center for Neutron Research. The team also provides support to the USNRC staff responsible for RTR licensing. An example of the latter work has been the review of the safety reports for conversion (HEU to LEU fuel) of the USNRC-licensed university reactors. Dr. Diamond has been asked to chair various international panels addressing safety issues. Dr. Diamond received his Ph.D. from the Massachusetts Institute of Technology, and he is a fellow of the American Nuclear Society (ANS) and a recipient of the ANS' Tommy Thompson Award recognizing contributions to nuclear installation safety.

Valentin B. Ivanov is chief research scientist at the RAS Institute of Ore Deposits, Petrography, Mineralogy, and Geochemistry. He graduated from the Samara Technical University with a degree in electrical engineering and received his doctorate of technical sciences in Moscow from Institute of Radiation Techniques in 1991. His sphere of professional interests includes the nuclear fuel cycle and spent nuclear fuel management. From 1963 to 1998, he worked at Research Institute of Atomic Reactors (RIAR), for the last nine of those years serving as its director general. From 1998 to 2002, he served as first deputy minister for atomic energy of the Russian Federation. In 2003, he was elected to the Russian State Duma, where he served as a member of the parliamentary Committee on Energy, Transport, and Communication until 2008.

Boris F. Myasoedov is deputy secretary general for science of the Russian Academy of Sciences (RAS), head of laboratories at both the RAS Vernadsky Institute of Geochemistry and Analytical Chemistry and the RAS Frumkin Institute of Physical Chemistry and Electrochemistry. His scientific activity covers such fields as the fundamental chemistry of actinides, fuel reprocessing, partitioning of radioactive waste, and environmental protection. He has authored more than 500 publications and serves as editor of the journals *Problems of Analytical Chemistry* and *Radiochemistry*. Academician Myasoedov graduated from D.I. Mendeleev Chemical-Technology Institute in Moscow in 1954 and earned a Ph.D. in radiochemistry from the Vernadsky Institute in 1965 and his full doctorate in 1975 from the same institute. He was elected to the Russian Academy of Sciences in 1994 and has been awarded two State Prizes for his research on the chemistry of transplutonium elements (1986 and 2001), the Khlopin Prize for his studies of the chemistry of protactinium (1974), and the Ipatiev Prize of the RAS Presidium in 2003.

James L. Snelgrove retired from Argonne National Laboratory (ANL) as senior physicist in February 2007. During his first 10 years, he worked in the areas of fast reactor critical experiments and test reactor analysis and design. He worked on the Reduced Enrichment for Research and Test Reactors (RERTR) program from its inception in 1978 until he retired, mainly in the areas of high-density fuel and Mo-99 target development and testing. He led the fuel development and testing effort from late 1981 until mid-2004 and coordinated the program's collaboration on fuel development with the Russian RERTR program from 1996 until his retirement. From 2005 through 2008, he coordinated the International Atomic Energy Agency's (IAEA) effort to produce a document on "Good Practices for Qualification of High Density LEU Research Reactor Fuels," which was published as a Nuclear Energy Series document in 2009. Since late 2009, he has been coordinating preparation of another IAEA document on the properties of uranium molybdenum alloy research reactor fuels. Currently he works part time at ANL for the RERTR program as a senior advisor for research reactor fuels, and he occasionally consults with agencies and companies around the world in the area of research reactor fuel development and qualification. Dr. Snelgrove received his B.S. in physics from Tennessee Technological University in 1964 and his M.S. in physics in 1966 and Ph.D. in experimental nuclear physics in 1968 from Michigan State University.

Anatoly Zrodnikov is scientific leader of the State Scientific Center of the Russian Federation, the Institute for Physics and Power Engineering (IPPE) in Obninsk (since 2010), IPPE director general (1996-2010), and head of Department of National Research Nuclear University "Moscow Engineering & Physics Institute" (since 2005). He joined the IPPE in 1969 after graduation from the Moscow Power Engineering Institute (Technical University) with an M.S. in applied physics, and he received his Ph.D. in nuclear engineering in 1975 and his D.Sc. in physics and mathematics in 1994. His scientific interests are in the areas of neutronics, thermal and plasma physics, direct energy conversion, perturbation theory, nuclear power engineering including the space one, fast neutron reactors, and nuclear pumped lasers. Dr. Zrodnikov was the president of the Russian Nuclear Society in 2001-2003, a member of the Government Committee of the Russian Federation on Science and Innovation Policy (2003-2005), chairman of the Obninsk City Scientific and Technical Council (since 1996), and president of the Kaluga Regional Scientific Center. Also he is a member of editorial boards of scientific journals, including *Laser and Particle Beams, Atomic Energy,* and *Nuclear Power Engineering,* and he is author and co-author of more than 300 scientific publications. He holds the title of Honored Scientist of the Russian Federation and was awarded an Order of Honor of the Russian Federation, and Honorary Citizen of Kaluga region.

Committee Consultant

Robert A. Bari is a senior physicist at the U.S. Department of Energy's Brookhaven National Laboratory. He has been involved in the design and safety assessments of complex, high-technology facilities since he joined the applied programs at the Laboratory in 1974. He has worked on projects and issues regarding nuclear safety and nonproliferation technologies, nuclear waste management, development of advanced nuclear reactors, and other related technologies. During the 1980s, at the request of the U.S. Nuclear Regulatory Commission (USNRC), Dr. Bari created and led a team of experts in the area of probabilistic risk assessment (PRA). This team expanded PRA methodologies in areas of importance to safety of nuclear power plants. In addition to his work for the USNRC, Dr. Bari led a four-laboratory team in a year-long evaluation of the impact of fuel enrichment on the performance of the Advanced Neutron Source, formerly planned for operation at Oak Ridge National Laboratory. His current research involves energy resources, national security, and reliability of the national electrical grid. Dr. Bari has lectured internationally on risk assessment and nuclear safety and has authored more than 100 papers and key reports in these areas. Dr. Bari earned an A.B. in physics in 1965 from Rutgers University and a Ph.D. from Brandeis University in 1970.

Staff

Yuri Shiyan is the director of the Russian Academy of Sciences Office for North American Scientific Cooperation. He has worked in this capacity for more than 25 years, facilitating collaborative efforts and exchanges between international partners and Soviet/Russian scientists, engineers, and medical professionals. In 2004-2005, he served as IAEA expert for the Nuclear Fuel Subcommittee, and since 1981 he has served as the coordinator of the Russian Academy of Sciences Committee on International Security and Arms Control. For the past four years, he has served as coordinator of the RAS-NAS Committees on Counterterrorism and Non-Proliferation. Further, he has assisted in several joint U.S.–Russian projects focusing on various aspects of the nuclear fuel cycle, including the storage of nuclear spent fuel. His knowledge of English and professional experience gained through assignments at several international posts have contributed to his success as an international scientific liaison.

Sarah Case (Study Director) is currently senior program officer in the Nuclear and Radiation Studies Board of the National Research Council, where she has worked since 2007. She currently manages a portfolio of consensus studies and workshops focused on technical issues related to nuclear security and non-proliferation. Previous projects have focused on nuclear security but have also addressed issues related to nuclear energy, electrical transmission and distribution, and the health effects of radiation. Dr. Case received her Ph.D. in physics from the University of Chicago and her A.B. in physics from Columbia University.

Kevin D. Crowley is senior board director of the Nuclear and Radiation Studies Board (NRSB) at the National Research Council–National Academy of Sciences in Washington, D.C. He is responsible for managing the NRSB's work on nuclear safety and security, radioactive-waste management and environmental cleanup, and radiation health effects. He is also the principal investigator for a long-standing cooperative agreement between the National Academy of Sciences and the U.S. Department of Energy to provide scientific support for the Radiation Effects Research Foundation in Hiroshima, Japan. Dr. Crowley's professional interests and activities focus on safety, security, and technical efficacy of nuclear and radiation-based technologies. He has directed more than 20 National Research Council studies on these and other topics, including Safety and Security of Commercial Spent Nuclear Fuel Storage (2004, 2006); Going the Distance? The Safe Transport of Spent Nuclear Fuel and High-Level Radioactive Waste in the United States (2006); Medical Isotope Production without Highly Enriched Uranium (2009); America's Energy Future: Technology and Transformation (2009); and Analysis of Cancer Risks in Populations near Nuclear Facilities (in progress). Before joining the National Academies staff in 1993, Dr. Crowley held teaching/research positions at Miami University of Ohio, the University of Oklahoma, and the U.S. Geological Survey. He holds M.A. and Ph.D. degrees, both in geology, from Princeton University.

Appendix C

Statement of Task

The U.S. National Academies and Russian Academy of Sciences will organize a joint symposium to discuss progress, challenges, and opportunities for conversion of research reactors from highly enriched uranium to low enriched uranium fuel. The symposium will address the following topics:

- Recent progress on conversion of research reactors, with a focus on U.S.- and Russian-origin reactors.
- Lessons learned for overcoming conversion challenges, increasing the effectiveness of research reactor use, and enabling new reactor missions.
- Future research reactor conversion plans, challenges, and opportunities.
- Actions that could be taken by U.S. and Russian organizations to promote conversion.