International Conference on Fifty Years of Nuclear Power - the Next Fifty Years

27 June – 2 July 2004
Moscow/Obninsk, Russian Federation

Organized by the
International Atomic Energy Agency

Hosted by the
Government of the Russian Federation

Book of Extended Synopses

IAEA-CN-114

THE SYNOPSES PRINTED
IN THIS BOOK ARE LISTED
IN THE ORDER OF THE PROGRAMME

TIME TABLE

Sunday, 27 June 2004

12.00	Registration of Participants
15.00	Inaugural Session
17.00	Press Conference
18.00	Reception
20.00	Travel to Obninsk

Monday, 28 June 2004

13.00	Registration (cont'd)
14.00	Plenary Session 1
15.15	Coffee Break
15.45	Plenary Session 1
17.00	Round Table 1
18.00	Obninsk City/IAEA Reception

Tuesday, 29 June 2004

09.00	Media Panel
10.30	Coffee Break
11.00	Plenary Session 2
12.15	Lunch Break
14.00	Plenary Session 2
15.15	Coffee Break
15.45	Plenary Session 2
17.30	Round Table 2

Wednesday, 30 June 2004

09.00	Parallel Sessions A, B and D
10.30	Coffee Break
11.00	Parallel Sessions A, B, D and Panel B
12.00	Lunch Break

Wednesday, 30 June 2004 (p.m.)

14.00	Parallel Sessions A, C and D
15.00	Coffee Break
15.30	Parallel Sessions A, C and D
16.30	Panel D
17.00	Panels A and C

Thursday, 1 July 2004

09.00	Parallel Sessions E, F and G
10.30	Coffee Break
11.00	Parallel Sessions E, F and G
12.00	Lunch Break
14.00	Parallel Sessions E, F and G
15.00	Coffee Break
15.30	Parallel Sessions E, F and G
16.30	Panel G
17.00	Panels E and F
18.00	Reception

Friday, 2 July 2004

09.00	Plenary Session 3
10.30	Coffee Break
11.00	Round Table 3, Concluding Remarks
12.00	Lunch Break
14.00	Press Conference

<u>SUNDAY, 27 JUNE 2004</u>

Venue: Russian Academy of Sciences

12:00 – 15.00 **Registration of Participants**

15.00 – 17:00 **INAUGURAL SESSION**

17.00 – 18.00 **Press Conference**

18.00 – 20.00 **Reception**

20.00 **Departure for Obninsk**

<u>MONDAY, 28 JUNE 2004</u>

Venue: Conference Centre, Obninsk

13:00 **Registration of Participants (cont'd)**

14.00 – 17.00 **PLENARY SESSION 1** **INTERNATIONAL PERSPECTIVES**

Chairpersons: Bouchard, J., France;
Rumyantsev, A.Yu., Russia

No of Paper IAEA-CN-114/	*Name*	*Designating Member State/ Organization*	*Title of paper*
P1-1 14.00 – 14.25	Kochetkov, L.A.	Russia	World's First Nuclear Power Plant and its role in nuclear power development
F-1 14.25 – 14.50	**Rachkov, V.I.** Turin, A.V.	Russia	The role of nuclear power in Russia's power strategy
P1-4 14.50 – 15.15	Bouchard, J.	France	Nuclear Fuel Cycle and Sustainable Development: Strategies for the Future
15:15 -15:45		**Coffee Break**	
P1-3 15.45 – 16.10	Sokolov, Y.A.	IAEA	The IAEA's Role in Nuclear Energy – Past, Present and Future
P2-3 16.10 – 16.35	Maeda, H.	Japan	Nuclear Energy in Japan, Current Status and Future
P1-5 16.35 – 17.00	Johnson, S.	USA	U.S. Perspectives on Nuclear Energy

17:00 – 18:00 **ROUND TABLE 1:** **INTERNATIONAL PERSPECTIVES**

Moderator:	Bouchard, J	France
Members:	Sokolov, Y.A.	IAEA
	Maeda, H.	Japan
	Chang, I. S.	Korea Rep. of
	Kochetkov, L.A.	Russia
	Rachkov, V.I.	Russia
	Johnson, S.	USA

18:00 **Reception hosted by the City of Obninsk and IAEA**

TUESDAY, 29 JUNE 2004

09.00 – 10.30 **MEDIA PANEL on Media Perceptions of Nuclear Power**

Moderator:	Von Randow, G.	Die Zeit, Gemany
Members:	Kempf, H.	Le Monde, France
	Berlinck, D.	O Globo, Brazil
	Hong Choi, Y.	The Korean Times
	Kirby, A.	BBC, UK
	Charbonneau, L.	Reuters News Agency, Austria

10.30 – 11.00 **Coffee Break**

11.00 – 17.00 **PLENARY SESSION 2 History and Future of Nuclear Power**

Chairpersons: Cirimello, R.O., Argentina;
Saraev, O.M., Russia

No of Paper IAEA-CN-114/	*Name*	*Designating Member State/ Organization*	*Title of paper*
P2-11 11.00 – 11.25	Saraev, O.M.	Russia	Status and Prospects of Russian Nuclear Power Programme
P2-5 11.25 – 11.50	Kakodkar, A.	India	Nuclear Energy in India - Retrospect and Prospects
P2-10 11.50 – 12.15	Brockman, K.	IAEA	IAEA Safety Standards and Future Developments
12.15 – 14.00		**Lunch Break**	
P2-2 14.00 – 14.25	**Steinberg, N.,** Babak, M., Bronnikov, V., Kopchinsky, G., et al	Ukraine	Nuclear Power of Ukraine - History, Present Status and Prospects
P2-9 14.25 – 14.50	Han, P.S.	Korea Rep. of.	Nuclear Power in Korea, Current Status and Future
P2-6 14.50 – 15.15	Yu Zhuoping	China	Nuclear Power in China: Current Status and Future
15:15 -15:45		**Coffee Break**	
P2-4 15.45 – 16.10	**Cirimello, R.O.,** Ward, A., Bergallo, J., Taboada, H.	Argentina	Past Present and Future of Nuclear Energy and Nuclear Fuel Cycle in Argentina
P2-8 16.10 – 16.35	**Soetrisnanto**, **A.Y.,** Suharno, Hastowo, H., Soentono, S.	Indonesia	Status of Nuclear Energy Program in Indonesia
P2-7 16.35 – 17.00	Barré, B.	France	The Nuclear Reactors from a "Natural History" Perspective
P2-12 17.00 – 17.25	Marchese, C. J.	UK	UK Nuclear History and Thoughts about the Future

17.30 – 18.30 **ROUND TABLE 2:** **Lessons Learnt and Requirements for Future Nuclear Power Growth**

Moderator:	Cirimello, O.	Argentina
Members:	Yu Zhuoping	China
	Barré, B.	France
	Brockman, K.	IAEA
	Kakodkar, A.	India
	Soetrisnanto, A.Y.	Indonesia
	Han, P.S.	Korea Rep. of.
	Saraev, O.M.	Russia
	Marchese, C.J.	UK
	Steinberg, N.	Ukraine

WEDNESDAY, 30 JUNE 2004

09.00 – 17.00 **PARALLEL SESSION – A** **Operating Experience and Planned Projects**

Chairpersons: Dieguez, J.A.D., Brazil;
Kochetkov, L.A, Russia

No of Paper IAEA-CN-114/	*Name*	*Designating Member State/ Organization*	*Title of paper*
A-7 09.00 – 09.30	Omoto, A.	IAEA	IAEA Perspective on Future of Nuclear Power
A-2 09.30 – 10.00	**Oshkanov, N.N.**, Poplavski,V.M.	Russia	Operating Experience of BN-600 Fast Neutron Reactor and BN-800 Reactor Design
A-10 10.00 – 10.30	**Guidez, J.**, Martin, L.	France	PHENIX: Thirty Years of Operation for Research, Reactor Renovation Overview and Prospects
10.30 – 11.00		**Coffee Break**	
A-4 11.00 – 11.30	Gevorgyan, A. A.	Armenia	Perspectives of Nuclear Energy Further Development in Armenia
A-5 11.30 – 12.00	**Dieguez, J.A.D.** Barroso, A.C.O.	Brazil	Nuclear Power Energy Development in Brazil - Future Perspectives
12:00 -14:00		**Lunch Break**	
A-1 14.00 – 14.30	Abagyan A.A	Russia	Experience gained with inspection of metal state in thermal and mechanical equipment for extending operation time of power units of the first generation NPPs
A-6 14.30 – 15.00	Holz, R.	Germany	20 Years of Experience in Steam Generator Replacement
15:00 -15:30		**Coffee Break**	
A-9 15.30 – 16.00	Yan Jiapeng	China	Programme of Self Assessment and Corrective Actions at Qinshan NPP
A-3 16.00 – 16.30	Dragunov Yu, **Toshinsky,G.I.**	Russia	Lead-bismuth Reactor Technology Conversion: from Nuclear Submarine Reactors to Power Reactors and Ways of Increasing the Investment Attractiveness of Nuclear Power Based on Fast Reactors
A-8 16.30 – 17.00	Roche, B.	France	The Continuous improvement of nuclear safety and competitiveness

17.00 – 18.00 **PANEL – A: Operating Experience and Planned Projects**

Moderator:	Dieguez, J.A.D.	Brazil
Members:	Gevorgyan, A. A.	Armenia
	Yan Jiapeng	China
	Roche, B.	France
	Holz, R.	Germany
	Omoto, A.	IAEA
	Abagyan, A.A.	Russia

WEDNESDAY, 30 JUNE 2004

09.00 – 10.40 **PARALLEL SESSION – B** **Non-Electricity Developments**
(Meeting Room 2)

Chairpersons: Shukla, D.S., India;
Baranaev, Yu.D., Russia

No of Paper IAEA-CN-114/	*Name*	*Designating Member State/ Organization*	*Title of paper*
B-2 09.00 – 09.20	Baranaev, Yu.D., Mishanina, Yu.A., **Romenkov, A.A.,** Lisitsa, F.D.	Russia	The use of pool type nuclear reactor RUTA for municipal district heating and desalination of seawater
B-5 09.20 – 09.40	**Shukla**, **D.S.,** Tewari, P.K.	India	Advances in Nuclear Desalination in BARC, India
B-1 09.40 – 10.00	**Kostin, V.I.**, et al	Russia	The Results of Russian Nuclear Civil Ships Operation and the Experience of their Specified Lifetime Extension
B-4 10.00 – 10.20	**Martins Jr**, **L.,** Dutra Filho, Z., Qassim, R.Y.	Brazil	Optimal Design of Nuclear Desalination Multiple Effect Evaporation Process Plant
B-3 10.20 – 10.40	**Ponomarev-Stepnoy,** **N.N.**, Stolyarevsky, A.Ya.	Russia	Major aspects of strategy of hydrogen-base power development with nuclear energy sources

10.40 – 11.00 **Coffee Break**

11.00 – 12.00 **PANEL – B: Non-Electricity Developments**

Moderator:	Shukla, D.S.	India
Members:	Martins Jr, L.	Brazil
	Baranaev, Yu.D.	Russia
	Kostin, V.I.	Russia
	Ponomarev-Stepnoy, N.N.	Russia

12.00 – 14.00 **Lunch Break**

WEDNESDAY, 30 JUNE 2004

14.00 – 17.00 **PARALLEL SESSION – C** **Public Communication**
(Meeting Room 2)

Chairpersons: Bazille, F., France;
Ivanov, A.P., Russia

No of Paper IAEA-CN-114/	*Name*	*Designating Member State/ Organization*	*Title of paper*
C-2 14.00 – 14.30	Bazille, F.	France	How to Achieve Greater Acceptance of Nuclear Industry
C-5 14.30 – 15.00	Shepherd, J.	UK	The Importance of International Nuclear Communications
15:00 -15:30	**Coffee Break**		
C-1 15.30 – 16.00	Prasertwigai, T.	Thailand	How to Present the Technical Messages to the Public
C-3 16.00 – 16.30	Sandquist, G.	USA	Revisiting the Nuclear Enterprise
C-4 16.30 – 17.00	Von Randow, G.	Die Zeit, Germany	A newspaper is not a heat exchanger: thoughts about media and nuclear power

17.00 – 18.00 **PANEL – C: Public Communication**

Moderator:	Bazille, F.	France
Members:	Von Randow, G.	Die Zeit, Germany
	Prasertwigai, T.	Thailand
	Shepherd, J.	UK
	Sandquist, G.	USA

WEDNESDAY, 30 JUNE 2004

09.00 – 17.00 **PARALLEL SESSION – D (Meeting Room 3)** **Nuclear Safety and Security Developments**

Chairpersons: Teller, A., Belgium; Bagdasarov, Yu.E., Russia

No of Paper IAEA-CN-114/	*Name*	*Designating Member State/ Organization*	*Title of paper*
D-9 09.00 – 09.30	Hagemann, A.	IAEA	International Nuclear Security
D-4 09.30 – 10.00	**Grenèche, D.,** Cazalet, J., Delaune, Ph.	France	Proliferation Resistance Assessment: An Illustration through the French Fuel Cycle
D-1 10.00 – 10.30	Malyshev, A.B.	Russia	Current Status and Development Perspectives of State Nuclear and Radiation Safety Regulation in the Russian Federation
10.30 – 11.00		**Coffee Break**	
D-5 11.00 – 11.30	**Choi, K. W.,** Jo, J.C., Ann, S.K., Kim, H.J.	Korea, Rep.of	Regulatory Approach for the Lifetime Management of the Nuclear Power Plants in Korea
D-2 11.30 – 12.00	**Sidorenko, V.A.,** Asmolov, V.N.	Russia	Nuclear Power Safety: The Present and Safeguards for the Future
12:00 -14:00		**Lunch Break**	
D-6 14.00 – 14.30	Teller, A.	Belgium	A Tale of two Theories
D-3 14.30 – 15.00	**Grinev, M.P.,** Ilyin, L.A., Kochetkov, O.A.S., et al	Russia	Radiation Safety for Personnel and Public in the Use of Nuclear Power. Experience gained, Problems, and Ways to resolve them
15:00 -15:30		**Coffee Break**	
D-7 15.30 – 16.00	Yang, M.S., **Park, S.W.**, Park, H.S.	Korea, Rep.of	Prospect of the Proliferation Resistant Fuel Cycle Technology Development in Korea
D-8 16.00 – 16.30	Pu Jilong	China	Safety Culture - the Way of Thinking

16.30 – 18.00 **PANEL – D: Nuclear Safety and Security Developments**

Moderator:	Teller, A.	Belgium
Members:	Pu Jilong	China
	Greneche, D.	France
	Hagemann, A.	IAEA
	Choi, K.W.	Korea, Rep. of.
	Park, S.W.	Korea, Rep. of.
	Bagdasarov, Yu.E.	Russia

18.00 – 19.00 **POSTER SESSION (see list of posters at the back)**

THURSDAY, 1 JULY 2004

09.00 – 17.00 **PARALLEL SESSION – E** **Design and Development of Advanced Nuclear Systems**

Chairpersons: Park, J.K., Korea Rep. of;
Toshinsky, G.I., Russia

No of Paper IAEA-CN-114/	*Name*	*Designating Member State/ Organization*	*Title of paper*
E-1 09.00 – 09.30	**Dragunov, Yu. G.,** Ryzhov, S.B., Denisov, V.P.	Russia	Evolution of WWER reactor plants for nuclear power plants
E-8 09.30 – 10.00	**Park, J.K.** Song, C.H.	Korea, Rep. of	Thermal-Hydraulic Experiments and Evaluation for Safety Features in APR1400
E-2 10.00 – 10.30	**Kodochigov, N.G.,** Kostin, V.I., Ponomarev-Stepnoi, N.N., Shenoy, A., et al	Russia/USA	GT-MHR International project of high-temperature helium cooled reactor with direct gas-turbine power conversion cycle
10.30 – 11.00		**Coffee Break**	
E-6 11.00 – 11.30	Bonhomme, N.	France	The European Pressurized Water Reactor, a Safe and Competitive Solution for Future Energy Needs
E-7 11.30 – 12.00	**Berbey, P.**, Rousselot, O.	France	European Utility Requirements: Common Rules to Design next LWR Plants in an Open Electricity Market
12:00 -14:00		**Lunch Break**	
E-3 14.00 – 14.30	Velikhov, Y.P.	Russia	ITER – the first experimental thermonuclear reactor
E-10 14.30 – 15.00	**Carré, F.**, Bernard, F.	France	R&D for the future nuclear systems: stakes, challenges and international cooperation
15:00 -15:30		**Coffee Break**	
E-5 15.30 – 16.00	**Poplavski, V.M.,** Vasiliev, B., Ershov, V.	Russia	Perspective Sodium Fast Reactor BN-1800
E-9 16.00 – 16.30	**Steur, R.,** Kupitz, J.	IAEA	International Cooperation for the next 50 Years: The International Project on Innovative Reactors and Fuel Cycles (INPRO)
E-4 16.30 – 17.00	Orlov, V.V.	Russia	Nuclear contribution to the power industry in the 21st century: Electricity, Fast Reactors, BREST Concept

17.00 – 18.00 **PANEL – E: Design and Development of Advanced Nuclear Systems**

Moderator:	Park, J.K.	Korea, Rep.of
Members:	Carré, F.	France
	Czech, J.	Germany
	Steur, R.	IAEA
	Poplavski, V.M.	Russia
	Velikhov, Y.P.	Russia
	Orlov V.V.	Russia

THURSDAY, 1 JULY 2004

09.00 – 17.00 **PARALLEL SESSION – F (Meeting Room 2)** **Nuclear Energy for Sustainable Development and an environmental future**

Chairpersons: Chidambaram, R., India; Ousanov, V.I., Russia

No of Paper IAEA-CN-114/	*Name*	*Designating Member State/ Organization*	*Title of paper*
F-2 09.00 – 09.30	**Bolshov, L.A.,** Arutyunyan, R.V., Linge, I.I.	Russia	The Environmental Risks and Prospects for Large-scale Nuclear Power Development
F-3 09.30 – 10.00	Saint-Pierre, S.	France	Radiological Impacts from Nuclear Industrial Facilities on the Public and the Environment - Their Current Magnitude and the next 50 years Forecast
F-6 10.00 – 10.30	Grover, R.B.	India	Technology and Knowledge Management in and by the Department of Atomic Energy, India
10.30 – 11.00		**Coffee Break**	
P2-1 11.00 – 11.30	Belyaev, L.S.	Russia	Russia's Energy Resources in the 21st century with account for their role in the World Energy
F-4 11.30 – 12.00	**Grenèche**, **D.,** Capus, G., Nigon, J.L.	France	A key Point for Nuclear Power Development, Sustainability of Uranium Resources
12:00 -14:00		**Lunch Break**	
F-7 14.00 – 14.30	Vyshnevskyi, I.M.	Ukraine	Role of Nuclear Energy in Ukraine
G-10 14.30 – 15.00	Proust, E.	France	Nuclear Energy: Sustainability and Economic Competitiveness
15:00 -15:30		**Coffee Break**	
F-8 15.30 – 16.00	**Clark, C.R.**, Kazennov, A., Kossilov, A., Mazour, T., Yoder, J.	IAEA	Achieving excellence in human performance through leadership, education, and training in nuclear power industry
F-9 16.00 – 16.30	Jinchuk, D.O.	Argentina	Power Generation Alternatives for the XXI Century
F-10 16.30 – 17.00	Sarici, E.L.	Turkey	The perceived benefits for Turkey adopting nuclear energy, and the lessons learned from the attempts made to introduce a nuclear programme

17.00 – 18.00 **PANEL – F: Nuclear Energy for Sustainable Development and an environmental future**

Moderator:	Chidambaram, R.	India
Members:	Jinchuk, D.O.	Argentina
	Saint-Pierre, S.	France
	Clark, C.R.	IAEA
	Grover, R.B.	India
	Bolshov L.A.	Russia
	Sarici, E.L.	Turkey
	Vyshnevskyi, I.M.	Ukraine

THURSDAY, 1 JULY 2004

09.00 – 17.00 **PARALLEL SESSION – G** **Nuclear fuel cycle and waste management**
(Meeting Room 3)

Chairpersons: Mathieson, J., UK;
Tsiboulia, A.M., Russia

No of Paper IAEA-CN-114/	*Name*	*Designating Member State/ Organization*	*Title of paper*
G-13 09.00 – 09.30	Volk, V.I., **Vatulin, A.V.,** Vakhrushin, A.Yu., Poluektov, P.P., Sukhanov, L.P.	Russia	New Radiochemical Technologies of Reprocessing of Spent Nuclear Fuel
G-9 09.30 – 10.00	Risoluti, P.	IAEA	The Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management. A move toward a global safety
G-8 10.00 – 10.30	**Afanasyev, A.,** Gromok, L., Pavlenko, V., Steinberg, N.	Ukraine	Radioactive Waste Management in Ukraine: Status, Problems, Prospects
10.30 – 11.00		**Coffee Break**	
G-12 11.00 – 11.30	Mathieson, J.	United Kingdom	Achieving Legitimacy in the UK's Radioactive Waste Management Programme
G-6 11.30 – 12.00	**Gillet, P.,** Yoneda, M., Ohtou, M., Arslan, M., Gillet, M.	France	JNFL and COGEMA: A Wide and Innovative Cooperation in the Fields of Spent Fuel Reprocessing and MOX Fuel Fabrication
12:00 -14:00		**Lunch Break**	
G-7 14.00 – 14.30	Van der Sanden, J.	Germany	More than Twenty Years of Storage Experience with Casks
G-5 14.30 – 15.00	Paliukhovich, V.M.	Belarus	Safe Storage of Spent Fuel From Mobile NPP
15:00 -15:30		**Coffee Break**	
G-1 15.30 – 16.00	**Mayorshin, A.A.,** Grachev, A.F., Skiba, O.V., et al	Russia	Power Reactor Fuel Cycle Based on Non-aqueous Technologies
G-11 16.00 – 16.30	Sicard, B., et al.	France/Russia	Use of Weapons-Grade plutonium from dismantled nuclear weapons for the peaceful objective of electric power generation - AIDA/MOX 2

16.30 – 18.00 **PANEL – G: Nuclear fuel cycle and waste management**

Moderator	Mathieson, J	UK
Members:	Paliukhovich, V.M.	Belarus
	Gillet, P.	France
	Van der Sanden, J.	Germany
	Risoluti, P.	IAEA
	Vatulin A.V.	Russia
	Mayorshin, A.A.	Russia
	Afanasyev, A.	Ukraine

FRIDAY, 2 JULY 2004

09.00 – 10.30 **PLENARY SESSION – 3** **Reports from Panel Discussions A-F**

Chairpersons: Cirimello, R.O., Argentina;
Poplavsky, V.M., Russia

10.30 – 11.00 **Coffee Break**

11.00 – 12.00 **ROUND TABLE 3** **Measures for a Healthy Growth of Nuclear Power**

Moderator:	Cirimello, R.O.	Argentina
Members:	Yu, Zhuoping.	China
	TBN	France
	Sokolov, Y.A.	IAEA
	Kakodkar, A.	India
	TBN	Japan
	Park, J.	Korea, Rep. of.
	Ilyin, L.A.	Russia
	TBN	USA

12.00 **Concluding remarks**

Closing of Conference

14.00 **Press Conference**

POSTER PRESENTATIONS: Wednesday, 30 June 2004, 17.00 – 18.00

No of Paper IAEA-CN-114/	*Name*	*Designating Member State/ Organization*	*Title of paper*
1p	**Falcao Martins**, **N. S.,** Peres, S.S.	Brazil	Estimated Source Terms and Operational Experience Applied to Angra 1 NPP
2p	Solignac, Y., **Rehbinder, S.**	France	Spent Fuel Transport: a Continuous Improvement
4p	Balcytis, C.	Lithuania	Tendencies of Occupational Exposure at Ignalina NPP
5p	Cesna, B.	Lithuania	A Historical Survey of the Ignalina NPP
6p	Jovanovic, S.	Serbia & Montenegro	Nuclear Power - A View from Small "Non-Nuclear" Country
7p	**Mirsaidov, U.,** Salomov, J., Hakimov, N.	Tajikistan	From History of Reception of Native Uranium
8p	**Babenko**, **V.V.,** Kazimirov, A.S.	Ukraine	Control of Inhalation Component of Human Internal Irradiations by Means of Whole Body Spectrometer "SICH-EXPRESS"
9p	**Kazimirov**, **A. S.,** Babenko, V.V., Isaev, A.G.	Ukraine	Methods and Instrumentation for Measurement of Radioactive Waste
10p	Courtois, Ch.	France	HLILW Management: Stakes for the Future
11p	Coe, R. P.	USA	Effectively Managing Nuclear Risk Through Human Performance Improvement
12p	Gurbanov, M.	Azerbaijan	Hydrogen Radiolytic syntesis by chain decomposition on hydrogen sulphide
13p	**Dragunov**, **Yu. G.,** Ryzhov, A.B., Denisov, V.P.	Russia	Reactor plant WWER-1500 for nuclear power plants
14p	**Samoilov, O.B.** Lavkovsky, S.A. Baranaev, Y.D., Shadrin, A.P.	Russia	Floating NPPs based on ship propulsion rector technologies - reliable and safe energy source for autonomas energy supply
15p	**Ivanov, A.P.,** Khorasanov, G.L., Blokhin, A.I., Demin, N.A.	Russia	Chromium-nickel steels depleted of nickel stable isotope Ni-58 as a material for fast reactor claddings
16p	Aksenov, P.M.	Russia	Fabrication of nuclear fuel at OAO MSZ. Current status and prospects for development

No of Paper IAEA-CN-114/	*Name*	*Designating Member State/ Organization*	*Title of paper*
17p	Drozdova, S.V.	Russia	Transportation of fresh nuclear fuel from OAO "Mashinostroitemny Zavod"
18p	**Sidorova, E.,** Asatiany, I.	Russia	Contribution of OAO MSZ into nuclear fuel production and performance results
19p	Eniushina, E.A.	Russia	VVER-1000 fuel rods manufactured by OAO MSZ
20p	**Kostin, V.I.,** Ponomarev-Stepnoy, N.N.	Russia	VBER-300 Reactor Plant on the Basis of Ship Technologies
21p	**Arzhaev, A.I.,** Evropin, S.B., Petrov, A.A.	Russia	Ensuring of NPP with RBMK-1000 MCC components safe operation based on Break Preclusion Conception
22p	**Kuznetsov, Yu. N.,** Gabaraev, B.A., Shishkin, V.A.	Russia	The use of nuclear energy for district heating. The Branch Program of activities. Nikiet Design Efforts on the Advanced nuclear co-generation plant with VK-300 reactor, the Ruta nuclear heating plant and small power units
23p	**Bychkov, A.V.,** Lykinykh, A.N., Babikov, L.G., Lavarinovich, Yu. G.	Russia	Processing and Immobilization of Pyrochemical Residues and PU-containing Waste
24p	**Grachyov, A.F.,** Ochnnikov, V.A.	Russia	The main directions in testing fuel rods for improved NPP fuel cycles in the Mir Reactor
25p	**Efimov, V.N.,** Korolkov, A.S., Mayorshin, A.A.	Russia	BOR-60 reactor as an instrument for experimental substantiation of fuel rods for advanced NPPs
26p	**Pavlov, D.V.,** Grigoriev, A.S., Kosourov, K.S.	Russia	Technical Proposal on development and outfitting of NPP having WWER-1000 reactor with fission materials monitoring system
27p	**Bykov, V.P.,** Bogoyavlenskii, R.G., Grishanin, E.I.	Russia	Perspectives of safe and cost effective APP power unit creation with a shell-type micro fuel elements reactor under supercritical pressure of light-water of the heat transfer medium
28p	**Dragunov, Yu. G.** Ryzhov, S.B. Rogov, A.M.	Russia	Advanced reactor plant WWER-1000
29p	**Dragunov, Yu. G.,** Boucharov, L.A, Belousov, V.D.	Russia	Design and delivery of the equipment for the first NPP in the world

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30p	**Dragunov, Yu. G.,** Stepanov, V.S., Zrodnikov, A.V.	Russia	Conversion of Lead-bismuth Reactor Technology: from NPS Reactors to Power Reactors
31p	**Delnov, V.N.,** Gabrianovich, B.N.	Russia	The Hydrodynamic Features in the Header Systems of Tank Reactors and Heat Exchangers
32p	**Nikolotov, A.M.,** Vikulov, V.K., Yermoshin, F. Ye.	Russia	A Concept of a Pressure-tube Power Reactor with Supercritical Water Coolant
33p	Toporov, Yu. G.	Russia	Production of radionuclide preparations and ionizing sources of medical use
34p	**Briskman, B.A.,** Borson, E.N.	Russia	ISO CD 1856. Guideline for Radiation Exposure of Nonmetallic Materials
35p	**Kascheev, M.V.,** Kuznetsov, I.A.	Russia	Computation analysis of fuel melt confinement processes in fast reactors
36p	**Petrova, E.V.,** Mescheryakov, V.I., Zakharova, E.V., Silin, B.G.	Russia	Studies on radioactive contamination of spent reactor graphite from PUGRs of Siberian Group of Chemical Enterprises with the purpose of further management planning
37p	**Portjanoj, A.G.,** Voznesenski, R.M., Vyunnikov, N.V.	Russia	Development of passive devices for emergency protection of fast reactors
38p	**Vigovsky, S.B.,** Strashnych, V.P., Bogachek, L.N.	Russia	The utilization of the program complex PROSTOR in calculation investigations concerning the applicability of coolant natural circulation regime with the expanded parameters scale of NPP with VVER-1000 providing violation of normal operation regimes
39p	**Kalin, B.A.,** Solonin, M.I., Konovalov, I.I.	Russia	Prospective materials and technologies of new materials for Atomic Power Engineering
40p	**Zabud'ko, A.N.,** Zrodnikov, A.V., Ionkin, V.I.	Russia	Space Nuclear Power in Views: 50 Years ago and Prevision for 50 Years
42p	**Ryabov, N.O.,** Semenov, A.A.	Russia	Algorithm of Spatial Xenon Oscillations Identifications
43p	**Semenov, A.A.,** Vigovsky, S.B., Tchernacov, V.A. Schukin, N.V.	Russia	Applications of the "PROSTOR" WWER Core Simulator
44p	**Parisi, C.** Auria, F. D’,Mazzini, M., Sollima, C.	Italy	An overview of the TACIS project (R2.03/97) dealing with WWER-1000 and RBMK technologies

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45p	Dolgov, V.V.	Russia	Bilibino NPP: Thirty-Year Operating Experience and Lifetime Extension
46p	Filippov, G.A., **Grishanin, E.I.,** Koukharkin, N.E. Tsiklauri, G.V., et al	Russia	Perspectives in development of a vessel-type nuclear reactor with once-through steam superheating
47p	Ivanov, E.A.	Russia	Formalization of systematic analysis of innovative development of nuclear engineering
48p	Lebedev, V.M.	Russia	Nuclear fuel cycle in Russia - Status and Perspectives

PLENARY SESSION 1

INTERNATIONAL PERSPECTIVES

WORLD'S FIRST NPP AND ITS ROLE IN NUCLEAR POWER DEVELOPMENT

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On 26th of June 2004, it will be 50 years anniversary of start-up of the World's First Nuclear Power Plant (FNPP). This small size nuclear power plant that was constructed on the site of Laboratory "V" (now State Scientific Center of the Russian Federation - Institute for Physics and Power Engineering) has become a symbol of peaceful use of nuclear energy.

As long ago as before the Second World War and during the War, proposals on the use of nuclear energy for peaceful purposes including electricity production, were made by Academicians V.I.Vernadsky, S.I.Vavilov and P.L.Kapitsa. Based on these proposals, a Decree was issued by the Council of Ministers of the USSR in December 1946, in which the objectives of peasuful application of nuclear energy were first stated. However, target-oriented, intensive work on the designs of power reactors began after successful nuclear weapon test in 1949. Academician I.V.Kurchatov, manager of Nuclear Project of the USSR, became scientific supervisor of this work, while L.P.Beriya, Head of Special Committee and B.L.Vannikov, Head of the First General Board were administrative leaders.

In May 1950, the Governmental Decree was issued on the construction at Laboratory "V" of "V-10" facility including three reactor units and separate machine building with of 5000 kW turbine generator. AM facility, the first nuclear power plant, was determined as the first priority. Initial evaluation of AM reactor neutronics was performed at Laboratory 2 under the leadership of I.V.Kurchatov, reactor design was developed at the SKB-5 NIIKHIMMASH under the leadership of Professor N.A.Dollezhal, and the LPI (A.I.Gutov) was the General Architect of the Project. On the proposal of I.V.Kurchatov, in March 1951, Laboratory "V" became the scientific supervisor of justification of the First NPP physical characteristics.

The First NPP was designed and constructed in 4 years and a half. Under condition of permanent lack of time and hard administrative control, it was necessary to solve very difficult problems related to the design of the fuel elements, reactor channels, the main components, control and safety system, justification of the core neutronics and reactor safety. Reactor construction was started on the stage of its design, nevertheless, the Government had to postpone trice the NPP start-up. On 9 May 1954, AM reactor reached criticality. On 26 June 1954, the turbogenerator of the First NPP was connected to Mosenergo grid. This day was acknowledged as a birthday of Nuclear Power. The main problem of the early stage of operation was caused by the water leaks into the graphite brickwork of the reactor heated up to 500^0C - 600^0C through the cracks in the tubes of the fuel and control rod channels, where coolant (water) pressure was 10 MPa. Another problem was caused by numerous shutdowns of the reactor by the safety system due to false signals of the system of the coolant flow rate control in the fuel channels. Reactor was shut down for repair of the equipment and after its completion, rated power was reached on October 25, 1954.

Later on, experimental studies were started on the First NPP reactor in addition to power generation. Comprehensive experimental studies were conducted on the reactor for justification of the future NPP designs, as well as special purpose facilities including studies on the fuel element characteristics and plant transients. Results of these studies were used in the designs of power units of Beloyarskaya NPP with in-reactor steam superheaters, Bilibinskaya NPP with natural coolant flow, transportable NPP "TES-3", icebreaker "Lenin" and reactor of the space unit "Topaz". New coolants, structural materials and control systems were tested in this reactor, studies on reactor neutronics and solid body physics were carried out, and production of different isotopes was arranged.

Training of specialists has been important activity field of the FNPP staff, as well as the other IPPE employees. Future staff of the Beloyarskaya and Novo-Voronezhskaya NPP, icebreaker "Lenin", the first teams of the nuclear submarines, specialists from GDR, CPR, CSSR, SRR were trained at the FNPP. Starting from the first days of its operation, FNPP became accessible for the numerous delegations from our country and from abroad.

Significant political resonance is not the only result of the FNPP creation; it makes the concrete scientific and technical contribution in the solution of the problems of Nuclear Power, being powerful stimulus of development and implementation of the programs of the NPP creation in many countries, including USSR.

Shortly after FNPP start-up, Sibirskaya NPP, transportable NPP "TES-3", two units of the Beloyarskaya NPP, two units of Novo-Voronezhskaya NPP, NPP VK-50, and BOR-60 in Dimitrovgrad were created.

Of course, on the stages of creation and operation of the FNPP, numerous defects of the equipment and errors of personnel were not avoided, however there was no dangerous cases of personnel overdose, and radioactivity release to the environment including the town located at 1.5-3.5 km distance from NPP, has been within natural background value. On 29 April 2002, the reactor of the FNPP was stopped. Preparation work for its decommissioning was started and the experience of this work will become additional important result of operation of the World's First NPP.

IAEA-CN-114/F-1

THE ROLE OF NUCLEAR POWER IN RUSSIA'S POWER STRATEGY

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Nowadays, nuclear power is the most rapidly developing sector of the Russian power industry. The growing power demand in 1999-2003 (about 17 billion kW·h annually) was partly (50%) covered by the increase of NPP-generated power at the annual rate of 9 billion kW·h, or 5%.

The role of nuclear power in Russia's fuel & power complex (FPC) is very important: it acts as a system-forming, fuel-balancing, tariff-stabilizing, and environment-protecting element.

The primary tasks in the advance of national power energetics are as follows:

- optimization of the fuel & power balance including, in particular, the development of NPP, gas substantiation and increased share of coal at thermal power plants (TPP);
- compensation for NPP and TPP with the expired service period (extension of service life, reproduction);
- 50% cover of power demand growth by NPP-generated power;
- enhanced efficiency of investment in FPC by creating such power complexes as NPP – hydro accumulating power plant, electric heat accumulation and nuclear heat plant (NHP), energobio complexes of low-grade heat, NPP - aluminum production;
- reduced power generation costs, in particular, through the optimal use of generating capacities (at the maximal loading of NPP);
- reduced man-caused impact on the environment.

The optimal scenario of the Russian economy development envisages the growth of NPP-generated power to 200 billion kW·h by 2010 and to 300 billion kW·h by 2010. In this case, the NPP total installed capacity shall reach 28 GW by 2010, and 40 GW by 2020, the capacity factor being up to 85%.

The enhancement of nuclear power efficiency shall be achieved through the use of both inner (reduced production costs) and outer (expansion of atomic energy sales markets) reserves.

Additional growth of NPP production shall be boosted by arranging heat supply to Balakovo, Volgodonsk, and Kursk, as well as by constructing NHP in such large towns as Archangelsk, Voronezh, etc.

Enhanced efficiency of capital investment is the key question for nuclear power competitiveness. Depreciation charges, profits, and credits are considered the basic investment sources.

NUCLEAR FUEL CYCLE AND SUSTAINABLE DEVELOPMENT: STRATEGIES FOR THE FUTURE

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The decision to build a nuclear programme implies to have a long-term vision. On the other hand, such a long-term vision is also required by the tension on natural resources and the necessity to prevent climate change.

Since its beginning, the French nuclear programme has been situated in a long-term perspective, particularly with the closed-cycle choice, then with the researches on fast breeder reactors, began in 1967 with the research reactor Rapsodie.

By now this perspective continues to drive our strategy of research and development, with the involvement of France in the "Generation IV Initiative Forum", to make nuclear power a main contributor to sustainable development. The "GIF" has defined criterias to answer to the long-term and worldwide nuclear expansion challenges. These challenges can be summed-up as follows:

- Enhance public confidence in nuclear safety,
- Increase economical competitivity of nuclear systems (cost advantage over other energy sources)
- Address the optimal use of limited space in geological disposal and achieve the benefits of a closed fuel cycle: minimizing the final waste volumes, saving the natural resources,
- Increase physical protection against terrorism and unattractiveness of the new systems for the nuclear proliferation.

The importance of sustainable development criterias has led to select for the GIF programmes a majority of closed-cycle and fast neutron spectra systems.

French nuclear sector has a great expertise about the sodium fast reactors, with Phenix and even Superphenix, which has permitted to reach important R&D results. These results are useful in the frame of the GIF R&D programmes.

Today the researches go on to improve the recycling technologies and to reduce the volume and the radiotoxicity of waste: in 2001 the CEA has demonstrated the feasibility of partitioning of minor actinides, and the future programmes encompass technological demonstration of partitioning.

The feasibility of transmutation in fast neutron reactors was already demonstrated. So, the XXIst century will probably have to appeal to evolutive recycling systems and will have to invent the best combination between the actual technologies and the integral recycling aimed by the GEN IV systems.

In parallel, the French nuclear researches are focused on the gas-cooled reactor technologies, with the VHTR and the GFR, which belong to the GEN IV selection. The advantages of the GFR system is to permit an homogeneous recycling of actinides with a benefit of fast breeding generation. On the other hand, the interest of VHTR system (with thermal spectrum and open fuel cycle) is to answer to the future needs, like hydrogen production by thermochemical process, or desalination by co-generation.

Anyway, in the XXIst century one will have to manage in France but elsewhere in developed and developing countries, the combination of several "generations" of nuclear systems (light water reactors, fast spectra reactors…) and we have to imagine and to plan the best distribution of these technologies in space and in time.

IAEA ROLE IN NUCLEAR ENERGY – HISTORY AND FUTURE

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The first section of the paper reviews the history of nuclear power and the history of the IAEA as it relates to nuclear power. The paper describes the range of technological initiatives in the 1950s and 1960s, the rapid growth in global nuclear electricity generation in the 1970s and 1980s and the subsequent slowdown to match the pace of overall global electricity growth in the 1990s. Agency initiatives in safety, small and medium size reactors, and nuclear desalination of seawater are highlighted. The paper's second section summarizes the status of nuclear power around the world today, including both operating plants and those under construction, and emphasizes the global buildup of spent fuel and progress on long-term waste repositories. The third section compares medium term projections for nuclear power with two long-term sustainable development scenarios. It explores possible reasons for the divergence between the medium term and long term results, what might be done to close the gap and how the Agency can and does contribute.

NUCLEAR ENERGY IN JAPAN, CURRENT STATUS AND FUTURE

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Based on the principle of peaceful use only, Japan has pursued development of nuclear power for the last fifty years and now fifty-two reactors are in commercial operation with a total installed capacity of forty-five GW. Nuclear has established itself as one of the major sources of energy production, supplying one-thirds of electricity generation. In order to better utilize uranium resources and to enhance independence of energy supply, Japan has also embarked on development of nuclear-fuel-cycle program as a national policy. Progresses have already been achieved in some fields of fuel cycle activities including uranium enrichment and nuclear waste management. A large-scale reprocessing plant is now ready for test operation. However, in the last ten years, various troubles and disgraces have happened at power plants and nuclear fuel facilities, which have led to a heavy loss of trust by public in the nuclear power. Facing such difficulties, the government and industry are making every effort to recover public trust and understanding in nuclear by maximizing nuclear safety and enhancing transparency and accountability. Nuclear power will play an important role in securing sustainable energy supply and in preserving global environment. Japan is determined to pursue safe and peaceful utilization of this precious source of energy in the twenty-first century.

U.S. PERSPECTIVES ON NUCLEAR ENERGY

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PLENARY SESSION 2

HISTORY AND FUTURE OF NUCLEAR POWER

STATUS AND PROSPECTS OF RUSSIAN NUCLEAR POWER INDUSTRY DEVELOPMENT

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In 2003 there were 30 nuclear power units with total (gross) capacity of 22,2 GW (e) in operation at ten Russian NPPs (Balakovo, Bilibino, Beloyarsk, Kalinin, Kola, Kursk, Leningrad, Novovoronezh, Smolensk and Volgodonsk). They have generated 148,6 billions kW·h, or ~16,5% of total energy production by all Russian power plants, whereas the nuclear plant share of total installed capacity was equal ~11%.

Starting from 1998 the nuclear power plants provide an average annual growth rate about 9 billions kW·h having covered the electricity demand increase in European Russia up to 50% in average.

The electricity and heat production in 2003 is equivalent to a saving of more than 40 billions $м^3$ of gas (30% of gas consumption by Russian power industry).

The long-term balance of Russian fuel resources envisages enhancement of the energy generation framework, in particular, by advance raise of electricity generation by nuclear power plants and by more comprehensive utilization of potential of hydropower industry.

As a result of accomplished multi-factor optimization of the fuel resource balance the priorities for territorial siting of power generating facilities have been established: thus, in European Russia it is expedient to develop electric power industry by means of technical retrofitting of existing thermal power plants, creation of combined steam/gas installations of necessary capacity and maximum possible development of nuclear power plants, which would substantially cover the electricity demand increase for this region.

In terms of moderate option of the economic development the electricity generation demand may reach 230 billions kW h in the year 2020. Additional increase of the electricity generation by nuclear power plants (up to 230 billions kW h) will require creation of the "nuclear-hydroaccumulating" power complexes. The increase of thermal power production by nuclear plants (up to 30 millions Gcal) is related with siting of NPPs and nuclear thermal and electricity generating plants at thermal power consuming regions.

The long-term industrial technological policy provides for 2010-2030 period an evolutionary implementation of new nuclear technology of the 4th generation being based on the fast reactors with the nuclear fuel cycle closing and uranium-plutonium fuel. This will eliminate the restrictions regarding raw fuel material for observable perspective.

In view of the total electricity production growth rate more than 2% per year for Russia, the nuclear power industry is required to ensure power production increase to be more than 4% per year with the growth of electricity generation up to 8 billions kW h / year and thermal production – up to 1,5 millions Gcal / year.

The main objectives of Russian nuclear power industry development are nowadays as follows:

- Further safety and reliability level raising;
- load factor increase up to 85-90%;
- modernization and extension by 10–20 years the operational lifetime of existing NPPs;
- commissioning of new power units (it is planned to put into operation new power units at Kalinin, Kursk, Volgodonsk and Balakovo NPPs before 2010);
- creation of new complex facilities for NPP radwaste and spent fuel reprocessing;
- acquisition of advanced reactor technology (BN-800, WWER-1500, etc.) supported by development of respective fuel cycle;
- extensive capacity reproduction (with rate of ~1 GW per year) and backlog of construction works for future periods.

Because the objective of new power unit construction takes a long period of time and requires significant investments, and also taking into account that design lifetimes for some Russian power units expire starting from 2004, the goal of operational lifetime extension beyond the expired design lifetime for existing NPP units becomes one of the most important challenges of current nuclear power development.

Results of the activities aimed to enhance nuclear plant safety and reliability, to extend power unit operational lifetimes and to construct new power units prove actuality of achieving established goals.

Russian nuclear power industry demonstrates reliably performance and dynamic development, and nowadays poses an important pillar ensuring energy security of the state.

NUCLEAR ENERGY IN INDIA – RETROSPECT AND PROSPECTS

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In his presidential address at the first International Conference on the Peaceful Uses of Atomic Energy in Geneva in August 1955, Homi J Bhabha, traced the growth of the civilization, correlating it with increase in energy consumption and the development of new energy sources. He emphasized that the acquisition by man of the knowledge of how to release and use atomic energy must be recognized as the third epoch of human history. Bhabha and his colleagues took up a detailed study on the needs and merits of India taking up R&D in atomic energy to meet its developmental needs. The study concluded that any substantial rise in the standard of living in this region – that can be sustained in the long term – will only be possible on the basis of very large imports of fuel or on the basis of atomic energy. This conclusion is as true today as it was in mid-fifties and led to the setting up of R&D facilities to pursue development of nuclear power and setting up of nuclear power plants. Further taking cognizance of India's nuclear resource profile, Homi Bhabha formulated a three-stage nuclear power programme. Pressurised Heavy Water Reactors (PHWRs) were chosen for the first stage of the nuclear power programme.

The decision to adopt PHWRs for India's nuclear power programme was based on best utilistion of country's limited uranium resources, higher plutonium yield which is necessary to lay the foundation of fast reactors to be set up in the second stage, freedom from import of enriched uranium which is necessary for light water reactors, and the then existing industrial capability in India to manufacture components required for the reactor systems.

Having conceived a three-stage programme for setting up nuclear power reactors, the Atomic Energy Establishment was set up at Trombay in 1957 and was renamed as Bhabha Atomic Research Centre in 1967. Additional research centers and industrial units were progressively set up over the years by the Department of Atomic Energy (DAE) and DAE has been quite successful in deploying the results of R&D for the benefit of the nation.

These efforts encompass every aspect of nuclear fuel cycle and as a result India now has a solid base to expand its nuclear power programme. During the year 2002-03, nuclear power plants generated 19.358 TWh of electricity. In percentage terms, this may not be very significant, but does signify the fact that we have been able to master this advanced technology. We are now poised for a rapid growth of nuclear generating capacity in the country. In around four years from now, we would reach an installed capacity of around 4500 MWe with pressurized heavy water reactors, the main stay of the first stage of our indigenous nuclear power programme, and another 2320 MWe with light water reactors making a total of around 6800 MWe as against the present capacity of 2770 MWe. In September 2003, the Government of India has approved construction of a 500 MWe prototype Fast Breeder Reactor marking the launching of the second stage of the nuclear power programme. This will open up a vast source of energy for the development of the country.

At the time of independence in 1947, total installed electricity generation capacity was 1,363 MWe. It rose to 30,214 MWe in 1980-81, 66,086 MWe in 1990-91 and 136970 MWe in 2002-03. The average growth rate over the entire period thus has been an impressive 8.6%/yr. In spite of this impressive growth, per capita energy consumption is still very low. The electricity generation in the fiscal year

2002-03 was about 532 billion kWhr from electric utilities and depending upon what capacity factor one assumes, additional 104 to 127 billion kWhr were generated by the captive power plants. On per capita basis, this works out to be about 610 kWhr per year. It is an order of magnitude lower than advanced countries. Assuming that energy intensity of GDP will continue to decline as in the past several decades and the fact that being a tropical country, India does not need energy for heating, India would have to plan to reach a modest target of electricity availability of 5000 kWhr per year per capita so as to provide a decent quality of life to its citizens. India's population could rise to 1.45 to 1.5 billion by the year 2050. This would call for a total electricity availability of 7250-7500 billion kWhr per year. This is an order of magnitude higher than the generation in the fiscal year 2002-03 and calls for developing a strategy for growth of electricity generation based on a careful examination of all issues related to sustainability including abundance of available energy resources, diversity of sources of energy supply and technologies, security of supplies, self sufficiency, security of energy infrastructure, effect on local, regional and global environment, health externalities and demand side management.

This is a gigantic task, but considering the way India has developed since independence, it is doable. However, we have to tap every available energy resource including fossil, hydro, nuclear and non-conventional. We have to take an integrated view with regard to energy planning. A recent study carried out in DAE examines all available energy resources in the country, prospective plans of the various Departments of the Government of India and the imperative to keep the energy import at about the present level and constructs a scenario that brings out relative contribution of various energy sources in the middle of the century. This study takes an optimistic view of all resources and plans. The study concludes that coal would continue to dominate the energy scene and coal fired power plants would contribute about half the electricity in the year 2052, while the contribution by nuclear would be about one quarter.

To generate quarter of electricity by nuclear, considering India's nuclear fuel resource, it is necessary to take recourse to Fast Breeder Reactors (FBRs) in a big way as the fast breeder technology has the potential of meeting India's electricity requirement for several decades. To tap this potential fully, we have to work on several new developments. In the case of FBRs, the aim has to be to develop short doubling time fuel and corresponding fuel cycle facilities to ensure fast growth in FBR installed capacity along with reduction in the capital cost, O&M costs, fuelling cost and improvement in safety.

The paper will summarize past achievements and outline future plans.

IAEA SAFETY STANDARDS AND FUTURE DEVELOPMENTS

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NUCLEAR POWER OF UKRAINE - HISTORY, PRESENT STATUS AND PROSPECTS

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The history of the nuclear power development in Ukraine has more than 50 years. Nuclear power rank leading position in insuring of Ukraine energy needs. The share of the nuclear power is supported close to 45 % of general electricity generation in Ukraine. Nuclear power securing safety of the electricity supply of Ukraine in out-of-the date conditions of the convention power plants. It also supports the electricity price at a level providing competitiveness of the Ukrainian production. Without doubts, the leading role of nuclear power will be kept and in the future of the state.

The history of a nuclear power in Ukraine has not only the light sides. The largest accident in practice of nuclear power usage for peace has taken place at the Chernobyl NPP - the first-born of nuclear energy in Ukraine. The reasons and circumstances of this accident, action on its consequences over comings steel is integral and sad compound the experience has gotten by global nuclear community.

Report contains the data describing stages of Ukraine nuclear power development, its current status, problems and prospects of the further development. The special attention is given to mutually advantageous cooperation of Ukraine and Russia, and also other CIS countries in the field of peace use of a nuclear power.

NUCLEAR POWER IN KOREA, CURRENT STATUS AND FUTURE

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Korea, dependent on the importation of 97% of its total energy demand due to a shortage of domestic natural resources, has extended nuclear development as a reliable alternative energy source. Currently, Korea has 18 nuclear units in operation as a result of efforts in the nuclear power development.

After experiencing the oil shocks of the 1970s, Korea overcame chronic power shortages through the expansion of its nuclear power program. Nuclear power has played a key role in the economic development of the country since the 1980s by securing long-term stability both in power supply and electricity tariffs.

The installed nuclear capacity is 15,716MW as of the end of 2003, representing more than 28% of the country's total installed capacity for electrical generation. Also, the nuclear power generation in 2003 reached about 129.7 billion kWh or 40.2% of the country's total electricity generation.

In line with launching the restructuring of the electric power industry in Korea, the 1st Basic Plan for Electric Power Supply and Demand was established in 2002. According to this plan, a total of eleven nuclear units, amounting to 12.6 GWe, including three units under construction, will be added to the grid by 2015. At that time, the share of installed nuclear capacity in overall electricity generating facilities will reach 34.6%. In addition, the share of nuclear electric production will increase to 46.1%, enlarging the role of base-load handling, thus positioning nuclear power as a central energy source in the 21st century.

The future direction of Korean nuclear energy development is the continued safe operation of existing plants and a realistic level of continued construction for new nuclear power facilities in anticipation of increased environmental regulations and uncertainty in the worldwide energy supply.

As a result of Korean self-reliance program in nuclear power technology formulated in 1984, the first Korean Standard Nuclear Power plant (KSNP), Ulchin 3 and 4 began commercial operation in 1998 and 1999, respectively. Since then, continuous improvements have been realized in the design, construction, operation and maintenance of the KSNP based on the lessons learned from previous units.

The standardized design for the Advanced Power Reactor 1400 (APR1400), developed as a result of the national advanced technology development project, was approved by the regulatory authority in 2002. The APR1400, an evolutionary PWR-type reactor, generating electric power of 1400 megawatts, was developed domestically over a ten-year span through the Korean Next Generation Reactor (KNGR) project as a main type of nuclear power plant in Korea.

Korea is also performing the mid- and long-term researches on the future nuclear power systems, implemented mainly by Korea Atomic Energy Research Institute (KAERI). One of them is to develop the small and medium sized reactor for multi-purpose uses. Since July of 1997, KAERI (Korea Atomic Energy Research Institute) has been developing an advanced reactor called SMART (System integrated Modular Advanced ReacTor). SMART is a 330MWt integral type pressurized water-cooled reactor that can be used for cogeneration, district heating and seawater desalination as well as electricity generation. Safety enhancement and economic improvement are emphasized as the most important considerations in the design of the SMART. The pilot plant construction project has already been launched to demonstrate its technology and will be completed by 2008.

Korea actively participates in an international collaboration on Gen IV (Generation IV) development as a chartered member of GIF (Gen IV International Forum). Since October 2002, Korea has made a national plan for an effective collaboration on Gen IV development with GIF member states. In the result of the planning, two main directions to Gen IV development are proposed; one is to develop Gen IV reactor with its fuel cycle technologies for electricity generation, and the other for hydrogen energy production.

Nuclear power is a large part of the solution to diverse energy concerns. It plays an increasing role in providing energy for sustainable development of Korea, taking into account its scarce energy resources, and rapidly growing electricity demand. Nuclear power is one of the few proven technologies that can, on a large scale, contribute to sustainable development, and it will continue to have a valuable role to play in the future energy supply in Korea.

IAEA-CN-114/P2-6

NUCLEAR POWER IN CHINA: CURRENT STATUS AND FUTURE

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PAST, PRESENT AND FUTURE OF NUCLEAR ENERGY AND THE NUCLEAR FUEL CYCLE IN ARGENTINA

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Argentina started its nuclear activities early in the fifties with Uranium Exploration, Research Reactors design and construction and Research Reactors Fuel development and manufacturing.

At the end of the sixties and after the construction of three research reactors, a decision was made on the construction of a series of NPP of D_2O-Natural Uranium type. Two of them, Atucha I and Embalse are in operation and the third is under construction waiting for the decision of its completion.

Three stages can be shown as an evolution of the Nuclear Fuel Technology in Argentina. The first one was the acquisition of basic knowledge in material properties, metallurgical processes and research reactor fuel manufacturing. This stage was developed late in the fifties and during the sixties. The second covers the Fuel Engineering and Power Reactor Prototype fuel manufacturing capability. This step was carried out from late 60´s to the end of the seventies. The third was the establishment of a Nuclear Fuel Cycle Industry that was developed during the eighties and beginning of the nineties.

CNEA developed a group of Companies that operate installations and are responsible for the supply and goods of the Nuclear Fuel Cycle. CNEA remains as the technological support for this industry and participates as technological partner in the shear of the stock.

We face now a new challenging step characterised by the needs of a very high-density research reactor fuel and highly reliable and low cost fuel for power reactor.

STATUS OF NUCLEAR ENERGY PROGRAM IN INDONESIA

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The demand for energy, especially in the form of electricity, continues increasing in developing countries like Indonesia. Electricity is indeed the main fuel for industrialization and socio-economic development in any country. From another perspective, the existence of energy demand can be considered as the most important driving force for installing or introducing a new power generating plant. Indonesia has considerable reserves of primary energy resources, even though not abundant. At present the Indonesian energy consumption per capita is relatively low even as compared to other ASEAN countries. The increase of population, especially in the rural areas those are not yet have an adequate access to the electric grid, is an indication of an expected high-growth rate of electricity demand.

Presently, Indonesia has operated 3 (three) research reactors. The first reactor was commissioned in 1964. It was TRIGA MARK type with capacity of 2 MWt. The second research reactor was also TRIGA MARK type that commissioned in 1979 located in Yogyakarta Nuclear Facility Area. The power capacity of this reactor was 100 KWt. The third research reactor due to the function was called the Multi-purpose reactor that can be utilized for fuel and material testing, radioisotope production, conducting the experiment utilizing neutron beam for neutron radiography and for basic research. The power of Multipurpose reactor is 30 MWt that suitable for material testing facility. The purpose to establish these nuclear facility besides to support technologically to the introducing of NPP, it was also to prepare and provide the adequate man power development related to number and capability or skill for facing the project on establishment of NPP in Indonesia.

At the present time, there are no power reactors in the country. But the first idea to have a nuclear power plant in Indonesia occurs in 1956 coming from the university circle in Bandung and Yogyakarta in the form of seminars. And then until 1996 many activities on the preparation of NPP introduction including the feasibility study have been done without any promising result. The recent study of long energy and electricity planning "*Comprehensive Assessment of Different Energy Sources for Electricity Generation in Indonesia*" prepared by Indonesian Team and supported by IAEA has already finished in 2001. The result of this study shows that the Indonesia energy demand is projected to increase in the future. The introduction of NPP on Java-Bali electricity grid will be possible in 2016 for 2 GWe and it will reach more than 6 GWe in 2024, using proven reactor PWR1000 (1000 MWe) with 85% capacity factor and investment cost $2000/kWe. The study continued on environmental aspect and externalities studies for year 2002, and the environmental constraints lead apparently to an earlier introduction of nuclear power plant. The general prospect for nuclear power is relied on four main driving forces. They are:

(1) Public Acceptance
(2) Energy Planning and Environment
(3) National Energy Policy
(4) Financing

In recognition of the need to develop a viable nuclear regulatory infrastructure in order to proceed with the development of nuclear power, the government of Indonesia has issued the new basic nuclear energy act on April 1997 (Act No. 10 of 1997). In this new Act, the authority in executing and regulating nuclear energy is separated into two different institutions to guarantee the control of nuclear energy to be more credible in order to suffice the nuclear safety. The responsibility to promote the application of nuclear energy is vested to the “Promotional Body” (National Nuclear Energy Agency or BATAN) and the responsibility to regulate and control is vested to the “Regulatory Body” (Nuclear Energy Control Board or BAPETEN).

Regarding the International engagement, the Preamble of the Indonesian Constitution stipulates among others, that Indonesia will take part in maintaining the world order based on freedom, eternal peace and social justice. Therefore when the Treaty on Non Proliferation (NPT) entered into force in 1970 the Government of Indonesia immediately acceded to the Treaty on the same year, as part of its commitment to take part in maintaining the world order based eternal peace. The Treaty was then ratified in 1978, in the form of Indonesian Law No. 8 of 1978. The ratification was then followed by signing Safeguards Agreement with IAEA in 1980. Finally, to answer the need for more transparency in peaceful uses of nuclear energy, by 29 September 1999 Indonesia signed and entered into force the Protocol Additional to the Safeguards Agreement, joining others to become the first seven states to implement the Additional Protocol.

THE NUCLEAR REACTORS, FROM A "NATURAL HISTORY" PERSPECTIVE

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From the discovery of fission and the divergence of the Chicago Pile, the evolution of the genus *nuclear reactor* can be viewed from a "palaeontology" perspective. It started with the unbelievably ebullient creativeness of the pioneer age, where almost every possible combination of reactor ingredients was dreamed of and often designed, built and operated, be it for a short period of time. The pioneer era was followed by a stringent selection process. Factors involved in the selection have been numerous and diverse: technology, safety, geopolitics, and finally economics. The weight of this factor was not identical from country to country, nor was it constant over the period. This "natural selection" resulted in the survival of a handful of the "fittest" reactor species, dominated by the two subspecies of Light Water Reactors.

But the reactor genus is still very young, and evolution does not stop. The changing environment may, and probably will, favour the emergence or re-emergence of other species better adapted to the new selection criteria, like those identified by INPRO and Gen-4 (competitiveness, sustainability, safety, resistance to proliferation, waste management, etc.). These new fission reactors and their evolving successors will for a long time cohabit with the early fusion reactors in a future where rarefied and expensive hydrocarbons shall be reserved for more valuable uses than mere combustion in power plants… but this is another story.

UK NUCLEAR HISTORY AND THOUGHTS ABOUT THE FUTURE

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PARALLEL SESSION A

OPERATING EXPERIENCE AND PLANNED PROJECTS

IAEA PERSPECTIVES ON THE FUTURE OF NUCLEAR POWER

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This paper will discuss the development of the IAEA's implementation of its Statute to "encourage development of atomic energy for peaceful uses throughout the world". The paper will look at the achievements after initial establishment following United Nations support. It will discuss some of the difficulties, following the boom in nuclear power plant construction in the 1960's and early 1970's. It will present how information and guidance on nuclear power planning and operation has been developed. The paper will conclude with a view of the prospects for nuclear power in light of energy demand and increased environmental awareness. This will include considerations of economic competitiveness, international trading arrangements, growing indications of new countries wishing to adopt nuclear power as part of their energy mix, and the Agency role in ensuring the availability of appropriate technology and infrastructure to encourage developments.

OPERATING EXPERIENCE OF BN-600 FAST NEUTRON REACTOR AND BN-800 REACTOR DESIGN

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Experience gained in Russia (USSR) in R&D work in the area of sodium cooled fast reactors in the period of 1950-1970-ies has been used in the design of NPP with the BN-600 reactor. Since its start-up in 1980, BN-600 reactor has demonstrated operating characteristics, which are unique for this nuclear technology. Average load factor value for 23 years of operation is near 74%, its values in 2002 and 2003 being respectively 77.35% and 75.7%. Release of inert radioactive gases is within 0.3% of reference value, while average collective dose rate of personnel is about 0.3 man. Sv per year.

In the course of operation of NPP with the BN-600 reactor, effectiveness of steam generator protection system was demonstrated in 12 cases of small and large water-into-sodium leaks. Besides, unique experience was gained in confining either radioactive and non-radioactive sodium fires in case of sodium leaks from the circuits. Radioactive sodium leak from the primary auxiliary circuit occurred in December 1994 is a typical example. Total amount of sodium released from the circuit was about 1000 kg, and protection system was capable of confining sodium ignition nucleation site and limiting radioactivity release to the atmosphere by 10 Ci value. This release has had almost zero effect on radiological conditions of the NPP controlled area.

BN-800 reactor design is the next stage of development of sodium cooled fast reactor technology. Fourth power unit with the BN-800 reactor is now under construction on Beloyarskaya NPP site.

Innovative design approaches have been used in the BN-800 reactor in order to further improve safety of fast reactors with sodium coolant. Among these innovations are as follows:

- Additional «passive» safety system using three absorber rods hydraulically suspended by the sodium flow;
- Passive decay heat removal system using sodium–air heat exchangers;
- Device for collection and retaining of the core debris in case of its disruption under conditions of beyond design accidents; etc.
- BN-800 reactor design is important also from the standpoint of increase of competitiveness of fast reactors.

Application of such innovative design approaches as the increase of reactor power from 1470 MWth (BN-600) up to 2100 MWth (BN-800) within almost the same reactor vessel, use of “mono-unit arrangement” (one turbogenerator per one reactor), steam-steam reheater, modification of auxiliary systems and decrease of their number, etc. has resulted in significant reduction of specific metal consumption of reactor facility (from 13 tons/MWe in the BN-600 reactor down to 9.7 tons/MWe in the BN-800 reactor).

IAEA-CN-114/A-10

PHENIX: THIRTY YEARS OF OPERATION FOR RESEARCH, REACTOR RENOVATION OVERVIEW AND PROSPECTS

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The French fast reactor prototype PHENIX was put on commercial operation in 1974. The total operation time of the plant is about 100 000 of grid connected hours representing 3900 Equivalent Fuel Power Days (EFPD).

The plant has achieved the objectives of demonstration of fast breeder reactor technology which were set at the time of construction, including the following significant examples: fuel optimisation, closing of the fuel cycle, reliable operation with high thermal efficiency of 45.3%, large feedback experience on sodium components, low staff dosimetry.

From 1992, the role of PHENIX as an irradiation facility has been emphasised, particularly in support of the CEA R&D programme on long-lived radioactive waste transmutation.

This new objective required an extension of the planned reactor lifetime. A major renovation programme was carried out on the plant from 1994 to 2003, involving safety up-grading, component inspections and repairs and the ten yearly statutory maintenance.

The main works consisted of:

- the seismic reinforcement of the plant buildings,
- the addition of a safety control rod to the reactor,
- the partitioning of the secondary sodium circuits in the Steam Generator building to improve protection against sodium fires,
- the installation of an anti-whip system on the high pressure steam pipes,
- the construction of two redundant seismic resistant emergency water cooling circuits,
- the repair of the superheater and reheater SG modules,
- special inspections of the reactor internal structures: core support conical shell and core cover plug.

An extensive plant requalification programme was carried out following the renovation works and the plant resumed power operation in June 2003.
Six operating cycles are planned representing a total irradiation time of 720 EFPD equivalent to about 5.5 years of operation until 2008.

A first series of experimental irradiations on transmutation of minor actinides and long-lived fission products have been loaded into the core. They involve:

- Inert matrices for heterogeneous mode transmutation,
- Americium targets located in special moderator carriers,
- Technecium 99 metal pins,
- Isotope irradiations for cross-section measurement.

The return to power operation of the reactor is also significant in the context of the GENERATION IV Forum. It will contribute to gain further operating experience on sodium cooled fast reactors, one of the six concepts selected by the Forum, in the frame of strong international collaboration.

It will provide irradiation capability for material and fuel R & D on some other types of reactor such as gas-cooled fast reactor.

PERSPECTIVES OF NUCLEAR ENERGY FURTHER DEVELOPMENT IN ARMENIA

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Nuclear energy is one of the most important options of electric energy generation in Armenia. The future role of nuclear energy can be determined only if the further development of the whole energy sector is analyzed in detail based on the future country's energy demand forecast. Such an integrated approach was used, and a set of methodologies was applied when, in the frame of the IAEA TC Program, the "Energy and Nuclear Power Planning Study for Armenia up to 2020" was carried out. The objective of the study was to analyze the future energy demand in various possible scenarios of Armenian economy development, and to develop economically and socially acceptable plans for the electricity generating system expansion to meet that demand, with the thorough assessment of the role that nuclear energy could play within those optimal programs.

Two scenarios of Armenian energy sector development were considered during the study - with the new nuclear capacities, and without them. According to the first scenario, after the ANPP decommissioning, it would be replaced by the new nuclear units, and the second scenario assumed not to use the nuclear energy any more. In particular, according to the second scenario, the energy sector would be developing with the new combined cycle plants implementation.

A thorough analysis has proved that, although the nuclear option is slightly more expensive, it would have the advantage of fuel diversification with a reduced dependency on imported fossil fuel. Therefore, in view of a country's future energy independence and energy supply security, the Armenian energy sector expansion plan with the new nuclear units seems to be more preferable than the option with the combined cycle plants, and also, this option is more attractive because it would make it possible to reduce the gas supply dependence for Armenia and enable it to escape the possible energy crisis in the future.

NUCLEAR POWER ENERGY DEVELOPMENT IN BRAZIL - FUTURE PERSPECTIVES

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This paper deals with the implementation of nuclear energy in Brazil from the 50's up to present time. The actual status of nuclear energy and its future perspectives are also considered.

Although Brazil has one of the largest hydroelectric potentials in the world, the interest in nucleoelectrical energy comes from the end of the 50's. From 1955 to 1960 – the Institute de Pesquisas Radioativas de Belo Horizonte – IPR – established the "Thorium Group" to demonstrate and establish the technical basis for the development of a nuclear reactor fueled with thorium in the country, since Brazil had one of the largest reserves in the planet. At that time scientists, decision makers and politicians were often involved in exasperating and provoking discussions in which the supporters of the natural and enriched uranium usually expressed strong opposing opinions, both based on political, technical, and economical considerations.

By the end of the sixties and first half of the 70's the country - through the National Nuclear Energy Commission, CNEN, especially by means of its research institutes IPEN (Instituto de Pesquisas energéticas e Nucleares) in São Paulo and IEN (Instituto de Engenharia Nuclear) in Rio de Janeiro - established a strong research programs related with gas reactors (HTGR), liquid metal reactors (LMFBR) and light water reactors (PWR). These programs gave orign to important and highly capable research groups in knowledge areas related to thermohydraulics, materials, reactor physics, nuclear technology and design, instrumentation and control, safety analysis, structural analysis and many others. At same time, fundamental research facilities have been implemented such as thermal loops, fuel processing and fabrication installations and research reactors.

Due to a mix of technical, political and economical considerations, in the early 70's, a turn-key contract with Westinghouse Electric Corporation of the United States of America was signed to install in Angra dos Reis, half way between São Paulo and Rio de Janeiro – the largest cities in the country – the first Brazilian nuclear power reactor – ANGRA 1, a 626 Mw(e) PWR reactor. ANGRA 1 construction started in 1971, and the first criticality was achieved eleven years later; since then more than 41,200 GWh of energy have been produced.

In an effort to become self-sufficient in nuclear power generation, a comprehensive agreement was signed with Federal Republic of Germany in 1975 to built eight 1,300 Mw(e) PWR reactors and all needed installations for a full technology transfer package. The first two units (ANGRA 2 and ANGRA 3) were scheduled for construction on the following years with most of their components imported from Kraftwerk Union's (KWU) shops in Germany. For the remaining plants it was aimed to reach a level of 90% Brazilian-made components. The Empresas Nucleares Brasileiras (NUCLEBRAS) was then created as the Brazilian stated-owned nuclear holding company to be responsible for this enterprise which together with several joint companies should promote nuclear technology transfer from Germany on all aspects of PWR reactors and fuel cycle technology.

Due to several problems the technology transfer program didn't succeed properly and the construction of ANGRA 2 and ANGRA 3 was interrupted several times. After many setbacks construction of ANGRA 2 nuclear power restarted on the second half of 90's and finished in July 14, 2000. However the construction of the third nuclear station (ANGRA 3) is still pending of governmental decision although 70% of the imported major components are stored on site. Beyond ANGRA 3 there is no real commitment of plants being ordered, but the feeling is that the next reactor after ANGRA 3 will be another generation.

Envisioning to define the most promising technologies as well as their R&D needs to achieve deployment within the next 30 years, some international initiatives have been established. Two of them deserve to be mentioned due to their importance. The first one, launched in 2000 under the leadership of USA, is the Generation IV International Forum – GIF. Ten countries have been participating in this Forum, including Brazil. The second initiative was set for by the International Atomic Energy Agency – IAEA – in 2001 and was named INPRO (International Project on Innovative Nuclear Reactors and Fuel Cycle). This project has the participation of 12 countries, including Brazil also. In short, both initiatives try to address the challenges for the future of nuclear energy which are: (a) to prove that nuclear energy is economically competitive in an environment ruled by market forces and (b) to get the public acceptance concerning safety, waste deposition, environmental and proliferation issues.

Brazil is also participating in the IRIS program. IRIS (International Reactor Innovative and Secure) is a small-to-medium power (335 MWe) integral type pressurized water reactor, which has the significant characteristics of simplicity, enhanced safety, improved economics, proliferation resistance and waste minimization. The research institutes of CNEN are participating in specific design activities. A decision will have to be made, concerning the strengthening of this participation and the making of more firm commitments to the future taking in account the present budget constrains.

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EXPERIENCE GAINED WITH INSPECTION OF METAL STATE IN THERMAL AND MECHANICAL EQUIPMENT FOR EXTENDING OPERATION TIME OF POWER UNITS OF THE FIRST GENERATION NPPS

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The report is devoted to the topical problem of the present time – the extension of service life of the first generation power units. Solving of this problem is impossible without an estimation of technical conditions of the equipment and pipelines. In this way the survey of metal conditions is the primary importance. Taking into account a fact, that works regarding the service life extension of power units were carried out for the first time, it has required not only the elaboration of the methodology of the assessment of technical conditions and residual life, but also the elaboration of new methods intended for metal conditions inspection and its practical application on-site. At the same time an important moment was the necessity of development of the normative document base providing both technical and law support of the defined problem.

This report considers the methodology, the newly developed methods and tools for metal inspection and the standard ones which being applied. Also the definite results of VNIIAES activities regarding the technical survey of the equipment and pipelines metal for the purpose of the first generation power units service life extension are presented.

The codes and normative documents elaborated by VNIIAES is considered, which determines the procedure and the order of work for assessment of the technical conditions and residual lifetime of power units concerning its thermal and mechanical equipment elements and pipelines (see fig.).

The results of complex metal researches of pipelines and equipment of the first generation power units executed by VNIIAES in cooperation with other industry’s organizations are submitted. These researches have included the application of new inspection methods and metal researches methodology, elaborated by VNIIAES and other organizations, which allow to give the forecast for planned operational terms for NPPs outside of 30 years limits.

There are also submitted the summary results of work regarding the assessment of the technical conditions and residual lifetime of the inspected technological systems elements of the first generation power units. The results being received made up the base for obtaining the licenses for operation of the first generation power units above the 30 years limits. Works regarding the assessment of the metal of the thermal and mechanical equipment with the purpose of service life extension are proceed now both in a way of improving the methods and equipment for metal inspection, and in direction of practical application of the research methods being submitted.

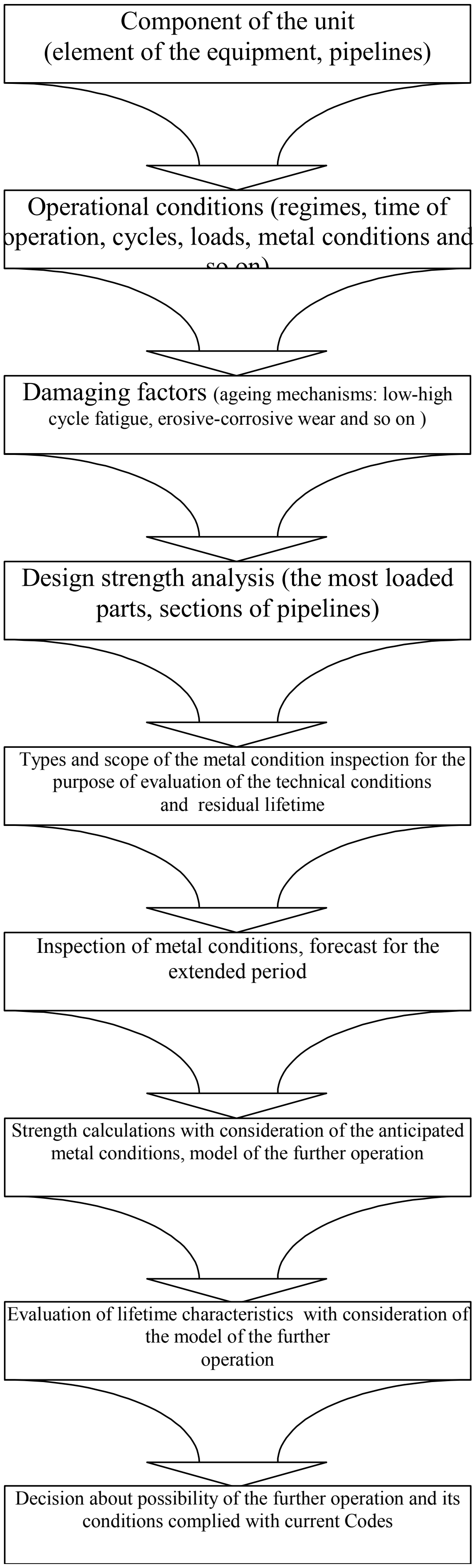
Component of the unit
(element of the equipment, pipelines)
Operational conditions (regimes, time of operation, cycles, loads, metal conditions and so on)
Damaging factors (ageing mechanisms: low-high cycle fatigue, erosive-corrosive wear and so on)
Design strength analysis (the most loaded parts, sections of pipelines)
Types and scope of the metal condition inspection for the purpose of evaluation of the technical conditions and residual lifetime
Inspection of metal conditions, forecast for the extended period
Strength calculations with consideration of the anticipated metal conditions, model of the further operation
Evaluation of lifetime characteristics with consideration of the model of the further operation
Decision about possibility of the further operation and its conditions complied with current Codes

FIG. 1.

20 YEARS OF EXPERIENCE IN STEAM GENERATOR REPLACEMENT

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Framatome ANP has 20 years of experience in the Replacement of Steam Generator components. Starting with the first SGR in the Nuclear Power Plant Obrigheim in 1983, Framatome ANP has been involved in over 50% of SGR projects worldwide, becoming the leader in turnkey SGR projects. Full range replacements have been performed in 31 Nuclear Power Plants with a total number of 89 replaced SGs including the SG types disigned by W, CE, B&W and Framarome ANP.

There will be given an overview about the way Framatome ANP approaches SGR projects. Various aspects of project planning, engineering, management, licensing, qualifications of personnel and techniques used including site implemetation will be presented.

IAEA-CN-114/A-9

PROGRAMME OF SELF ASSESSMENT AND CORRECTIVE ACTIONS AT QINSHAN NPP

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LEAD-BISMUTH REACTOR TECHNOLOGY CONVERSION: FROM NUCLEAR SUBMARINE REACTORS TO POWER REACTORS AND WAYS OF INCREASING THE INVESTMENT ATTRACTIVENESS OF NUCLEAR POWER BASED ON FAST REACTORS

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The report presents an innovative nuclear power technology (NPT) based on usage of modular type fast reactors (FR) (SVBR-75/100) with heavy liquid metal coolant (HLMC) i.e. eutectic lead-bismuth alloy mastered for Russian nuclear submarines' (NS) reactors [1].

Use of this NPT makes it possible to eliminate a conflict between safety and economic requirements peculiar to the traditional reactors.

Physical features of FRs, an integral design of the reactor and its small power (100 MWe), as well as natural properties of lead-bismuth coolant assured realization of the inherent safety properties. This made it possible to eliminate a lot of safety systems necessary for RIs of operating NPPs and to design the modular NPP which technical and economical parameters are competitive not only with those of the NPP based on light water reactors (LWR) but with those of the steam-gas heating electric power plant [2].

A conservative approach was used for development of the RI SVBR-75/100 design. This approach uses the primary and secondary circuits' mode parameters mastered in practice, it uses to the maximal possible extent the mastered fuel and structure materials and developed principal solutions to the equipment elements and RI design.

Adhering to this principle reduces the execution terms, amount and cost of the R&D, assures reliability of the RI and its operating safety, reduces the investment risk.

Multipurpose usage of transportable reactor modules SVBR-75/100 of entirely factory manufacture assures their production in large quantities that reduces their fabrication costs.

Those "standard" reactor modules can be used:

- for renovation of the NPP units with thermal neutron reactors (TR), which reactors have expired their lifetime. As the realized technical and economical investigations concerning the Novovoronezhskaya NPP have revealed, this is two times cheaper being compared with construction of the NPP replacing unit;
- for construction of nuclear heating electric power plants (NHEPP);
- for construction of modular power units of different power NPPs;
- for construction of nuclear desalinating power complexes.

The proposed NFT provides economically expedient change over to the closed nuclear fuel cycle (NFC). At the first stage when the cost of natural uranium is low, reactors operate by using the oxide uranium fuel with postponed reprocessing. Further change over to the closed NFC using MOX fuel fabricated by using the pyro-electro-chemical method in the chlorides melt at reprocessing the own spent nuclear fuel (SNF) will be realized. At this, the amounts of SNF reprocessing in terms of 1 t of plutonium are reduced several times as compared with those of conventional closing of the FR NFC, in which plutonium is extracted at reprocessing the LWRs' SNF and where its concentration is low.

At the same time in the proposed NFC the TRs' SNF is used directly after thermal-chemical reprocessing (similarly to the DUPIC-technology) as make up fuel instead of waste pile uranium. This assures step-by step utilization of TRs' (both VVER and RBMK) SNF without its reprocessing.

Use of proposed NPT makes it possible to considerably increase the investment attractiveness of nuclear power (NP) with fast neutron reactors even today at low costs of natural uranium.

The report reveals that use of RI SVBR-75/100 for renovation of the NPP units, which reactors have expired their lifetime, makes it possible to considerably increase the pace of NP development due to its own investment potentials. It is especially viable for the approaching phase of NP development when the opportunities of "cheap" increase (LF increase, extending of the service lifetime of high and medium availability units) are expired and the additional expenses are required for construction of replacing power capacities and withdrawing the "old" units from operation.

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THE CONTINUOUS IMPROVEMENT OF NUCLEAR SAFETY AND COMPETITIVENESS

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The liberalization of electricity markets in Europe that will be totally effective in 2007 obliges utilities to become more and more competitive. On the other hand, utilities that operate nuclear reactors are of course responsible of nuclear safety, which costs money. A superficial analysis seems to show that safety is opposed to competitiveness; an in depth analysis shows that with some conditions it is possible to reconcile these two concepts.

First at all improving safety performances will protect the public: it will secure public acceptance and help to protect investment from destructive accidents.

Secondly, it is a fact that, when the management of a NPP is efficient, the results in term of economics and nuclear safety are generally excellent: WANO indicators shows that the plants belonging to the first quartile for safety are generally the best for availability or capacity factors and thus for competitiveness.

It is also possible to master O and M costs without decreasing safety expenses: for example, improvement of procurement mechanisms or limitation of tertiary expenses don't influence safety, but reduce the costs. EDF also developed new maintenance methodologies named "reliability centered maintenance" that allows to carry out these component repairs only when objectively needed.

In same cases technological progress allows to improve both safety and competitiveness: a good example is the recent evolutions of fuel assemblies claddings: the new alloys (M5 or Zirlo) allow to increase the burn-up – which is good for competitiveness – while giving increased safety margins.

The continuous adaptation – day-by-day, hour-by-hour –of electricity production to market needs implies to be flexible and to anticipate constraints, included regulatory ones. Consequently, safety regulations most allow some degree of flexibility and must be predictable.

These previous considerations shows that it is possible to improve both safety and competitiveness if some conditions are met, especially flexibility and predictability of safety regulations.

PARALLEL SESSION B

NON-ELECTRICITY DEVELOPMENTS

THE USE OF POOL TYPE NUCLEAR REACTOR RUTA FOR MUNICIPAL DISTRICT HEATING AND DESALINATION OF SEAWATER

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Since the beginning of the nuclear era it's obviously that there are no principle technical obstacles for the usage of nuclear reactors as heat supplying sources for district heating, and almost all types of reactors can be used for such purposes. By the time being, the significant experience is accumulated in this field [1], [2].

In the last years the signs of increasing interest in the sphere of non-electric applications of nuclear energy are appearing [3].

In Russia this is connected with a matter of fuel supplying of far isolated regions and started reformation of municipal district heating systems.

One of the possible heat sources for district heating are single-purpose nuclear heating plants (NHP). Such application of nuclear energy realized by this moment only in frames of experimental and demonstration projects [4].

Perspective project in this field is RUTA reactor (Reactor Unit for Thermal Applications), developed both SSC RF – IPPE and ENTEC specifically for district heating application.

RUTA reactor is a low-temperature water-cooled pool-type reactor with natural convection of coolant in primary circuit. There are different designs of RUTA reactor, power ranges from 10 to 70 MWt [5].

Safety, as well as reliability and simplicity of construction of RUTA reactor is determined in the first place by absence of the surplus pressure of water in the primary circuit (reactor pool). NHP with such reactors possess the inherent self-protection and safety and can be deployed in close proximity of heat consumers.

NHP with RUTA reactors are intended for heating of small towns or city districts with population ranging from 10 to 100 thousand.

Marketing studies show that there are many regions in Russia, where district heating can be provided by NHP with RUTA reactors [6].

The most suitable site for the first NHP deployment is Obninsk, where the necessary infrastructure, scientific and operational personnel are abundant, which allows to realize the project with minimal cost and time.
So the SSC RF – IPPE carried out the feasibility study for improvement of Obninsk district heating system by introduction of NHP with RUTA reactor [7].

Estimated capital investments in the NHP construction in Obninsk (power 70 MWt) is $14 700 000. The cost of heat produced by NHP RUTA is $5.2 for Gkal.

The results of feasibility study for Obninsk show real prospects for advanced district heating technology, based on the low-temperature pool-type RUTA reactors not only in Far North regions encountering lack of fossil fuel, but also in European part of Russia, which currently doesn't experience such troubles.

Another option for use of pool type RUTA reactors is the desalination of seawater.

Nuclear desalination plants (NDP) based on RUTA reactors possess exclusive reliability and environment safety, which is very important for fresh water production for household consumption.
Such NDP can use typical distillation facilities DOU GTPA, adapted for thermal range of heat provided by RUTA reactors. Production of fresh water from NDP with one RUTA-70 reactor is 30 000 m^3/day. Estimated cost of fresh water ranges from $0.9 to $1.3 for cubic meter, depending on local economic circumstances.

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ADVANCES IN NUCLEAR DESALINATION IN BARC

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As a part of our programme to improve the quality of life of our large population by systematic induction of nuclear energy, BARC has been engaged in R&D activities on desalination since 1970s.The desalination activities were part of a programme of setting up a number of demonstration plants for the energy intensive processes such as desalination of seawater, electrolytic production of hydrogen and electro thermal production of phosphorus. These activities are presently termed by IAEA as 'Non Electrical Application of Nuclear Energy'. Over a period of time, BARC has successfully developed desalination technologies based on multi-stage flash (MSF) evaporation, reverse osmosis (RO) and low temperature evaporation (LTE). In the field of thermal desalination, efforts are directed towards utilizing the low-grade heat and the waste heat as energy input for desalination. In membrane desalination, work is being carried out on newer pre-treatment methods such as use of ultra filtration, energy reduction and higher membrane life.

Based on operational experience of MSF and RO plants at Trombay, BARC has undertaken establishment of the Nuclear Desalination Demonstration Project (NDDP), Kalpakkam. NDDP consists of a hybrid MSF-RO desalination plant of 6300 m^3/d capacity (4500 m^3/d MSF and 1800 m^3/d Sea Water Reverse Osmosis (SWRO)) coupled to 2 x 170 MWe Pressurised Heavy Water Reactors (PHWRs) at MAPS, Kalpakkam. The requirements of seawater, steam and electrical power for the desalination plants are met from MAPS I & II which are around 1.5%, 1.0% and 0.5% of available at MAPS. The hybrid plant has provision for redundancy, utilization of streams from one to other and production of two qualities of products for their best utilization.

The 1800 m^3/d SWRO plant (Fig.1), which is already commissioned in August 2002, is designed to operate at relatively lower pressure (51.5 bar during 1st year and 54 bar during 3rd year) to save energy, employs lesser pre-treatment (because of relatively clean feed water from MAPS outflow) and aims for longer membrane life resulting in lower water cost. The MSF plant which is in advanced stage of completion is designed for higher top brine temperature with Gain Output Ratio (GOR) of 9 and utilizes less pumping power (being long tube design).

Fig. 1 View of SWRO Plant building and membrane modules

The desalination plant can meet the fresh water needs of around 45,000 persons @ 140 liters per capita per day (l.p.c.d.) There is a provision of augmentation of product water capacity by blending the low TDS product of MSF plant with brackish ground water/moderate salinity permeate from SWRO plant. This will then serve the need of larger population. Useful design data are expected from the plant on the coupling of small and medium size reactors (SMR) based on PHWR. It will further enable us to design large size commercial plants upto 50,000 m^3/d capacity.

Efforts at BARC are also directed towards the utilization of waste heat. The Centre has been studying the possibility of use of waste heat of nuclear reactors for seawater desalination using low temperature evaporation (LTE) technology. The know-how utilizing waste heat was developed and a 30 cubic metre/day pilot plant was installed. The LTE plant has been connected to CIRUS reactor at BARC, Mumbai for demonstration of coupling to a nuclear research reactor . The product water from this plant after minor polishing meets the make up water requirement of the research reactor (Fig. 2).

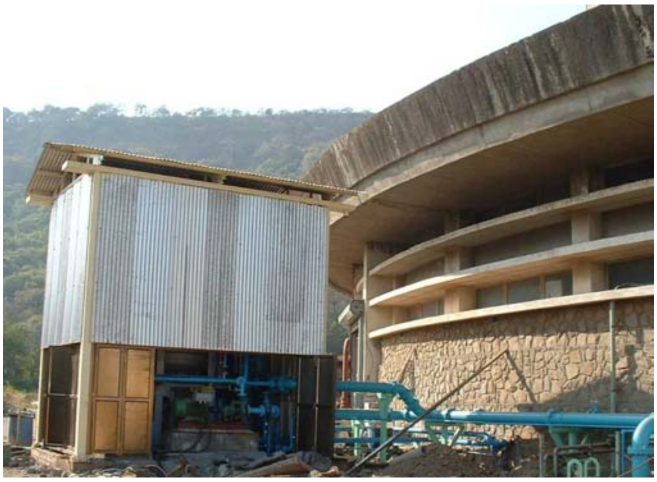

Fig. 2: LTE Plant at CIRUS

The desalination industry is witnessing numerous technological innovations so that these are available to the population in the water scarce areas. The following new projects on "Desalination Technology Studies and Development" have been taken up to incorporate some of the important innovative features; (i) Desalination by Centrifugal Reverse Osmosis (CRO), (ii) Low Temperature Evaporation (LTE) Desalination Plant with Cooling Tower, (iii) Multi-Effect Distillation Vapor Compression (MED-VC) Desalination Plants, (iv) Continuous Thin Film Composite (TFC) Membrane Casting Assembly and (v) Barge Mounted Reverse Osmosis (RO) Unit.

The technological innovations in desalination would lead to its large-scale application and provide opportunities for the socio-economic development of water scarcity areas and large coastal arid zones of the country. India has been sharing the experience of nuclear desalination with the member states of International Atomic Energy Agency (IAEA).

THE RESULTS OF RUSSIAN NUCLEAR CIVIL SHIPS OPERATION AND THE EXPERIENCE OF THEIR SPECIFIED LIFETIME EXTENSION

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The report presents short results of almost thirty-year operation of nuclear ships. The main causes influencing upon the failures, damageability and lifetime of RP systems and equipment units and elements are analyzed. Main measures and recommendations decreasing the damageability of the most loaded units and elements of RP systems and equipment are given.

The situation in Russian nuclear civil fleet after the termination of specified reliable lifetime of RP systems and equipment is analyzed. In spite of long operation time the state of main NSSP equipment is good.

Instead of the provided factory repair the method of stage-by-stage lifetime extension to 90-100 thousand hours and service life extension to 25 years of RP systems and equipment (steam generators, reactor control and protection system drives, primary circuit circulation pumps) was developed.

At the meetings of Transport Ministry and Atomic Energy Ministry in March, 1999 and with the First Deputy Chairman of RF Government in September, 1999 the decision was made on the development of the Program of operating ship-based NPPs lifetime extension and nuclear ice-breakers service life increase.

In this connection OKBM has developed two programs of RP equipment and systems durability extension for the nuclear icebreaker “Arctika” (175000 hours lifetime and 32 years service life) and for the other nuclear ships (150000 hours lifetime and 30 years service life). Special programs for the performance of activities directed to RP systems and equipment lifetime and service life extension are based on these two programs.

OKBM and MMP together with the other organizations and enterprises realize the measures, provided by these programs and first of all at the nuclear icebreaker “Arctika”, where practically the whole complex of the activities has been performed. The decision on the extension of nuclear icebreaker “Arctika” NSSP lifetime to 175000 hours and service life to 32 years was made. It allowed to continue the ice-breaker operation at Sevmorput routs successfully.

At present the activities on the other nuclear ice-breakers (in particular ice-breakers “Taimir” (2005), “Vaigach” (2003), and “Rossiya” (2005)) lifetime extension are performed.

OPTIMAL DESIGN OF NUCLEAR DESALINATION MULTIPLE EFFECT EVAPORATION PROCESS PLANT

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The interest in the use of nuclear reactors as an energy source for the desalination of seawater started in the Nineteen Sixties. The urgent necessity to provide potable water has evoked a number of reports; e.g., Refs. [1-3]. The International Atomic Energy Agency (IAEA) has shown intense interest in nuclear desalination since 1989 [4]. In particular, water scarcity has become a reality in the arid and semi-arid parts of the northeastern region of Brazil, and its variability from season to season, makes much of water supply unavailable when it is most needed. This is the reason why nuclear desalination may be an alternative for fresh water supply and power generation in this region of the country. There exist three concepts of nuclear desalination having technico-economic feasibility; i.e., multiple-effect evaporation (MEE), reverse osmosis (RO), and multiple-stage flashing (MSF). There is an extensive literature, both academic and industrial, covering diverse aspects of the aforementioned concepts; e.g., Refs. [5-10]. However, it appears that the question of the optimal design of nuclear desalination process plant has not been addressed so far. It is the objective of the work reported in this paper to start to fill this gap. The focus of the present work is on the optimal design of nuclear desalination process plant consisting of a pressurised water reactor (PWR) and a multi-effect evaporator (MEE). In particular, the PWR-MEE conventional coupling scheme is considered, whereby vapour extracted from one (on more) turbine stage (5) is used to preheat seawater [10]. The optimal design problem is formulated and solved as a mathematical programming model, in a somewhat similar fashion to work reported in Ref. [11] for a combined cycle coupled to a MSF system. Results obtained so far are encouraging, although vapour consumption rates are 2-3 times higher than those suggested by EURODESAL work [10]. Work is in progress to ascertain that operating conditions are identical to those of EURODESAL [10] with a view to validating the mathematical programming model developed in the present work.

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IAEA-CN-114/B-3

MAJOR ASPECTS OF STRATEGY OF HYDROGEN-BASE POWER DEVELOPMENT WITH NUCLEAR ENERGY SOURCES

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Hydrogen - highly effective and ecologically clean fuel. Large-scale use of hydrogen is mastered in industrial chemical processes and rocket engineering. Manufacture of hydrogen in the world has exceeded 50 MT H2 (6.4 EJ) and quickly grows.

In the report the concept of Nuclear-hydrogen power (NHP) as strategy of production and use of hydrogen on base of "clean" technologies and, first of all, modern innovative high-temperature nuclear energy source such as GT-MHR, modified under hydrogen production application is considered with steam reforming of methane (SRM). The increase of a nuclear energy share in a global energy balance is capable essentially to affect structure of on organic fuel consumption, and, hence, on resulting parameters on hydrocarbon world flows and CO2 emissions.

Under the various forecasts (IIASA, IAEA, IEA, EPRI etc.) in 21 century the sharp growth of demand hydrogen is expected in connection with transition of various base technological branches to mainly intensive methods of qualitative products output with increase of processing depth of petroleum industry, greater release of ammonia and methanol, refinery processing (for example, from heavy oil or bituminous sand) or synthetic (first of all, - from coal) liquid fuel, increase of direct production of qualitative sponge iron etc.

At the same time, greatest contribution to perspective growth of world demand on hydrogen is necessary to expect from a vehicle sector and systems of the dispersed power supply, in which the hydrogen acts as energy carrier, capable to collect and to be transported similarly to natural gas, but not having, as against methane, restrictions on resource base and not having effluents of greenhouse gases in an atmosphere.

For the various scripts of global economy development and parameters of an expected level of power supply forecasts of market potential estimation are changing from low volumes of hydrogen production (in 1.5 - 2 times exceeding a present consumption level of this product equal approximately 6 EJ) up to priority strategy of hydrogen economy with an output of hydrogen on 300-400 EJ by 2100.

The analysis of modern lines and prospects of the strategy allowing to specify nuclear option in hydrogen economy is given.

PARALLEL SESSION C

PUBLIC COMMUNICATION

HOW TO ACHIEVE GREATER ACCEPTANCE OF NUCLEAR INDUSTRY

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In the 90's, the acceptance of nuclear energy in France has decreased, as in other nuclear countries amongst the developed nations. All the surveys indicated that two main anxieties progressed – the fright of another « Chernobyl » accident, and the questionings about the high-level long life waste management (the responsibility towards future generations). Moreover, a large majority of people - more than 70% - thought that nuclear industry had a lack of transparency.

There was in these years a vicious circle between the leaders'point of view (politicians, but also nuclear industry managers) which was that they had no interest to speak on nuclear matters because there was a bad acceptance of nuclear energy in the large public, and the feeling, in the large public, that nuclear was an energy in decline, which was shown by the silence of governments on the future of nuclear. The trust in nuclear energy declined, slowly but regularly.

This trend has began to change in France with some foreign countries' announcements of interest for nuclear energy (USA, Finland, China…). But a decisive change arose in 2003 with the Government's decision to launch a « national debate on the energies », to enhance the democratic foundation of the future energetical choices, regarding the next 30 years, including the replacement of a part of the nuclear fleet. This decision was due to the consciousness that such choices had to be explained and discussed, not only at the Parliament to produce an outline law, but also with the large public, to give a more participative framework to the decisions. Anyway, this « national debate » had not the purpose to result in a referendum: the political decision stays in Parliament's hand, but the debate had to guarantee a high level of public information and to allow questions to experts and politicians.

This debate has led to give to some people a better knowledge of each primary energy's advantages and drawbacks and of the fact that that all the energetic choices are made under many environmental and economical constraints. It allowed the nuclear actors to promote the solutions proposed by the nuclear R&D for the future – closed cycle with preservation of natural resources and minimization of waste, hydrogen production, desalination – and to situate the future prospects of different energies, demystifying the role of renewables.

There have been few feedbacks of this debate in the national medias, and few people, except the professionnals and the militants, have participated in the national meetings (much more in the local ones). But the surveys realized after this debate indicate that about 25 to 30% of « inhesitant » people have changed their point of view on nuclear, and that the consciousness of the main stakes of energetic choices for the XXIst century – greenhouse gas emissions, growth of electricity demand especially in developing countries – has progressed.

The conclusion of this analysis is that one must not take into account the question of public acceptance as a real obstacle to the development of nuclear energy, because public opinion can change and can be convinced by making things clear with the future economical and environmental stakes.

IAEA-CN-114/C-5

THE IMPORTANCE OF INTERNATIONAL NUCLEAR COMMUNICATIONS

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We live in an age where 24 hours news reporting is the norm where events on one side of the world can be shaping headlines on the other side of the globe in seconds.

For the nuclear industry, recognising the dangers - and benefits - of working in an information intensive global society is essential.

A single, non-safety related incident - anywhere - can spark a 'chain reaction' of ill-informed reports that has the potential to do more harm to the nuclear industry in terms of public and investor confidence than a serious accident.

It is important to harness that flow of nuclear news and for the nuclear industry to ensure the nuclear case is put factually and effectively - reporting with authority and clarity when nuclear events unfold.

Today, as we reflect on experiences and lessons learned from the first 50 years of nuclear power, the need for the nuclear community to communicate effectively and with transparency - within the industry itself as well as the general public - is of greater importance amid renewed interest in the benefits that nuclear power are likely to bring over the next half-century.

HOW TO PRESENT THE TECHNICAL MESSAGES TO THE PUBLIC

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The use of nuclear energy as a source for electricity generation in Thailand has been introduced since 1967 when electricity demand increased considerably for the first time as a result of the economic growth. Its viability had since been assessed several times during the early seventies. Lack of a national firm intention and the uncovering of promising indigenous resources had been seen as reasons for the plans being turned down.

In July 1996, A national committee to review the nuclear energy option was established and chaired by the Minister of Science, Technology and Environment. It is assigned with the tasks of studying and making recommendations to the cabinet for decision. Four sub-committees were subsequently formed to establish the facts surrounding the introduction of nuclear energy. The sub-committees cover technology and safety, economic feasibility, environment impact assessment and public relations. The committee is also expected to highlight areas where policy decisions will be needed, the options, which are available, what they mean and the contexts in which they should be considered.

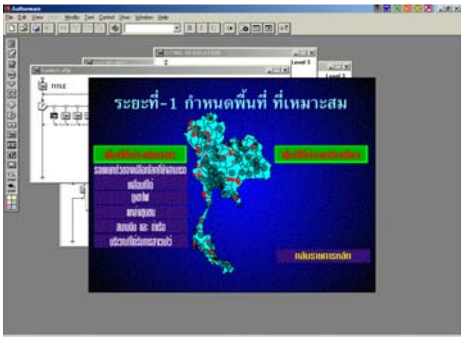
Almost the technical messages seem to be the great problem for general public to understand. EGAT had carried out a preliminary site selection for the Nuclear Power Project in three phases since 1981, based on the IAEA guidelines. As a result, in 1995 EGAT identified five candidate sites, namely Ban Bang Boet, Ban Laem Thaen, Ban Laem Yang, Ban Thong Ching and Ban Klong Muang. Why did EGAT have to select our areas?, the question from the publics in those areas, EGAT have to answer and explain the procedure of site selection that include a lot of technical terms and messages to those public. The problem was how to make the understanding to them with easy words in a short time. The Multimedia Software such as Authorware was considered to be the tool for the Site Selection Presentation, due to the properties of an icon based multimedia, object-oriented authoring system and so forth. The presentation programme was created as a simple prototype, included with the principle and procedure of Site Selection. Each procedure was explained by pictures and animation in stead of technical messages, by this way the audient could see the image during the presentment.

The question was answered when Site Selection Programme was presented to the representative of the public in the target areas, but the next question was how to make sure about the safety of Nuclear Power Plant. Explaination with technical messages still was the problem because it was very difficult to understand, they could not have image about safety in their mind. Visiting the Nuclear Power Plant might be the one option to be considered.

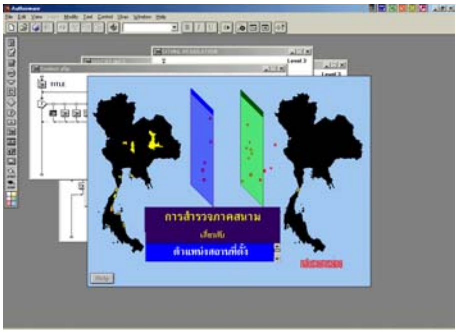
REFERENCE

U.S.N.R.C., Regulatory Guide 4.7, Rev. 1, “General Site suitability Criteria for Nuclear Power Stations”, November 1975, Washington, D.C.

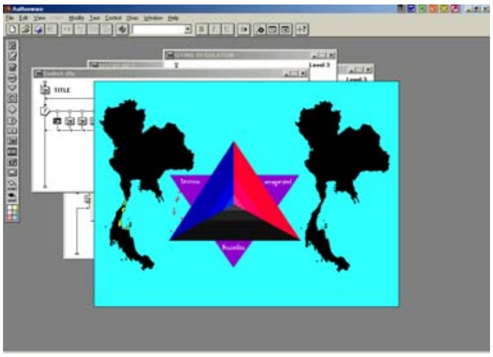
PHASE1: Identification Candidate Zones



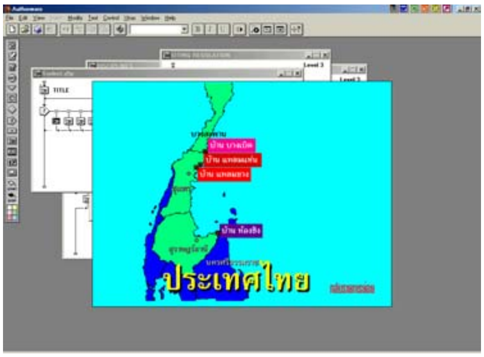
PHASE2: Identification of Potential Sites.



PHASE3: Identification of Candidate Sites.



PHASE4: Evaluation of Candidate Sites.



REVISITING THE NUCLEAR ENTERPRISE

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The peaceful, industrial and commercial nuclear enterprise in the US and the world has been virtually dormant for almost three decades now. Many national and international causes and events have contributed to this situation. The TMI nuclear plant accident in the US in 1979 and the Chernobyl accident in Russia in 1986 all have wrought international and political concern and doubts over the safety of and need for the nuclear enterprise.

The increase of terrorist activities throughout the world has heightened concern over the possibility of use of nuclear and radiation weapons. Countries throughout the world have been put on heightened alert and military actions by the US and others have exacerbated concerns.

The increase in the number of nations with actual nuclear arsenals and others with access to these nuclear arsenals has contributed to concerns. Even the association of nuclear power plants and possible production of nuclear weapon materials has been contentious. National and International Safeguards and Proliferation Programs have not constrained public concern. Weapons of mass destruction are particularly alarming, with particular fear over rogue nuclear weapons and so-called "dirty bombs."

The cleanup and management of radioactive wastes and dismantling of excess nuclear weapons materials have also contributed to the fear and hostility by many within government, business, and the public towards any activities associated with the nuclear enterprise.

Nevertheless, the "Atoms for Peace Program" as President Eisenhower presented before the United Nations General Assembly on 8 December 1953, still has great promise and unique, potential benefits for the U.S. and the world. Like many discoveries in science and new technologies, both risks and benefits have accrued with development and implementation of the nuclear enterprise. Can the nations and their people learn to successfully harness the atom and truly put it to beneficial use for mankind?

This paper attempts to review and assess the major positive benefits that can and should result from the peaceful nuclear option be fully exploited throughout the world. Focus is made upon the economic, environmental, and social benefits that could result with careful planning, development, public education, management, and international supervision.

The major benefits that will be addressed include the nuclear enterprise as a

- primary, economical, and essentially inexhaustible energy source based upon the use of existing resources of natural uranium and large inventories of depleted uranium;
- practical means of actually eliminating existing inventories of SNM in weapons as MOX fuel in power reactors;

- practical means for reducing the problems of long term spent fuel disposal through the recycle and burn up of transuranics;
- only major available primary energy source for the total elimination of greenhouse gases;
- economical and technically feasible means for significant production of large quantities of fresh water via desalinisation;
- most economical and mass production method for hydrogen as a liquid fuel for the eventual replacement of oil and natural gas without greenhouse gas release.

[1] US DOE, Energy Information Agency, www.eia.doe.gov.
[2] INTERNATIONAL ENERGY AGENCY, Paris, France, www.iea.org and www.worldenergyoutlook.org

A NEWSPAPER IS NOT A HEAT EXCHANGER: THOUGHTS ABOUT MEDIA AND NUCLEAR POWER

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The talk shall challenge the notion of the media as a system of conveyor, or pipelines, belts, or heat exchangers, or telephone lines. Media are not machines whose task would be the transport of information or THE TRUTH. Instead, media are a social system, which reproduces itself and its environment, and it does so by fulfilling the following functions:

- they help their customers to shape their understanding of the world
- they entertain their customers.

Print journalists, for instance, are agents of their principals: their readers. Journalists investigate and write stories.

This is not exactly what the nuclear community likes. It has a culture of secrecy and a cult of „facts“, i.e. impersonal knowledge. A knowledge, that is contaminated if any kind of personal interest or viewpoint is added. This notion of truth is outdated, and - interesting enough - the community itself gives an example for that.

Journalism could (and mostly did not) help the public to understand nuclear power. This requires to be open, honest and sceptical towards all players in the field: the nuclear community, politics, and the media themselves. But journalism often does not live up to this standard. Many journalists lack technological knowledge, follow mainstreams and/or partition the world in black and white.

PARALLEL SESSION D

NUCLEAR SAFETY AND SECURITY DEVELOPMENTS

INTERNATIONAL NUCLEAR SECURITY

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The potential theft or diversion of nuclear material has been an international concern since the beginning of the peaceful use of nuclear energy. In 1972 the Director General of the IAEA convened a panel of experts to develop recommendations regarding physical protection of nuclear material in use, storage and transport. The resulting document, "Recommendations for the Physical Protection of Nuclear Material", commonly known as the "Grey Book", was published under the auspices of the IAEA in June 1972. These recommendations addressed only theft and were described as an essential supplement to the State System of Accounting and Control (SSAC).

A first revision of the document in 1975 resulted in upgraded recommendations for physical protection. Three levels of protection were introduced by identifying three categories of nuclear material with correspondingly increased levels of protection. The potential danger of radiation to health and safety of personnel the public and environment by exposure to radiation or release of radioactive substances as a result of a deliberate act against a nuclear material has been considered by addressed threats of sabotage and plutonium dispersal. The revised recommendations were published by the IAEA as INFCIRC/225 in September 1975.

After 1975, INFCIRC/225 has undergone four revisions; in 1977, 1989, 1993 and 1998[1]. The revision in 1998, was the result of a comprehensive review made by Member States experts under the auspices of the IAEA. INFCIRC/225/Rev. 4 (Corr), *Physical Protection of Nuclear Material and Nuclear Facilities* underlines the responsibilities of the State to establish and maintain a State physical protection system and introduces more detailed recommendations to clarify the role and responsibilities of the national competent authority and those of the operators. INFCIRC/225/Rev. 4 (Corr) gives new emphasis on the protection against sabotage of certain nuclear facilities.

The Convention on the Physical Protection of Nuclear Material (CPPNM)[2] was adopted in 1979 and entered into force in 1987. After the end of the Cold War the number of illicit trafficking cases in nuclear materials triggered an awareness of the need to strengthen the international physical protection regime. In 1999 an open-ended group of experts was convened by the IAEA Director General to examine the need to strengthen the Convention on the Physical Protection of Nuclear Material. The work of the group of legal and technical experts to draft an amendment to the CPPNM in 2003 resulted in a report proposing possible amendments to include: the extension of the scope to cover

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[1] Issued in 1999.

[2] INFCIRC/274/Rev.1

nuclear material in domestic use storage and transport as well as the protection of nuclear material and facilities from sabotage.

The cases of illicit trafficking in the 90's have shown that international cooperation is a must to establish an effective system of prevention, detection of, and response to inadvertent and illicit trafficking. The events of September of 2001 in the USA demonstrated a new scale, dedication and organization of terrorist groups. The acquisition of a nuclear weapon or related materials remains the gravest concern. The threat of a radiological dispersal device (RDD) or sabotage of a nuclear facility or transport became also serious concern. The potential consequences of terrorist attacks resulting in a release of radioactive substances, which could affect neighbouring countries point to a transnational dimension of nuclear security.

Thus in the post-9/11 period nuclear security must consider the potential of: a) the theft of a complete nuclear weapon; b) the theft of nuclear material for the purpose of constructing a crude nuclear explosive device with or without the active involvement of a State, c) the theft of nuclear and other radioactive materials to construct a radiological dispersal device (RDD); and d) attacks directed against nuclear facility or a nuclear transport. A global nuclear security regime is the frame to counter nuclear terrorism by measures of prevention and detection of and response to theft, sabotage, unauthorized access, illegal transfer or other malicious acts involving nuclear material, other radioactive material or their associated facilities.

The IAEA's mandate, technical competencies, extensive experience, and global reach make it a well-suited international organization to effectively assist States in improving their nuclear security systems. To confront the post-9/11 nuclear security threats and to provide nuclear security assistance to States, the IAEA Board of Governors, in March 2002[3], approved a *Plan of Activities for Protection Against Nuclear Terrorism* and assigned the highest priority to its coherent and effective implementation. The Plan covers the three lines of defence: prevention, detection and response, supplemented with activities in support of information management and co-ordination. While the responsibility for nuclear security within a State responsibility rests entirely with the Government of the State, the IAEA provides advice and assistance to States in strengthening their nuclear security systems.

[3] GOV/2002/10, 5 February 2002

PROLIFERATION RESISTANCE ASSESSMENT: AN ILLUSTRATION THROUGH THE FRENCH FUEL CYCLE

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Proliferation Resistance has become an essential issue of public interest and more generally for the acceptance of nuclear energy systems. It is an integral part of the requirements defined by the INPRO project launched by the International Atomic Energy Agency as well as other projects such as Generation IV. It fair to recognise that proliferation resistance has been implemented since a long time and is currently implemented everyday. Proliferation Resistance Assessment Methodologies are being developed in several places in the world, making best use of past and recent works. Proliferation Resistance is achieved through a combination of technical features, which are defined as “intrinsic” to the technology and institutional and other measures, where safeguards inspection plays an important role, defined as "extrinsic" measures. Work is still on going to design a comprehensive methodology which will account for all elements and provide an integrated assessment for the proliferation resistance of a given nuclear energy system (a nuclear energy system comprises the reactor and the fuel cycle, and encompasses the whole life cycle). However, work already performed is providing the building blocks of the assessment of the proliferation resistance of a given nuclear energy system. And decades of practical implementation offer concrete examples. The building blocks include the “quality” of the nuclear material, the “attractiveness” of the facilities, the safeguards inspection, the export control regime or the localisation and size of the fuel cycle services and facilities. This paper is reviewing the most significant of those indicators, and illustrates their use against real life examples coming from the French nuclear fuel cycle industry.

CURRENT STATUS AND DEVELOPMENT PERSPECTIVES OF STATE NUCLEAR AND RADIATION SAFETY REGULATION IN THE RUSSIAN FEDERATION

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The state radiation safety regulation in Russia stems from the establishment in 1946 of the state radiation monitoring service on the basis of Laboratory N 2 (the RNC "Kurchatov Istitute" nowdays). The nuclear safety was subjected to control later – following the setting up of corresponding divisions in the Obninsk Physics Power Institute and Nuclear Power Institute (the RNC "Kurchatov Institute") in Moscow. Since 1963 NPP safety was brought under oversight by various organisations.

Since that time a series of organizational changes in bodies involved in controlling nuclear and radiation safety (establishment of new bodies, their convergence and separation) took place. The safety oversight was implemented both inside the former Soviet Union " nuclear " agency and by various organisations belonging to other agencies. It should be noted that the main emphasis was laid on controlling nuclear safety.

Gosatomnadzor of Russiaa was shaped as it is now in 1991 (first as the State Committee, then – the State Committee under the President of the Russian Federation, then – the Federal Service and finally -. The Federal Authority). Gosatomnadzor of Russia has become not only a supervision, but also regulatory body. The shere of competence of Gosatomnadzor of Russia was extended to cover, along with NPPs and nuclear reactors, nuclear fuel cycle facilities and national economy organizations as well as military facilities. Later, by Decrees of the President of the Russian Federation (1993, 1995) the Ministry of Defense and a number of Minatom's facilities and organizations were taken out from the Gosatomnadzor of Russia's oversight.

Presently, basic areas of activities of Gosatomnadzor of Russia being the state authority for nuclear and radiation safety regulation, spring from the Federal Law on " Atomic Energy Use" [1] and are defined by the " Statute of the Federal Nuclear and Radiation Safety Authority of Russia" approved by the Government of the Russian Federation in 2002.

Among these areas are the following:

- Development, approval and enactment of federal rules and regulations in the area of atomic energy use;
- Licensing with the aim of ensuring safety of activities in the area of atomic energy use;Organization of safety expert review of nuclear installations, radiation sources and storage facilities;Oversight for compliance with nuclear and radiation safety rules and regulations as well as license conditions;Conduct of inspections by Gosatomnadzor of Russia's inspection departments with regard to their authority.

Now Gosatomnadzor of Russia regulates more than 2000 enterprises and organizations with various forms of ownership**.** They operate more than 6000 objects, which use atomic energy (including NPPs, fuel cycle facilities, marine power installations, research reactors, facilities for storage of nuclear materials, spent fuel and radioactive waste, etc).

In 2003 Gosatomnadzor of Russia developed and approved 8 federal rules and regulations pertaining to atomic energy use (in addition to 41 which were developed earlier), 1 safety guide and 33 regulatory guidelines

During this period 1416 licenses were issued for enterprises and organizations operating in the area of atomic use (currently, there are more than 4300 licenses) and 3 870 permissions to work in the nuclear area were given to personnel of facilities using atomic energy .

11 211 inspections of enterprises and organizations operating in the area of atomic energy use were carried out. In the result of the inspections 12051 cases of non- compliance with federal rules and regulations or license conditions were revealed and prescribed to be eliminated. In regard to more severe cases of non-compliance, 27 licenses were suspended until corrective measures were undertaken and 111 prescriptions were made to halt works.

Well-planned actions for implementing the state oversight and enforcement, enhancement of requirements from the inspector's side resulted in 2003 in a considerable decrease of non-compliance cases- from 12 294 in 2002 down to 12 051 in 2003. The number of prescription requirements enforced by Gosatomnadzor of Russia, but not fulfilled has become much lower – from 157 in 2002 to 77 in 2003. This allows to conclude that the nuclear and radiation safety of facilities using atomic energy is maintained at an adequate level and tends to be continuously improved.

Among the main challenges that should be overcome in the upcoming years, the following should be highlighted:

- With regard to upcoming transition of the safe use of atomic energy regulation to the regulatory framework where technical regulations will be applied in the Russian Federation [2], it is necessary to complete very soon the development of federal rules and regulations in the nuclear field (including regulations on the production and use of MOX fuel) and concentrate efforts on producing within stringent time limits a number of technical regulations, including the generic technical regulation " On Nuclear and Radiation Safety". It will be also needed to introduce relevant changes to some federal laws and legal acts of the Russian Federation. Substantial assistance in this respect is rendered by experts of the G-8 Nuclear Safety and Security Working Group, EC regulatory bodies and international organizations (IAEA, OECD/NEA).
- According to the " Fundamentals of State Policy Pertaining to Ensuring Nuclear and Radiation Safety in the Russian Federation for the Period up to Year 2010 and Further Perspective", approved by the President of the Russian Federation on 04 December 2004, it is necessary to solve a number of significant tasks aimed at enhancing the state safety regulation of atomic energy use, including the improvement of the licensing and oversight effectiveness.
- It has become imminent to develop the emergency response concept in the system of Gosatomnadzor of Russia and the concept of state regulation of nuclear and radiation safety for facilities produced by peaceful nuclear explosion.

[1] Federal Law "On Atomic Energy Use" № 170-FZ, 21 November, 1995 г.
[2] Federal Law "On Technical regulation". № 184-FZ, 27 December 2002 г.

REGULATORY APPROACH FOR THE LIFETIME MANAGEMENT OF THE NUCLEAR POWER PLANTS IN KOREA

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There are 18 NPPs in operation and 2 NPPs under construction in Korea, since the first NPP started its commercial operation in 1978. 9NPPs have been operating more than 10 years, and Kori unit 1 which is the oldest PWR plant approaches its design life of 2008. As the number of aged plants is increased, public concern has been increased on the safety of operating NPPs.

Systematic and comprehensive operational safety assessment and plant life management are necessary to maintain a high level of safety, taking into account improvements of safety standards and practices, the cumulative effects of plant aging, operating experience, and the evolution of science and technology.

In Korea, operating license is issued without a fixed term and design life is specified in Final Safety Analysis Report (FSAR). Institutional scheme for the plant life extension should be devised in the very near future, considering the remaining life of Kori unit 1 and the time required for the safety assessment.

The technical and economic feasibility studies for the lifetime management of NPPs have been carried out for last 10 years. However the licensing basis for aging management (mainly apply to the long-lived and passive components) should be prepared for the institutionalization of the plant life extension of NPPs. Also different opinions between the regulatory authorities and stakeholders should be converged.

On the other hand, Periodic Safety Review (PSR) system was introduced and established with sound legal basis in order to evaluate the comprehensive and systematic safety of operating plants. The PSR review scope is based on the safety factors suggested by IAEA [1]. The results of PSR with focusing on aging assessment can be utilized for the life extension of NPPs. Also Nuclear Safety Commission recommended to use the results of PSR in the life extension of NPPs.

In this study, the PSR system enforced in Korea would be compared with the current periodic inspection system and the License Renewal of the USA in the aspect of procedural requirements on the plant life extension. Then basic factors [2] for life management are considered in PSR and License Renewal.

Considering the present status and the comparison results on the life extension of the NPPs in Korea, three possible models considered on the sound legal basis are:

- License renewal: limit license term and renew a license term (less than 20 years) beyond the established the licensed term. Typical example is the USA formulation.

- Approval for continued operation beyond design life: license a fixed term (10~20 years) by the legal approvement. Typical example is provided by Swiss or Hungarian regulation.

- Continued operation with PSR results without any procedural requirement: license every 10 years according to the PSR results. Typical example is provided by the UK or French regulation.

In view of the safety assessment for the lifetime management of the NPPs in Korea, the above models would be seriously discussed in terms of characteristics, merits and demerits for each other. For any models, PSR results can provide useful information for the decision of continued operation.

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NUCLEAR POWER SAFETY: THE PRESENT AND SAFEGUARDS FOR THE FUTURE

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Specific factors of danger emanating from a new power engineering – the employment of power from nuclei fission – from the very beginning determined basic points of special attention to the safety assurance of its matters: a chain fission reaction having a high potential for uncontrolled power generation as well as radiation and possible spread of radioactive materials.

The natural priorities of safety measures are as follows: understanding and mastership of a controlled chain reaction and tightened requirements imposed on the quality and reliability of design statements and equipment as well as the creation of new forms in the organization of works in this new power engineering.

At the initial stage of atomic power engineering formation in the Soviet Union, the factor of a distant location from densely populated regions was of significant importance for the population protection from possible accident situations. That factor determined differences in design statements for technical barriers preventing accident spread of radioactive materials.

On the progress in the scale of atomic power engineering, the unification of approaches and requirements to the technical means of safety assurance developed all over the world took place in the USSR. This process was completed during the period of mitigation of consequences from the accident at the 4-th unit of Chernobyl NPP in the course of which technical and organizational disadvantages in domestic approaches to safety assurance of nuclear power plants were revealed.

The system of safety assurance formed by the present time and combining the efforts of all countries is logically closed and transparent. This system allows to overcome the public distrust to atomic power arisen out of large accidents.

The reduction in the initial danger of a nuclear facility by the improvement of the facility inherent self-protection properties and development of passive methods for an accident control represent the principle of safety assurance being of importance and accepted by the international nuclear community. The central place in the safety concept supplementing the strategy of the defense-in-depth takes a danger management realized on developing means for the accident management. This produces safety assurance supported by the knowledge base formed by the whole nuclear community.

A TALE OF TWO THEORIES

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Attempts to assess the consequences of living with high-risk technologies fall into two categories, leading to opposing conclusions. High Reliability Theory claims that it is possible to devise measures that will effectively guard against the occurrence of significant accidents [1]. Normal Accident Theory holds these as an inherent, and therefore inescapable, feature of high-risk technologies. As far as the civil use of nuclear power is concerned however, neither theory has given a satisfactory account of the past accident record. The 'Normal Accident' vision leads to overestimate the likelihood of significant accidents. On the other hand, High Reliability Theory did underestimate it in the early days of nuclear power. It is therefore legitimate to ask which of these theories will provide the more appropriate framework for assessing future performance.

On the face of it, the objections raised by Normal Accident Theory are damning [2]. Human operators are expected to cope with failures in the design, maintenance and managerial framework of highly complex, tightly coupled systems. Defects in these areas might have been sown in the system long before the occurrence of the mishap that will reveal their presence. System automation and defence in depth make the situation only worse by rendering the responses of the facility more opaque and unpredictable. The operators are left to their own devices to handle precisely those situations that have not been foreseen by the system designers. To do so, they must rely on mental tools notoriously poorly adapted to such a task.

However convincing these objections raised by Normal Accident Theory might look on paper, they do not give a faithful depiction of what is happening in practice. The measures implemented after Three Mile Island and Chernobyl address much more than the particular circumstances that gave rise to these accidents. Probabilistic safety assessments, advances in material science and non-destructive testing, emphasis on safety culture beyond mere Quality Assurance and promotion of best practice all hold great potential for the eradication of failures. The emerging picture is one in which most of the Normal Accident Theory objections are countered as postulated by High Reliability Theory. The one residual danger lies in the insufficient or imperfect use of the abovementioned tools.

This brings us back to an observation put forward by the chairman of WANO, in which overconfidence and complacency were identified as the main threats to the nuclear industry [3]. Successfully monitoring these twin pitfalls, however, requires overcoming two difficulties. The first one is that the said pitfalls become apparent only with the benefit of hindsight. The second one is that they might constitute a 'none of the above' explanation to be invoked when no other cause appears to apply. An indicator, or a set thereof, capable of giving an early warning regarding lapses in the safety performance would therefore be most welcome.

It is suggested that comparing the safety-related pattern of activities of two categories of nuclear facilities during well-chosen periods of time might provide a promising avenue for research:

- the facilities that had apparently achieved a high level of safety *before* this was denied by the occurrence of an accident; and
- the facilities recognised for having greatly improved their safety performance *after* an accident.

What differentiates their patterns of activities might well provide an objective basis for the identification of complacency and overconfidence and therefore avoid any hindsight bias. The said differences might then be translated into one or several safety performance indicators. If this can be achieved, it is submitted that High Reliability Theory will definitely provide a convincing framework for the safety assessment of the next fifty years.

[1] For want of a generally accepted definition, this wording is used here to refer to any event higher than level 4 on the International Nuclear Event Scale (INES).
[2] J. Reason, Human Error, Cambridge University Press, 1990. See especially Chapter 7, Latent Errors and System Disasters.
[3] Nucleonics Week, 16 October 2003: Complacency, Negligence Threaten Nuclear Industry

RADIATION SAFETY FOR PERSONNEL AND PUBLIC IN THE USE OF NUCLEAR POWER. EXPERIENCE GAINED, PROBLEMS, AND WAYS TO RESOLVE THEM

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Development at the end of the 40-th years in USSR of a nuclear industry has caused necessity of creation of the specialized medical-sanitary service, and accordingly, organization of state sanitary - epidemiological supervision. For the decision of the state program of medical-sanitary maintenance of creation of the nuclear weapon and atomic power engineering, in USSR in 1947 was created the 3-d Senior Management, transformed now into Federal Management "Medbioextrem".

The principle of a priority of safe working conditions and protection of an environment was fixed as a basis of activity system of the 3-d Senior Management.

The limits of irradiation dozes of the personnel, maximum permissible concentration of radionuclides in air of working premises, allowable pollution levels of working clothes, equipment etc. were put in force in 1948 at the first enterprises of PO "Mayak".

Coming-to-be of the Russian state from the end of the 80-th years has pushed development of state regulation of safety in all industries, and for all that the special attention was given and is given to radiation safety of the personnel, population, and environment at use of an atomic energy in the peace and defensive purposes. In short term a number of the laws is developed, the bodies of state regulation of safety are created, and the basic Federal normative documents of the sanitary legislation are developed.

From the point of view of supervision functions performance in all the nuclear industry history, Health Ministry of Russia has coped successfully with its tasks. The following facts testify to it. The emissions of working nuclear stations in Russia create negligibly small dozes on the population (< 2 mkSv per one year), the emissions of the enterprises of Atomic Ministry cause an irradiation of the population of ZATO in 4 times below than doze quota of emission for nuclear stations.

At the same time the radiation failures that occurred in 1957 – 1997 have resulted in human victims and brought significant economic, political, social damage to the country.

Each of these failures has required putting in force the special emergency rules of radiation-hygienic aspect, which were aimed at the restriction of irradiation of the population undergone to emergency irradiation.

Putting in force the rigid limits of an irradiation of the personnel and population, high requirements of bodies of State supervision and control, the political realities put before a leadership and experts of Atomic and Health ministries of Russia a complex and crucial task of solving of technical, organizational, medical-hygienic, and social problems of modernization of the senescent enterprises Atomic ministry, applying new technologies and kinds of nuclear fuel in, creation of an infrastructure for manipulation with INF and radioactive wastes, physical protection of nuclear and radioactive materials in a context of the prevention of radiological terrorism.

PROSPECT OF THE PROLIFERATION RESISTANT FUEL CYCLE TECHNOLOGY DEVELOPMENT IN KOREA

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Nuclear energy plays a major role of supplying stable and sustainable energy in Korea. Currently, 18 nuclear power units with 15,716 MWe capacities are in operation, and the nuclear share in power generation is about 40 percent. As the utilization of nuclear energy increases, the management of the spent fuel becomes imminent task to be resolved. While the ultimate policy of the back end fuel cycle is still under consideration in Korea, the main direction of the research is to develop the proliferation resistant fuel cycle technology to reuse the spent fuel for effective management of the accumulated spent fuel in a proliferation resistant way.

Since Korea is developing the improved nuclear reactor system utilizing the oxide fuel for the near term deployment, and innovative nuclear system utilizing the metallic fuel for the long term application, the proliferation resistant fuel cycle technology development has mainly two directions of researches, that is, DUPIC (Direct use of spent PWR fuel in CANDU reactors) technology for the oxide fuel system and the pyroprocessing technology for the metallic fuel system, respectively.

There has been a significant achievement in the DUPIC technology development since its beginning of 1991. The basic concept of the DUPIC fuel cycle is to directly fabricate the CANDU fuel from the spent PWR fuel by using thermal/mechanical processes in hot cell without the separation of fission products and transuranic elements. KAERI has successfully fabricated several DUPIC fuel elements at hot cells remotely, and the performance evaluation through a series of irradiation tests at the Hanaro research reactor is under way. DUPIC technology is developed by international joint researches among Canada, the USA with participation of the IAEA, and is internationally acknowledged as a typical proliferation resistant fuel cycle technology. The current status of the development and the requirements for practical application will be discussed in the presentation.

The development for the pyroprocessing technology in Korea has recently been embarked on. The main objectives of the research are to develop the electrolytic reduction method of the spent oxide fuel, and to fabricate the metallic fuel for use in the future innovative nuclear system of fast neutron spectrum. It will be performed in close relations with the international Generation IV research program, and the basic research using inactive surrogate material is currently under way. The current status and future plan of the development will be discussed in the presentation.

As both fuel cycle technologies are well recognized in the viewpoint of the proliferation resistance aspect, the requirements of the future fuel cycle technology for the enhanced proliferation resistance with improved performance and economy will be discussed.

SAFETY CULTURE -- THE WAY OF THINKING

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The Safety Series No. 75-INSAG-4 published by the International Atomic Energy Agency in 1991 points out, in clear-cut terms, that "the human mind is very effective in detecting and eliminating potential problems, and this has an important positive impact on safety. For these reasons, individuals carry heavy responsibility. Beyond adherence to defined procedures, they must act in accordance with a 'Safety Culture'. The organizations operating nuclear plants, and all other organizations with a safety responsibility, must so develop safety culture as to prevent human error and to benefit from the positive aspects of human action".

This paper showcases the implementation of nuclear safety culture at China Guangdong Nuclear Power Holding Co., Ltd. (CGNPC) in relation to the operations of the Guangdong Daya Bay Nuclear Power Station (GNPS) and the Ling Ao Nuclear Power Station (LNPS) comprising 4 Pressurized Water Reactors of 1000 MWe in Shenzhen, Guangdong, China. It emphasizes that the cultivation of a nuclear safety mentality or mindset is essential to the safe and stable operations of the nuclear plant. To this end, compliance with the rules and practices fully consistent with the nuclear safety culture requirements must be put into place at all times such that it becomes second nature of the staff of the nuclear plant. As a result, human errors can be forestalled to avoid maloperations or malpractices in operating the nuclear plant. This paper highlights the need for a keen sense of nuclear safety culture from top to bottom at the nuclear plant, and emphasizes that, without safety, there is no viable nuclear plant, as there would be no economic return for the operator. A safe nuclear plant is undoubtedly an economically competitive plant. To ensure safety, operators must operate the plant in a way that is free from human errors, viz. nuclear safety culture has become a way of thinking tantamount to second nature for the operating crews.

As is well known, making mistakes or committing errors is part of the growing up experience. The challenge is to prevent such mistakes from occurring in the first place, and the ideal solution is to rely on a built-in safety culture within the organization. Operators of the GNPS and LNPS have, from the outset, risen up to the above challenge by cultivating in the management and staff a full set of nuclear safety culture awareness, as evidenced by several specific cases outlined in the paper.

PARALLEL SESSION E

DESIGN AND DEVELOPMENT OF ADVANCED NUCLEAR SYSTEMS

EVOLUTION OF WWER REACTOR PLANTS FOR NUCLEAR POWER PLANTS

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Improvement of WWER reactor plants and their basic equipment (reactors, steam generators, reactor coolant pumps and other RP components), safety systems and process systems was systematically performed at all the stages of development of OKB "Gidropress" projects with participation of traditional partners (General designers –"Atomenergoprojects" Institutes, Scientific leader RCC "Kurchatov Institute" within the period of 1955-2003.

Improvement was performed on the basis of results of WWER operation at NPP with regard for development of requirements of home and foreign regulatory documentation.

Both design bases, in developing WWER of the next generations, and design solutions in terms of design and materials of the equipment, process systems, and safety systems have been improved. As a result, it was corroborated that a series of design solution laid down in the first WWER were original, they stand the test of long-term operational experience and have become traditional for all generations of WWER reactor plants including present-day projects (WWER-640, WWER-1000, WWER-1500).

The results of comparative analysis between WWER and PWR have shown that WWER are found to be at the level of foreign PWR by a number of objective data and design solutions, and they are higher than this level in terms of separate characteristics.

THERMAL-HYDRAULIC EXPERIMENTS AND EVALUATION FOR SAFETY FEATURES IN APR1400

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Thermal-hydraulic experimental program and typical test results for the evaluation or verification of new safety system design features adopted in the APR1400 (Advanced Power Reactor 1400) are presented for major test items.

Development of the APR1400, an evolutionary type of pressurized water reactor (PWR), has been completed at the end of 2001 and succeedingly got the approval of its standard design on May 2002. It has a capacity of 4,000 MWth with 2x4 loop arrangement of the reactor coolant system (RCS) and 60 years of the design lifetime. APR1400 incorporates many advanced design features to enhance its safety. New design features adopted in the APR1400 include, among others, four trains of the safety injection system (SIS) with DVI (Direct Vessel Injection) mode and passively operating SIT (Safety Injection Tank), the In-containment Refueling Water Storage Tank (IRWST) and the Safety Depressurization/Venting System (SDVS). These new design features has been evaluated to provide significant improvements in the performance and safety, and they are major contributors to the reduced core damage frequency and improved performance against severe accident.

Since these new design features, which were not adopted in the existing nuclear power plants, are required to evaluate and verify their performance for ensuring their contribution to the safety enhancement, experimental activities for ensuring their performance and safety enhancement capabilities have been started in April 1999 for three years at KAERI under the national nuclear mid- and long-term R&D projects. They include the LBLOCA ECCS performance evaluation test for the DVI mode of safety injection, performance verification test of the fluidic device as a passive flow controller and performance evaluation test of the steam sparger for SDVS. And additional experimental works are still on-going focusing on new multi-dimensional thermal-hydraulic phenomena such as DVI line break (SBLOCA) and boron mixing relate to the DVI and thermal mixing and stratification in a large pool related to the IRWST under the governmental support (MOST & MICIE).

In this paper, the thermal-hydraulic test and evaluation program is presented, which includes the overview, test objectives and experimental and evaluation method, and some of typical test results also are given for each research item.

Key Words: APR1400, Thermal Hydraulic, Test and Evaluation, DVI, Fluidic Device, Sparger, IRWST, Safety Injection, Safety Depressurization

GT-MHR INTERNATIONAL PROJECT OF HIGH-TEMPERATURE HELIUM COOLED REACTOR WITH DIRECT GAS-TURBINE POWER CONVERSION CYCLE

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The international project of gas-turbine modular helium-cooled reactor (GT-MHR) presented in the report is a realization of high-temperature technology and is based on the experience with helium-cooled reactors with ceramic fuel particles and on innovative solutions concerning power conversion system with closed gas-turbine cycle and turbomachine with electromagnetic bearings. The international GT-MHR Project is currently being jointly developed by USA and Russia for disposition of excessive weapon-grade plutonium. For commercial electricity generation, the reactor will use uranium fuel. The GT-MHR combines a gas-cooled modular helium reactor (MHR) and a highly efficient integrated gas-turbine power conversion system (Brayton cycle) with expected cycle efficiency up to 48%. The reactor and power conversion unit are located in an underground concrete silo.

The GT-MHR technical characteristics and design features assure:

- high level of passive safety that completely prevents core melting in accidents with any scenario, including full loss of inert coolant;
- low level of thermal and radiation impact to the environment;
- capability to use various fuels in the core (e.g. low-enriched uranium, mixed uranium-thorium and uranium-plutonium fuel, or plutonium fuel) without modifying the core design;
- meeting the non-proliferation requirements through technology and properties of ceramic fuel particles;
- capability to achieve coolant temperatures of up to 1000 °C, which is needed for various industrial processes;
- high efficiency of electricity generation.

A whole complex of research and development activities is being carried out by RRC KI, OKBM, VNIINM, "Lutch", and other Russian organizations in order to support key design solutions, primarily on fuel, turbomachine with electromagnetic bearings, structural materials, vessels and computer codes.

At present, the GT-MHR capability to generate high-grade heat at temperatures up to 1000 °C makes it the only existing nuclear technology that can assure supply of high-temperature heat for thermal processes used for production of hydrogen (including hydrogen production from water), synthetic fuels, fertilizers, etc.

THE EUROPEAN PRESSURIZED WATER REACTOR, A SAFE AND COMPETITIVE SOLUTION FOR FUTURE ENERGY NEEDS

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1. EPR Design Philosophy

The EPR design philosophy is governed by three essential targets:

- Improving the safety level compared to existing plants by deterministic and probabilistic considerations.
- Mitigating of hypothetical severe accidents by restricting their consequences to the plant itself.
- Providing economic power generation at costs competitive with other primary energy sources.

The economic requirements have led to a large electrical output of about 1600 MW, with which the EPR will reach outstanding specific investment costs. Operation costs are reduced by a high average fuel burn-up.

The design allows for cycle lengths between 12 and 24 months.

An average availability target for the design has been set at 92 % over the whole 60 year lifetime of the plant. Therefore, preventive maintenance features are incorporated in the design to optimize outage durations.

2. Safety Approach

A twofold strategy is pursued for the EPR safety requirements:

- To improve the preventive measures against accidents.
- To mitigate Severe Accidents consequences, even if their probability has been further reduced. This is achieved by implementing features to ensure containment integrity. Thus, it can be demonstrated that the need of stringent countermeasures are restricted to the immediate vicinity of the plant.

The safety approach includes a strong deterministic basic complemented by probabilistic analyses in order to improve the prevention of accidents, as well as their mitigation.

Accident Prevention measures are enforced by:

- Simplification of the safety systems.
- Elimination of common mode failures by physical separation and diverse back-up functions for safety functions.

- Increase of grace periods for operator actions by designing components (e.g. pressurizer and steam generators) with larger water inventories to moderate transients.
- Less sensitivity to human errors by an optimized man-machine interface by digital instrumentation and control systems and information supplied by modern operator information systems.

Low probability events with multiple failures and coincident occurrences up to the total loss of safety-grade systems are considered in addition to the deterministic design basis.

Representative scenarios are defined for both, core melt prevention and the prevention of large releases.

3. Technical Features

The Reactor Building is located in the center of the plot plan. The Containment is surrounded by the Safeguard Buildings and the Fuel Building, which contain the safety systems. All safety systems are designed in a four-fold redundancy and located in physically separate divisions.

Each division comprises a Low Head Safety Injection/Residual Heat Removal System with the related intermediate cooling system, a Medium Head Injection System and an Emergency Feedwater System. The related electrical systems as well as the instrumentation and control systems are also allocated in these divisions but on a higher building level.

The inner Containment is constituted by a pre-stressed concrete cylindrical wall with elliptical head and a reinforced concrete basemat. The leak tightness is assured by a metallic liner.

The outer Containment is formed by a reinforced cylindrical wall, resting on the same basemat, and covered by a reinforced concrete dome which serves as protection against external hazards (airplane crash).

Since the Severe Accident Mitigation approach imposes special requirements on the Confinement function of the Containment, systems for isolation, retention and control of leakages are proved. Leakages through the inner Containment wall are collected, filtered and released via the Annulus Air Extraction System.

Conclusion

The EPR with its innovative design features will be the trendsetter for the future of nuclear power plant engineering, the strong support of important entities and organizations of the European nuclear industry ensure a serious and reliable approach for the realization of nuclear power plant projects with EPR technology. On the 18th of December 2003 the first EPR was ordered by TVO, Finland.

EUROPEAN UTILITY REQUIREMENTS: COMMON RULES TO DESIGN NEXT LWR PLANTS IN AN OPEN ELECTRICITY MARKET

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The major European electricity producers want to keep able to build new nuclear power plants and they believe 3rd generation LWRs would be the most adapted response to their needs in the first decades of this century. Producing a common European Utility Requirement (EUR) document has been one of the basic tasks towards this objective. In this common frame, standardized and competitive LWR NPPs could be developed and offered to the investors. This idea is now well supported by all the other actors on the European electricity market: vendors, safety regulators, HV transmission grid managers and regulators, administrations.

The EUR document today includes all the parts that were foreseen when the EUR works started in 1991. Two sets of generic requirements have been developed: one dedicated to LWR nuclear islands, the other one to power generation plants. Another part (volume 3) deals with the application of the EUR generic requirements to those LWR designs that may be offered in Europe. The EUR generic requirements have undergone detailed reviews by peer utilities worldwide, as well as by vendors and regulators. After ten years of development and checking the EUR document is now complete and operational. It has been used as the base for the call for bids of the fifth Finnish nuclear unit in 2003.

In the competitive and unified European electricity market that is emerging, the electricity producers' stakes are increasingly different from the other electricity business actors'. The next term objectives of the electricity producers involved in EUR are focused on negotiating common rules of the game, together with the different rule makers. This covers the nuclear safety approaches, the conditions requested to connect a plant to a HV grid, as well the design standards. Discussions are going on between the EUR organization and all the corresponding bodies to develop stabilized and predictable design rules that would meet the constraints of nuclear electricity generation in this new environment. From these exchanges and from internal EUR works, a revision D of the EUR volumes 1 & 2 will be developed that will fit the new European environment. Its release is foreseen in 2007.

There cannot be competition without competitors. The EUR organization has already proven to be a good place to establish trustful relationship between the vendors and their potential customers. The main common tasks have been to assess the level of compliance of the designs vs. the utility needs in the frame of the EUR volume 3. This will be continued and developed with the main vendors present in Europe, to keep alive a list of 4 to 6 designs "qualified" vs. EUR. In particular, the EUR organisation has agreed to work out two new subsets of their EUR volume 3, one dedicated to the Westinghouse's AP1000, the other one to the AEP Moscow's VVER AES92. These two new subsets should be available in 2006.

ITER – THE FIRST EXPERIMENTAL THERMONUCLEAR REACTOR

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R&D FOR THE FUTURE NUCLEAR SYSTEMS: STAKES, CHALLENGES AND INTERNATIONAL COOPERATION

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In the XXIst century, the stakes related to power production – increasing demand, particularly in developing countries, and environmental concerns, with the necessity to reduce greenhouse gas emissions – are major and global challenges. The world's population is expected to expand from about 6 billion people to 10 billion people by 2050; international energy projections anticipate energy consumption increasing by 100% and more in the same period; at present 2 billion world inhabitants do not have access to electricity, and 1 billion lack decent access to drinking water. The world faces both necessities of increasing the energy supply and of fulfilling the sustainable development targets, and reducing the prospect of global warming.

Energy savings and renewable energy sources will and must be able to contribute to these objectives, however it will not be enough and nuclear energy should provide a sustainable solution for the very high stakes involved: long-term resources (several thousands of years, if the fuel use is optimised) and no greenhouse effect or toxic gas emissions. The industrial operators and markets are becoming more and more global, and so is becoming R&D, in order to gather skills and share costs for developing the key innovative technologies for the future of nuclear energy.

In the year 2000, was launched the "Generation IV International Forum" (GIF) at the initiative of the USA, to face-up to these long-term challenges and to build sustainable nuclear systems, including reactors and fuel cycle. The six GEN IV systems were selected by GIF with the help of leading international experts because of their significant potential to advance the sustainability, safety, economics and proliferation resistance of future nuclear systems. As well as electricity generation the plants offer potential for the generation of hydrogen from water for use in transport and for water desalination. All are considered deployable by at least 2030, with some possibility available as early as 2020.

In the same period, the INPRO initiative, from IAEA, is defining the socio-economical requirements for the nuclear systems to be built in the next 50 years.

In parallel, experts are working in a European framework, in order to identify common requirements and R&D goals for the future nuclear systems. The presentation will describe the main features of the 4th generation nuclear energy systems, the priority scientific and technology challenges for R&D, and the related international cooperation.

The feasibility of each system generally depends on two or three breakthroughs, so the R&D efforts are focused on projects dedicated to reach these breakthroughs.

This paper will also analyse how these international processes have permitted to develop the vision in perspective of the fuel cycle options.

PERSPECTIVE SODIUM FAST REACTOR BN-1800

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Review of experience gained in design, construction and operation of BR-10, BOR-60, BN-350, BN-600, Phenix and Super-Phenix reactors, as well as design of BN-800 and EFR has shown that technology of sodium cooled fast reactors offers considerable reserves in the improvement of their technical, economical and safety characteristics.

Proposals have been worked out by the Minatom of Russia on large size advanced sodium cooled fast reactor design.

Basic design of BN-1600 reactor was taken as a basis, and considerable modifications were made to the arrangement of the components, parameters and design approaches. The following results were achieved:

- owing to the increase of sodium temperature and use of supercritical steam parameters, it became possible to significantly improve thermodynamic parameters of the steam cycle assuring 46%/43% gross/net efficiency;
- use of integral primary circuit arrangement with all primary components located in the reactor vessel, thus assuring near zero probability of radioactive sodium leak, application of passive safety systems and further expansion of use of inherent safety features (including advantages of nitride fuel) have made it possible to significantly improve reactor safety;
- use of high density nitride fuel assuring high burn-up has resulted in the increase of reactor refueling interval (1.5-2 years) and, taking into account increase of reliability of the components based on the BN-600 reactor experience, achievement of ~0.9 load factor value;
- use of innovative design approaches as applied to reactor refueling system, steam generator, auxiliary systems, etc., increase of the fuel burn-up and load factor along with high efficiency has resulted in the improvement of technical and economical characteristics of the power unit.

Comparison of the main parameters of BN-1800 reactor NPP and those of VVER-1000 (V-392 Project) and VVER-1500 (on condition of two power units on site) has shown that technical and economical characteristics of large size advanced sodium cooled fast reactor are similar to those of VVER reactors and much better than those of previous designs of sodium cooled fast reactors (Table 1).

Table 1. Comparison of specific characteristics of sodium cooled fast reactors

Parameters	BN-600 in operation (BNPP)	BN-800 under construction (BNPP)	BN-1800 (conceptual design studies)
Power unit capacity, • thermal, MWth • electric, MWe	1470 600	2100 880	4000 1780
Technical and economical characteristics (as compared to those of the BN-600 reactor) • specific metal consumption (ton/MWe) of reactor facility • specific capital cost of NPP with one power unit (roubles'91/kWe)	1 1	0.7 0.9	0.33 0.48

Use of radiochemical technologies without extraction of plutonium and minor actinides, as well as technologies with only partial extraction of fission products would significantly decrease environmental impact by the nuclear fuel cycle of fast reactor.

There is also significant improvement of characteristics of reactor and fuel cycle from the standpoint of non-proliferation of nuclear materials.

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INTERNATIONAL COOPERATION FOR THE NEXT 50 YEARS: THE INTERNATIONAL PROJECT ON INNOVATIVE REACTORS AND FUEL CYCLES (INPRO)

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During the last fifty years remarkable results are achieved in the application of nuclear technology for the production of electricity. Looking ahead to the next fifty years it is clear that the demand for energy will grow considerably and also the requirements for the way the energy will be supplied.

For INPRO the future is of the energy demand and supply is explored and several scenarios identified. A leading requirement is coming up and will play a crucial role: sustainability of the way the energy supply will be realized. Fulfilling the growing need for energy in developing countries is as well a crucial issue.

Based on these scenarios for the next fifty years a inventory of requirements for the future of nuclear energy systems has been collected as well a methodology developed to assess innovative nuclear systems and fuel cycles. On the base of this assessment the need for innovations and breakthroughs in existing technology can be defined.

To facilitate the deployment of innovative nuclear systems also the infrastructure, technical as well institutional has to be adjusted to the changes in the world such as the globalisation.

In the contribution to the conference the main message from INPRO will be presented.

NUCLEAR CONTRIBUTION TO THE POWER INDUSTRY IN THE 21ST CENTURY: ELECTRICITY, FAST REACTORS, BREST CONCEPT

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Electromagnetism discovered in the 19th century gave us electricity – a form of energy most convenient for utilisation, as well as for centralised industrial-scale generation, network distribution and long-distance transmission. Because of the high cost due to considerable consumption of expensive fossils, however, it accounts for a mere ~1/6 of total energy consumption in households, industry and transport.

Fission of heavy nuclei discovered in the 20th century showed a physically and technically simple way of producing cheap electricity at inexpensive nuclear power plants, using cheap, available and inexhaustible nuclear fuel, without greenhouse gas emissions. However, the neutron-deficient thermal reactors that originated in military engineering (E. Fermi, I. Kurchatov), are incapable of solving the problems of a large power industry, such as fuel balance, etc. Considerable neutron excess in fast reactors (E. Fermi, A. Leipunsky) is the main inner physical resource for settling the aggregate problems in a safe and convincing way. Therefore, fast reactors open the way to a "nuclear era" in energy supply - the real era of cheap electricity, with cogeneration of heat and use of different thermal reactors whenever profitable.

The experience of the 1970s – 1980s shows that the issue of plant safety and its closely associated problem of cost, the issues of waste and proliferation, were underestimated, while the high breeding rate in fast reactors has lost its importance in view of considerable plutonium stockpiles.

Nuclear plant safety improvement in the wake of serious accidents made these facilities 2 to 4 times more expensive than conventional power plant, having "swallowed up" the benefits of cheap fuel. It was already after the TMI accident that A. Weinberg highlighted the need for basically new reactor concepts, centered on the principles of "inherent safety". This appeal found proper response in the 1980s in Russia, where a search was started for a new concept of a fast reactor, with Weinberg's principles extended to cover the problems of waste and nonproliferation and referred to as "natural safety". This effort led the researchers to the BREST concept and has recently culminated in the design of a demonstration BREST-300 reactor. In the late 1990s, the USA initiated research programmes for "innovative" reactors of the IV Generation meant to replace the present-day NPPs and to ensure "survival" on the energy market.

Without urgent need for new energy sources, generations of nuclear engineers, trained to work and think in terms of technical improvement of the reactor concepts dating back to the 1950s-60s, are carrying on the optimisation effort in order to keep the "nuclear option" available as an energy diversification factor.

Studies on the BREST concept make us confident that a nuclear power plant with a fast reactor of natural safety may be created within the first decades of the 21st century, without going too far from the technologies developed for NPPs and nuclear submarines. This offers a sound opportunity of gradually entering, in this century, a new era in the development of the power industry – the era of nuclear electricity. This presentation discusses the results of the recent studies on the BREST concept and design.

PARALLEL SESSION F

NUCLEAR ENERGY FOR SUSTAINABLE DEVELOPMENT AND AN ENVIRONMENTAL FUTURE

THE ENVIRONMENTAL RISKS AND PROSPECTS FOR LARGE-SCALE NUCLEAR POWER DEVELOPMENT

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In the last few years more and more attention has been focused on both the environmental population health risk factors and safe environmental conditions. 50-year period of the world's first Nuclear Power Plant (NPP) operation allows stating that all important energy sources including heat energy, hydropower and nuclear power have a similar - historically-comparable - life period through which a wide range of negative environmental and health effects has emerged.

The report considers principal environmental implications related to operation of different-type power generation facilities and also the factors of their long-term prospects in the context of resolving power supply-related tasks under the concept of sustainable economic development.

Priority environmental problems which solution deems indispensable for large-scale development of nuclear power systems have been determined.

To justify large-scale nuclear power development, demonstration of its potentialities in order to resolve many technological & environmental challenges is of prime importance. Among them most important are:

- Management of produced radioactive wastes which cannot be involved into the Nuclear Fuel Cycle (NFC); and
- Prevention of severe accidents at both power reactor facilities and NFC nuclear & radiation risk sites.

In the area of setting and resolving the tasks related to nuclear power positioning to ensure environmental safety of the Russian Fuel & Energy Complex, a recent trend towards enlarging investigations on multi-purpose environmental risks analyses – including those related to different-type power producers - has exhibited. Ministry of Energy of the Russian Federation jointly with interested agencies has begun the implementation of a long-term project known as "An estimate of the actual status and prospects for reaching strategic goals in the State energy policy related to environmental safety of power system operation". The solution of priority tasks aimed at ensuring environmental safety of nuclear power systems taken together with the implementation of an expanded multi-purpose environmental analysis, when considering power system development, will make it possible to formulate some "reference points" on the path to strategic goals put by in the power environmental safety area for the period up to 2020.

To estimate prospects for power system development in the very long run, a more consistent approach to specific radiation safety areas with consideration for environmental safety of non-nuclear power technologies is necessary. Unfortunately, contradictory trends in the present-day approaches to power technology environmental safety estimates are prevailing. This means that, on the one hand, one has to do with underestimates of non-nuclear environmental risks, but, on the other hand, unjustified toughening of radiation regulation on the background of some - rather unproductive - scientific efforts is observed.

The latter includes, in particular:

- Attempts to revise the basic radio ecological paradigm;
- Ultra long-run forecasts (up to 10 to 100 thousand years ahead) using the concept of collective population exposure dose to examine alternatives of fuel cycle development based on already available technologies as well as on those which can be developed within few coming decades; and
- Setting the tasks of discovering low-dose effects on the population health in the close-to-NPP areas under specific conditions that in reality these effects are not very low but negligibly low.

Some recent positive tendencies in the concerned area (e.g., R. Clark's "Memorandum", the Chernobyl Forum, etc.) along with new problem issues are also considered in the report. The elaboration of a unified scientific attitude to these challenges deems to be an important condition of large-scale nuclear power development in the future.

RADIOLOGICAL IMPACTS FROM NUCLEAR INDUSTRIAL FACILITIES ON THE PUBLIC AND THE ENVIRONMENT – THEIR CURRENT MAGNITUDE AND THE NEXT 50 YEARS FORECAST

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Since their inception, nuclear industrial facilities (front-end cycle, reactors, back-end cycle) have been designed and operated with such a high level of containment that under normal operations, only a minuscule fraction of the radioactive materials used by these facilities has been released into the environment.

The best indicator of the radiological impacts from nuclear industrial facilities on the environment is "radiation dose to the public". This integrated indicator accounts for the radiation hazard (radiotoxivity) from each radionuclide, the transfer of each radionuclide into the environment, and the exposure scenarios for the group of individuals which are potentially the most exposed among the public. Potential radiation risk for humans is expressed in units of milliSievert (mSv).

From the "radiation dose to the public" indicator, regulatory authorities have set limits to strictly control air and liquid radioactive releases from nuclear industrial facilities into the environment. Until the last decade or so, regulatory limits for radioactive releases were derived from a public dose limit of 5 mSv per year. In practice though, it is emphasized that nuclear facilities were required to comply with much lower operational limits (often about a few hundred times smaller).

Over the last decade or so, the nuclear industry brought forward further significant improvements in effluent controls that resulted in significantly lower radioactive releases. A corresponding reduction of the resulting public dose impacts by about a factor 10 is not unusual. In parallel with these improvements, the regulatory framework has been updated in most countries on the basis of a reduced public dose limit of 1 mSv per year. This means that currently, public dose impacts from nuclear facilities are fort the most part about 100 times smaller (and sometimes even smaller) than either the public dose limit or background radiation, and about 10 times smaller than the local variation of background radiation. At such a low level of individual exposure, it is noteworthy to mention that the aggregation of doses over distance and time to a large group of people (i.e. collective dose) and the hypothetical theoretical radiation risk that can be calculated from it, becomes meaningless and of no significant value in societal decision-making.

With this background, the next 50 years forecast in terms of radiological impacts from nuclear industrial facilities might evolve as follows:

- Normal operations - Public dose impacts will continue to be very small and effluent controls will continue to improve though at a slower rate than in the past decade as the rationale for this decreases.
- Accidents - Steady improvements in the safety of containment of radioactive materials are expected, thus reducing the potential occurrence and severity of public dose impacts.

This paper also provides a perspective on the radiological impacts on the fauna and flora – an emerging topic on the international scene.

TECHNOLOGY AND KNOWLEDGE MANAGEMENT IN AND BY THE DEPARTMENT OF ATOMIC ENERGY, INDIA

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Indian Department of Atomic Energy (DAE) has been working right from its inception for harnessing nuclear science and technology for the development of the nation based on a well-planned strategy involving simultaneous pursuit of basic research and technology development with equal rigour. Over the years, a robust institutional framework for exploiting nuclear science and technology has been put in place. Now, in its golden jubilee year, the DAE runs major research centers, academic institutions and industrial units.

Basic strategy for the development of nuclear energy programme in India was formulated at a time when India hardly possessed any infrastructure to nurture any hi-tech activity. Keeping this in view, a large R&D establishment was progressively set up. This establishment, known as Bhabha Atomic Research Centre (BARC), consists of research reactors and other facilities for research and technology development. While setting up various other institutions, the Department has ensured that an organic linkage between all the institutions is maintained, and research and development lead to deployment of technologies. To achieve this objective, the DAE, besides setting up research centres, has also set up closely linked industrial units. The resulting synergy between research, technology development and industrial application has benefited all the agencies involved. As a result of all these efforts, several radiation and isotope technologies have been developed and deployed and India is self-reliant in all aspects of nuclear fuel cycle, starting with prospecting and mining of uranium and ending with the back-end of the fuel cycle, which involves reprocessing of the spent fuel and nuclear waste management.

All possible mechanisms for technology management have been adopted by the DAE to harness nuclear science and engineering. Technologies developed by R&D centers have been deployed in-house. Technologies have also been transferred to industrial organizations and many innovative methods have been used for such transfer. In the process of developing technologies, many organizations outside the DAE have been involved and this has led to strengthening of technology base of those organizations and strengthened technology base has been used by them for other sectors of the economy. Technology cross over has been very helpful in areas such as quality assurance, precision manufacturing, remote handling and robotics and design in general. The DAE has also adopted a proactive approach in transfer of technologies to other organizations and providing consultancy in hi-tech areas.

Several technologies have direct societal applications and latest initiative taken by the department involves implementing those applications by involving other government agencies.

A key element of working of the DAE has been the stress on human resource development. The working of various institutions under the department is so organized that while the research centres focus more sharply on technology and product development, the academic institutions concentrate relatively more on basic research. In the process, the research centres and the academic institutions have provided high calibre manpower to the Department but for which India’s spectacular strides in the field of nuclear sciences and their applications would not have been possible.

This paper looks at the strategies adopted in the past and outlines recent initiatives.

RUSSIA'S ENERGY RESOURCES IN THE 21ST CENTURY WITH ACCOUNT FOR THEIR ROLE IN THE WORLD ENERGY

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The long-term prospects (to the end of the 21st century) of Russia and world's energy development were studied to determine the extent of Russia's provision with energy resources, estimate their cost, production scale, final use and export, as well as the competitiveness of energy technologies. Bearing in mind an important role of energy resources export Russia's energy was considered as a part of world energy.

The currently available data on the world explored reserves and speculative resources of oil, natural gas and coal were systematized. According to the classification by accessibility, reliability, geological conditions of occurrence, natural quality, technical recoverability, economic efficiency of use and readiness to exploitation all the resources were divided into 8 cost categories with an extraction cost of USD 1–20 /GJ.

A long-term forecast of consumption of 4 types of final energy (electric, thermal, mechanical and chemical) was made for 13 regions of the world to 2100 based on the empirical relationships between the GDP growth rate, change in the energy/GDP ratio and the economic development level achieved. According to the forecast by the year 2100 the final energy consumption of the world will increase 2–4 times and that of Russia – 1.5–2 times as compared to the current level.

A 13-region model of the world energy system [1] was employed to calculate the optimal structure of energy technologies. The model described the energy in the form of interrelated processes (technologies) of primary energy production, its conversion to the secondary energy carriers and production of final energy. The model uses the data on several hundreds of energy technologies. The structure of energy was optimized based on the economic criterion with account for the constraints on extraction of resources, production of final energy, schedules of power supply, emissions of harmful substances, etc. The former USSR is represented in the model by 4 regions, including two regions of Russia (European and Asian parts). The scenarios that differ in energy consumption and constraints on nuclear energy use were studied.

The results obtained allow the following conclusions:

1. World traditional resources of oil (200–280 billion t) and natural gas (280–430 trillion cub. m) will be largely depleted by 2050–2075. In the second half of the century the use of more expensive non-traditional resources of oil (oil-bearing shale, heavy oil, crude bitumen) and natural gas (methane from coal seams and thick sandstone, and, probably, dissolved gas from the zones of abnormally high pressures and gas hydrates) is inevitable.

2. Russia is a world leader in natural gas and coal resources. Besides, Russia is well provided with heavy oil and natural bitumen resources. However most of them are located in the eastern areas of the country with severe climate, which will call for essential investments to convert these resources to explored reserves and to develop an infrastructure in the new production areas.

3. In the 21st century Russia will be a major oil and natural gas exporter. Oil export volume will be kept at the current level to the middle of the century with its drop by the end of the century. Natural gas export will be kept at the current level or will increase somewhat in the future.

4. The calculations show that despite a great extent of Russia's provision with fossil fuel resources the nuclear energy development is economically efficient (to the level of 70 GW in 2050 and 100–140 GW in 2100). The nuclear energy development in the 21st century will decrease the internal natural gas consumption for electricity production and will maintain optimal volumes of its export. In the second half of the century after depletion of cheap natural gas resources the nuclear energy will become competitive with natural gas in Russia's electricity market.

[3] BELYAEV L.S., MARCHENKO O.V., FILIPPOV S.P. et al. World Energy and Transition to Sustainable Development, Kluwer Academic Publishers, Dordrecht, Boston, London (2002).

A KEY POINT FOR NUCLEAR POWER DEVELOPMENT, SUSTAINABILITY OF URANIUM RESOURCES

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The purpose of this paper is to provide the results of a recent assessment carried out in COGEMA on the potential duration of uranium resources for nuclear energy generation; it relies upon numerous data from various sources: the "Red Book", an international joint report of IAEA and OECD; technical documents prepared by Generation IV working groups on the uranium consumption in various types of nuclear reactors ; recent and older publications – mostly from Japan – considering the extraction of uranium from sea water ; and energy generation scenarios developed by the World Energy Council.

The combination of demand scenarios with nuclear electricity generation scenarios leads to uranium consumption histories, which take the type of reactors into account. In the continuation of the present generation of reactors, thermal advanced reactors will probably represent the major part of the reactor fleet during the next thirty years, and most likely for more than that. Uranium prices may increase during this period; U and Pu recycling as well as recovery of U from tails may be considered. It is generally accepted that fast reactors should probably progressively replace thermal reactors during the second half of the century. Furthermore, as soon as economical trends are derived in association with the scenarios, non-realistic situations may be eliminated. For example, there is no reason to develop simultaneously on an industrial scale fast breeder reactors and extraction of uranium from seawater.

We are convinced that new systems will become available around the middle of the 21st century, either fast breeders, or new sources of uranium. Thus the uranium based nuclear energy generation –i.e. fission of U and Pu generated during irradiation- is sustainable with regard to resources. However we strongly recommend to improve our present assessment of uranium resources, to validate the dates at which we shall have assurances with regard to future systems feasibility. Economical aspects should be considered simultaneously. The delays in our business are so long that we have to lighten the scene for more than a half century.

ROLE OF NUCLEAR ENERGY IN UKRAINE

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The structure of energy in Ukraine was performed during the decades and was determined by the energy policy of the former USSR in order to create a unique energy system. Ukraine can comply the energy demand only partially due to its own organic fuel. To cover the energy deficit a certain course of the nuclear energy development on Ukraine was undertaken. Atomic energy of Ukraine commenced from Chornobyl Nuclear Power Plant (ChNPP) where the first nuclear power unit was put into operation at 1977. Further, the construction and the operation of other units and new NPPs were realized. General information about NPPs in Ukraine at present is given at the table below.

NPPs title	Unit number	Reactor type	Installed energy capacity (MW)
Zaporizhzhya	1	WWER - 1000/320	1000
	2	WWER - 1000/320	1000
	3	WWER - 1000/320	1000
	4	WWER - 1000/320	1000
	5	WWER - 1000/320	1000
	6	WWER - 1000/320	1000
South Ukraine	1	WWER - 1000/320	1000
	2	WWER - 1000/338	1000
	3	WWER - 1000/320	1000
Rivne	1	WWER - 440/213	415
	2	WWER - 440/213	420
	3	WWER - 1000/320	1000
	4*	WWER - 1000/320	1000
Chornobyl	1**	RBMK - 1000	800
	2**	RBMK - 1000	1000
	3**	RBMK - 1000	1000
Khmelnitsky	1	WWER - 1000/320	1000
	2*	WWER - 1000/320	1000

* – unit on the stage of construction completion
** – finaly decomissioning unit

We have 4 running NPPs (Zaporizhzhya, Rivne, Khmelnitsky, South Ukraine) where 13 nuclear power units are operating. The last reactor of Chornobyl NPP was abandoned in December 2000. The ChNPP is decommissioning. Ukraine is gaining the 8-th place in the world and the 5-th in Europe on installed capacities (11.880 MW). Nuclear energy complex ensure sufficient part of total electricity generation (more than 40 %).

Elaboration of Atomic Energy Strategy Development (AESD) in Ukraine for the period till 2030 and further perspectives was completed in 2003. The crucial point of AESD is the prolongation of reactor's operational resource for 10 – 15 years after 30 years of designed term. The second important element is the position where after commissioning of the 2^{nd} unit of Khmelnitsky NPP and 4-th unit of Rivne NPP. The total installed capacity of NPPs will be kept on the level around 14000 MW till 2030. Due to the maintenance prolongation period of the running units from 5 to 9 new reactors will be operating.

New reactors of improved safety will be selected on tender base. Taking into account the existing infrastructure and operating experience the advantage is given to the reactors of WWER type.

It is rational to construct new reactors on the existed NPP sites. It is foreseen in the strategy that due to the reliability and effectiveness of running NPPs to reach the mean worldwide level coefficient of installed capacity utilization (85 %) and authorized coefficient (1 person / MW) till 2014. AESD includes all processes and problems, which accompany the nuclear energy development. Among them: nuclear fuel cycle; handling of radioactive wastes; decommissioning; scientific, engineering, design supervision of nuclear energy complex; personal for atomic energy.

The report contains the information about the basic directions of solutions stated in AESD till 2030. The principle of nuclear energy self-finance for the concerned period is specified in the strategy.

At least, for the period of 2030 – 2050 years the stability of NPPs installed capacity level is predicted. Another version of the development predicts the increase of installed capacity to 25 % till 2050.

Thus, nuclear energy plays and will play the important role to ensure the electricity generation in Ukraine.

NUCLEAR ENERGY: SUSTAINABILITY AND ECONOMIC COMPETITIVENESS

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A growing number of prospective studies envision that nuclear energy, because it is carbon-free, will play an important, possibly essential, role in the world energy mix of the 21st century, would mankind succeed in embarking on a sustainable development path.

Among the challenges that must be taken up for this vision of a large expansion of nuclear power to occur, economics, from a liberalized electricity market perspective, is sometimes put forward as the major one, ahead of waste and natural resources management.

The present paper relies on the most recent European economic studies on the comparative costs of electricity generation to show that nuclear power is an economically competitive choice in the European electricity market, and may even prove to be 20% cheaper than cycle gas turbine (CCGT) units for base-load generation. Thus, the most recent French detailed study (DIDEME 2003), essentially performed from an investor standpoint, concludes that, for base-load generation units starting operation around 2015, nuclear power, with a levelised generation cost of 28.4 €/MWh, is more competitive than CCGTs (35 €/MWh, not including the cost of CO2 emissions). Moreover, the prospects of internalization of the greenhouse-gas emission cost in the total generation cost will boost even further the competitiveness of nuclear against gas-fired plants in Europe. In fact, in view of the huge investments required in the power generation sector in Europe and worldwide, the fundamental challenge rather relates to competitive electricity markets: how to arrive at making them function in such a way as to provide generators with incentives to invest “at the right time” with reasonable returns for investors, and not to impede investments that are economically viable.

From a longer-term perspective, the main challenge for enabling a large and global nuclear expansion relates to sustainability and obviously also includes a major economic dimension: nuclear energy must remain competitive while minimizing the volume and toxicity of radwaste it generates and making an optimal use of natural resources through spent fuel reprocessing and recycling.

Based on a synthesis of relevant detailed French and OECD economic studies on the cost assessment of the fuel cycle back-end, this paper shows that, until 4th generation fast spectrum systems implementing a closed fuel cycle be brought to industrial maturity, spent fuel reprocessing and plutonium recycling, as currently implemented in LWRs, is a economically affordable strategy providing major benefits in terms of sustainability. Indeed, this strategy only induces a 2 to 6% increase in the total kWh cost compared with the open once-through fuel cycle option. Thus, the current economic calculation carried out by the French utility EdF shows, in balance sheet/provisions terms, an accounting value of less than 1.5 €mills/kWh for future additional discounted back-end liabilities, including both spent fuel interim storage and reprocessing, and high level waste storage and final disposal. This evaluation does not take into account any further optimization of the whole back end process that could be expected in the future, with a goal towards 1 €mills/kWh.

ACHIEVING EXCELLENCE IN HUMAN PERFORMANCE THROUGH LEADERSHIP, EDUCATION, AND TRAINING IN NUCLEAR POWER INDUSTRY

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In order to achieve and maintain high levels of safety and productivity, nuclear power plants are required to be staffed with an adequate number of highly qualified and experienced personnel who are duly aware of the technical and administrative requirements for safety and are motivated to adopt a positive attitude to safety, as an element of safety culture. To establish and maintain a high level of human performance, appropriate education and training programmes should be in place and kept under constant review to ensure their relevance.

As the nuclear power industry continues to be challenged by increasing safety requirements, a high level of competition and decreasing budgets, it becomes more important than ever to maintain excellence in human performance and ensure that NPP personnel training provides a value to the organization.

Nuclear industry managers and supervisors bear the primary responsibility to assure that people perform their jobs safely and effectively. Training personnel must be responsive to the needs of the organization, working hand-in-hand with line managers and supervisors to ensure that human performance improvement needs are properly analyzed, and that training as well as other appropriate interventions are developed and implemented in the most effective and efficient way possible.

The International Atomic Energy Agency together with its Member States has provided for coordinated information exchange and developed guidance on methods and practices to identify and improve the effectiveness NPP personnel training. This has resulted in: plant performance improvements, improved human performance, meeting goals and objectives of the business (quality, safety, productivity), and more effective training programs.

This article describes the IAEA activities and achievements in the subject area for systematically understanding and improving human performance in nuclear power industry. The article also describes cooperation programmes between IAEA and national institutions (for example, the US DOE) in the field of training and simulators that have led to sustainable improvements in the personnel performance of operation, maintenance and other personnel. Current trends in the area of human performance improvement will also be discussed.

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POWER GENERATION ALTERNATIVES FOR THE XXI CENTURY

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Keywords: Power generation, nuclear power, Fossil fuels, Carbon Dioxide, CO_2, Greenhouse effect, global warming, environment.

Forecasts from different specialized sources indicate that the power consumption in the world will continue to increase. In Argentina it is expected that for the year 2020 the consumption will double the present values.

In 2003, in our country, fossil fuels – carbon, oil, gas – contributed approximately 48% to the power generation, while hydroelectricity was 43% and nuclear power 9%.

Fossil fuels have some advantages, main ones are present low cost and easy transport, but they have also many disadvantages in terms of contamination, environmental effects and non-renewable resources. The Carbon Dioxide (CO_2), that is produced when burning fossil fuels, is considered as one of the major sources of global warming in earth (Greenhouse effect), with devastating climate consequences in certain regions as dry seasons, floods etc.

In Argentina total CO_2 emissions in year 1998 (last measured) were 114 million Tons, Figure 1 shows the distribution according to different sources.

Although absolute emission values are not high, when compared with those of certain developed countries, some mitigation measures could be adopted.

Emissions due to transport are diminishing thanks to a strong reconversion of public and private vehicles to run on natural gas instead of gasoline or diesel.

But what are we going to do to optimize Power Generation lowering fossil fuels consumption?

Some environmental NGO´s insist that the only solution is to use the "so called" renewable energies – Solar, Wind, Biomas, Geothermal – but these sources contribute only with less than 0,03% to Power Generation in our country. Figures provided by the World Energy Council shows that only 2% of Power used commercially all around the world comes from "renewables". Although this output could be increased in the future, WEC estimates very difficult to reach even 5% for year 2020

Solar energy is employed successfully in some countries to heat water for household purposes or to produce little amount of electricity for specific purposes. It is tempting to think that wind and sun, that are everywhere are free, could be a limited source of energy free of CO2.

Unfortunately these sources have several disadvantages that prevent them to be used efficiently for base load Power Generation.

Another alternative, available right now, affordable and safe that proved to be reliable and economically competitive is Nuclear Power.

In Argentina, thanks to Power generated by the Atucha and Embalse Nuclear Power Plants, we avoid contaminating the air with 70.000.000 tons of Carbon Dioxide.

Although we cannot assert that Nuclear Power alone will solve the Greenhouse effect, what we can conclude is that without an increasing participation of it, the global warming effect will not have a positive solution in the next century.

THE PERCEIVED BENEFITS FOR TURKEY ADOPTING NUCLEAR ENERGY, AND THE LESSONS LEARNED FROM THE ATTEMPTS MADE TO INTRODUCE A NUCLEAR PROGRAMME

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The International Atomic Energy Agency (IAEA) was formed in 1957.It had two major objectives. It aimed at promoting nuclear energy to make the maximum contribution to world energy needs. And it aimed to prevent the spread of nuclear weapons.

Turkey became an IAEA Member State in 1957,a year after the establishment of the Turkish Atomic Energy Commission which later was re-established as the Turkish Atomic Energy Authority (TAEK). TAEK is a governmental organization and has served as a driving force for all institutions enhancing and broadening the fields of interest thereby fostering nuclear oriented activities in many spheres.

Turkish Electricity Generation and Transmission Corporation (TEAŞ) was a state owned enterprise established for the performance of services for production and transmission of electrical energy in conformity with general economy and policies of the State under the Decree of Board of Ministers.

Interaction with the Agency and Technical Co-operation in particular became a very essential component of the widespread programme within the general framework of nuclear applications in the country. Most of the assistance to Turkey was related to nuclear applications.

Turkey had been trying for many years to establish a comprehensive energy development plan where nuclear power shall have an important share. Turkey has tried to launch a nuclear power plant project in Akkuyu third times in years 1976,1983 and 1993 and these attempts have ceased in a couple of years, because of some "surrounding reasons".

Integration of the nuclear power plants into the grid system of Turkey will bring the following profits:

- Long-term reliability of the plant (around 40 years of life-time)
- Introducing a diverse fuel into the Turkish electricity production system beside fossile fuels like coal, natural gas and fuel oil.
- With only one fuel loading, months of reliable operation from the aspect of fuel,
- During operation, environmental friendly behaviour of the plant.

Short, medium and long-term nuclear policy of TEAŞ was in brief:

- Realization of nuclear projects in "Turn-key"basis for the beginning,
- By know-how and technology transfer, in the second phase, to realize nuclear power plants in "Packages" augmenting indigenous share,
- Maximizing the local share and managing the project on "Component" basis.

Least but not the last, Turkey's and TEAŞ's policy includes the strong conformity to all signed treaties and agreements by the Country and in addition,to respect all safety and environmental standards accepted.

From the three attempts Turkey has gained the followings:

- Site selection capability. There is one site already having the site permit.
- Experienced staff to prepare the bid specifications both in package-vice and in turnkey basis.
- Technical, administrative, commercial and economical evaluation experience.
- Capability of carrying out contract negotiations, and preparation of contract documents both in package-vice and in turnkey basis for conventional type contract.
- Experience in Built-Operate-Transfer type contract negotiations.
- Trained and experienced technical staff to start and carry the project

However because of the delays imposed, the followings are the losses for the country and TEAŞ:

- The nuclear program could not start effectively although there have been three attempts,
- Turkey has lost some of her industrial production capacity from time to time because of electricity insufficiency,
- Know-how and technology transfer in high amounts have been disrupted since the project could not be started,
- Invested money on the site and on the staff have partly been lost,
- Last but not the least, the energy problem of the country is still in the agenda and seems to continue.

When the decision will be taken to prefer one of the bidders, it will be the day of right economical conditions for both the Project and the Turkish economy.

This paper intends to give brief information on the Turkish experiences, lessons learned and the surrounding problems of the three attempts on the bidding process and summarizing the situation for the time being probable future problems which can be encountered are mention in the paragraph as well.

PARALLEL SESSION G

NUCLEAR FUEL CYCLE AND WASTE MANAGEMENT

NEW RADIOCHEMICAL TECHNOLOGIES OF REPROCESSING OF SPENT NUCLEAR FUEL

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New approach to the reprocessing of spent nuclear fuel (SNF) of power reactors, which was proposed earlier, is based on the following sequence of procedures:

- alkaline voloxidation of fuel (oxidation by air or oxygen in the presence of sodium hydroxide at 350-400^{0}C and confinement of tritium in the small amount of condensate);
- leach of oxide-salt mixture (removal of sediment-forming compounds, first of all, molybdenum, and primary rectification of radionuclides);
- extraction of ^{134}Cs, ^{99}Tc and ^{129}J nuclides from the alkali solution;
- nitric-acid dissolution of voloxidized fuel;
- evaporation of solution and crystallization of uranyl nitrate hexahydrate;
- extraction of uranium and plutonium accompanied by separation of TPE/REE fraction;
- conditioning of recycle fractions (U, U+Pu and Pu) for transfer for the fuel fabrication;
- conditioning of recycle fractions (Tc, J, Np, Am and Cm) for transfer for fabrication of transmutation targets;
- conditioning of Cs + Sr fraction for controlled storage;
- conditioning of REE and FP fractions for eternal disposal.

According to preliminary estimates, implementation of this technology would make possible factors of decrease of the amount of evaporated solutions, consumption of reactants and mass and overall dimensions of the main components respectively equal to 10, 2 and 6.

THE JOINT CONVENTION ON THE SAFETY OF SPENT FUEL MANAGEMENT AND ON THE SAFETY OF RADIOACTIVE WASTE MANAGEMENT. A MOVE TOWARD A GLOBAL SAFETY

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The Joint Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management (the *Joint Convention*) is the first international legally binding instrument to address the issue of safety on a global scale. It was drafted by a group of experts from July 1995 to March 1997, adopted by a Diplomatic Conference in September 1997 and opened for signature on 29 September 1997. The Convention entered into force on 18 June 1998, and to date has been signed by 44 States, of which 34 have formally confirmed, thus becoming Contracting Parties.

A first Review Meeting of the Contracting Parties was held in November 2003 in Vienna, through which the Convention has become fully operational.

The Joint Convention applies to spent fuel and radioactive waste resulting from civilian application. Spent fuel and radioactive waste from military application are covered by the Convention only when and if these materials are transferred permanently to and managed by civilian programs.

The Convention's main objective is "to achieve and maintain a high level of safety worldwide in spent fuel and waste management." The obligations of the Contracting Parties are largely based on the international safety standard developed by the IAEA, in particular on the principles contained in the IAEA Safety Fundamentals document "The Principles of Radioactive Waste Management" published in 1995.

The Convention is relevant not only for those countries having nuclear energy programs, but for any country where activity generating radioactive waste is carried out or planned, included medicine and research.

This paper will describe the origin of the Convention, its content, the potential benefits from being part to it, and summarize the findings of the first Review Meeting.

RADIOACTIVE WASTE MANAGEMENT IN UKRAINE: STATUS, PROBLEMS, PROSPECTS

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One of the main problems of the successful development of Nuclear Power Industry doubtless is the management of radioactive waste, which will be arising both during nuclear power plant (NPP) operation and during NPP decommissioning.

There are a different sides from the same problem - decommissioning of NPPs, rehabilitation the territory Estrangement Zone and radwaste Chernobyl origin within the 30-km Zone, maintenance of 8000 firms of the country's economic complex, which use of ionizing radiation sources, treatment of 6000 м3 solid radioactive waste by 13 operating units at four Ukrainian NPPs, liquidation of the storages former rocket -nuclear complex.

It is leading the analysis of all generators of radioactive wastes, including such specific one as the destroyed unit 4 of Chornobyl NPP and enterprises of a former military-industrial complex of the Soviet Union. The problem of a radioactive waste management is multilevel.

The tasks and their solutions at national, branch, industrial levels are considered. It is marked that in Ukraine based on the nuclear legislation and as the independent part of the national law is developing currently. The work under adaptation of Ukrainian nuclear legislation to European has amplified.

The infrastructure is unified at a branch level: container park, transportation, system of the radioactive waste calculation, process engineering.

The National Center on the management of low- and intermediate level radwaste and disposal are building.

The construction of the plant on extraction of radwaste from Chornobyl storages is finished. The construction of the plant on sorting and handling the solid and liquid radioactive waste, storages of radioactive waste arising from Chornobyl units decommissioning is begun.

The high level of international integration on radwaste management by Chornobyl's origin is marked.

Problems in a field of a radwaste management in Ukraine are:

- Outdated and inadequate legislation;
- Deficient of staffs for routine work on radwaste management;
- Availability of numerous targets;
- Lack of the complete technological schemes;
- Availability of parallel structures;
- Lack of market relations in a field radwaste management;
- Absence of a Fund on radwaste management.

The Programs of radioactive waste minimization at Ukraine NPPs stipulate the implementation of technological solutions developed recently:

- Remotely - controlled of complexes on a radioactive waste treatment;
- Systems of automatic monitoring behind transition of the radwaste and ionizing radiation sources;
- Installation on discharging of ionizing radiation sources;
- Parametrical set of containers.

On an example of the plants (Kharkov specialized factory and Zaporozhye NPP) the prospects of technology development and technical tools in the field of radwaste management are considered.

ACHIEVING LEGITIMACY IN THE UK'S RADIOACTIVE WASTE MANAGEMENT PROGRAMME

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This paper provides an update on the current status of radioactive waste management in the United Kingdom (UK) from the perspective of Nirex, the organisation responsible for providing safe, environmentally sound and publicly acceptable options for the long-term management of radioactive materials.

In summary, it proposes that:

- the waste exists and must be dealt with in an ethical manner;
- legitimacy is the key to public acceptance of any attempt to solve the waste issue; and
- credible options and a new political will allow, and indeed, compel this generation to deal with it.

The paper takes account of a number of announcements and ongoing developments in the UK nuclear industry, in particular:

- the announcement that Nirex is to be made independent of industry;
- the Department of Environment, Food and Rural Affairs and Devolved Administrations' Managing Radioactive Waste Safely consultation exercise;
- the creation of the Committee on Radioactive Waste Management to oversee the next stage of consultation;
- the creation of the Nuclear Decommissioning Authority to manage the civil nuclear site clean-up programme;
- the future of BNFL
- proposals for improved regulation of Intermediate Level Waste conditioning and packaging; and
- proposals by the European Commission for a new radioactive waste Directive.

These institutional and policy changes amount to an evolution of the back-end of the fuel cycle that represents the most radical transformation in the UK nuclear industry for many years. This transformation was made necessary by past failures in trying to impose a solution to the radwaste problem on the general public. Therefore, in order for these changes to provide a successful long-term radioactive waste management programme, it is necessary to pay as much attention to political and social concerns as to scientific and technical ones. It is crucial for all interested parties to act in an open and transparent manner so that the decisions made achieve a high degree of legitimacy and thus public acceptance.

JNFL AND COGEMA: A WIDE AND INNOVATIVE COOPERATION IN THE FIELDS OF SPENT FUEL REPROCESSING AND MOX FUEL FABRICATION

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Cooperation for the smooth and safe start-up of the Rokkasho Mura Reprocessing Plant (RRP)

For more than 15 years, COGEMA Group is working with JNFL and Japanese Makers for implementing the Technology Transfer Agreement signed in 1987 between JNFL and SGN (COGEMA subsidiary), by transferring the La Hague UP3 plant technology to RRP.

A few years ago, JNFL and COGEMA decided to deeply strengthen their cooperation, in particular in the fields of chemical, uranium and active tests, but also in the fields of preparation for operation, maintenance, and radioprotection, for the purpose of a smooth and safe start-up of RRP. This global assistance to be performed by COGEMA to JNFL is very innovative and can be summarized as follows:

- An important team of COGEMA engineers, 50 in average, are working with JNFL at RRP, for ensuring on line witnessing, assistance, guidance and advice, and training services during the performance of the tests. Around 80 to 100 COGEMA engineers are working at La Hague as back-up of this COGEMA RRP team.
- Implementation of specific 3 months operation training sessions on UP3 plant using the tutoring method, 1 JNFL trainee and 1 COGEMA UP3 operator acting as a tutor: such training is performed in the control room of UP3 on the key process units of UP3 facilities. A dedicated reprocessing program is prepared for each training session, so-called « Model Campaign » including, several unit start-up and shut-down phases, several changes of spent fuel types (BWR and PWR), drifts analysis, 7 Model Campaigns are scheduled from September 2001 to May 2004 corresponding to a reprocessing activity of 600 tons of spent fuels.

After more than 2 years, the results of the cooperation is satisfactorily implemented seen from JNFL and COGEMA:

- 6 Model Campaigns already implemented at La Hague plant, and corresponding to the training of about 100 JNFL personnel.
- The training at RRP is fruitful and direct exchanges between JNFL operators and COGEMA tutors in the control room of RRP is very efficient.
- The assistance of the COGEMA RRP team and of the COGEMA La Hague back-up team for the performance of the chemical test phase is positive and reactive, as well as for the

preparation of the uranium and active test phases. For this last point, and beyond the assistance that is usually implemented for such cooperation, JNFL and COGEMA decided to launch several working groups on topics such as “safety culuture”, the purpose of such working groups being to confirm that not only RRP, in terms of equipment and facility, is tested and validated, but also JNFL, in terms of organization, expertise, training, is in a good position to pass the next phases, in particular the comprehensive uranium test phase.

Cooperation for the Japanese MOX fuel fabrication plant (J-MOX)

The cooperation between JNFL and COGEMA on RRP has been extended to the Japanese MOX fuel fabrication plant, J-MOX, by the transfer of the technology for powder preparation and pellet fabrication used in the French MOX fuel fabrication plant MELOX. It includes technical support by COGEMA for the implementation of tests, which have fully demonstrated that the MIMAS process as used at MELOX plant is compatible with the PuO_2/UO_2 powder which will be produced at Rokkasho Mura Reprocessing Plant.

The presentation will explain and detail this wide and innovative cooperation between JNFL and COGEMA on the field of the back-end of the fuel cycle.

MORE THAN TWENTY YEARS OF STORAGE EXPERIENCE WITH CASKS

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History of the Base Material

Over the River Severn in Western England, which flows in the direction of Liverpool, there is an intricate construction spanning the river since 1781, located just before Birmingham. This is the first cast iron bridge in the world, the so-called "Iron Bridge", near the village of Coalbrooksdale. The cast iron parts, weighing nearly 400 tons altogether, appear at closer look like filigree and have nevertheless withstood every high water until now. In the meantime, the Iron Bridge is a symbol of the region and even of the early coal industry of England.

The Beginning of the CASTOR@ Story

Over twenty years ago, the CASTOR@ cask technology was developed and is now used on four continents of the world. The cask, which is made from ductile cast iron, fuJfi1Is an important purpose in the nuclear fuel cycle by providing a safe system for nuclear, spent fuel management. With its special feature, the double barrier lid system and conferred with the licenses of competent authorities, the cask is successfully produced by GNB, and serves both for transport and interim storage of spent fuel.

On November 30, 1978 at 10 o'clock, experts from around the world were invited to participate in the spectacular safety tests on the newly developed dual-purpose cask. The CASTOR@ la was quite conspicuous in the middle of the BAM test site, not only because of its bright yellow colour, but also due to its dimensions: weight approx. 70 tons, six meters long, and nearly two meters in diameter.

The First Cask Generation

The initial designs for CASTOR@ s included the la, Ib and Ic casks, which were designed to accommodate spent fuel with approx. one or two year's cooling time after discharge from the reactor, and had capacities of, for example, 9 PWR fuel assemblies or 16 BWR elements. The first storage cask loading worldwide took place in 1983 at the Paul Scherrer Institute in Switzerland, where the CASTOR@ Ic-Diorit was loaded with fuel from the Diorit reactor where. The cask was so large that the fuel had to be loaded through a transfer system; for this a smaller cask was loaded initially where the fuel was stored, and reloading was done through an airlock system into the CASTOR@ cask. The fuel is still stored in the cask in Switzerland today.

The Second Cask Generation

The next generation of casks was made for higher capacities and longer cooling times of the fuel. This includes the CASTOR@ V /21, for example, which holds 21 PWR assemblies of the Westinghouse type with five years of cooling time. This cask type was first loaded at the Surry Power Station in Virginia in an open-air interim storage facility there. For German fuel, the V/l9 was made for PWR fuel, and the V/52 for BWR fuel. One cask type was also made for canisters of vitrified high-Jewel waste from reprocessing, the CASTOR@ HA W 20/28C.

The New (Third) Generation of Casks

In order to accommodate the higher enrichments and bum ups of fue1 used in commercial reactors nowadays, a new generation of CASTOR@ casks has been designed. The CASTOR@ Va, for example, has been designed to hold 21 PWR assemblies, including up to 8 MOX assemblies, all with maximum enrichment of 5% U-235 and bum up to 75 GWd/MTU. The maximum heat output of the cask is 48 kW. For BWR fuel, the Vc can accommodate 61 elements with parameters the same as in the Va, and a heat output of 52 kW. For these higher parameters, new materials and design features had to be found.

SAFE STORAGE OF SPENT FUEL FROM MOBILE NUCLEAR POWER PLANT

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The report addresses issues of safe storage of spent fuel from mobile nuclear power plant (MNPP) "Pamir-630D" with 630 kW that was in operation during three years. That nuclear power plant consisted of gas-cooled reactor (weight of ≈76 500kg), gas turbine-driven set (≈76 000kg), two control units (≈2×20 000kg), and auxiliary unit (≈20 000kg). The reactor and turbine-driven set were supposed to be put on transport platforms and carried by tractors. The control and auxiliary units were set on track beds. "Pamir-630D" was constructed and tested in appropriate building. Set-up time was no greater than six hours after all units of the MNPP had reached the site. "Pamir-630D" was ready to be moved to the other site in 30 hours after the shut down. Service lifetime of "Pamir-630D" was 10 years: 7 years of storage and 3 years of operation. Operational lifetime was no less than 10 000 hours (non-stop operational period was no longer than 2000 hours). Dose rate at the boundary of restrictive area was no more than 65 μSv/h at the time of reactor operation. The dose rate was no greater than 3 mSv/h on side surface of biological shielding and 10 mSv/h on end surface of biological shielding of the reactor in 24 hours after the shut down.

Two pilot prototypes of "Pamir-630D" were constructed. The first one of "Pamir-630D" was put into operation in 1985 and was in operation 3668 hours. The second one was not put into operation at all.

The reactor unit included: vessel reactor cooled by gas "nitrin" based on N_2O_4, biological shielding, pipelines with valves. The reactor vessel and other auxiliary equipment were made of stainless steel, moderator and reflector - of zirconium hydride.

The reactor core (height of 0.5 m and diameter of 0.506 m) consisted of 106 fuel assemblies, each of them contained 7 fuel rods surrounded by stainless steel claddings of wall (thickness 4×10^{-4}m and diameter of 6.2 $\times10^{-3}$m). Fuel spherical particles of UO_2 enriched to 45% ^{235}U were embedded in Ni-Cr matrix. The share of nickel and chrome in fuel composition was 40%. Weight of ^{235}U in reactor core was 18.7 kg; weight of the reactor core - 5700 kg.

The mobile nuclear power plant "Pamir-630D" was shut down 26.11.1987 and up to 24.06.1988 the reactor was cooled by gas-liquid coolant circulation in main loop. After that time the reactor was cooled by liquid coolant circulation through auxiliary loop of accident cooling system of the reactor. The coolant was removed from all circuits 22.05.1989 and the reactor was filled in nitrogen and cooled by it until removing the fuel (January 1991).

The decommissioning plan included a short period of preparation for disposal followed by reactor and turbine units of "Pamir-630D" dismantling and safe short-term keeping of spent fuel in an appropriate temporary storage facility. Special equipment included tank for temporary keeping of radioactive pieces of reactor, turntable of biological shielding and turntable with devices and tank (height of 5.3m, a diameter of 2m) for the removing of the fuel assemblies, which were designed and constructed to unload the reactor core under water. A gas tightness of the assemblies was checked under water. The removed fuel assemblies were placed into the flasks under water after a gas tightness checking. The fuel assemblies with cladding defects were placed in addition into sealed pills.

The temporary storage facility consisting of two pools (volume of 2×28 m^3) with a water shielding (thickness of 3.1m) and a concrete shielding (thickness of 1.8m) was built in the reactor hall. Since then the spent fuel assemblies have been kept under water. Surveillance, monitoring and inspections are carried out to ensure that the spent fuel storage remains in good condition. The water quality is maintained in accordance with appropriate chemistry requirements. The flasks with the fuel assemblies are tested for leak proofness by the check weighting under water quarterly if its weight is stable and weekly in case the weight changes. The flask is removed for testing and correcting the leak proofness in hot cell in accordance with appropriate procedure if the reference weight increasing is 0.1kg.

At the time a decision was taken to shut down and decommission the MNPP it was foreseen that the spent fuel would be shipped to Russia afterwards. Unfortunately, due to the break-up of the Soviet Union, the spent fuel was left in the territory of the Republic of Belarus.

Since Belarus does not have substantial nuclear programme, there is neither appropriate dry storage facility no reprocessing plant for spent fuel. Belarus also lacks financial resources for long-time storage or reprocessing spent fuel. Moreover, even the reprocessing technology for this kind of nuclear fuel has not been developed mostly due to the fact that this sort of fuel was designed only for “Pamir-630D” and was not used in any other type of reactor. At the same time, there is no reason to develop such technology, design and construct the reprocessing plant for 43 kg of spent fuel. More than that, soon it will be necessary to move the spent fuel from the temporary wet storage facility to a dry storage facility given the fact that the flasks with fuel assemblies which originally were not designed and constructed for the purposes of long-time storage can lose tightness any time.

Besides, under the Convention on the Physical Protection of Nuclear Material this fuel has become category I material due to the decrease in the radiation level. It requires application of strengthened physical protection measures.

The best solution of this problem is to return the spent fuel from “Pamir-630D” to a reprocessing plant of Russian Federation as it had been planned before decommissioning of MNPP in 1987. A positive decision on its return to the Russian Federation will contribute to strengthening the physical protection as well as reducing its maintenance expenditures.

The Republic of Belarus would highly appreciate IAEA assistance in facilitating the return of the spent nuclear fuel to the Russian Federation within the framework of the Trilateral Initiative of the Agency, the Russian Federation and the United States of America.

POWER REACTOR FUEL CYCLE BASED ON NON-AQUEOUS TECHNOLOGIES

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For the last 30 years, RIAR has been conducting research on development of the advanced closed fuel cycle using pyrochemical reprocessing of irradiated fuel and obtaining of granulate that is immediately suitable for fabrication of fuel pins by the vibropacking method. Pyrochemical reprocessing and the vibropacking technology allow implementation of the process of granulated fuel production and fuel pin fabrication in the remote and automated mode. These capabilities ensure cardinal improvement of all basic performance of the fuel cycle: economy, safety, and ecological effect on the environment, which provides real background for manufacturing application of the new production processes.

By now, RIAR specialists have formulated and experimentally validated the basic principles of the advanced closed fuel cycle of fast reactors that is based on mutual compatibility of the U-Pu fuel regeneration technology and the fuel pin fabrication technology:

- employment of "dry" pyrochemical processes for reprocessing of irradiated fuel with production of granulated oxide fuel;
- employment of the vibropacking method for fabrication of fuel elements from granulated fuel;
- application of remotely controlled automated equipment to fuel reprocessing, fuel pin and FA fabrication.

The experimental basis for working through of the above-stated principles is the Pilot Research Complex (OIK), which includes: the Facility for granulated fuel fabrication and the Facility for fuel pin and FA fabrication. The fabricated fuel pins and FAs are tested in the BOR-60 and BN-600 reactors.

USE OF WEAPONS-GRADE PLUTONIUM FROM DISMANTLED NUCLEAR WEAPONS FOR THE PEACEFUL OBJECTIVE OF ELECTRIC POWER GENERATION - AIDA/MOX 2

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Following previous bilateral cooperative activities, the Russian/French/German studies conducted from June 2, 1998 to June 1, 2002 under a Trilateral Intergovernmental Agreement (AÏDA/MOX 2 program) have produced a detailed description of the facilities to be built in the Russian Federation to transform W-Pu into MOX fuel and of the modifications to implement in Russian reactors so they can be loaded with MOX fuel in compliance with safety requirements. These studies, which include a cost estimation of the whole project, are described along with the main results and achievements.

POSTER PRESENTATIONS

ESTIMATED SOURCE TERMS AND OPERATIONAL EXPERIENCE APPLIED TO ANGRA 1 NPP

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Operation of nuclear power plants results on discharges of radioactive effluents to environment. Dealing with the basic objective of radioactive waste management in order to protects human health and environment now and in the future without imposing undue burden on future generation is made a previous analysis of the source terms discharge. These source terms are essential issues for environmental impact analysis. This paper compares the results of calculated source terms discharged from Liquid Waste Processing System and those measured during the operation of Angra 1 Nuclear Power Plant.

1. Introduction

During the normal operation of nuclear power plants there are several radioactive materials released to the environment. Those radioactive materials have been arising in the nucleus of the reactor or in your immediate neighborhood. The estimation of the medium amount of radioactive materials released from the liquid and gaseous waste treatment systems are done based on calculation models and on the operational experience. The principal parameters used in the calculations of the source terms have been compiled of the operational experience of the several plants in operation. This work presents a comparison of the expected source terms to be discharged on effluents streams from the wastes treatment systems of Angra 1 NPP and the actual activities measured after 20 years of operation of the plant. The results presented together with the other available operational data can be used to update principal parameters values to forecast calculations of source terms to another similar reactors.

2. Source Terms Assessment and Measurement

Releases rates of noble gases and iodines to the primary coolant are calculated using diffusion models. For other fission products, whose diffusion mechanisms cannot be appropriately specified, the release rates are derived from operational typical values and adopted as design base. In the same way, the activities concentration of the activation products are also based on similar plants operational experience [1].

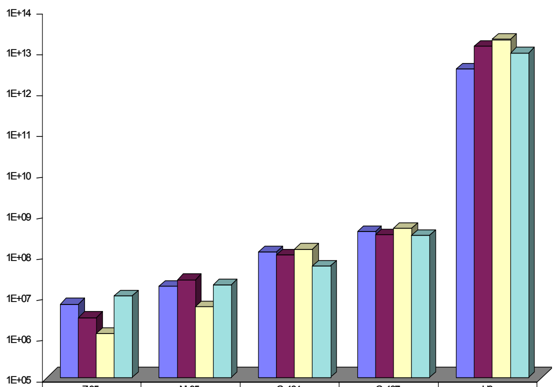
The main source of liberation of radioactive material to the environment is the Liquid Waste Processing System (LWPS), through the discharges of the Waste Monitor Tanks (WMT's) [2]. The fission products inventory in the primary coolant was calculated starting from the activities of noble gases and iodines present in reactor core after 650 consecutive days operation on full potency [3-6]. The corrosion products inventory in the reactor coolant is based on operational measures of similar reactors [7]. In this case it is considered that the plant already operated during your halth lifetime of about 20 years, taking into account the consume of the components. All liquid waste generated during the normal operation is collected and processed by LWPS before be discharged into the environment.

On the other hand, all potentially radioactive liquid effluents released from LWPS is sampled and analysed in order to verify the adequacy of effluent processing to meet discharge limits to unrestrict areas. The results of effluent sampling program are reported on regular bases to the CNEN (8).

3. Source Terms Comparison

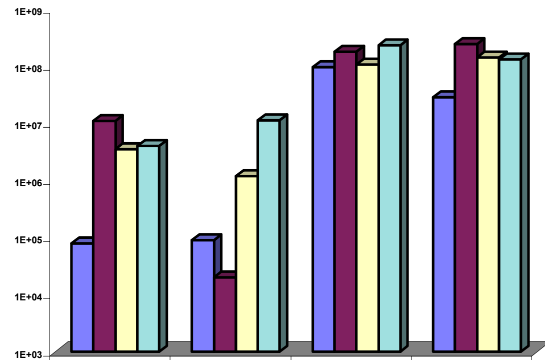
The estimated releases from LWPS and the principal design parameters used for the assessment are presented in ref. [2]. The monthly total activities of each isotope released with LWPS effluent and used to comparison are presented on Effluent and Waste Reports from years 1999, 2000 and 2001 [9,10,11,12,13,14].

As it can be observed in Fig.1, the activities of the certain fission products estimated during the licensing and measured in effuents are on the same order of magnitude. The differences among 3H activities measured and calculated are about an order of magnitude. Corrosion products leakage rates estimate based on operation reference plant data base and using principal design parameters of the plant get to be two orders of magnitude lower than the measured as 54Mn and 59Fe. Fig.2 displays comparison bethween activities measured and estimated.



	Zr95	Nb95	Cs134	Cs137	H3
FSAR	6,29E+06	1,70E+07	1,18E+08	3,70E+08	3,56E+12
1999	2,93E+06	2,45E+07	9,79E+07	3,18E+08	1,28E+13
2000	1,20E+06	5,28E+06	1,38E+08	4,43E+08	1,86E+13
2001	9,72E+06	1,84E+07	5,39E+07	2,98E+08	8,74E+12

Fig 1 Fission produts release from LWPS



	Mn54	Fe59	Co58	Co60
FSAR	7,78E+04	8,88E+04	9,62E+07	2,81E+07
1999	1,09E+07	2,00E+04	1,77E+08	2,40E+08
2000	3,49E+06	1,18E+06	1,06E+08	1,41E+08
2001	3,98E+06	1,13E+07	2,32E+08	1,30E+08

Fig. 2 Corrosion produts release from LWPS

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SPENT FUEL TRANSPORT: A CONTINUOUS IMPROVEMENT

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Since the 70's, approximately 30,000 tons of spent fuel has been safely transported to COGEMA-La Hague plant from French Nuclear Power Plants (NPP) and foreign nuclear operators involving more than 5,500 shipments.

COGEMA LOGISTICS made this possible by the continuous development of adapted transport casks duly licensed according to the regulations in force, the procurement of dedicated transport equipment such as wagons, trucks and ships, and an efficient transport organization providing a comprehensive door-to-door service.

New markets are under development implying new routes and organization. This paper is aimed at presenting our approach to meet the future challenges.

TENDENCIES OF OCCUPATIONAL EXPOSURE AT IGNALINA NPP

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Throughout the world, occupational exposures at nuclear power plants have been steadily decreasing for over a decade. Regulatory pressures, technological advances, improved plant designs, and improved operational procedures have contributed to this downward trend. However, with the ageing of the nuclear power plants the task of maintaining occupational exposures at low levels sometimes may become increasingly difficult.

Lithuania has one nuclear power plant – Ignalina NPP, which contains two RBMK-1500 reactors (actual thermal power output – 4200 MW, electrical power capacity – 1500 MW). The first Unit of Ignalina NPP went into operation at the end of 1983, the second Unit in August 1987. Due to unique construction, the occupational exposure doses for the reactors RBMK type are one of the highest if comparing with reactors of other types. Due to effective implementation of the radiation protection programme, ALARA programme, using of work management (work planing) programmes and modernization of equipment used at the Ignalina NPP occupational exposure doses are decreasing during the period from 1997 to 2001 (from total collective dose in 1997 – 18.51 man Sv to 6.27 in 2001). In 2002 the total collective dose was 7.33 man Sv, and 2003 – 8,53 man Sv. Last two years a small increase of collective dose due to modernization and implimentation of safety measures and larger amount of repair works is observed at first Unit of Ignalina NPP.

The main works which contributed to the highest collective doses during last two years were:

- in the reactor vessel: maintenance, repairs, replacement of the reactor fuel channels; installation of the system of the additional emergency reactor protection;
- main circulation circuit: inspection of the primary system pipes d=300mm, d=800 mm, of drum separators, repairing of the primary system pipes;
- repairing of the reactor equipment and refuelling;
- insulation works;
- installation of the temporary shielding;
- rooms decontamination;
- other general works.

Proper implementation of works cannot be guaranteed without effective implementation of valid radiation protection programme. Basic regulation which establishes requirements for radiation protection of workers working at the nuclear power plant and for radistion protection of members of the public during the nuclear power plant operation, is the Lithuanian Hygiene Standard HN 87:2002, Radiation Protection at Nuclear Power Plant" [1]. It was approved by the Order of the Minister of Health and came into force on 1 April 2001.

According to this Hygiene Standard, practices at the nuclear power plant shall be authorized and conducted in accordance with the basic radiation protection principles: justification of the operation of radiation sources, optimization of exposure and limitation of doses. The license holder shall ensure that doses of NPP workers do not exceed limits of occupational exposure determined by [2] and [3], with the exception of cases, when special conditions are applied. The daily effective dose constraint established by the license holder for Ignalina NPP workers is 0.2 mSv.

According to the requirements set out in [1], the radiation protection programme shall be established at the plant. Following items shall be included in the programme:

- classification of working areas and access control;
- local rules, measures of supervision of safety at work and order of organization of work;
- procedures of monitoring of workplaces and individual monitoring of workers;
- individual protective equipment and rules for their application;
- main premises, control systems for assurance of radiation protection;
- requirements for management of radioactive waste;
- radiation protection in the case of accident;
- application of optimization principle (ALARA) and measures on exposure reduction;
- programs of health surveillance;
- mandatory training of workers and their instructions.

The charts, dose analysis and implemented measures for reducing occupational doses will be presented in the poster.

A HISTORICAL SURVEY OF THE IGNALINA NPP

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When the boom of nuclear power industry began in the former Soviet Union, the idea of constructing the Ignalina NPP occurred to the circles in Moscow's central institutions at the turn of the 1970&. The nuclear power plant remained a facility under all-union jurisdiction supervised by the Ministries of Atomic Energy and Medium-Machine Building of the USSR from September 16, 1971, when the Central Committee of the Communist Party of the Soviet Union and the Council of Ministers of the Soviet Union adopted the resolution regarding the beginning of its construction, until Lithuania regained independence in 1990.

Nuclear power is the basis of Lithuania's power industry. The Ignalina NPP is a product of the former Soviet Union. Two reactors of RBMK-1SOO type are operational at the Ignalina NPP. Admittedly. This is the most advanced and the most recent version of the RBMK reactor design series (only two reactors of this type have ever been built). The power plant was built as part of the Soviet Union's North-West Unified Power System rather than to meet Lithuania's needs. The first unit of Ignalina NPP was commissioned in late1983, and the second one in the August of 1987. A total of four units with RBI\1K-1S00 reactors were to be built. However, due to political and safety motives the construction of the third unit was suspended as early as 1989.

After Lithuania declared independence in 1990, the Ignalina NPP was still guarded by Soviet troops and KGB operatives, and remained under the jurisdiction of the Soviet Union until the August of 1991. Supervision was carried out by that country's regulatory authority, the State Nuclear Power Supervision Inspection (Gosatomnadzor). It was only after the political events of the August of 1991 in Moscow that the Ignalina NPP finally came under the authority of the Lithuanian Republic. It is now controlled administratively by the Lithuanian Ministry of Economy, and its supervision is carried out by the Lithuanian State Nuclear Power Safety Inspectorate (V AlESI).

In this paper will be present brief information about the Ignalina NPP, information how the Ignalina NPP came into being, about prerequisites for the construction of the Ignalina NPP, about work on constructing the NPP and protests of the academic community, about speeding up the construction of the Ignalina NPP with an energy crisis imminent and the debate on termination of Unit 3 construction.

The material on appearance of the Ignalina NPP and problems related to it is collected at the Special Archive of Lithuania, and the Central State Archive of Lithuania. Special Archive of Lithuania has been using fund No.1771 of the Central Committee of the Lithuanian Communist Party whose documents mostly reflect the activities of the Central Committee of the Lithuanian Communist Party in supervising the construction of this AI1- Union facility. The fund contains, among other things, documents of correspondence with different AII-Union institutions reflecting Moscow's position o~ the issue. Resolutions of the

Central Committee of the Communist Party of the Soviet Union and the Council of Ministers of the Soviet Union whose second copies are stored in the archive have also been used for the present survey. Only the minutes of the sittings that were held in Moscow were sent to Lithuania; the shorthand records and primary material are absent. Nearly all the quoted documents were marked as top secret, which shows that the Ignalina NPP was a facility of extreme importance to both Moscow and Vilnius, as in the 1970s and 1980s most of the Communist Party's documents were marked simply as secret.

Fund No. R- 754 of the Managing Department of the Council of Ministers of the Lithuanian SSR was-also studied in the Central State Archive of Lithuania. Unfortunately, very few documents related to the subject under consideration remain there. The files of the 1976-87 period on construction of the Ignalina NPP and issues related to it are completely absent, although nearly all the resolutions of 1974-75 remain. The originals of the bulk of the documents are also missing, only their copies have been preserved.

The situation is even worse at the archive of the former State Planning Committee of the LSSR, whose documents were not handed over to Central, State Archive of Lithuania after 1963. This archive mostly contains the correspondence with the State Planning Committee of the USSR.

NUCLEAR POWER – A VIEW FROM SMALL "NON-NUCLEAR" COUNTRY

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Mastering nuclear fire, one of the greatest, deepest and most ingenious human achievements since mastering wood fire, remains however an asset not affordable for the major part (85%) of the countries existing in the world at present – the fact which will likely be continued in the foreseeable future. Whether small, or poor, or both - are they to be exculded from the story? Surely not, for many reasons, to mention just a few.

(i) Electricity market, decisively shaped by non-conservable character of electric power, and strongly influenced in positive manner by stable and reliable NPP production, is *sui generis* open to **all**, and hence highly international and co-operative in its nature.

(ii) Scientific and technological progress, which grew on nuclear power research, development and exploitation, led to many both nuclear and non-nuclear applicative inventions for the benefit of **all**, e.g. in the fields of medicine, industry, agriculture, geology, hydrology, analytical instrumentation and methods, etc.

(iii) Various risks of NPP operation, including accidents, misuse of nuclear (fissionable) and radioactive materials and waste issues (disposal, transport, storage) cannot be kept within borders of the producing countries, but should be met by **all**, again.

Republic of Montenegro, now in the State Union with the Republic of Serbia, belongs to this group of small, developing and "non-nuclear" countries. Formerly part of ex-Yugoslavia, and thus among founders of the IAEA in 1957, Serbia and Montenegro were re-admitted to the Agency in 2001.

Montenegro is short for some 30% of the electricity it consumes, buying it from the countries in the region, including NPP-clubbers Bulgaria, Romania and Hungary. As to the applications, there are departments for nuclear spectrometry and for radioecology at the University of Montenegro and Ministry of the Environment, respectively, which are quite active and pretty well staffed and equipped.

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FROM HISTORY OF RECEPTION OF NATIVE URANIUM

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Tajikistan is mountain country. In its recourses may found almost all chemistry elements of periodic system. Not a secret that in practical solving of problem of uranium Soviet country in 40^{th} and after years important role play uranium resources of the Tajikistan. Academic V. Vernandskiy in his diary rouse an alarm for work state on proceeding for uranium in Soviet Union. He was entirely aware of important appeared in world, particularly, in war period in connection to open possibility of carrying out of nuclear chain reaction.

He not agreed the decision to close works in Taboshar uranium mine (North Tajikistan) taking all possible actions to destroy this decision. V.Vernandskiy write that physicists "direct all efforts for study nuclear and its theory, and here (e.g. Kapitsa, Landau) make a lot of important – but life order ore-chemical direction", which means that task of extraction of isotope of uranium-235 from uranium ore. [1]. It should mention that aim-directed search of uranium ores in Tajikistan appeared in after-war years and result with openness of a number of mines, from 1926 was known Taboshar uranium mine, from ore of which, periodically was found radium. Discovery of number of uranium mines in that region did that region as with priority on organization of their industry manufacture and proceeding.

With Decision of created 30 June 1941 emergency party-state body – State defense commission (SDC) from 27 November 1942 in Tajikistan was organized mining of uranium ore and its proceeding up to concentrate. Implementation of those jobs was ordered to Ministry of color metallurgy of USSR, and after two years Order of SDC from 8 December 1944 No. 7102 this industry transferred to People Secretariat on internal affairs of USSR (NKVD USSR). By order of SDC from 12 May 1945 was created in region of Leninobod-city the specialized mining plant No.6 (from 1967 Leninobod mining plant, and from 1990 State enterprise "Vostokredmet").

On base of local raw material in territory of Leninobod region (now Sogd) was constructed two-test plant of proceeding uranium ores. One in Taboshar settlement, other in Chkalovsk. Those plants proceeded 20-40 tons of ores every 24 hours. Altogether those plants proceed 5392 tons of ores in 1945 and was get first uranium concentrate in USSR for creation of atomic bomb. All processes of receiving concentrate of uranium in those plants were manual. To compare may be note that in 80^{th} of last century hydro-metallurgic plant lonely proceed till one million tons of ore and produced 1.2-1.3 thousand tons of high quality concentrate in year. All process was automated and contacts for personnel with production of uranium was minimized.

On 1948-50 on territory of Chkalovsk was constructed powerful plant for proceeding of uranium ores and on 1963 he was expended several times, which proceed except local ores, ores and solutions from Russia, Kazakhstan, Uzbekistan, and Kyrgyz Republic. Thus, on 40^{th} of last century on base of plant No.6 in Chkalovsk, near Khujand-city (former Leninobod) was constructed industry of uranium concentrate, which played primarily role on creation of not only atom weapon but also creation of atomic power engineering of Soviet Union. And it is pleasant that Obninsk Atomic-power station was started on Tajik uranium.

Accounting large scientific-technical and social-economical potential of plant in a field of mining and proceeding of uranium Leninobod mining plant became center of staff training for new mining plants of other USSR republics.

Leninobod mining plant, which is head organization in this field and directed a lot of plants and mines on whole territory of Soviet Union after USSR disintegration was practically without job. Today Tajikistan has not its own paying mine for uranium production which cause that last years the plant does not proceed uranium and it is oriented on proceeding of other products and raw material.

Unfortunately, from past uranium industry of Soviet Union on territory of Sogd region of Tajikistan were left large territories of million tons of radioactive wastes and “tails”. Their state threat not only local population and environment but also neighbour countries. According to data of Ministry of nature Protection of Tajikistan the volumes of accumulated “tails” (conserved and acting) today consist more than 200 million tons.

Only one acting waste field (Degmay settlement) which has area 90 hectares and where accumulated 39.8 million of tons of wastes, gamma radiation reach 200-250 micro-curie per hour. Most of “tails” does not have sufficient cover and has a great ecological threat. We are in always contact with IAEA specialists and now are seeking for ways of solution of these problems. But for entirely solution of these problems, we look for cooperation of other international organizations and foundations as well.

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CONTROL OF INHALATION COMPONENT OF HUMAN INTERNAL IRRADIATIONS BY MEANS OF WHOLE BODY SPECTROMETER "SICH-EXPRESS"

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The whole body spectrometer of human irradiation "SICH-EXPRESS" is developed for the control of inhalation component of personnel internal irradiations.

For detection of an inhalation component is used the principle of "linear geometry" by which detectors are located on a longitudinal axis one under another, and for reduction of dependence of registration from redistribution of activity in a body and maintenance of given LLD the multidetector system of registration is used. In structure of SICH-EXPRESS consists of 2 detectors NaJ (Tl) ∅ 12 x 8 sm on the back part and chest detector NaJ (Tl) ∅7.6 x 7.6 sm. Inside each block there is a high-voltage power supply, the preamplifier and the amplifier. Signals from an output of the detecting block arrive on appropriate MSA input of the spectrometer processor block. For registration of ^{131}I in a thyroid gland is used the chest detector with replaceable collimator. The software for management of a spectrometer and automatic processing of spectra, record the received information and its transfer it to databases is developed.

At designing of such multidetector system we based upon that the minimal area in which it is necessary to provide isometric sensibility, is that part of a body where are placed those organs, in which can be located radionuclides. The relation of activities, detected by top and bottom detectors, also will serve as the indicator of an inhalation component. For decrease (reduction) of deep dependence and reduction of an error the spectra measured on the breast and on the back are summarized.

Basic technical characteristics

Range of registered energies, MeV	0,1÷3
Lead protection shields for detectors, thickness, mm	50
Energy resolution in the line 0.661 MeV, %	6.4
Integral non-linearity, %	0,9
Whole body MSA for 300 sec. P = 0.95	Bq
^{60}Co	75
^{134}Cs	72

	^{137}Cs	90
	^{40}K	916
Lungs MSA for 300 sec.P =0.95	^{60}Co	25

The used software, on the basis of the developed algorithms of annual receipt size calculation by the measured values of the contents of radionuclides in the body of the patient, estimate radiation exposure on three models (models of the single receipt, repeating receipts and chronic receipt).

METHODS AND INSTRUMENTATION FOR MEASUREMENT OF RADIOACTIVE WASTE

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According to nuclear legislation of Ukraine radioactive waste (RAW) by acception for disposal (storage), data about radioisotope contents of waste, specific and total activity of each important radionuclide have to be noted into RAW index card for the purpose of its certification.

New requirements stimulate the elaboration of methods and spectrometric devices for certification of RAW, which can ensure reliability and necessary accuracy of the values being measured.

The report represents a survey of methods for determination of radionuclide contents and specific activity of RAW. The methods are based on measurement of exposure rate on the surface of RAW container or package and on analysis of gamma spectra from volumetric sources. A method of calibration of gamma-spectrometers for determination of specific activity of RAW developed by the authors is offered. The method is suitable for scintillation and semiconductor detectors. Calibration coefficients are got in laboratory conditions using point standard sources, which are later used for software calculation of specific activity of radionuclides in RAW. Material and density of RAW matter, dimensions and material of container or package are also taken into account by calculation. The method is tested in real conditions on Rivnenska NPP.

A problem of application of spectrometry equipment for RAW certification is considered, as well as software, which is used for acquisition and analysis of gamma-spectra.

HLILW MANAGEMENT: STAKES FOR THE FUTURE

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The production of nuclear energy in France has been associated, since its inception, with the optimization of radioactive waste management, comprising the partitioning and the recycling of recoverable energetic materials, volume reduction, the conditioning, and the disposal of the end waste.

The public's concern regarding its long-term management made the French Government in 1990 to delay the implementation of the geological disposal and to prepare a law, passed on December 30, 1991, requesting in particular the study of solutions and processes for:

- minimizing the quantity and the hazardousness of waste, via partitioning and transmutation,
- either reversibly or irreversibly disposing the waste in deep geological formations,
- waste conditioning and long-term interim storage.

The law stipulates that by 2006 the research results obtained should be documented in the form of a general report to be submitted to members of parliament.

The law also created the National Evaluation Commission to supervise the research work on an annual basis.

At the time of the implementation of measures comprised in the Law, the Government commissioned CEA to conduct research in the area of Line 1 (partitioning and transmutation) and Line 3 (conditioning and interim storage) and Andra on Line 2 (geological disposal).

We will present here an update on the progress made by the research conducted on Lines 1 and 3.

The feasibility of partitioning did not appear easily accessible at the time the research began. Its feasibility was demonstrated in 2001 following a series of tests conducted on actual solutions of dissolved spent fuel, in the CEA's Atalante installation at Marcoule.

The 2002-2005 program encompasses technological demonstration, with representative process equipment, and economic evaluation of industrial implementation of partitioning.

Studies on transmutation, which were initiated before the 1991 Law, rapidly led to concluding that transmutation of minor actinides (Americium, Curium, and Neptunium) was feasible in particular in fast neutron spectra.

Work on transmutation is now focusing on technical elements necessary for the demonstration of its technological feasibility.

Developments made in the area of waste treatment and conditioning were targeted at ensuring the availability of qualified processes that could be applied to historic waste to be recovered or to improve (volume reduction) a number of existing treatment processes.

The objective of the conditioning being to ensure durable confinement for all the steps in the package management, it is necessary to establish scientific and technical ground for the prediction of the package long-term behavior, and to confirm that the relevant functions are provided, in particular confinement, handling, and recovery.

Interim storage is a mode of package management ensuring, by design, the protection of waste package and their recovery at a later date, under safe and technically established conditions

The work done in the context of Line 3 of the 1991 Law has comprised the identification and the development of long-term interim storage concepts.

Important results are now available, on the one hand concerning the possibility of significantly reducing the quantity and the radiotoxicity of long-lived waste, and also relating to the modes of waste conditioning and long-term interim storage facilities.

With these results combined with the results get by Andra on geological disposal, technical solutions that could be implemented in a progressive way, will be presented in 2006 for HLILW management. These scientific achievements must be accompanied by a democratic and political debate at the French Parliament.

In this respect, one will have also to combine the French institutional process with the international experiences and the European requirements, for example the “nuclear package” proposed by the European Commission. Whatever will be the French parliament’s decision in 2006, one will need long-term and worldwide solutions in the XXIst century to permit the development of nuclear power.

The French experience can contribute to make progress in the search of sustainable and acceptable solutions

EFFECTIVELY MANAGING NUCLEAR RISK THROUGH HUMAN PERFORMANCE IMPROVEMENT

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The US commercial nuclear industry has just completed an outstanding decade of plant performance. Safety levels and electric production are at unprecedented high levels and continue to exceed even high industry goals. Nuclear energy continues to keep the highest priority on performance improvement programs and highly trained/qualified people that maintain its record setting safety and reliability of operations. While the industry has maintained a consistently high level of performance, the advent of deregulation and the consolidation of NPP ownership, as well as the current climate of concern about both rising energy costs and availability of power, has raised the standard for nuclear energy's level of competitiveness in today's market place. The resulting challenge is how to more effectively manage risk and improve performance even further in a generally high performing organization. Newer technology and more training by themselves are not the answer. Rather, the answer will lie in the human side of the organization and management's ability to tap into the unused potential of employee commitment and productivity. It is people who offer the greatest potential for organizational success. Given the fact that human performance has been demonstrated to yield higher rates of return than physical capital, it makes good business sense to determine how to encourage the behaviors in the workplace to manage the risk that will accompany efforts to boost the nuclear industry to new heights of excellence. This means effectively developing a performance improvement culture through identifying measurable performance indicators and determining how behaviors can best be influenced to improve those indicators. It also means seeing a culture of performance improvement and risk management as a strategic planning tool rather than a solution to a particular problem. One of the most effective ways to develop this culture of performance improvement and effectively managing risk is to apply the principles of Human Performance Technology, or HPT, to the nuclear workplace. HPT is a relatively new field that has been emerging over the several decades. Its principles are derived from the research and practice of behavioral and cognitive psychologist, instructional technologists training designers, organizational developers and various human resource specialists. Relying heavily on general systems theory as applied to organizations, HPT takes a systemic approach to risk assessment and performance analysis/ change as opposed to making piecemeal interventions which often happens in designing training programs intended to be the only fix specific organizational behavior problems in the short term. Specifically, HPT methodology emphasizes examining any given problem in relation to the more global aims of the setting or environment within which the problem is identified. Its consistent driver is measurable performance and the structuring of elements within the system to improve and reward desirable performance and effective risk management. In general, HPT is more than another way to look at training, HPT is a systemized process that combines selection, analysis, design, development, implementation, and evaluation of programs to most cost effectively influence human behavior and accomplishment. By taking a systems view of organizations rather than discreetly focusing on pockets of concern within an organization, it seeks to link the actions and interventions of all the organizational elements that affect overall performance. It does this through identifying, among other things, current performance levels, required performance levels for effective risk management, exemplary performance, and developing measurements for each step in the process. It examines such things as the external and organizational environments as well as the influences that directly impact individual performance. Direct application to the nuclear industry is evident when considering the temper of the times. For nuclear organizations to stay viable performance improvement processes will have to be part of their culture in order to better manage risk and to achieve and maintain ongoing success. Greater plant performance will be directly related to greater human performance. To be successful, the nuclear industry, like any other organization, must clearly understand its purpose, structure itself to achieve that purpose, structure appropriate internal relationships, establish a realistic incentive/reward system, establish efficient work processes and assure knowledgeable and supportive leadership. Using the principles of Human Performance Technology can provide a structure for helping the industry in general as nuclear facilities individually take steps to accomplish these goals. Taking strategic steps to develop these elements will result in higher levels of production efficiency, better risk management, higher customer service/satisfaction and an ability to successfully meet ever changing environmental and business demands. Human Performance Technology is proven effective methodology to help in the successful development of these strategic steps. This session will offer a discussion of the basics of HPT, as well as some specific examples of how it can be applied to nuclear facilities to effectively manage risk and drive enhanced performance.

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*HPT has its roots in what was the National Society for Programmed Instructions (now known as the International Society for Performance Improvement – ISPI). Its principles are currently supported by such professional organizations as the American Society for Training and Development (ASTD) and the International Federation of Training and Development Organizations, as well as in such professional journals as Training.

HYDROGEN RADIOLITIC SYNTESIS BY CHAIN DECOMPOSITION OF HYDROGEN SULPHIDE

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The two stages method including of radiolysis process of H_2S-CO systems was studied:

$H_2S + CO == COS + H_2$

$COS \overset{T}{==} CO + 1/n\ S_n$ or $COS + O_2 \overset{T}{==} CO + SO_2$

The radiation-chemical yields of H_2 at various concentration of H_2S in the gas mixtures of H_2S-CO and temperatures in the range of 313-773K were determined. Radiolysis was conducted under gamma radiation and electrons beam. Results are given in the Table 1.

Table 1. Radiation-chemical yields of H_2 at the radiolysis of gas mixtures CO-H_2S

H_2S,% // T	313	473	573	673	773
0,9	0,60+0,19	3,0+0,6	6,7+0,7	19+5	50+5
2,1	1,7+0,5	-	-	-	89+22
5,3	2,1+0,4	3,5+0,4	7,7+0,5	19+6	85+17
9,3	-	-	-	19+6	120+27
20	-	-	-	-	130+21
26,5	4,3+1,1	4,3+0,6	8,4+0,6	30+4	204+18
53	5,3+1,2	-	-	41	102

The energy activation (E_a) of the hydrogen formation process is calculated and given in table 2. Two fields (regime) are distinguish at the temperature dependence G(H_2) in Arrhenius coordinates. Initial values of energy activation E_1 were increased at high temperatures.

Table 2. Energy activation (E_a) of hydrogen formation in the H_2S-CO systems

Contents of H_2S,vol.%	0,9	5,3	26,5
E_1 (kJ/M)	12,6+3,8	4,2+0,2	0
E_2 (kJ/M)	40,3+2,0	48,3+2,4	63,3

The obtained results showed a chain mechanism of hydrogen formation at the radiolysis of H_2S-CO mixture.

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REACTOR PLANT WWER-1500 FOR NUCLEAR POWER PLANTS

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On 14 January 2004, the Board of Russian Federation Ministry of Atomic Energy has taken a decision as to the term of design development of NPP with forerunner power unit WWER-1500 proceeding from an advisability of NPP commissioning in 2012. This decision is taken on the basis of the results of development of conceptual designs of NPP and reactor plant WWER-1500, completion of a large amount of the research and development work adopted by the scientific and technical council of Russian Federation Ministry of Atomic Energy on 22.05.03.

The paper gives brief information as to the design concept, basic technical characteristics of reactor plant WWER-1500, list of principal R&D work in order to justify the design. It is noted that the activities initiated in 2000 and related to designing of reactor plant WWER-1500 are based substantially on the basic technical solutions of RP WWER-1000 of new generation, especially as to justifications of technical solutions on safety.

FLOATING NPPS BASED ON SHIP PROPULSION REACTOR TECHNOLOGIES - RELIABLE AND SAFE ENERGY SOURCE FOR AUTONOMAS ENERGY SUPPLY

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Small-sized floating NPPs of 3 to 10 MW in electric capacity seem to be rather attractive for energy supply in Republic Sakha (Jakutiya) and some other regions of Russian north and north-east. High level of reliability, radiological and ecological safety, acceptable economic characteristics, long-term operation without refueling (about 10 years) make them advantageous for application in these remote areas.

Commercial advantages are based on the following features of the floating NPPs:

- Floating variant of small NPP is relatively not expensive, does not require development of special supporting infrastructure (warehouses, transportation, maintenance and repair facilities etc.). Floating NPP tows to a site as a shop-assembled ready for operation unit. In 10-12 years of operation it is to be replaced by “fresh” one. At the end of life (~40 years) the floating NPP will be simply towed to the factory for decommissionig thus leaving a “green lawn” at site.
- Shallow draught (not more than 2.5 m) provides wide choice of location sites along rivers and in sea shore.
- It is possible to arrange single or twin-reactor design of a floating NPP and operate it as dedicateded electricity only or dual-purpose (electricity and heat) sourse.
- Relatively low construction cost which is about 20 M$ for 3 MWe.
- Low number of operating staff (~ 20 persons) reduses operation cost essentially.
- Availability of operating prototypes of all main equipment of nucear power unit and long-term experience in construction and operation of nuclear vessels is a guarantee of high quality of design, excludes costly R&D, demonstration tests and unpredictable design modifications.
- Not any expenditures are required for development new facilities for construction of floating NPPs, reactor refueling equipment, radwaste management and NPP decommissioning.

- Implementation of floating NPPs eliminates critical problems und annual “headache” with delivery of vast amount of fossil fuel to remore regions.

Preliminary assessments reveal promissing propects for application of floating NPPs in a number of sites in Jakutija such as Nakyn, Ebelekh-Tomtor, Kuchus, setelments Chokurdakh and Batagay.

Utilisation of ship-propulsion reactor technology with 6000 reactor-years of operating experience as well as avalability of advanced project of ABV reactor, involvement of experienced scientific and design institutions and enterprises of Minatom allows to construct and supply of small floating NPPs to consumer in short term and provide full scale life-cycle technical support for them.

CHROMIUM-NICKEL STEELS DEPLETED OF NICKEL STABLE ISOTOPE Ni-58 AS A MATERIAL FOR FAST REACTOR CLADDINGS

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It is well known that mechanical properties of materials used commonly for fast reactor (FR) claddings set a limit for reaching high burn-up of a nuclear fuel. Standard cladding steels, such as the Russian steel grade "TchS-68" with 15% of nickel content, become ballooned and unfit for further service after the burn-up fraction of 9-10% heavy atoms.

It is generally assumed that steel ballooning is caused by the combined action of radiation damage and helium accumulation in the irradiated material. In this case, nickel stable isotope, Ni-58 (its content in natural nickel is equal to 68%) is a source of helium creation. After neutron capture via (n,g) reaction Ni-58 is transmuted to radionuclide Ni-59 ($T_{1/2}$=7.6·10^4 years), which has a high value of (n,α) reaction cross-section for thermal and epithermal neutrons. For neutron energies of 0.5 eV – 100 keV its resonance integral of (n,α) reaction is equal to 18 –20 barns while for other nickel isotopes it is less than 10^{-4} - 10^{-5} barns. This unique nickel isotope property can be used within a concept of creation of materials with controlled isotopic composition for the nuclear engineering as it is proposed in Ref.1. Calculations performed show that cladding steel depleted of Ni-58 produces smaller quantities of helium and hydrogen, and it can be considered as a material suitable for reaching higher burn-up fractions up to 14-15% h.a. is required now.

It should be mentioned that at the RNC "Kurchatov Institute" – Institute of Molecular Physics a new, based on a gaseous centrifuge, technology of obtaining nickel isotopes in kilogram quantities is successfully utilized [2].

For the realization of the proposed concept to create slightly ballooning claddings, relatively small quantities (tens or hundreds kilograms) of nickel depleted of Ni-58 are required taking into account that in modern FRs, such as the Russian BOR-60 or BN-600, thin-walled claddings of steels with 15-45% nickel content are used.

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FABRICATION OF NUCLEAR FUEL AT OAO MSZ. CURRENT STATUS AND PROSPECTS FOR DEVELOPMENT

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Public Joint Stock Company (OAO) “Mashinostroitelny Zavod” is situated in the town of Electrostal, 55 kilometers to the east of Moscow, the capital of the Russian Federation. The company is known to be one of the biggest suppliers of nuclear fuel for different nuclear reactor types and designs.

Research activities related to uranium were initiated by the company in the mid 40 –s of last century and as a result, in 1953 the fabrication of first fuel assemblies was started thus providing with fuel the first nuclear power plant to be commissioned in Obninsk, Kaluzhskaya oblast, Russia.

Nowadays the company possesses the technologies allowing to manufacture various fuel designs for several types of nuclear reactors operated in Russia, member states of the Commonwealth of Independent States (CIS) and in Europe.

State-of the art automated production lines equipped with all the necessary instrumentation and test equipment have been commissioned by the company in order to cover the needed production capacities. The company has developed and implemented the quality management system certified by independent TUV authority according to international standards of the ISO 9000 series.

The increase of the number of customer requirements to the fabricated fuel induces the company to improve production technologies, develop and commercialize new types of production. The company has been successfully using recycled uranium and has implemented the technologies of fabricating nuclear fuel containing several types of burnable absorbers. Based on these fuel technologies fuel assemblies for nuclear power plants operating RBMK, VVER-440, VVER-1000 reactors are manufactured. The fuel pellet enrichment loaded into the finished fuel rods to be used in VVER reactors is measured by means of several types of non-destructive testing equipment having different operation principles.

Starting from the mid 90-s the company has been cooperating with “Framatome” ANP manufacturing different fuel designs for BWRs and PWRs located in the countries of Western Europe, for example in Germany, Switzerland, the Netherlands and Sweden.

In 2002 the Dry Conversion Plant (DCP) was commissioned by the company. The DCP is used for converting enriched urainum hexafluoride into uranium dioxide of a ceramic grade and has the installed capacity making 400 t/year, allowing to increase the volumes of uranium dioxide powder production – basic component for manufacturing reactor fuels.

The company is an integral part of TVEL corporation – the Russian nuclear fuel supplier. Keeping in touch and cooperating with leading design and scientific organizations in Russia the company is constantly carrying out the activities related to the fuel performance improvement and reduction of production costs.

The long and successful experience of cooperating with customers as well as production modernization, traditional reliability, implementation of new designs and new customers form a solid and sound basis for the further development of the nuclear fuel producing company and face the future with confidence.

TRANSPORTATION OF FRESH NUCLEAR FUEL FROM OAO "MASHINOSTROITEMNY ZAVOD"

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OAO "MSZ" is one of the biggest Russian companies involved in production and transportation of nuclear fuel by means of land transportation (railway, motor and water transport) as well as by air and sea vessels.

Since 1966 OAO "MSZ" has been developing design documentation for transport packages (TYK) and supervising the process of manufacturing and operating transport packages (TYK).

The following types of fuel were transported based on the designed documentation:

a. fuel assemblies for research reactors;

b. fuel assemblies for VVER-440 reactor;

c. fuel assemblies, control and scram rods for RBMK-1000 and RBMK-1500, BN-350 and BN-600, EGP-6, VK-50;

d. fuel assemblies for VVER-1000 reactor ;

e. PWR and BWR cores for the marine force.

Safety and reliability of TYK design during transportation is confirmed by strength calculations and tests performed at a specialized test area of OAO MSZ in accordance with the requirements of IAEA Regulations-96 and OPBZ-83.

Based on test results conclusions and substantiations are prepared and sent to FGUP «GI» VNIPIET» (Federal state unitary enterprise "Leading research institute" All Russian scientific and research institute of power engineering technologies, Saint Petersburg) or RFNC VNIIEF (Russian Federal Nuclear Center All Russian scientific research institute of experimental physics, Sarov) for issuing release certificates for TYK design and transportation.

By the present moment OAO MSZ has been granted 77 release certificates by the competent authority of Russia.

OAO "MSZ" prepares all the necessary documents for fresh fuel deliveries.

CONTRIBUTION OF OAO MSZ INTO NUCLEAR FUEL PRODUCTION AND PERFORMANCE RESULTS

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Open type joint stock company "Mashinostroitelny zavod" is a leading industrial enterprise in the Nuclear industry of the Russian Federation, integrated into the TVEL corporation, one of the major international suppliers of nuclear fuel.

At present the nuclear fuel manufactured by OAO Mashinostroitelny zavod is operated in 59 commercial reactors of 14 countries of Europe and Asia that is practically in every eighth reactor of the world. Skilled personnel, considerable production capacities, special equipment and accumulated experience enabled OAO Mashinostroitelny zavod to become one the leaders in the nuclear fuel production and conclude contracts with the Russian and foreign customers.

The cooperation range has extended lately. Besides the longstanding partners the OAO MSZ started cooperating with the Chinese company of the Nuclear energy industry (CNEIC) in 2000. The perspectives of this cooperation are core design and supply of fuel assemblies to the Chineze fast research reactor. Cooperation with the company Siemens started in 1995 and in 2001 with the company "Framatome ANP". Within the cooperation frames the fuel assemblies are manufactured for the PWR and BWR reactors, designed by French and German specialists.

Nuclear fuel is delivered to the NPPs of Armenia, Bulgaria, Hungary, China, Lithuania, Slovakia, Ukraine, Finland, Czeck republic, Germany, Sweden, Switzerland and the Netherlands.

In Russia OAO MSZ delivers the nuclear fuel to the Novovoronezh NPP, Kola, Kalinin, Beloyarsk, Leningrad, Kursk, Smolensk and Bilibinsk NPP as well as produces the nuclear fuel for the research reactors and for ships.

OAO MSZ also produces the fuel assemblies for the cores of energy reactors of different types: VVER-440, VVER-1000, RBMK-1000, RBMK-1500, BN-600, PWR and BWR (Fig.1).

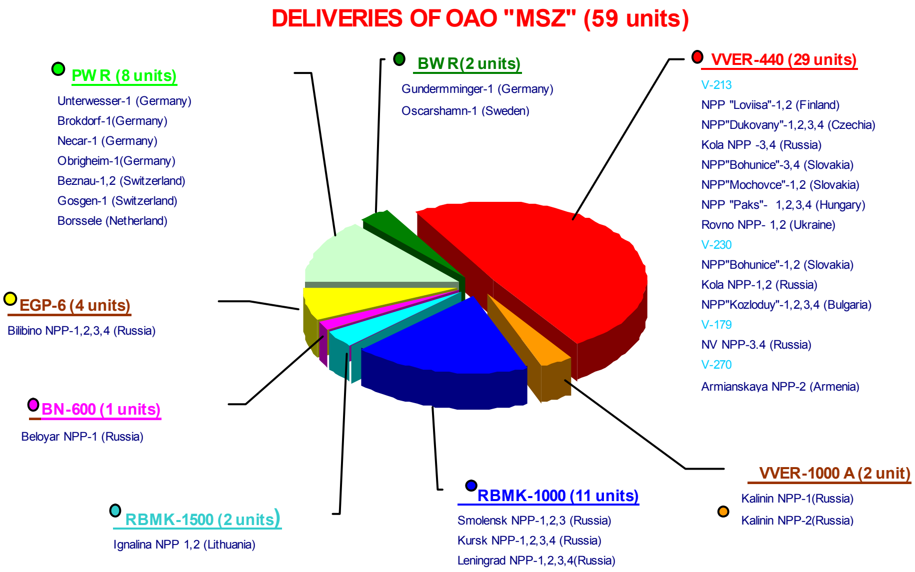

Fig.1

The fuel of OAO MSZ features high performance reliability. One of the main indices accepted all over the world to assess the reliability of the nuclear fuel is the leakage factor of fuel rods for an operational period.

As for the VVER-440 reactors of the second generation (type B-213), operating in the mode of commercial operation, this value corresponds to the best world indices and for the last five years has made up $1*10^{-6}$ (calculation was done according to [1]). Leakage factor at the NPP VVER-1000 (AFA) corresponds to the average European level and makes up $\sim 2*10^{-5}$. The cooperation period with the company «Framatome ANP» has not seen a single leaking fuel assembly manufactured by the OAO MSZ.

The use of the nuclear fuel of OAO MSZ at the VVER-440 NPP enabled the operators to increase the burn-up of the unloaded fuel assemblies for the last 5 years by 10% in average, achieve the load factor within the limits of 80-90%, to implement the operation modes which are needed for the operating utilities and to operate with 100% power. Transition to new types of fuel will provide for more efficient fuel cycles and for a higher burnup of the unloaded fuel assemblies.

After many years of operation the OAO Mashinostroitelny zavod has deserved the reputation of a highly respected, reliable and forward-looking partner. Cooperating on a long-term basis with the domestic and foreign companies the OAO Mashinostroitelny zavod is always open for new business and efficient joint activity.

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VVER-1000 FUEL RODS MANUFACTURED BY OAO MSZ

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OAO “Mashinostroitelny Zavod” is known to be one of the major fuel manufacturers in the world producing fuel for nuclear reactor cores of different designs.

Taking into consideration present-day requirements of the nuclear fuel market as well as customer requirements, engineering departments of OAO MSZ, having a long-term cooperation partnership with research and design institutes, have implemented a lot of improvements aimed at extending the life time and enhancing fuel reliability.

A number of performed activities resulted in the design upgrade and fuel improvements. The following modified fuel rod designs have been prepared based on improvement activities:

1. The fuel rod having the length increased by 150 mm compared to the standard design by means of adding two blanket zones (the lower zone is 90 mm, the upper one - 60 mm), being loaded with pellets of 0,5 % enrichment. The bottom end plug of the fuel rod is equipped with a latch allowing to remove irradiated fuel rods. Blankets are used to avoid the distortion of the powder density field. In the future, these low-enriched pellets are planned to be replaced with pellets having regular enrichment, thus, leading to the increase of uranium quantity in the fuel and extension of the fuel campaign.

2. The fuel rod with the length increased by 17 mm compared to the standard design by means of using a longer insert. This design modification will give an opportunity to use fuel rods for nuclear power plants abroad, for example, for Temelin NPP in Czechia. The bottom end plug of the fuel rod is equipped with a latch allowing to remove irradiated fuel rods.

3. The fuel rod loaded with fuel enriched to 5 % in U-235. The availability of fuel pellets having a hole with the inner diameter of 1,2 mm will increase the weight of UO2 up to 500,6 kg per a fuel assembly. Fuel assemblies containing fuel rods of the described design are expected to be operated in reactors during 5 years.

4. Fuel rods containing regenerated fuel.

The above fuel rod modifications can be characterized by the following parameters:

- four or five year fuel cycle;
- increase of the fuel burn up to 60 MWday/kgU;
- improvement of performance characteristics of the fuel cycle.

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VBER-300 REACTOR PLANT ON THE BASIS OF SHIP TECHNOLOGIES

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Russian technologies of nuclear shipbuilding are based on many outstanding achievements in science, engineering and industrial technologies, unique experience of ship nuclear reactors construction and operation.

Modular ship reactors with water under pressure, developed in OKBM (more than 460 nuclear reactors), together with VVER-type power reactors are the most proven reactor technology, tested and confirmed by the successive experience of ship nuclear power plants operation. Total experience of different purpose ship reactors operation exceeds 6000 reactor-years.
The long-lived experience of ship plants development, construction and operation and the results of the performed R&D for the projects substantiation, process base and staff potential of Russian enterprises form the basis for the creation of highly reliable power sources for nuclear power.

The report gives the description of average power VBER-300 reactor plant, developed on the basis of modular ship reactors and nuclear power stations with RP VBER-300 in ground-based and floating design.

The main engineering solutions of VBER-300 reactor plant are the following:

- use of 300 MW(e) vessel-type PWR, the most proven in the world practice;
- block arrangement of the main equipment;
- leak-tight primary circuit;
- use of once-through steam generators of coiled type;
- cassete-type core with VVER-type fuel and reduced fuel rating corresponding to the proven technologies of VVER nuclear fuel cycle;
- use of passive safety systems.

Safety of the nuclear power station with VBER-300 reactor plant together with realization of the reactor inherent self-protection is provided by defense in depth, it provides minimum influence upon the personnel, population and the environment. Indices of radiation safety allow the location of NPP in the immediate vicinity of the consumer.

VBER-300 reactor plant design is evolutionary relative to modular ship reactor plants; increase of the reactor plant thermal power up to 850 MW is provided owing to the increase of overall dimensions at the maximum reservation of the reactor plant image and main design decisions on the reactor unit.

VBER-300 reactor plant developed on the basis of nuclear ship building technology may be considered as basic for “nuclear district heating” of Russian cities and towns.

Calculations of main technical and economical indices of the plant and estimation of financial (commercial) efficiency of investment outlays show that NCP with two VBER-300 reactor plants is economically profitable and efficient.
Application of VBER-300 RP for floating NPPs is extremely attractive. Economic efficiency of floating NPP is provided owing to: flexible scheme of construction at the ship building plant, low steel intensity, reduction of specific capital outlays, transportation to the consumer, simplicity of removal from operation, possibility of floating power unit utilization at the specialized enterprise.

The design of floating NPP with two-loop VBER-300 reactor plant is a prospective option. At 100-150 MW(e) RP a long-term cycle of nuclear fuel operation without refueling (8 years) is provided, it allows to perform refueling, operations with spent fuel and maintenance at special enterprises – service centers. Spent fuel storage is not necessary at the floating power unit.

Construction of such floating NPPs is possible at Russian ship building plants with the provision of floating power unit transportation by the Volga-Don channel to the potential consumers.

So the performed design developments confirm the possibility of floating NPPs with 100-600 MW(e) power range creation, using unified equipment of VBER-300 reactor plant for electric energy, heat generation and sea water desalination.

The supposed power sources have commercially attractive characteristics and high export potential.

ENSURING OF NPP WITH RBMK-1000 MCC COMPONENTS SAFE OPERATION BASED ON BREAK PRECLUSION CONCEPTION

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Application of Leak-Before-Break Methodology to Main Circulating Circuit (MCC) large diameter components was included in Plant Life Management (PLIM) Conception in 1998 as one of main tasks.

Regulatory guidance RD 95 10547-99 (Leak Before Break (LBB) Conception application to NPP piping) have been added by methodology of more general Break Preclusion Conception (BPC) to consider peculiarities of MCC austenitic piping. Development of such approach was undertaken taking into account results of IAEA Extrabudgetary Programme (2000-2002). Such BPC utility guidance RD EO 0513-03 was adopted for application at NPPs by RF Gosatomnadzor in January 2004.

Technical analysis based on LBB & BPC have been performed for the following components:

- Main Circulation Pump (MCP) piping dia800 mm, main steam and feedwater piping (in reactor compartment), MCC headwaters dia900 mm;
- Steam separators, MCP casings, gate valves;
- Dia300 austenitic piping and headers.

Structural integrity diagrams have been developed for the components mentioned on the basis of methodology from regulatory guides such as RD 95 10547-99 and RD EO 0513-03. These diagrams were used in defect tolerance assessment. Performance of BP and LBB technical analysis were based on results of comprehensive LNPP Unit 1 component state inspection additionally undertaken in year 2002. Actual mechanical properties of MCC piping material have been studied on cut-off samples after 200 000 hours of operation.

Results of LBB/BP technical analysis made it possible to:

- Reassess requirements ton ISI and LDS and also to justify their current capabilities;
- Substantiate lack of additional supports preventing pipe whips.

Activities on BP/LBB application at LNPP Unit 1 can be treated as a generic approach for all RBMK-1000 NPP Units to ensure safe MCC operation.

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THE USE OF NUCLEAR ENERGY FOR DISTRICT HEATING. THE BRANCH PROGRAM OF ACTIVITIES. NIKIET DESIGN EFFORTS ON THE ADVANCED NUCLEAR CO-GENERATION PLANT WITH VK-300 REACTOR, THE RUTA NUCLEAR HEATING PLANT AND SMALL POWER UNITS

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District heating is among the top priorities of the state economic and energy policy of Russia and is the largest and expanding sector of the national power industry. The nuclear sources of energy are regarded as the promising option for this sector of the power industry.

The branch program of activities which is being implemented is intended for developing the policy and program of nuclear district heating. The priority task is to provide co-generated heat from the NPPs and nuclear co-generation plants to the amount of 30 mln Gcal/year by 2020 as specified in the Energy Policy of Russia for the period until 2020.

NIKIET named after N.A. Dollezhal has been developing the special purpose reactor facilities for the power units of the nuclear co-generation plants and nuclear heating plants.

The detailed design of the power unit with the simplified passive boiling water reactor VK-300 has been developed for the nuclear co-generation plant (NCP) intended to be deployed in the large-scale power industry. It has been demonstrated that NCP with VK-300 reactor is competitive with respect to the operating and advanced fossil thermal co-generation plants. It is envisaged to construct the four-unit first of-the-kind NCP with VK-300 reactor in Arkhangelsk region.

The nuclear heating plant based on the pool RUTA reactors operating under atmospheric pressure is being developed for the small towns. It is planned to construct the pilot plant of such kind on the site of RF State Research Center FEI, Obninsk.

In the frame of conversion of the defense-oriented works NIKIET has developed the UNITHERM reactor facility for a small NPP to be located in the distant and difficult-to-access regions of Russia.

To provide heat and electricity to the small communities, meteorological observatories, lighthouses and radio navigation stations in a reliable and safe way, it is possible to use non-attended small nuclear power plants based on the self-regulating water-water reactor and thermoelectric energy conversion. Serviceability of such plants is justified by the Gamma nuclear thermoelectrical facility operated in Russian Research Center Kurchatov Institute since 1981.

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PROCESSING AND IMMOBILIZATION OF PYROCHEMICAL RESIDUES AND PU-CONTAINING WASTE

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For more than 30 years, RIAR has been conducting complex research on the fast reactor fuel cycle including development and experimental validation of pyroelectrochemical regeneration process of fast reactor irradiated fuel, production of granulated uranium and uranium-plutonium oxide fuel, and fabrication of fuel pins and FAs by the vibropacking method. The pyrochemical process under investigation was approved in the course of reprocessing of irradiated BOR-60 and BN-350 fuel, fabrication of large batches of uranium-plutonium oxide fuel as well as conversion of "weapons" plutonium.

An important direction of this research is reprocessing of spent products and studying of the preparation and controlled storage methods for high-level solid waste that is formed at different stages of the fuel cycle.

Recycle products may be conventionally classified into three groups:

- evaporated salts after washing of granulate and bulk filters, sweepings of the product;
- pyrographite products that were in contact with molten salts and those with expired service life, packaging and cleaning materials (polyethylene and polyvinylchloride films, cloth, paper), cloth of in-box aerosol exhaust filters and process filters of the system for gases removal from apparatuses, rubber (hoses, gaskets, gloves);
- chlorine-containing off-gases.

Recycle products of the first group go directly to the head apparatus of the process. Recycle products of the second group are reprocessed by means of burning in a special apparatus, and the ash residue returns to the cycle or is sent for immobilization, depending on the content of nuclear components. The chlorine-containing off-gases are purified from chlorine by the cryogenic method, and the chlorine returns to the process.

The wastes from the process of pyroelectrochemical regeneration and granulation of nuclear fuel, phosphate concentrates and chloride salts, are materials that are suitable, by their properties, for long-term controlled storage in stainless steel containers. This was validated by investigations with real waste. To prepare the waste for the ultimate storage, it is necessary to create an extra safety barrier, e.g. inclusion or conversion of the waste into glass-like or ceramic matrices. The method of phosphate residues inclusion into alumofluorinephosphate glass was tested as well as the method of their conversion into ceramic mineral-like blocks with monazite structure. Investigations were also conducted on conversion of spent electrolyte salt into various types of phosphate glasses and ceramic with the structure type of cosnarite mineral. The properties of the obtained matrix materials with waste meet the requirements for solidified waste forms suitable for long-term or ultimate storage.

THE MAIN DIRECTIONS IN TESTING FUEL RODS FOR IMPROVED NPP FUEL CYCLES IN THE MIR REACTOR

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The MIR reactor provides for the following directions in testing fuel for water-cooled reactors: burn-up increase, implementation of power maneuvering mode, use of claddings made of G-110, G-635, E-635 alloys, testing of experimental fuel rods containing MOX, vibropacked, or cermet fuel, testing under emergency.

VVER fuel. Comparative testing of experimental fuel rods with different modifications of uranium dioxide pellets, zirconium alloy claddings (E-110, E-635, G-110, G-635), re-irradiation of instrumented re-fabricated and full-size standard fuel rods up to the burn-up of 70-80 MW*day/kgU are carried out. Thirteen power ramp experiments were performed using both full-size fuel rods (the length is up to ~3.9m) and instrumented re-fabricated ones to deep burn-ups, as well as two tests of the instrumented re-fabricated fuel rods with the burn-up of ~40-50 MW*day/kgU under the power ramp mode. The LOCA program is being performed at present, in particular 7 experiments with deeply burnt fuel rods have been performed, and the experiments with impulse power change simulating behavior of a deeply burnt fuel rod under the design accident mode, as well as with leaky fuel rods are being prepared.

Fuel rods containing vibropacked uranium dioxide, pelletalized and vibropacked mixed fuel including gadolinium additives are tested. The burn-up of ~65MW*day/kgU was reached.

RBMK fuel. Two experiments with single or multiple power ramps using full-size and instrumented re-fabricated fuel rods to different burn-ups were carried out.

Fuel rods containing cermert fuel of different modifications in encladded into E-110 alloy tubes (diameter - ~9.1mm) were tested under the VVER conditions up to the burn-up of ~ 65 MW*day/kgU.

Fuel rods of different design containing uranium dioxide composition dispersed in silumin matrix are also tested. A fragment of a fuel assembly containing dispersion fuel rods based on uranium intermettalide was tested under the LOCA conditions.

SSC RF RIAR performs almost all the work required for designing and manufacture of experimental devices, special transducers and some types of experimental fuel rods, preparation of irradiated full-size or re-fabricated fuel rods from reference and standard FAs for NPPs, interim and post-irradiation examination.

The MIR reactor enables simultaneous implementation of several testing programs at different thermal neutron flux density (the maximum is ~5 x 10^{14} 1/cm^2*s, the difference is 5-10 times), in particular, power ramp testing with 100% power increase within 5-10 minutes and longer, testing of mockup fuel rods containing low enriched fuel (up to 20%) for research reactors at a specific energy release of 15-20 kW/cm^3. The reactor loop systems provide maintaining of water-chemical conditions, continuous leak testing of the fuel rods, on-line displaying and processing of the information from the transducers with a registration frequency of 20 Hz (and higher, if necessary).

BOR-60 REACTOR AS AN INSTRUMENT FOR EXPERIMENTAL SUBSTANTIATION OF FUEL RODS FOR ADVANCED NPPS

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The BOR-60 fast test reactor is actually the only facility of this type in the world that has been in reliable and continuous operation for about 35 years. One of the principle reactor tasks is irradiation of advanced fuel and structural materials in different conditions. Inside the reactor the materials can be irradiated in any core and reflector cell except seven cells used for control rods. The number of fuel assemblies loaded into the reactor can vary from 85 to 124 depending on the burnup, core configuration and fuel properties. Due to the reactor design, the core dimensions can be widely changed allowing accommodation of no less than 20 experimental assemblies in different reactor cells. The neutron flux value in individual cells can vary more than 3 times at the maximum value of $3.7 \cdot 10^{15}$ n/cm^{2}s. Thus various fuel compositions can be loaded into the reactor and practically any burnups can achieve.

Based on the long-term investigation of the reactor characteristics, we studied the reactor behavior in different conditions, developed a set of the verified codes and different procedures for the on-line reactor maintenance and performance of the wide scope of experiments.

A set of specialized testing facilities consisting of capsule units and dismountable assemblies are used for irradiation of the wide range of materials and items at different conditions. The advantages of these facilities are their simplicity and possibility of installation in any core and reflector cell. In addition to the precision calculations of the irradiation conditions there is also a possibility for monitoring the neutron flux and temperature. A special thermometric channel available in the core allows accommodation of the experimental facilities and output of information of the irradiation conditions by 30-50 communication lines. It was required to develop a series of independent instrumented capsule-loops, special instrumented fuel assemblies etc. to be used in the channel. The principle task was to provide the required temperature conditions on specimens. This was achieved through the use of the thermal insulation gaps, intense cooling or additional heating at the expense of radiation energy release or fuel fission. As a result the reactor is used for testing various advanced types of fuel and structural materials at high thermal loads (100kW/m), temperatures (100^{o}C), burnups (33% h.a.) and fluences (1.8·1023cm-2 with E>0.1MeV). In case of necessity, the temperature can be stabilized by changing the thermal resistance in the heat transfer or heat removal intensification scheme using the liquid metal kept in the boiling condition. These units also provide the specified height and azimuthal temperature nonuniformity. The experimental facilities can be used for testing the fuel rods of up to 15 mm in diameter placed in different grids (triangular, square etc.) and environments (sodium, lithium, lead, various gases etc.).

Due to the availability of the experimental facility for reprocessing of irradiated fuel and production of fast reactor fuel and fuel rods, the reactor is used for the experiments related to the closed fuel cycle, such as testing of refabricated fuel with involvement of minor actinides and long-lived fission products into the fuel cycle. The BOR-60 demonstrated an effective operation as a MA burner as well as a power and weapon grade burner. This allows us to solve the important tasks of the nuclear power engineering, in particular to reduce the fuel cost and the quantity of radioactive waste and improve the environmental situation.

The BOR-60 reactor has great experience in irradiation of oxide, metal, ceramic, carbide and nitride fuel compositions for reactors of different purposes, in particular for fast sodium reactors. Such fuel properties as regularities of gas release, shape change and structure formation were studied. The results obtained allowed us to substantiate the use of fuel rods for the BN-350 and BN-600 fast reactors as well as for the other types of reactors. From 1969 to 1981 the BOR-60 used the pellet uranium fuel and since 1981 it has been using the vibropac oxide mixed fuel using the power grade plutonium and for the last years the weapon grade plutonium. Thus, the BOR-60 demonstrated the possibility for mass testing of various types of fuel. The experiments with failure of vibropac fuel rods carried out in a special capsule-loop demonstrated the possibility for testing fuel not only in steady state and transient but in the accident conditions.

Investigations of various fuel compositions form a basis for the development of the fuel cycle for the advanced fast reactors with the inherent safety, such as, the BREST-OD-300 reactor with nitride fuel and some other advanced reactors. Carbide and nitride fuel has high density in fissionable atoms and high thermal conductivity in contrast to oxide fuel and thus it allows for higher thermal loads and significant improvement of the safety parameters.

The first series of the BREST model fuel rods was tested using the loop unit in the lead environment under the irradiation parameters close to the design ones. Burn up achieved was 0.5% h.a. It is planned that the tests with the other experimental units will be continued.

Today the BOR-60 is still in operation and widely used for the research purposes. The reactor lifetime has been extended till 2010. It is planned to perform the reactor reconstruction in order to extend its experimental capabilities and to prolong the lifetime. Thus all the prerequisites are available for further work related to the substantiation of the closed fuel cycle of advanced reactors

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TECHNICAL PROPOSAL ON DEVELOPMENT AND OUTFITTING OF NPP HAVING WWER-1000 REACTOR WITH FISSION MATERIALS MONITORING SYSTEM

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Nowadays more than ten power-generating units with water-moderated reactors WWER-1000 are being operated in Russia and other countries. Several power-generating units of this type are being under construction. As is well known, spent fuel consists of various fissile substances, including plutonium isotopes. When operating nuclear power plants (NPP), it is necessary to exclude a possibility of theft (unauthorized use) of either spent or fresh fuel. Besides, the NPP, due to a number of factors, is a quite attractive target for acts of sabotage and terrorism.

RRC “Kurchatov Institute” has accumulated a great experience in the sphere of development and operation of nuclear materials monitoring systems of various types as well as nuclear materials accounting and physical protection systems. Based on this experience, with the use of recent developments, we propose to create a “hybrid” system for the NPP with water-moderated reactor. This system is to provide an advanced reliability control over the fissile materials, guarantying in this way security of the fuel cycle.

The main task of the proposed system is to collect, archive and analyze information (video data mainly) on precise location and status of fuel assemblies during the whole period of their being at the NPP. This system is designed as an in-site information network based on TCP/IP protocol. The system includes a set of “intelligent” monitoring video cameras, light sources, sensors of different types, communication lines, monitors as well as firmware to transfer and process data providing high level protection against intrusion.

The structure of the proposed system was developed for the NPP with water-moderated reactor WWER-1000, but it can be easily adapted for use at other nuclear plants.

Creation of such a system will make it possible to introduce the following at the NPP:

- Video monitoring with digitalized video data transmitted through the networks of Ethernet and Internet types;
- Automatic and computerized method of detection, recording and registration of events; a well-ordered system to record video frames on digital data carriers;
- Long-term data security, data protection against unauthorized access, subsequent computerized analyses, for instance with the purpose of an accident investigation, etc.

In addition, implementation of the proposed monitoring system will open way to a ubiquitous use of network digital technologies with their advantages in the industrial television systems.

PERSPECTIVES OF SAFE AND COST EFFECTIVE APP POWER UNIT CREATION WITH A SHELL-TYPE MICRO FUEL ELEMENTS REACTOR UNDER SUPERCRITICAL PRESSURE OF LIGHT-WATER OF THE HEAT TRANSFER MEDIUM

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In the proposed report some results of conceptual study of safe and cost-effective APP power unit with a shell-type micro fuel elements by reactor under supercritical pressure light-water heat transfer medium are presented. The one-loop case channel reactor with a radial flow of the heat transfer medium consisting of 37 fuel assemblies has been considered. The height of the fissile region - 3,5 m, diameter ~3,9 m. the fuel load ~42 ton. The thermal rating of reactor makes 3500 MWt. The efficiency ~ 44 %. Temperature of feeding water – 280^0C, temperature of the heat transfer medium on output – 550^0C. Pressure in the contour - 25 Pa.

The spherical fuel element consists from fuel centre mark UO_2, diameter of 1,5 mm covered by two-layer protection from pyrocarbon of different density and stratum of silicon carbide. An aggregate diameter of fuel element - 1,8 mm.

The step of a lattice of fuel assemblies is equal to 460 mm. The construction of fuel assemblies has multilayer on radius structure with two fuel coats, separated porous membranes, zone of heat transfer medium intermixing, assembly and distributing collectors.

On the base of heat-hydraulic and neutron -physical calculations the analysis of the different constructive outlines of fuel assemblies with consideration of adopted restrictions for permissible parameters and statistical deviations was conducted.

The results of analysis and comparing with existing analogs are presented. The basic features of such reactors are considered. Some aspects of safety of the providing offered type of reactor are surveyed.

ADVANCED REACTOR PLANT WWER-1000

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The concept and main directions for development of RP WWER-1000 design are given in this report. The main requirements (Russian and international) for modern reactor plants are indicated:

- achieving the service life equal to 60 years by the main equipment (reactor, steam generator and etc.):
- providing average value of fuel burn up equal to 50 MWday/kgU and more;
- decrease in duration of outages and increase in availability factor and etc.

Feasibility of realization of the above mentioned requirements in WWER-1000 reactor plant designs is considered. The main problems are determined and proposals for their possible solution are given. Changes in separate structural units of equipment are covered. Configuration of the advanced WWER-1000 RP design, meeting the modern requirements, is designated:

- modernized reactor vessel keeping the design of the vessel upper part, that is the flange and the nozzles zone as well as keeping bottom geometry with the aim of using the approved process of stamping: internal diameter of the cylindrical part is increased to decrease fluence and requirements for chemical composition of steel are made more strict to provide service life equal to 60 years;
- modernized steam generator with internal diameter of 4200 mm and in-line layout of tube bundle with increased service life;
- main equipment of the reactor plant: MCP, ECCS accumulator, PRZ, RCP set casing, pipelines of ECCS and PRZ – evolution changes and justification of the increased service life;
- CPS drive SHEM-3 with one sealing unit "position indicator-housing";
- reactor core on the base of FA-2.

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DESIGN AND DELIVERY OF THE EQUIPMENT FOR THE FIRST NPP IN THE WORLD

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Decision on construction of the first nuclear power plant (NPP) in the world with electrical power of 5000 kW was made in 1951 under the initiative of I.V.Kurchatov.

The territory near a small scientific town where there was a Laboratory “V” of Academy of sciences of the USSR (now – RNC FEI) near the railway station “Obninskoe” of Moscow-Kaluga railway was chosen as the site of NPP. Scientific leadership of creation of NPP was entrusted to D.I. Blokhintsev and A.K. Krasin. N.A. Dollezhal – the Head of NII-8 (now, NIKIET) was assigned as Chief Designer of the reactor plant.

A significant contribution to creation of the first NPP in the world was made by OKB “Gidropress” and ZiO. For the first time for power reactor plants a simple reliable steam generator (SG) was developed whose main design solutions were further used in SG for reactor plants of various types. In addition to SG, all heat exchanging equipment and primary and secondary side pipelines were developed as well as layout of the main heat exchanging equipment in the room was made.

In 1953 all the equipment was manufactured at ZiO, and it was mounted by the workers of ZiO under supervision of ZiO and OKB “Gidropress” specialists. The first NPP in the world was commissioned in operation on the 27-th of June 1954 having opened the path of the mankind for peaceful use of the energy of the nucleus of the atom that has marked the beginning of new era in power engineering.

CONVERSION OF LEAD-BISMUTH REACTOR TECHNOLOGY: FROM NPS REACTORS TO POWER REACTORS

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Russia has gained unique experience in developing and operating reactor plants (RP) with lead-bismuth heavy liquid metal coolant (HLMC) as applied to power plants of nuclear-powered submarine (NPS).

At the beginning of the 50s the USA and the USSR almost at the same time started developing RP for NPS. In both countries the work was performed for two types of RPs – with water-cooled water-moderated reactors and reactors using LMC. In the USSR there was selected entectic alloy of lead and bismuth as LMC.

Altogether eight NPSs with RP with lead-bismuth HLMC were built. The first prototype NPS of 645 project had two reactors. The rest of 7 NPSs of 705(705K) project had one reactor. Besides, there were built and operated two full-scale surface reactor bench-prototypes: in FEI (Obninsk) and in NITI (Sosnovy Bor). During the last period of time all principle problems were solved, which took place in developing RP with HLMC including problems of heat exchange and hydrodynamics, technology of coolant, corrosion and mass transfer, ensuring of radiation safety during operation with equipment polluted by polonium-210. Regulations for operation, issues of RP repair and reactor refuelling were optimized. RP life amounted to about 80 reactor-years. For commercial application a new nuclear power technology, not having analogues in the world practice, was demonstrated.

At present the conditions are available for introducing this technology into civil nuclear energy.

As a result of cooperative work performed by FSUE OKB "Gidropress", GNC RF-FEI and the other organizations during last years, there was shown technical feasibility of creation of *unified reactor module SVBR-75/100* with fast neutrons core and lead-bismuth coolant in the primary circuit, application of which will permit to realize a module principle in creating nuclear power complexes of different purpose and power.

In developing SVBR-75/100 RP design there was accepted a conservative approach consisting in application of the primary and secondary parameters approved in practice and in maximum usage of approved fuel and structural materials and optimized principle decisions on equipment components and RP layout.

Compliance with this principle decreases due date, scope and cost of R&D work, provides RP reliability and safety of its operation, decreases investment risk.

This report deals with main structural features of RP with lead-bismuth HLMC and their evolution during conversion from NPS RP to RP for civil energy.

THE HYDRODYNAMIC FEATURES IN THE HEADER SYSTEMS OF TANK REACTORS AND HEAT EXCHANGERS

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The basic element of the flowing parts of tank reactors and heat exchangers is the distributing header systems. To construct economic reliable and safe reactors and heat exchangers, the optimum hydrodynamics of the flowing part of the header systems, namely, the given coolant flow rate (velocity) profile at the header exit, the minimum hydraulic losses for its pumping and maximum flow mixing in the header system, should be provided. In tank reactors, the distributing header systems with lateral flow supply and its central removal are widely used, and in heat exchangers – the distributing header systems with the central flow supply and its lateral removal.

In SSC RF IPPE, a wide complex of experimental researches has been performed to study the hydrodynamics of the flowing parts of the header systems.

As a result of the experiments conducted on the aerodynamic test facility using cylindrical models, the effect of the ratio of the header dimensions, its design, the coolant flow pattern in the flowing part of the system, the hydraulic resistance of the exit section of the system on the coolant flow rate (velocity) profile at the header exit and the hydraulic resistance coefficients for its flowing part were established. Besides, tracer concentration profiles at the header exit were obtained for the header systems of tank reactors as a result of investigation of coolant mixing.

In tests on hydroflume using plane models of the both types of the systems representing the longitudinal axial cross-section of the corresponding cylindrical systems, the extent to which the variation of the ratio of the header dimensions and its design may exert effect on the distribution of the water velocity at the header exit and the sizes of eddy and stagnant zones in the header and on the water flow pattern in the header system was established. Analysis of the results obtained made it possible to determine the typical coolant flow pattern in the flowing part of the systems.

The obtained experimental data and due account of the coolant patterns made it possible to develop a universal semi-empirical procedure for evaluation of the distribution of the coolant flow rate (velocity) at the exit of axisymmetric header systems of different types. .

The relative coolant velocity (flow rate) distributions on the radius and over the header exit width, obtained using cylindrical plane header system models, coincided only qualitatively.

The results of investigation of the hydrodynamics of the flowing parts in the header systems can be used in developing and verifying computer codes as well as in designing NPP and heat exchangers. The above investigations have been performed in co-operation with different design organizations available in MINATOM of Russia.

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A CONCEPT OF A PRESSURE-TUBE POWER REACTOR WITH SUPERCRITICAL WATER COOLANT

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A concept of a nuclear power plant with a once-through pressure-tube reactor with supercritical pressure coolant (PTR-SCP) has been proposed. The concept is based mainly on the proven technologies and design approaches applied for the traditional uranium-plutonium fuel cycle. Due to the specific features of the pressure-tube reactor design where the tight fuel element lattice is excluded the concept is envisaged for the thermal reactors.

The concept provisions have been developed to the extent of details sufficient for the subsequent feasibility study of the power unit and the evaluation of investments efficiency.

The results of neutronic and thermal hydraulic calculations have been used to justify the basic configuration of the reactor core with the minimum parasitic neutron absorption. Under supercritical coolant pressure the core of such design would remain competitive in terms of fuel cycle indicators as compared to other advanced reactors with traditional coolant parameters. At the same time the mandatory nuclear safety requirement is met, i.e. the neutronic characteristics of the reactor would ensure the stabilizing reactivity feedback on thermal parameters and, first of all, on void reactivity effect.

The fuel elements and fuel channels (FC) of two types have been analyzed as the main components of the core:

FC with a pressure-bearing zirconium pressure tube; thin-walled steel casing and the assembly of fuel rods of RBMK type with the fuel pellets of UO2 and fuel cladding of heat-resistant steel;

FC with an assembly of tubular fuel elements of AMB type with cermet fuel (UO2 middling in a metal matrix), outer casing and the central fuel element tube of heat-resistant steel placed in a gas environment.

Liquid neutron moderators have been considered: H2O, organic substance, i.e. ditolylmethane (DTM) and D2O. Solid moderators, in particular, graphite have not been considered because under SCP the graphite stack could not be cooled at required efficiency and maintained at acceptable temperature and, besides, utilization of graphite would present an additional problem. It was found that heavy water was the best candidate in terms of fuel characteristics.

The optimization calculations have revealed that the variant with cermet fuel in tubular fuel elements would offer better capabilities for PTR-SCP in terms of fuel cycle efficiency and neutronic characteristics related to safety. This variant has been taken as the basis for the concept.

The spatial two-dimensional reactor analysis has been made with account for fuel burnup in the equilibrium refueling. The results of the analysis were used to make the reactor core loading map with a square lattice of calandria tubes placed at a 190 mm pitch. The lattice of this type provides the best moderator to fuel ratio with account for some cells reserved for CPS channels. The reactor core loading consists of 880 cells, including 784 cells for FCs and 96 cells for CPS channels.

The schematic design of the reactor and design features of its main components have been proposed based on the developments. The mass and dimension characteristics of the reactors and the need of materials for its manufacturing have been estimated. Most design units of the reactor have been earlier manufactured and proven by operation.

The schematic flow chart of the power unit has been developed and the list of systems required for ensuring the normal operation of both the reactor facility and safety-related systems has been identified. The main stages of reactor startup and bringing to power have been analyzed. The major reactivity effects and coefficients have been determined. It has been demonstrated that CPS members could compensate for the reactivity margin in the reactor cold state, during startup and power operation.

The main aspects of the safety concept are provided and the set of the safety systems needed for the concept implementation is suggested. The design limits of fuel element damage have been determined.

The feasibility evaluation of the concept features has demonstrated that a power unit with PTR-SCP of 1960 MW(th) and turbine unit of 850 MW installed capacity with account for heat losses in D_2O($\approx$ 5%) and in-house electricity consumption (2.6%) could supply 828 MW(e) to a power network, i.e. the net efficiency of this plant is above 42%.

PRODUCTION OF RADIONUCLIDE PREPARATIONS AND IONIZING SOURCES OF MEDICAL USE

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Production of radionuclides and ionizing sources at Research Institute of Atomic Reactors (RIAR) has been started more than 30 years ago. This is founded on the unique reactor and engineering base of the Institute providing production of a wide range of high-specific activity radionuclides. Five research reactors are used for radionuclides production including SM high-flux reactor with the thermal neutron flux density up to $2 \cdot 10^{15}$ $cm^{-2}s^{-1}$. The specialized radiochemical facility containing of 13 hot cells, allowing reprocessing large quantities (up to 100.000 Ci) of radioactive materials, are used for isolation of radionuclides and their purification from impurities. A chain of hot cells with a set of equipment intended for encapsulation of radioactive materials (source active core), capsule sealing, source decontamination and certification, is used for production of ionizing sources.

A wide range of radionuclide preparations and sealed sources are currently produced at the Institute. Among them products of medical use take considerable part of the entire production. In particular preparations of ^{89}Sr, ^{131}I, ^{51}Cr, ^{188}W, $^{113,117m}Sn$, ^{106}Ru, ^{252}Cf neutron sources, ^{60}Co, ^{192}Ir, ^{153}Gd photon sources are produced and supplied to domestic and foreign Customers. Advanced technologies are developed for production of new radioactive products, like miniature ^{90}Y and ^{144}Ce beta-sources for vascular brachytherapy, ^{103}Pd sources for treatment of prostate cancer. A review of the current status of the production of radionuclide preparations and sealed sources used for medical applications operated by RIAR is presented in the report.

ISO CD 1856. GUIDELINE FOR RADIATION EXPOSURE OF NONMETALLIC MATERIALS

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In the framework of the International Organization for Standardization (ISO) activity we started development of international standard series for space environment simulation at on-ground tests of materials. The proposal was submitted to ISO Technical Committee 20 (Aircraft and Space Vehicles), Subcommittee 14 (Space Systems and Operations) and was approved as Working Draft 15856 at the Los-Angeles meeting (1997). A draft of the first international standard "Space Environment Simulation for Radiation Tests of Materials" (1st version) was presented at the 7th International Symposium on Materials in Space Environment [1]. The 2nd version of the standard was limited to nonmetallic materials and presented at the 20th Space Simulation Conference [2]. It covers the testing of nonmetallic materials embracing also polymer composite materials including metal components (metal matrix composites) to simulated space radiation. The standard does not cover semiconductor materials. The types of simulated radiation include charged particles (electrons and protons), solar ultraviolet radiation, and soft X-radiation of solar flares. Synergistic interactions of the radiation environment are covered only for these natural and some induced environmental effects. This standard outlines the recommended methodology and practices for the simulation of space radiation on materials. Simulation methods are used to reproduce the effects of the space radiation environment on materials that are located on surfaces of space vehicles and behind shielding. It is very important to use the ISO regulations for radiation test and selection of nonmetallic materials for spacecrafts with onboard nuclear reactor where serious synergistic effects of space and nuclear radiation are predicted [3].

It was discovered that the problem of radiation environment simulation is very complex and the approaches of different specialists and countries to the problem are sometimes quite opposite. To the present moment we developed seven versions of the standard. The last version is a compromise between these approaches. It was approved at the last ISO TC20/SC14/WG4 meeting in Houston, USA, October 2002. At a splinter meeting of Int. Conference on Materials in a Space Environment, Noordwijk, Netherlands, ESA, June 2003, the experts from ESA, USA, France, Russia and Japan discussed the last version of the draft and approved it with some notes. A revised version of the standard was presented this May at ISO TC20/SC14 meeting in Russia.

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COMPUTATION ANALYSIS OF FUEL MELT CONFINEMENT PROCESSES IN FAST REACTORS

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An accident with fuel meltdown in fast reactors is under consideration. It is assumed that the reactor core is destroyed partially or completely. Further accident process development could lead to either conservation of the reactor vessel (or its safeguard vessel) integrity, or wall melt-through. The BRUT (*in Russian* Fast Reactor, Fuel Confinement) code is purposed to study feasibility of the residual heat removal in the destroyed fuel rod debris and of the meltdown confinement in the reactor vessel.

The design model under consideration is complex and comprises 14 sub-regions. Heat-releasing layer, sodium layer above the heat-releasing layer; lower axial blanket; “gas cavity”; header modules; pressure chamber; in-vessel radiation shield; sodium-wetted intermediate heat exchanger; reactor-vessel and guard-vessel layer; support belt; heat-conducting layer of the lateral blanket FSAs survived; other in-core structures are simulated.

Mathematical simulation of the sub-regions as porous bodies is executed by implementing conservation laws for mass, momentum, and energy, which are stated as equations for continuity, motion, and energy in 2-D cylindrical coordinate system; they are solved with the corresponding boundary conditions.

For the first time the task of heat-releasing layer formation at the lower axial blanket has been solved. During changes of the state of layer component aggregation (melting, boiling), heat sinks are taken into account. Recalculation of thermal and physical properties of the layer with the change of its component volume concentration is under way.

The Stephen’s task is being solved at the melting of the lower axial blanket, “gas cavity” steel, header modules; upper plate of the pressure chamber.

The task of melt penetration through the channels of Control and Protection System has been solved analytically.

In the process of hydrodynamic calculation, a closed circuit of natural circulation is formed in the sub-regions, which provides cooling of the vessel internals. Heat sinks are taken into account in the sub-region with heat exchangers.

The BRUT code is applicable for velocity and temperature field computations in all the sub-regions of the design model under consideration.

Using the code it is possible to determine whether the melt goes to the vessel bottom or not and how it interacts with it.

The BRUT code can be used for computations of any kind of accident scenario.

Computations of two accident scenarios have been performed. In the first case it is assumed that 6 FSAs around the central shim rod are melted down. In the second case - 36 FSAs are melted down, i.e., four FSAs rows in the reactor core center, around the central shim rod. Thus, on the basis of the BRUT code calculation results the following conclusion can be made: there is no destruction of the delivery header, corium release to the vessel bottom, and corium interaction with reactor vessel observed. Consequently, it could be stated, that molten fuel is confined in the reactor vessel in both cases (destruction of 6 and 36 FSAs).

STUDIES ON RADIOACTIVE CONTAMINATION OF SPENT REACTOR GRAPHITE FROM PUGRS OF SIBERIAN GROUP OF CHEMICAL ENTERPRISES WITH THE PURPOSE OF FURTHER MANAGEMENT PLANNING[3]

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A strategic aim of radioactive wastes management, independent on their generation origin and parameters, is an exclusion of a possibility to contaminate the environment with radionuclides and another toxic materials contained in the wastes during full period of their potential threat. It corresponds with the internationally adopted principle that all the ecological problems unsettled now must not be left as an inheritance to the next generations.

For the prolonged period of nuclear reactors operation, huge amounts of spent graphite have been accumulated in several countries. The reactor-grade graphite was used as a neutron moderating and reflecting material. For example, in the United Kingdom, twenty-six Magnox-type reactors and fourteen commercial AGR-type reactors were erected and operated. Three nuclear power plants (NPP) with Magnox-type reactors were under operation in France. At Hanford site (USA) there are nine decommissioned reactors for production of weapons-grade plutonium. The uranium-graphite reactors have been built also in Spain, in Japan and Italy (Magnox-type reactors), in Germany and China. In the former Soviet Union, thirteen uranium-graphite reactors for production of weapons-grade plutonium, four power uranium-graphite reactors at Bilibino NPP, two power uranium-graphite reactors at Beloyarsk NPP and eighteen RBMK-type reactors have been erected and operated [1].

[3] The paper presents results of the studies performed specialists from MEPhI, SGCE and IPhCh RAS within the frames of three ISTC Projects ##561, 1409, 1722. during the period from 1996 to 2003.

The reactor-grade graphite represents the largest fraction of the solid radioactive wastes (SRAW) accumulated all over the world. So, search for optimal ways of spent graphite utilization is an urgent problem of international importance.

The purpose of this study is to evaluate radioactive contamination of the graphite sleeves from the decommissioned SGCE reactors (I-1, EI-2 and ADE-3) and to analyse a possibility for their utilization by means of incineration. Planning of actions on spent graphite management requires the reactor type, spent graphite parameters, features of graphite usage and storing should be taken into account.

It can be concluded that:

1. the most part of the SGCE storages where spent graphite sleeves are disposed of do not meet contemporary requirements;
2. the radioactive contamination level of spent graphite sleeves from the SGCE reactors allows their utilization by incineration at the SGCE territory; ^{14}C discharge into the atmosphere will not cause any significant risk of negative consequences for human health of cities Seversk and Tomsk population;
3. full cycle of spent sleeves incineration and subsequent RAW storing can not lead to appearance of nuclear dangerous situation;
4. there are available technologies and equipment for conduction of experiment on spent graphite incineration.

Taking all the aforementioned into consideration, the authors suppose that nowadays there are all the conditions for planning, preparation and conduction of full-scale experiment on incineration of some spent graphite sleeves taken from the SGCE storages. The results obtained in the test graphite incineration will allow to evaluate and, if necessary, improve the utilization technology and used equipment.

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DEVELOPMENT OF PASSIVE DEVICES FOR EMERGENCY PROTECTION OF FAST REACTORS

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In our time one of the most actual problems of nuclear engineering development is the increase of safety NPP. The researches conducted in different countries, have shown, that one of the most effective ways is using so-called of passive devices of emergency protection (PDEP). PDEP call also as "direct-acting" and "self-operated" devices (SASS).

The operation PDEP does not demand external sources, circuits of the control and direction and their operate descends at exceeding of the main operational parameters (usually temperature, coolant flow rate) under maximal possible values. On the basis of such devices there is a possible development nuclear power installations, internally steady against failures of means and errors of staff. The probability of sever accident development (such as ULOF, UTOP, ULOHS) fast reactors thus decreases practically on two order.

In SSC RF-IPPE the practical works on development PDEP for BN-800 started in 1988 under Yu.E. Bagdasarov management. In next years the main requirements shown to PDEP BN-800 were formulated by R.M. Voznesensky:

- the efficiency is close to effect power value (0,3%□K/K);
- lag is not more 10 s, that excludes eliminates boiling of sodium in reactor core ($T_{boil} \geq 900$°C);
- actuation is at achievement of sodium temperature on exit from reactor core, □650□C and at flow rate decrease of sodium of first loop □0,5G_{nom};
- constructional distinction from a nominal system reactor core with the purpose of exception of failure on the general cause.

For elapsed years in SSC RF-IPPE the following types of PDEP in a different degree were researched:

- hydraulically fluidized rod;
- magnetic on the basis of "Curie point";
- on the basis of hyperthermal memory effect of form;
- on the basis of nonconventional physical effects, including lyophobic.

The stored experience and condition of development of different types PDEP is analyzed. The perspective designs and characteristics of devices take into consideration.

The advantages and defects of different type of Russian and foreign devices of the given assigning are analyzed. Comparison of perfection degree of different types PDEP for fast reactors was executed by estimation expert method on the complex of generalized characteristics and guidelines on selection of the most perspective types PDEP for fast reactors are given.

THE UTILIZATION OF THE PROGRAM COMPLEX PROSTOR IN CALCULATION INVESTIGATIONS CONCERNING THE APPLICABILITY OF COOLANT NATURAL CIRCULATION REGIME WITH THE EXPANDED PARAMETERS SCALE OF NPP WITH VVER-1000 PROVIDING VIOLATION OF NORMAL OPERATION REGIMES

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The urgency of the complex mathematic model utilization in calculation equipment support of NPP is of primary importance. The best example of their practical adaptation in calculation support of NPP with VVER-1000 operation may be successfully found in such programs and program complexes as: “BIPR-7”, “PERMACK”, “KASKAD”, “PIR-VOPOL”, “IR”, “NOSTRA”, “RAGUGA”. In addition to this list one cannot but mention the computer analytical simulators processed by Federal Scientific Research Nuclear Power Plant Center (VNIIAES) - in the center of simulators making and mathematic modeling in Atomic Energy Institute (Obninsk) - in training applied to almost all NPPs. Among the program complexes have been mentioned the program complex PROSTOR developed by the group ENICO TSO as to the contract with KlnNPP being processed in different NPP services takes the certain place.

The main target of the program complex PROSTOR is to establish the coordinated calculations, both neutron-physical and thermo-hydraulic ones, of stationery and transient processes in Primary System NPP with VVER-1000 equipment and also it should be aimed at the integrate modeling of monitoring and protection NPP systems.

The most notable example of the program complex PROSTOR utilization on a large scale at NPP should be calculation investigations coolant natural circulation regime with the expanded parameters scale of NPP with VVER-1000 providing violation of normal operation regimes.
The investigation was initiated by Saint Petersburg Polytechnic University and was ordered by The Operating Organization. As the key direction of the state cutting edge nuclear energy technologies is PS with VVER with greater power (with VVER-1000, might be VVER-1500 in the future), the issue concerning energetic power rates in regimes with forced circulation losses.

Within these investigations, it was considered the more concrete task. It may be introduced certain changes in operational regulations of VVER-1000 with the purpose to avoid Emergency Protecting snap into action and to transfer the reactor to the critical state by the simultaneous 3-4 MCP shut off.

In the report the results of calculation investigations of certain accident regime are given. It is bound to be accompanied by the work-stop of all the MCP and PS work through 30% power compared with the nominal level in Coolant Natural Circulation Regime under the worst condition according to the Nuclear Core characteristics (the selection was carried out through 32 fuel loadings). The method made all the results about CNCR realizations more reliable.

Calculation results have shown that none of the limiting factors of PC has been exceeded in the one- phase CNCR. One may consider the question about regulation changes under normal operation violation involving the forced circulation losses.

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PROSPECTIVE MATERIALS AND TECHNOLOGIES OF NEW MATERIALS FOR ATOMIC POWER ENGINEERING

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The development tasks of new materials are determined by the development strategy of atomic power engineering. In the coming decades, their base will be water-moderated reactors (WWERs) [1]. At present, materials of the core region, reactor vessels, pipes and other facilities of power and transport reactors are developed and successfully functioning. Nevertheless, new materials are necessary to obtain higher burnups and develop power reactor facilities (PRFs) of new generation [2].

The directions of improving the tablet fuel are prospective: to form a grain of the maximum size to decrease gas release and improve the behavior under transition regimes, plasticizer additions, additions to increase thermal conductance. Prospective are cermet fuel based on micro-fuel elements, fuel with inert matrixes, vibrocompacted nuclear fuel, development of nitride and carbonitride nuclear fuel, as well as development of high-density metal fuel. The composite nuclear fuel and fuel with a gradient structure along the tablet radius deserve investigation [1,2].

Modification of zirconium alloys is carried out in the field of core region structural materials of thermal neutron reactors. Composite, multilayer, thin-walled cladding materials, bimetallic pipes are prospective to get super-high burnups and guarantee manoeurable regimes. The works being carried out on alloying the surface layers of E110 and E635 alloys under ion mixing, including those in MEPhI, are a definite step in this direction [1,2].

As to structural materials of core regions for fast neutron reactors at a burnup up to 17% f.a., investigations are being carried out to replace austenitic steels by ferritic-martensitic steels strengthened by nanosized oxides (of titanium, yttrium, and others – ODS steels) to increase high-temperature strength at temperatures up to 700^0C [1,2].

The main structural material of PRFs and APP-equipment are austenitic, ferritic/martensitic, martensitic, perlitic and other steels. To increase the workability of steel articles, it is necessary to form a uniform structure-phase state as a result of the production technology improvement, optimize the chemical composition, decrease the level of harmful impurities (Cu, P, S) in a metal, and broaden the production of powder steels [2].

Today the urgent problem is to obtain low activated materials of new generation that would possess a sufficient complex of technological and service properties and have, at the same time, a low induced radioactivity with short half-life periods of resulting isotopes.

PRFs of the future nuclear power engineering are thermal-neutron water reactor of supercritical parameters with an output temperature of 400 – 450^{o}C, fast liquid metal reactors with a coolant temperature more than 700^{o}C and an economic gas turbine cycle, high-temperature gas-cooled reactors with an output temperature up to 850^{o}C [1]. The necessity of high temperatures is also connected with the technological needs of the future industry and hydrogen power engineering.

Analogous tendencies under the development of the future type of reactors are being observed throughout the world (ultra-supercritical coal-fired power plant at 650^{0}C). In this case, higher parameters of the coolant stated in INPRO and GENERATION – 4 programs are being considered. At the same time, in spite of the good developmental work of the reactor physics, the materials science support of these projects is remaining behind.

Prospective high-temperature materials with low-induced activity are high-clean vanadium alloys, new carbonic radiation-resistant materials, for example, composites, high-temperature nanoceramics and composites on their base, intermetallic alloys.

The first wall and the divertor are the most stressed structural elements of the future thermonuclear reactor. The most advanced idea in the field of the first wall materials is the development of gradient composites (GC), i.e. materials with a composition changing along their thickness from the plasma facing surface to the surface that is in contact with the heat-removing alloy. Intermetallic compounds, as well as alloys on their base, obtained from metals with a relatively low atomic mass, for example aluminum and titanium, attract attention as a new, alternative armour material [2].

Thus, in conclusion it is necessary to point out that the efficiency and competitiveness of the existing and future atomic power engineering in many respects depend on the development of new types of fissionable and structural materials, the technologies of their fabrication and treatment, application of the achievements of powder technologies, with the use of speed-hardened materials and powders, to production, including mechanoactivation (ball milled in a high energy attritor) of powders before compacting and sintering.

Considering extremely severe and various demands to the properties of materials for structural units of atomics working under hard conditions, materials with gradient structure, composition, and properties, as well as composites and ceramics are prospective. The given materials can meet these requirements to a great extent.
Taking the positive influence of structural elements and hardening phases on dispersion properties in structural materials into consideration, now it is necessary to focus attention on investigation of nanostructural materials, their use for strengthening and improving the properties of materials.

The change-over to the alternative (existing) materials and technologies in atomic power engineering certainly depends on the economic expediency. Nevertheless, it is inevitable. Firstly, it is determined by the specific character of the work of materials; secondly, the obtained properties of traditional materials have attained their physical limit.

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SPACE NUCLEAR POWER IN VIEWS: 50 YEARS AGO AND PREVISION FOR 50 YEARS

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The second half of the XXth century became the age of the origin and formation of space nuclear power. During that time the potentialities and advantages of its application in large, medium and small nuclear rocket propulsions (NRP) were being earnestly demonstrated. The prototypes of different level NRP reactors were being tested in the USA and in the USSR during 1970÷1980.

Since 1956 the practical works on studying the opportunities of use the nuclear power installations (NPI) with direct methods of converting thermal power into electricity at the space vehicles have been launched. In addition to radio-isotopic space generators of current, reactor thermoelectric installations SNAP-10A of 0,5 kW, "Bouk" of 3 kW and thermo-emission installation "TOPAZ" of ~6 kW were proposed, designed and constructed. 32 "Bouk" installations were operating in space during 1970÷1988. Two "TOPAZ" installations successfully passed flight space testing in 1987÷1988.

An important contribution to design and construction of "Bouk" and "TOPAZ" installations was made by V.Ya. Poupko.

Simultaneously with the designs which reached their technical realization, the feasibility studies of the whole number of installations with different class reactors were carried out in the USSR (Russia) and USA. Brief descriptions of some of them are cited in the present Report. They were the modernized variants of thermo-emission and thermoelectric installations ("TOPAZ-2", SP-100) as well as the variants of combinations of the NPIs with multi-mode functioning, installations with power convert systems removed from the core, bi-modal installations using NRP and NPI solutions, installations based on the lithium-niobium technology and installations with machine methods of conversion.

However, in the end of the XXth – in the beginning of the XXIst centuries, depending on the economical expedience of the space NPIs, the higher requirements were presented to power (from several kW units to several hundred kW units), to the operating time (from 5-7 years to 15-20 years) and to assurance of nuclear and radioactive safety.

The results of the scientific-technical and feasibility-study works on the advanced space NPIs and their usage as parts of the propulsion-power modules are summarized in the Report. The brief description of works evolution, the current state of the NPIs and perspectives of their development are presented. The preference is given to the space NPIs with direct methods of power conversion.

In the end of the XXth century the new type of the converter of nuclear reactor thermal power into highly-organized power of photon radiation for the purpose to give momentum to the spacecraft in the interstellar space flight was proposed by V.Ya. Poupko. Maybe, construction of such nuclear photon propulsion will be a reality of the XXIst century.

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ALGORITHM OF SPATIAL XENON OSCILLATIONS IDENTIFICATION

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The goal of the paper is to propose original algorithm of neutronics model adaptation to reactor state.

The adaptation of the reactor kinetics model to the current reactor state is the first step of optimal control strategy identification. One algorithm of model setup is proposed in paper [1]. In this algorithm, the xenon and iodine concentration is calculated by reproduction of observed offset and reactor neutron power behavior. Experimental offset values was fitted by simulation of control rods movement from they observed positions. Reactor power was simulated by changing of bore concentration in core. The reduction of spatial model was realized by averaging of neutron flux contribution on up and down parts of core [2].

This approach causes two questions.

Described reduction method can be interpreted as power distribution convolution with sectionally-constant function. The optimal functional basis that is necessary to predict transients with setting precision is unknown. Necessary conditions and time of simulation convergence to reactor behavior is unknown too in this approach.

To solve these problems the follow approach was proposed.

1. The main component basis (MCB) [3] is calculated using the time sequence of xenon and iodine concentration relative deviations vectors obtained by transient simulation.
2. The utilized 3D model was projected to calculated basis. The projecting gives two systems of equations. The first relates xenon-iodine concentration projections with their derivatives and control rod positions. The second is the linear equation system that relates the neutron sensors signals and. concentration projections.
3. The search of model initial conditions was made so that simulated and observed signals are to be congruent. At that the simulated and observed control are congruent too.

The difference between the firs and second approaches is following. In first the model is setup by control rod movements. In second the initial conditions are corrected to secure the congruence of the simulation and experiment.

The freedom of the basis dimension choice makes possible to answer the question, what basis dimension is enough to simulate transients up to assigned precision.

Some numerical experiments were made to demonstrate the proposed methodology. The using experimental data was obtained by the contributed neutron model “MFA” of reactor VVER-1000.

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APPLICATIONS OF THE "PROSTOR" WWER CORE SIMULATOR

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Experimental Research and Design Company “Simulator Training Systems” and Simulator Systems Laboratory of MEPhI (SSL) have the ten years stage in sphere of computer simulators and training systems development for different types NNP. During this period SSL completed number of projects such these:

- computer simulator development for different work places in Kursk NPP;
- simulators for reactor and turbine departments in South Ukrainian – 3 NPP;
- simulators for core and primary coolant circuit in Ignalina NPP;
- neutronics and thermal hydraulics reactor core mathematical real time models for full scope Goesgen NPP simulator;
- reactor core and coolant circuits simulators for Kalinin NPP.

The computer code PROSTOR was developed on base of simulator for Kalinin NPP. The code is under certification procedure in GOSATOMNADZOR (Russian Regulatory Board) now.

Latest applications of Kalinin NPP simulators are:

- development of online reactivity control system for reactor start up procedure;
- investigation of coolant natural circulation mode acceptability in case main circulation pump failure.

The sphere of PROSTOR code implementation isn’t restricted by the items mentioned above.

Now the following problems are under consideration in SSL:

- the calculation analysis of existing techniques of reactor experiments performing and processing;
- experimental-calculation reactor core power distribution analysis;
- online reactor core equipment monitoring.

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AN OVERVIEW OF THE TACIS PROJECT (R2.03/97) DEALING WITH WWER-1000 AND RBMK TECHNOLOGIES

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The Department of Mechanics, Nuclear and Production Engineering (DIMNP) of the University of Pisa has always been involved since its institution, at the beginning of the sixties, in the development and in the advanced use of tools for the safety analyses of nuclear power plants. Experimental and theoretical activities were carried on in the fields of thermal-hydraulics, structural mechanics and radiation detection and protection as well as in the field of systems codes improvement and validation.

Nowadays, a broad part of the DIMNP is involved into the TACIS project (R2.03/97). This is a project instituted by the European Commission for the technical cooperation between European subjects (i.e. the DIMNP) and some Russian institutions (Rosenergoatom, Gosatomnazdor, NIKIET,) in the field of nuclear safety, having the main purpose of a technological diffusion. TACIS project started at the beginning of 2004 and it will last for the next two years.

The project is divided in two parts (A & B). Part A, focused on light water reactor technology (i.e. WWER-1000), deals with the execution of system thermal-hydraulics experiments aimed at understanding the physical scenarios and with the demonstration of the quality level of 'best estimate' computational tools adopted for safety and licensing applications like CATHARE, RELAP5/3.2, RELAP5-3D ©. The code accuracy evaluation will be performed with the Fast Fourier Transform Based Method (FFTBM), a tool proposed at the beginning of the nineties by the DIMNP at nowadays widely adopted in the international community. The scaling issue (i.e. the demonstration of similarity between experiments performed in differently scaled facilities and between measured phenomena and phenomena expected in the reference NPP) is also the other relevant purpose for the design, execution and analysis of these experiments.

Part B, focused on RBMK (graphite moderated, boiling water cooled) technology, deals with the development, improvement and the qualification of suitable modelling for accident analysis including severe accidents. The main tasks specified consist in connecting a number of different models already developed by Russian institutions and in demonstrating their quality in the light of Western standards and procedures. The final goal is the achievement of a computational tool for calculating complex scenarios that are peculiar to the RBMK with the specific capability of evaluating the performance of graphite and its interaction with other core materials.

From what reported above, it is clear that a great number of aspects of the nuclear technology are involved: codes assessment and validation, scaling issue, neutron-kinetics / thermal-hydraulic transients analyses, fuel pin mechanics, severe accidents study and accident management. Exchanging information and personnel training by experts of both parts is already in course and it will be intensified in the next future.

Therefore, TACIS project it is revealing a great opportunity for the Russian institutions for integrating themselves into the wider environment of the Western countries. At the same time, TACIS is also giving a great opportunity to the DIMNP, allowing it to maintain and increase its competences in the field of the nuclear power technology.

BILIBINO NPP: THIRTY-YEAR OPERATING EXPERIENCE AND LIFETIME EXTENSION

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Bilibino nuclear power plant (BiNPP) has been in operation since early 1974 near Bilibino settlemenmt located in Tchukotka Autonomous District. This NPP has demonstrated high effectiveness of use of nuclear energy sources under conditions of Far North-East region of our country.

BiNPP was designed as nuclear co-generation plant. Specific features of NPP site required the use of innovative design approaches. The correctness of these approaches has been confirmed by the BiNPP operating experience. This NPP consists of four similar power units. Installed electric power of the NPP is 48 MWe assuring 78 MWth heat supply. In the period of stable economic conditions in Russia (before 1991), load factor of BiNPP was 85%, availability factor being equal to 90-92%. Economical characteristics of the BiNPP were much higher than those of local fossil fuel power plants. Now this difference is even more significant because of the increase of cost of imported fossil fuel. Studies on reactor accidents resulting in standard operation of reactor safety system, as well as in case of safety system failure revealed high inherent safety characteristics of the reactor plant. Reliability of reactors has assured high stability of the BiNPP as power source under extreme conditions of Far North-East region of our country. Design lifetime of the first power unit of the BiNPP equal to 30 years expired in January 2004, while that of the fourth power unit will expire in December 2006. The issue of extension of the BiNPP lifetime arose in the view of impossibility of implementation (because of high cost) of the second stage of the BiNPP, design of which was completed in 1992.

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PERSPECTIVES IN DEVELOPMENT OF A VESSEL-TYPE NUCLEAR REACTOR WITH ONCE-THROUGH STEAM SUPERHEATING

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This work presents the analysis of main problems characteristic for once-through nuclear reactors with direct steam superheating and suggests a set of interrelated decisions ensuring feasibility of the concept of a vessel-type reactor of such a type.

Basic peculiarities of once-through reactors with direct generation of superheated steam are the following:

- essential nonuniformity of power distribution caused by great changes of coolant density;
- high sensitivity of coolant parameters to nonuniformity of power distribution (10-15 times greater than for other reactor types), that is caused by high enthalpy increase;
- high thermal-hydraulic sensitivity to bypass flows of underheated coolant;
- possibility of thermal-hydraulic instability of coolant flow;
- transport of activity of corrosion products and fission products into the turbine part;
- corrosion and stability problems for structure materials at high temperatures;
- ensuring the safe reactor behavior in severe accidents.

The paper studies the possibility to solve these basic problems using a set of the following correlated technical decisions:

- two-circuit scheme of the reactor installation;
- reactor core on the basis of coated fuel microparticles (CP) directly cooled by water coolant of supercritical pressure;
- continuous refueling with supplying the "fresh" coated fuel particles from above using the sand-glass principle
- upward flow of coolant (countercurrent with respect to CP movement).

(1) The use of the two-circuit scheme excludes the problem of activity transport into the steam-turbine part of the facility and, therefore, allows imposing realistic requirements to CP tightness at the presently achieved level. The technical and economic

characteristics of a two-circuit facility do not differ a lot from such characteristics of a one-circuit facility, since the contribution of steam generators and pumps is rather moderate. Use of a standard turbine with steam at 16 MPa pressure and 530-570^{0}C temperature allows for reaching 44-46% efficiency of turbine island.

(2) The use of a reactor core on the basis of coated fuel microparticles (CP) directly cooled by water coolant of supercritical pressure allows to solve the basic problems mentioned above and simultaneously ensure the safe power plant behavior in accidents.

- Large heat exchange surface practically eliminates the problems connected with heat transfer, including heat exchange crisis (DNB).
- Direct cooling of coated particles allows to design a reactor with continuous refueling with supplying the "fresh" coated fuel particles from above. When applied to a once-through reactor, continuous CP refueling in each FA provides the constant power density field not only during lifetime of CPs, but also during the whole period of reactor operation at power. Effects stipulated by the loading of a “fresh” FA in reactors with fixed fuel are excluded, thus eliminating the resulting local overheating of coolant.
- Estimations showed that for the chosen core design the temperatures of CPs and structure materials would not exceed 1000-11000C in LOCA.

(3) Continuous CP refueling permits to turn to advantage the known drawback of boiling reactors and reactors with steam superheating, namely, the drastic decrease of the coolant density in the upper part of the core, because in the proposed reactor design this part contains the coated particles with “fresh” fuel and, therefore, the maximum content of fissile material.

The key element of the active core concept is the fuel assembly (FA) intended for the placement of CPs, organization of coolant flow and movement of CPs at their continuous refueling. The FA design combines the cross and longitudinal coolant flows and makes use of zirconium alloys and steel thermal insulation where necessary. It allows to ensure low FA hydraulic resistance, in the same time having the acceptable non-uniformity of the coolant heat-up. In a FA there are three sections with cross coolant flow located one above another. In so doing, the output collector of an underlying section is connected with the input collector of an overlying section. At such a design the non-uniformly heated coolant mixes up in the collectors.

The FA design also includes decisions aimed at the increase in moderation ratio, such as placement of additional volumes for moderator (e.g. in the form of tubes filled with water), or reduction of the uranium fraction by reduction of the fuel kernel size. Even decreasing the total uranium loading in the core, the latter does impact the core availability factor due to the use of continuous refueling.

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FORMALIZATION OF SYSTEMATIC ANALYSIS OF INNOVATIVE DEVELOPMENT OF NUCLEAR ENGINEERING

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Key words: an innovative project, a nuclear power installation, a concept of sustainable development, a systematic analysis

They are considered several questions connected to formalization of procedure of qualitative estimation of potential of innovative designs in nuclear power.

As an example of development of methodology of systematic analysis of technology system it is considered INPRO project. In work it is prospected analogies between positions of methodology INPRO developed by the international commissions of experts and a classical formalism of the integrated system analysis.

There is brief description of basis of systematic analysis and it's applicable for studying and prognosis of development of power generating systems.

The Nuclear power is represented as an open subsystem of more large and powerful system. Groups of resources which are necessary for development of nuclear power are selected. Resources of accessible nuclear fuel and sharing materials, mineral resources, and also two sorts of the resources determined assimilation ability of environment, such as biological and earth's ones, and also adaptable abilities of a technological complex are selected actually.

It is shown, that in some cases critical resources can appear not fuel stocks, and resources, corresponded with a level of development of a technological complex.

As the offered analysis, first of all, is directed on a concrete definition of concepts of system criteria of sustainable development, then on the basis of the analysis of concept of sustainable development they are emphasized several social groups incorporated on interests and requirements which can be put forward to development of power engineering and, in particular, of nuclear power.

Utilisation of ship-propulsion reactor technology with 6000 reactor-years of operating experience as well as availability of advanced project of ABV reactor, involvement of experienced scientific and design institutions and enterprises of Minatom allows to construct and supply of small floating NPPs to consumer in short term and provide full scale life-cycle technical support for them.

NUCLEAR FUEL CYCLE IN RUSSIA - STATUS AND PERSPECTIVES

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A prerequisite for the nuclear power fuel cycle is the creation of the nuclear weapons industry including the fabrication of highly concentrated fissile materials with the use of uranium-graphite reactors to produce plutonium, radiochemical and metallurgical productions to extract plutonium as well as the fabrication of highly enriched uranium by gaseous diffusion and electromagnetic methods for isotope separation and the metallurgy of highly enriched uranium.

At those plants, problems on the fabrication of uranium fuel for ship nuclear power facilities, research reactors and first experimental-and-industrial nuclear power plants have been solved.

For the development of nuclear power engineering, capacities of mining enterprises were considerably increased for the most part on the basis of ore deposits in the Ukraine and Central Asia and capacities of the isotope separation complex of Russia continued to increase on the basis of gas centrifuge technology. Moreover, large-scale capacities to produce fuel for nuclear power plants were created.

The radiochemical industry still lags behind the demands. These suggest the construction of irradiated fuel storages.

In view of objective circumstances (disintegration of the USSR, absence of perspective uranium deposits on the Russian territory) the development of modern nuclear-power engineering in Russia is principally based on stockpiles of various kinds of nuclear materials. In the future, it will be also based on plutonium isolated from the nuclear weapons complex and probably on highly enriched uranium after its dilution up to energy concentrations.

Gas centrifuge technology for isotope separation ensures an essentially lower cost price of separation work in comparison with diffusion work. That is why there is no point in using highly enriched military uranium on the home market after its dilution that is substantially a loss of previous separation work.

The detailed analysis of stockpiles of raw nuclear materials and production capacities shows, that there are no special problems for the development of nuclear-power engineering up to the middle of the century. However, the lack of a reliable source of raw materials for the further development necessitates the world market integration. According to supply conditions, it can be compensated by the surplus and possibility to increase separation capacities. At the same time, the radiochemical industry should be developed because there are large stockpiles of irradiated fuel and it is necessary to use nuclear materials more efficiently. Besides, problems on the cycle closure must be reliably solved primarily from the ecological reasoning not to leave our problems to future generations.

In Russia, a large potential of perspective nuclear fuel cycle technologies is accumulated. Among these are uranium hydrometallurgy designed for processing of depleted raw materials using the most efficient processes, isotope separation by a gas centrifuge, rich experience in developing radiochemical processes and instruments. This allows to offer those technologies on foreign markets and convert nuclear technologies to other industry branches, first of all to technologies of the production of rare, non-ferrous and noble metals, stable isotopes, environmental protection technologies etc.