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Overview of Sandia National Laboratories and Khlopin Radium Institute Collaborative Radiological Accident Consequence Analysis Efforts

M.L. Young and D.D. Carlson
Sandia National Laboratories
Albuquerque, New Mexico USA

L.N. Lazarev, B.F. Petrov, and V.N. Romanovskiy
V.G. Khlopin Radium Institute
St. Petersburg, Russia

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Introduction

In January, 1995 a collaborative effort to improve radiological consequence analysis methods and tools was initiated between the V.G. Khlopin Institute (KRI) and Sandia National Laboratories (SNL). The purpose of the collaborative effort was to transfer SNL's consequence analysis methods to KRI and identify opportunities for collaborative efforts to solve mutual problems relating to the safety of radiochemical facilities. A second purpose was to improve SNL's consequence analysis methods by incorporating the radiological accident field experience of KRI scientists (e.g the Chernobyl and Kyshtym accidents).

The initial collaborative effort focused on the identification of:

- safety criteria that radiochemical facilities in Russia must meet.
- analyses/measures required to demonstrate that safety criteria have been met.
- data required to complete the analyses/measures identified to demonstrate the safety basis of a facility.

In addition, SNL staff presented a one week consequence workshop hosted by KRI. The workshop included an introduction to the MELCOR Accident Consequence Code System (MACCS). KRI developed MACCS sample problems of mutual interest to KRI and SNL in order to exercise the analysis methods presented in the workshop. KRI then ran the sample problems to gain experience in the application of radiological consequence analysis.

MACCS[1,2,3,4] was developed at Sandia National Laboratories (SNL) under U.S. Nuclear Regulatory Commission (NRC) sponsorship to estimate the potential offsite consequences of severe accidents at nuclear power plants (NPPs). MACCS, publicly released in 1987, models the transport and dispersion of plumes of radioactive material released to the atmosphere and the subsequent human dose, health, and economic consequences. Results generated by MACCS can be presented probabilistically, in the

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form of complementary cumulative distribution functions generated using a year of hourly meteorological data.

Identification of Safety Criteria and Analyses/Measures Required to Demonstrate Safety Basis of Russian Nuclear Facilities

Legal, engineering and technical, and ecological safety approaches and criteria applicable to Russian radiochemical facilities were examined. Conditions at the currently operating RT Mayak and the planned RT-2 Mining and Chemical Association (MChA) radiochemical plants were briefly reviewed with regard to the identified safety criteria.

Summary of Criteria for Establishing Safety Basis of Facilities
Six criteria were identified that Russian radiochemical facilities must meet in order to demonstrate the safety basis of facilities:

- compliance with established technical standards and legal requirements (i.e. dose limits), with consideration of more stringent standards and requirements that are likely to be implemented,
- prevention of adverse effects on workers, the public, or the environmental,
- implementation of technological approaches that conform to international standards in all stages of the radiochemical process,
- implementation of ecological monitoring systems,
- development of emergency response capabilities that serve to minimize accident consequences,
- development of technologies for facility decommissioning.

Legal Criteria
The main legal requirements relating to Russian radiochemical plants are the Radiation Safety Standards (RUS. NRB-76/87) and the Basic Rules (Rus. OSP-72/87). Additional Russian legal requirements regarding nuclear safety are contained within the following:

- Environmental Protection Law
- Law of Radiation Safety for Population
- Nuclear Power Law
- Law of State Politics in the Field of Radioactive Waste Management

Radiochemical processes at RT Mayak and RT-2 were implemented based on the dose limits specified in NRB-76/87, which are annual dose limits of 50 mSv for workers and 5 mSv for the public.

The dose received by personnel working in the most hazardous process areas of the RT Mayak facility is within 5 to 10 mSv per average year. The criteria established for RT-2 is that the NRB-76/87 worker dose limits are not to be exceeded by more than 10 percent. The annual effective dose to individuals in the surrounding population from normal operating conditions is expected to be between 0.1 to 0.15 mSv.

Engineering and Technical Approaches

Table 1 provides an overview of radioactive waste strategies implemented and planned for RT Mayak and RT-2 respectively as well as international analogues. Table 2 provides more detailed data on high-level waste (HLW) and intermediate-level waste (ILW) disposal strategies at RT Mayak and RT-2. The strategic directions for improving waste management technology at Russian radiochemical plants are moving toward bringing the plants up to world standards.

Additional examples of technological improvements that have enhanced facility safety include:

- implementation of technologies for shearing and dissolution of spent fuel,
- development and introduction of HLW partitioning flowsheets,
- significant decrease in discharge of low- and intermediate-level wastes into drainage systems at radiochemical plants,
- purification of water contaminated with radionuclides using ion-exchange and selective sorbents,
- pilot-industrial realization of vitrification process in the direct-heating furnace EP-500,
- transportation of spent fuel in containers that meet national and international safety standards.

Table 1. Comparison of Safety Related Technologies Implemented at Radiochemical Facilities

Technological Operation	RT Mayak Facility	RT-2 Facility	World Analogues
HLW partitioning	partial implementation of TUE	design complete	
Solidification of radioactive wastes	implemented	projected	implemented
Capture of Krypton-85		projected	
Concentration of Iodine-129		projected	
Localization of Tritium	under consideration as disposal option	under consideration as disposal option	
Interim storage of solidified radioactive wastes	facilities in service and projected	facilities projected	facilities in service and projected
Disposal of radioactive wastes	Mayak site options are under study	MChA region options are under study	pilot industrial studies

Ecological Approach

The existing overall environmental conditions surrounding facilities has generated criticism of these facilities because of inadequate attention to the development of technologies for reliable isolation of wastes, a lack of adequate environmental radiation monitoring systems, violation of standard permissible discharge limits, and environmental contamination resulting from accidental releases.

The ecological approach involves the evaluation of the impact of the facility on the environment, compliance with the discharge limits, accident mitigative action response planning, the rehabilitation of contaminated sites, and decommissioning.

Table 2. Comparison of Current RT Mayak and Projected RT-2 Waste Management Processes

Process	RT, Mayak	RT-2, MChA
HLW solidification 1/t of spent fuel	Production of phosphate glass in direct-heating furnace EP-500	One- and two-stage processes for production of borosilicate glass and mineral-like compositions
- 1st cycle raffinate	300 - 400 liters/ton	
- concentrate of Sr and Cs		40 liters/ton
- strip RE and TPE		30 liters/ton
- raffinate upon HLW reprocessing		50 liters/ton
ILW solidification	Bituminization process is not implemented. Expected amount 6 m^3/ton	Cementation process is projected, 3 - 4 m^3/ton

Radiological Consequence Analysis Workshop

During the week of October 16, 1995 the KRI hosted a week long radiological consequence analysis workshop that was developed and presented by SNL staff. Twenty Russian specialists from nine Russian organizations and agencies participated in the workshop.

Workshop topics included a general overview of both risk and consequence analysis methods, environmental transport, human exposure pathways and dose assessment methods, health effects modeling, accident response modeling, economic consequence analysis, and an overview of the MACCS program. In addition to lectures, a computer modeling workshop was conducted to provide Russian scientists with practical experience performing consequence analyses and applying the MACCS code. The class was successful in transferring SNL analysis capabilities to Russian scientists and transferring Russian field experience to the SNL staff.

Application of SNL Radiological Consequence Analysis Methods to Russian Nuclear Facilities

MACCS data input files were developed for hypothetical accidents at the planned RT-2 facility and the Leningrad Nuclear Power Plant (LNPP). The input files used in these analyses were based on the sample problem files included with the MACCS software. Only the sample input data relating to source term, site data, skin protection factors and cloud and groundshine shielding factors were modified for these hypothetical accident calculations.

Development of Data for RT-2 Fuel Storage Site

The following accident scenarios were considered for the RT-2 facility:

- process violations
- external threats leading to partial destruction of building, with failure of safety systems, e.g., an airplane crash.
- a spontaneous chain reaction (criticality) from random grouping of fuel containers
- heat-up, release of radionuclides beyond building, and water leakage from basin.

The most stressing scenario was a hypothetical airplane crash, a resultant fire, damage to the building and safety systems, resulting in the development of a fast-terminated criticality (pulse fission reactor). The number of fissions will not be higher than 1.0E18 due to fast heat-up and dynamic scattering of nuclear fuel. The radioactive release was conservatively estimated to not exceed 0.1 MCi.

An airplane crash may be followed by explosions. Because of the low amount of combustible material in the building constructions and the evaporation of water, the development of a high intensity fire that would result in a radioactive cloud with a high elevation was not considered plausible. The altitude of the radioactive cloud was conservatively estimated to be in the range of 50 m. The duration of the release from the failed fuel elements was estimated as 2 hours.

This postulated accident was assumed to involve one ton of ten year old fuel, from which original short-lived radionuclides are absent. It was assumed that the short lived radionuclides generated from the criticality event (10^{18} fissions) would not significantly contribute to the dose estimates. The source term was approximated by assuming release fractions for the radionuclide groups were identical to those included in the sample problem files for NPP calculations. This was a very artificial and conservative assumption due to the relatively low temperature of fuel.

Two primary limitations of applying the MACCS code to the RT-2 calculations were identified as follows:

- The MACCS Gaussian dispersion model is applicable only to dispersion of radioactive gases and aerosols. The main part of the explosion would consist of large fuel rod fragments. Ballistic calculations indicate that these fragments would be dispersed up to 1 km from the release site. It would be helpful if MACCS could accept externally generated ground contamination data for the EARLY and CHRONC calculations.
- The Gaussian dispersion model does not include the effect of wind fields generated by complex terrain, and the RT-2 facility is located in complex terrain.

LNPP Accident Analysis

The LNPP is of interest because of the proximity of St. Petersburg and the agricultural region surrounding the city. In addition, the LNPP is an RBMK reactor, the same as the Chernobyl Unit 4 reactor, and there have been safety incidents at the LNPP.

MACCS input parameters were modified to reflect the specifics of the LNPP site. A population grid for LNPP was developed. Growing season parameters and shielding factors were changed to reflect conditions for NW Russia. A meteorological file was generated. Regional economic data were approximated. Table 3 lists the radionuclide inventory defined for the RBMK reactor and compares the RBMK inventory to the pressurized-water reactor (PWR) inventory listed in MACCS Sample Problem A. The RBMK-1000 inventory was derived from the Handbook on Nuclear Fuel (1984) by D.E. Kolobashkin.

Comparison of Accident Scenarios
The primary difference between the LNPP and the RT-2 accident scenarios was in the source term defined for each scenario. The activity of the power plant source term was much greater than radiochemical plant source term. In addition, all radionuclides with short half-lives were omitted from the fuel storage accident because it was assumed the fuel was 10 years old and the short-lived radionuclides from the criticality event were an insignificant contributor to dose.

Comparison of the analysis results for the LNPP and RT-2 showed, as expected, large differences between the estimated consequences. The main contribution to dose during the acute phase for the LNPP scenario was from cloudshine and ingestion of radioactive iodine. The contribution of cesium isotopes for the first week was negligible. The RT-2 scenario showed no significant acute effects beyond 1 km. The dose impact of short-lived radionuclei produced by the criticality event was negligible in comparison with the release of ^{137}Cs, ^{90}Sr, and transuranics.

Conclusions
Collaborative efforts between the KRI and SNL relating to radiological consequence analysis efforts have advanced the knowledge, skills, and experience of both Russian and U.S. specialists. Insights have been gained into the potential consequences of accidents at radiochemical facilities. Limitations of the MACCS code for radiochemical plant consequence analyses have been identified.

The potential benefits for additional collaborative efforts are significant. The Russians have practical field experience gained from the accidents at the Chernobyl Unit 4 reactor and the 1957 Kystym plutonium separation plant accident. U.S. specialists have developed extensive experience with probabilistic risk assessment methods, radiological consequence assessment, and strategies for evaluating the safety basis of facilities. Future collaborative efforts to improve the MACCS energetic release modeling capabilities and to compare MACCS output to actual field data are currently in the planning stage.

Table 3. Comparison of Inventories defined for the KRI LLNP Sample Problem and the MACCS Sample Problem A.

Radio-nuclide Inventory	RBMK Activity (Bq)	PWR Activity (Bq)	Ratio PWR/RBMK Activities	Radio-nuclide Inventory	RBMK Activity (Bq)	PWR Activity (Bq)	Ratio PWR/RBMK Activities
CO-58	3.20E+16	3.22E+16	1.0072	I-134	4.98E+18	7.44E+18	1.4940
CO-60	2.40E+16	2.47E+16	1.0271	I-135	5.54E+18	6.39E+18	1.1538
KR-85	6.59E+16	2.48E+16	0.3756	XE-133	6.62E+18	6.78E+18	1.0245
KR-85M	5.18E+17	1.16E+18	2.2375	XE-135	1.72E+18	1.27E+18	0.7401
KR-87	7.72E+17	2.12E+18	2.7435	CS-134	8.06E+17	4.32E+17	0.5365
KR-88	1.42E+18	2.86E+18	2.0169	CS-136	1.43E+17	1.32E+17	0.9203
RB-86	3.88E+15	1.89E+15	0.4866	CS-137	5.78E+17	2.42E+17	0.4182
SR-89	2.50E+18	3.59E+18	1.4360	BA-139	3.86E+18	6.28E+18	1.6275
SR-90	4.00E+17	1.94E+17	0.4845	BA-140	5.68E+18	6.22E+18	1.0944
SR-91	2.94E+17	4.62E+18	15.7007	LA-140	3.84E+18	6.35E+18	1.6542
SR-92	2.72E+18	4.80E+18	1.7658	LA-141	4.62E+18	5.83E+18	1.2610
Y-90	4.14E+17	2.08E+17	0.5022	LA-142	3.58E+18	5.62E+18	1.5687
Y-91	3.44E+18	4.37E+18	1.2715	CE-141	5.28E+18	5.65E+18	1.0703
Y-92	3.46E+18	4.82E+18	1.3934	CE-143	4.60E+18	5.49E+18	1.1943
Y-93	4.04E+18	5.45E+18	1.3500	CE-144	5.06E+18	3.41E+18	0.6729
ZR-95	5.26E+18	5.53E+18	1.0506	PR-143	4.70E+18	5.40E+18	1.1479
ZR-97	4.92E+18	5.76E+18	1.1705	ND-147	2.08E+18	2.41E+18	1.1596
NB-95	5.48E+18	5.22E+18	0.9533	NP-239	1.00E+20	6.46E+19	0.6464
MO-99	5.86E+18	6.10E+18	1.0406	PU-238	8.60E+15	3.66E+15	0.4260
TC-99M	5.18E+18	5.26E+18	1.0160	PU-239	1.20E+15	8.26E+14	0.6886
RU-103	5.68E+18	4.54E+18	0.7996	PU-240	3.66E+15	1.04E+15	0.2847
RU-105	3.70E+18	2.95E+18	0.7984	PU-241	5.34E+17	1.76E+17	0.3287
RU-106	2.96E+18	1.03E+18	0.3486	AM-241	9.00E+16	1.16E+14	0.0013
RH-105	3.84E+18	2.05E+18	0.5328	CM-242	1.28E+17	4.44E+16	0.3466
SB-127	3.22E+17	2.79E+17	0.8655	CM-244	4.88E+15	2.60E+15	0.5320
SB-129	8.62E+17	9.87E+17	1.1452				
TE-127	3.24E+17	2.69E+17	0.8309				
TE-127M	5.94E+16	3.56E+16	0.6000				
TE-129	1.03E+18	9.27E+17	0.8997				
TE-129M	2.10E+17	2.44E+17	1.1633				
TE-131M	7.20E+17	4.68E+17	0.6500				
TE-132	4.62E+18	4.66E+18	1.0082				
I-131	3.34E+18	3.21E+18	0.9599				
I-132	4.72E+18	4.73E+18	1.0011				
I-133	6.48E+18	6.78E+18	1.0461				

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