Shielding Aspects of Accelerators, Targets and Irradiation Facilities – SATIF-14

Proceedings of the Fourteenth Workshop
30 October–2 November 2018, Gyeongju, Korea
Shielding Aspects of Accelerators, Targets and Irradiation Facilities – SATIF-14

Proceedings of the Fourteenth Workshop, 30 October-2 November 2018, Gyeongju, Korea

Please note that this document is available in PDF format only.
ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

The OECD is a unique forum where the governments of 38 democracies work together to address the economic, social and environmental challenges of globalisation. The OECD is also at the forefront of efforts to understand and to help governments respond to new developments and concerns, such as corporate governance, the information economy and the challenges of an ageing population. The Organisation provides a setting where governments can compare policy experiences, seek answers to common problems, identify good practice and work to co-ordinate domestic and international policies.

The OECD member countries are: Australia, Austria, Belgium, Canada, Chile, Colombia, Costa Rica, the Czech Republic, Denmark, Estonia, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Israel, Italy, Japan, Korea, Latvia, Lithuania, Luxembourg, Mexico, the Netherlands, New Zealand, Norway, Poland, Portugal, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission takes part in the work of the OECD.

OECD Publishing disseminates widely the results of the Organisation’s statistics gathering and research on economic, social and environmental issues, as well as the conventions, guidelines and standards agreed by its members.

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1 February 1958. Current NEA membership consists of 34 countries: Argentina, Australia, Austria, Belgium, Bulgaria, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Korea, Luxembourg, Mexico, the Netherlands, Norway, Poland, Portugal, Romania, Russia, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission and the International Atomic Energy Agency also take part in the work of the Agency.

The mission of the NEA is:
– to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally sound and economical use of nuclear energy for peaceful purposes;
– to provide authoritative assessments and to forge common understandings on key issues as input to government decisions on nuclear energy policy and to broader OECD analyses in areas such as energy and the sustainable development of low-carbon economies.

Specific areas of competence of the NEA include the safety and regulation of nuclear activities, radioactive waste management and decommissioning, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.
**Foreword**

The transport of radiation through shielding materials is a major consideration in the safety design studies of nuclear power plants, and the modelling techniques used may be applied to many other types of scientific and technological facilities. Accelerator and irradiation facilities represent a key capability in R&D, medical and industrial infrastructures and can be used in a wide range of scientific, medical and industrial applications. Intermediate-energy ion accelerators, for example, are now used not only in fundamental and applied research, but also for therapy as part of cancer treatment.

While the energy of the incident particles on the shielding of these facilities may be much higher than that in nuclear power plants, much of the physics associated with the behaviour of the secondary particles produced is similar, as are the computer modelling techniques used to quantify key safety design parameters, such as radiation dose and activation levels. Clear synergies exist, therefore, with other technical work being carried out by the Nuclear Energy Agency (NEA), and its Nuclear Science Committee (NSC) continues to sponsor activities in this domain.

One of these activities concerns “Shielding Aspects of Accelerators, Targets and Irradiation Facilities” (SATIF). A series of workshops have been held over the last 25 years: SATIF-1 was held on 28-29 April 1994 in Arlington, Texas; SATIF-2 on 12-13 October 1995 at CERN in Geneva, Switzerland; SATIF-3 on 12-13 May 1997 at Tohoku University in Sendai, Japan; SATIF-4 on 17-18 September 1998 in Knoxville, Tennessee; SATIF-5 on 17-21 July 2000 at the NEA in Paris, France; SATIF-6 on 10-12 April 2002 at the SLAC National Accelerator Laboratory, Menlo Park, California; SATIF-7 on 17-18 May 2004 at ITN, Sacavem, Portugal; SATIF-8 on 22-24 May 2006 at the Pohang Accelerator Laboratory in Pohang, Korea; SATIF-9 on 21-23 April 2008 at Oak Ridge National Laboratory (ORNL), Oak Ridge, Tennessee; SATIF-10 on 2-4 June 2010 at CERN in Geneva, Switzerland; SATIF-11 on 11-13 September 2012 at the High-energy Accelerator Research Organisation (KEK) in Tsukuba, Japan; SATIF-12 on 28-30 April 2014 at Fermi National Accelerator Laboratory (FNAL) in Batavia, Illinois; SATIF-13 on 10-12 October 2016 at Helmholtz-Zentrum Dresden-Rossendorf (HZDR) in Dresden, Germany.

The 14th workshop on Shielding Aspects of Accelerators, Targets and Irradiation Facilities (SATIF-14) took place in Gyeongju, Korea and was jointly organised by the Korea Multi-purpose Accelerator Complex (KOMAC), Pohang Accelerator Laboratory (PAL) and the Expert Group on Radiation Transport and Shielding (EGRTS) of the Working Party on Scientific Issues of Reactor Systems (WPRS) of the NEA.

The workshop was sponsored by KOMAC and PAL and co-sponsored by the NEA and its NSC. The current proceedings provide a summary of the discussions, decisions and conclusions as well as the text of the presentations made at the fourteenth workshop.
Acknowledgements

The following members of the SATIF-14 Scientific Committee and the Local Organising Committee are thanked for their contribution to shaping the technical programme and organising the workshop: S. Ban (KEK), F. Cerutti (CERN), A. Ferrari (HZDR), R. Grove (ORNL), J. Gulliford (NEA), H. Hirayama (KEK), G. Hughes (LANL), B. Kirk (honorary), H.-S. Lee (the general co-chairman of SATIF-14, PAL), N. Mokhov (FNAL), G. Muhrer (ESS), T. Nakamura (honorary), H. Nakashima (JAEA), S. Roesler (CERN), S. Rokni (SLAC), M. Silari (CERN), S. Tsuda (NEA), T. Valentine (ORNL), P. Vaz (IST), N.S. Jung (PAL), M.H. Kim (PAL), Y.S. Min (the general co-chairman of SATIF-14, KOMAC), S.G. Park (KOMAC).

Thanks are also extended to all participants who contributed valuable work and ideas to these proceedings.
Our friend and colleague Johannes “Hannes” Ranft passed away in 2018. Johannes was born in Saxony, Germany in 1933. After he earned his Ph.D in 1962 at Leipzig University, he visited at CERN and produced fundamental studies of a Monte Carlo code FLUKA: FLU(ktuierende) KA(skade), developing and improving it for many applications. He was a professor of theoretical physics at Karl-Marx-University Leipzig and worked with many colleagues at CERN, Rutherford Laboratory, INFN, etc. He was an outstanding physicist, with deepest knowledge and numerous contributions to the particle physics theory, playing the role of a “founding father” of the Monte Carlo approach in Europe, particularly in modelling extranuclear cascades in a matter with his FLUKA-family simulation code. He has made a great impact in the accelerator shielding field. He was an important participant of several SARE meetings happening in parallel with SATIF meetings, bringing invaluable contributions to the SATIF coverage domain. Hannes’s scientific accomplishments, personality and human qualities will be sorely missed.
# Table of contents

List of abbreviations and acronyms.................................................................................. 12
Executive summary .................................................................................................................. 16
Session I: Source terms and related topics ........................................................................... 19
Measurement of the energy spectrum of spallation neutron emitted to the most backward angle of 180° produced from thick mercury target irradiated with 3 GeV protons ............... 20
Dark current and radiation of KEK-Linac accelerating structure protons ......................... 21
Preliminary results of re-analysis of photoneutron experiments performed at PAL using MC codes........................................................................................................................................... 22
Monte Carlo simulations for the quantification of beta skin dose ......................................... 23
  Introduction......................................................................................................................... 23
  Description of the FLUKA simulations ............................................................................ 25
  Comparison of the results ................................................................................................. 28
  Conclusions....................................................................................................................... 31
  References......................................................................................................................... 31
Radiation gradient evaluation at the CERN CHARM mixed-field facility using RPL, RadMON and FLUKA simulation....................................................................................................... 32
Evaluation of future employment of boron carbide shielding at CERN High Energy AcceleRator Mixed Field (CHARM) Facility ........................................................................................................ 33
Measurement of neutron energy spectra after penetrating concrete shield at CERN/CSBF with 24 GeV/c protons on thick copper target .................................................................................... 34
  Introduction......................................................................................................................... 34
  Experiment......................................................................................................................... 35
  Data analysis ................................................................................................................... 36
  Monte Carlo simulation ................................................................................................. 38
  Results and discussions................................................................................................. 38
  Summary......................................................................................................................... 40
  Acknowledgements....................................................................................................... 41
  References....................................................................................................................... 41
Session II: Shielding and dosimetry ....................................................................................... 43
Shielding and skyshine of a 10 MeV electron accelerator enclosed in a large hall ............ 44
  Introduction......................................................................................................................... 44
  Operational parameters and shielding requirements....................................................... 46
  Initial shielding evaluation............................................................................................. 46
  Monte Carlo shielding and skyshine calculations......................................................... 47
  Conclusions.................................................................................................................... 54
  Acknowledgements...................................................................................................... 54
  References..................................................................................................................... 55
Session III: Beam-plasma and laser-plasma interactions and acceleration .......................... 120
Radiological protection studies on high intensity laser facilities........................................ 121
Radiation protection and shielding at current and planned laser facilities at SLAC .......... 122
  Introduction.................................................................................................................. 122
  Characterisation of hot electron source term with EPOCH ........................................ 123
  Calculation of bremsstrahlung dose yield with FLUKA ........................................... 126
  Upgrade of MEC to petawatt-class laser facility ....................................................... 130
  Other source terms from laser-matter experiments .................................................... 132
  Summary .................................................................................................................... 134
  Acknowledgements...................................................................................................... 134
  References.................................................................................................................. 134

A novel active detector for the Bremsstrahlung source term measurement in ultra-intense laser-plasma experiments ......................................................................................... 136

Session IV: Code status and medical and industrial accelerators .................................... 137
New analytical method to estimate systematic uncertainty in PHITS................................. 138
  References.................................................................................................................. 139

Stray neutron measurements in scanning proton and carbon ion therapy.......................... 140
  Acknowledgements...................................................................................................... 141

Radiological estimation and validation for the accelerator-based boron neutron capture therapy facility at the Ibaraki Neutron Medical Research Center ...................................... 142
  Introduction.................................................................................................................. 142
  Radiological estimation and design ............................................................................ 143
  Calculated neutron beam............................................................................................. 147
  Experimental validation .............................................................................................. 149
  Summary .................................................................................................................... 150
  Acknowledgements...................................................................................................... 150
  References.................................................................................................................. 150

Activation assessment of self-shielded and non-self-shielded PET cyclotrons.................... 152
  Introduction.................................................................................................................. 152
  Method ....................................................................................................................... 153
  Result and discussion ................................................................................................. 155
  Conclusion .................................................................................................................. 158
  Acknowledgements...................................................................................................... 159
  References.................................................................................................................. 159

Session V: Induced radioactivity ......................................................................................... 160
Activation zoning of the experimental areas of the CERN antiproton decelerator using ActiWiz 3 ...................................................................................................................................... 161
  Introduction.................................................................................................................. 161
  Activation studies ........................................................................................................ 162
  Activation of compound materials not directly exposed to the beam.......................... 163
  Summary and conclusions ......................................................................................... 165
  References.................................................................................................................. 165
### Experience with inner reflector plug exchange in SNS

<table>
<thead>
<tr>
<th>Section</th>
<th>Page</th>
</tr>
</thead>
<tbody>
<tr>
<td>Introduction</td>
<td>167</td>
</tr>
<tr>
<td>Methods and codes</td>
<td>167</td>
</tr>
<tr>
<td>Geometry</td>
<td>168</td>
</tr>
<tr>
<td>Results</td>
<td>169</td>
</tr>
<tr>
<td>Conclusions</td>
<td>173</td>
</tr>
<tr>
<td>Acknowledgements</td>
<td>179</td>
</tr>
<tr>
<td>References</td>
<td>179</td>
</tr>
</tbody>
</table>

### A Beam Dump Facility (BDF) at CERN: The concept and a first radiological assessment

<table>
<thead>
<tr>
<th>Section</th>
<th>Page</th>
</tr>
</thead>
<tbody>
<tr>
<td>Introduction</td>
<td>180</td>
</tr>
<tr>
<td>Radiological protection evaluation</td>
<td>182</td>
</tr>
<tr>
<td>BDF target prototype test</td>
<td>188</td>
</tr>
<tr>
<td>Summary and conclusions</td>
<td>189</td>
</tr>
<tr>
<td>Acknowledgements</td>
<td>189</td>
</tr>
<tr>
<td>References</td>
<td>189</td>
</tr>
</tbody>
</table>

### Standardisation of concrete composite for radiation shielding

<table>
<thead>
<tr>
<th>Section</th>
<th>Page</th>
</tr>
</thead>
<tbody>
<tr>
<td>Evaluation</td>
<td>191</td>
</tr>
</tbody>
</table>

### Evaluation of the effect of the airborne natural radioactivity near KOMAC

<table>
<thead>
<tr>
<th>Section</th>
<th>Page</th>
</tr>
</thead>
<tbody>
<tr>
<td>Session VI: Code benchmarking, and intercomparison</td>
<td>193</td>
</tr>
</tbody>
</table>

### Improvement of a high-energy fission model for spallation reactions

<table>
<thead>
<tr>
<th>Section</th>
<th>Page</th>
</tr>
</thead>
<tbody>
<tr>
<td>Introduction</td>
<td>194</td>
</tr>
<tr>
<td>The modified models</td>
<td>195</td>
</tr>
<tr>
<td>Results and discussions</td>
<td>195</td>
</tr>
<tr>
<td>Conclusion</td>
<td>199</td>
</tr>
<tr>
<td>Acknowledgements</td>
<td>199</td>
</tr>
<tr>
<td>References</td>
<td>199</td>
</tr>
</tbody>
</table>

### High-Energy Intra-Nuclear Cascade Liège-based Residual (HEIR) nuclear data for activation-transmutation studies

<table>
<thead>
<tr>
<th>Section</th>
<th>Page</th>
</tr>
</thead>
<tbody>
<tr>
<td>MARS15 code developments and applications to the LBNF/DUNE project</td>
<td>200</td>
</tr>
<tr>
<td>Introduction</td>
<td>201</td>
</tr>
<tr>
<td>MARS15 main features</td>
<td>201</td>
</tr>
<tr>
<td>Nuclear event generator</td>
<td>202</td>
</tr>
<tr>
<td>Three-body decays</td>
<td>202</td>
</tr>
<tr>
<td>Atomic displacements (DPA)</td>
<td>205</td>
</tr>
<tr>
<td>Geometry description and particle tracking</td>
<td>205</td>
</tr>
<tr>
<td>Platform, compilers and MPI</td>
<td>207</td>
</tr>
<tr>
<td>Recent benchmarking</td>
<td>208</td>
</tr>
<tr>
<td>LBNF/DUNE</td>
<td>209</td>
</tr>
<tr>
<td>Acknowledgements</td>
<td>210</td>
</tr>
<tr>
<td>References</td>
<td>215</td>
</tr>
</tbody>
</table>
MARS15-MADX integration and its application for design of the Fermilab Booster collimation system ......................................................... 217
  Introduction ............................................................................. 217
  MARS STREG library ............................................................ 218
  MARS–MAD-X coupling ....................................................... 222
  New Booster collimation unit optimisation studies .................. 225
  Conclusions ......................................................................... 232
  Acknowledgements ............................................................... 232
  References ........................................................................... 232

Intercomparison of particle production .......................................... 234
  Introduction ......................................................................... 234
  Problems for an intercomparison at SATIF-14 ......................... 234
  Summary of contributors ..................................................... 235
  Results and discussions ....................................................... 235
  Summary ............................................................................. 248
  Future themes ..................................................................... 248
  References ......................................................................... 248

Measurements and FLUKA simulations of bismuth, aluminium, indium and carbon activation at the upgraded CERN Shielding Benchmark Facility (CSBF) ...................... 250
  Introduction ......................................................................... 251
  Design of the CERN Shielding Benchmark Facility (CSBF) ....... 252
  Beam parameters and configurations .................................... 255
  Activation samples and their irradiation ................................. 258
  Comparison of FLUKA simulation results to measured production yields .................................................. 258
  Summary ............................................................................. 265
  Acknowledgements ............................................................... 265
  References ......................................................................... 265

Monte Carlo simulations of ISIS Target Station 1 using FLUKA code and comparison with the MCNPX reference model ................................................................. 267
  Introduction ......................................................................... 267
  FLUKA model of the upgraded target, reflector and moderator for ISIS TS1 .................................................. 268
  Results ................................................................................ 270
  Conclusions ....................................................................... 281
  Acknowledgements ............................................................... 281
  References ......................................................................... 281

Benchmarking tests and models for solid state diamond detectors at LCLS-II ................................................................. 283

Poster session ........................................................................ 284

Neutron source evaluation for the Neutron Data Production System (NDPS) at RAON ................................................................. 285

Shielding and radiological protection for a compact inverse Compton backscattering X-rays source, ThomX ................................................................. 286
  Introduction ......................................................................... 286
  Design and optimisation of the shielding ......................... 289
  Radiological protection for workers .................................. 291
  Protection of the environment ........................................... 293
Status and next steps ................................................................. 296
Acknowledgements .................................................................. 296
References .............................................................................. 296

Shielding design of a dedicated cabin for the conditioning of a second RF cavity prior to be installed in the BOOSTER ring tunnel of the Synchrotron SOLEIL ........................................... 298

Shielding and activation calculations for the MYRRHA Accelerator-Driven System design... 299

Thermal neutron profiles inside J-PARC main ring tunnel ..................................................... 301

Optimum design of neutron moderator assembly for accelerator-based boron neutron capture therapy using MCNPX code ................................................................. 302

Introduction ............................................................................ 302
Moderator optimisation .............................................................. 303
Results .................................................................................. 305
References ............................................................................. 307

Composition analysis of ordinary concrete to estimate residual isotopes in the decommissioning of particle accelerator ................................................................. 308

Introduction ............................................................................ 308
Process of composition analysis .................................................. 309
Result – EA, XRF, ICP-MS .......................................................... 312
Summary and conclusions ......................................................... 313
Acknowledgements .................................................................. 313
References ............................................................................. 313

A development of calculation module to analyse high-energy reactions in the AR2S code .... 315

Benchmarking experiments of displacement cross-sections for high-energy proton irradiation with cryogenic-sample ................................................................. 316

Introduction ............................................................................ 316
Experiments ............................................................................ 317
Experimental results ................................................................. 320
Summary ................................................................................. 322
Acknowledgements .................................................................. 322
References ............................................................................. 323

ELI Beamlines laser facility - Current status ......................................................... 324

Introduction ............................................................................ 324
Lasers and particles beamlines .................................................. 324
Radiological protection assessment ............................................ 325
Conclusions .............................................................................. 329
Acknowledgements .................................................................. 329
References ............................................................................. 329
### List of abbreviations and acronyms

<table>
<thead>
<tr>
<th>Abbreviation</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>ABNCT</td>
<td>Accelerator-based Boron Neutron Capture Therapy</td>
</tr>
<tr>
<td>ACMI</td>
<td>Accumulated Charge Monitor Interlock</td>
</tr>
<tr>
<td>AD</td>
<td>Antiproton Decelerator</td>
</tr>
<tr>
<td>ADS</td>
<td>Accelerator-Driven System</td>
</tr>
<tr>
<td>ADVANTG</td>
<td>Automated Variance Reduction Parameter Generator</td>
</tr>
<tr>
<td>AESJ</td>
<td>Atomic Energy Society of Japan</td>
</tr>
<tr>
<td>aka JLab</td>
<td>Thomas Jefferson National Accelerator Facility</td>
</tr>
<tr>
<td>ALARA</td>
<td>As Low As Reasonably Achievable</td>
</tr>
<tr>
<td>ANOVA</td>
<td>ANalysis Of VAriance</td>
</tr>
<tr>
<td>arc-dpa</td>
<td>athermal recombination-corrected displacement damage</td>
</tr>
<tr>
<td>ARIEL</td>
<td>Advanced Rare Isotope Laboratory</td>
</tr>
<tr>
<td>ARM</td>
<td>Area Radiation Monitor</td>
</tr>
<tr>
<td>BDF</td>
<td>Beam Dump Facility</td>
</tr>
<tr>
<td>bERLinPRO</td>
<td>Berlin Energy Recovery Linac</td>
</tr>
<tr>
<td>BNCT</td>
<td>Boron Neutron Capture Therapy</td>
</tr>
<tr>
<td>BNL</td>
<td>Brookhaven National Laboratory</td>
</tr>
<tr>
<td>BPE</td>
<td>Borated PE</td>
</tr>
<tr>
<td>BRS</td>
<td>Bremsstrahlung Stop</td>
</tr>
<tr>
<td>BSA</td>
<td>Beam shaping assembly</td>
</tr>
<tr>
<td>BST</td>
<td>Bremsstrahlung Stop</td>
</tr>
<tr>
<td>CERF</td>
<td>CERN-EU High Energy Reference Field facility</td>
</tr>
<tr>
<td>CERN</td>
<td>European Organization for Nuclear Research</td>
</tr>
<tr>
<td>CHARM</td>
<td>CERN High Energy AcceleRator Mixed Field</td>
</tr>
<tr>
<td>CH-cavities</td>
<td>Cross-bar H-type cavities</td>
</tr>
<tr>
<td>CSBF</td>
<td>CERN Shielding Benchmark Facility</td>
</tr>
<tr>
<td>CSDA</td>
<td>Continuous-Slowing-Down Approximation</td>
</tr>
<tr>
<td>CV</td>
<td>Core Vessel</td>
</tr>
<tr>
<td>DB</td>
<td>Databank</td>
</tr>
<tr>
<td>DCM in SST-2</td>
<td>Double Crystal Monochromator</td>
</tr>
<tr>
<td>DOE</td>
<td>US Department of Energy</td>
</tr>
<tr>
<td>DPA</td>
<td>Displacements per atom</td>
</tr>
<tr>
<td>DTL</td>
<td>Drift-tube linear accelerator (linac)</td>
</tr>
<tr>
<td>DUNE</td>
<td>Deep Underground Neutrino Experiment</td>
</tr>
<tr>
<td>DXTRAN</td>
<td>Deterministic transport method for variance reduction</td>
</tr>
<tr>
<td>EA</td>
<td>Elemental Analysis</td>
</tr>
<tr>
<td>ECR</td>
<td>Electron Cyclotron Resonance</td>
</tr>
<tr>
<td>ECX5</td>
<td>Beam dump located in the Super Proton Synchrotron</td>
</tr>
<tr>
<td>EES</td>
<td>Experimental end stations</td>
</tr>
<tr>
<td>EIC</td>
<td>Early injector commissioning</td>
</tr>
<tr>
<td>ELI</td>
<td>Extreme Light Infrastructure</td>
</tr>
<tr>
<td>ELIMAIA</td>
<td>ELI Multidisciplinary Applications of laser-Ion Acceleration</td>
</tr>
<tr>
<td>EPOCH</td>
<td>Relativistic particle-in-cell code for modelling laser-plasma interaction</td>
</tr>
<tr>
<td>EPU60</td>
<td>Elliptically Polarized Undulator</td>
</tr>
<tr>
<td>ERBSS</td>
<td>Extended range Bonner sphere spectrometer</td>
</tr>
<tr>
<td>ESPP</td>
<td>European Strategy for Particle Physics</td>
</tr>
<tr>
<td>ESS</td>
<td>Experimental end stations</td>
</tr>
<tr>
<td>Abbreviation</td>
<td>Full Form</td>
</tr>
<tr>
<td>--------------</td>
<td>-----------</td>
</tr>
<tr>
<td>FDG</td>
<td>F-18-tagged fluorodeoxyglucose</td>
</tr>
<tr>
<td>FE</td>
<td>Front End</td>
</tr>
<tr>
<td>Fermilab</td>
<td>Fermi National Accelerator Laboratory</td>
</tr>
<tr>
<td>FFAG</td>
<td>Fixed-Field Alternating Gradient</td>
</tr>
<tr>
<td>FM</td>
<td>Fixed mask</td>
</tr>
<tr>
<td>FOE</td>
<td>First Optical Enclosure</td>
</tr>
<tr>
<td>FRIB</td>
<td>Facility for Rare Isotope Beams</td>
</tr>
<tr>
<td>FWHM</td>
<td>Full Width at Half Maximum</td>
</tr>
<tr>
<td>GB</td>
<td>Gas bremsstrahlung</td>
</tr>
<tr>
<td>GEM</td>
<td>Generalized Evaporation Model</td>
</tr>
<tr>
<td>GM</td>
<td>Gifford-McMahon</td>
</tr>
<tr>
<td>GUI</td>
<td>Graphical-User Interface</td>
</tr>
<tr>
<td>GULL</td>
<td>Guillotine</td>
</tr>
<tr>
<td>HANARO</td>
<td>The High-Flux Advanced Neutron Application Reactor</td>
</tr>
<tr>
<td>HELL</td>
<td>High energy ELectron acceleration by Laser</td>
</tr>
<tr>
<td>HEX</td>
<td>High Energy Engineering X-ray</td>
</tr>
<tr>
<td>HZDR</td>
<td>Helmholtz-Zentrum Dresden-Rossendorf</td>
</tr>
<tr>
<td>IAEA</td>
<td>International Atomic Energy Agency</td>
</tr>
<tr>
<td>iBNCT</td>
<td>Ibaraki boron neutron capture therapy</td>
</tr>
<tr>
<td>ICP-MS</td>
<td>Inductively Coupled Plasma Mass Spectrometry</td>
</tr>
<tr>
<td>ICRP</td>
<td>International Commission on Radiological Protection</td>
</tr>
<tr>
<td>ICRU</td>
<td>International Commission on Radiation Units and Measurements</td>
</tr>
<tr>
<td>INCL</td>
<td>Intranuclear-cascade model</td>
</tr>
<tr>
<td>IRP</td>
<td>Inner Reflector Plug</td>
</tr>
<tr>
<td>IRR</td>
<td>Instrument Readiness Review</td>
</tr>
<tr>
<td>IS</td>
<td>Ion source</td>
</tr>
<tr>
<td>JAEA</td>
<td>Japan Atomic Energy Agency</td>
</tr>
<tr>
<td>JAM</td>
<td>Jet AA Microscopic Transportation Model</td>
</tr>
<tr>
<td>JENDL</td>
<td>Japanese Evaluated Nuclear Data Library</td>
</tr>
<tr>
<td>J-PARC</td>
<td>Japan Proton Accelerator Research Complex</td>
</tr>
<tr>
<td>JRR-4</td>
<td>Japan Research Reactor No. 4</td>
</tr>
<tr>
<td>JSBS</td>
<td>Japan Society for the Promotion of Science</td>
</tr>
<tr>
<td>KAERI</td>
<td>Korea Atomic Energy Research Institute</td>
</tr>
<tr>
<td>KEK</td>
<td>High Energy Accelerator Research Organization</td>
</tr>
<tr>
<td>KOFONS</td>
<td>Korea Foundation Of Nuclear Safety</td>
</tr>
<tr>
<td>KOMAC</td>
<td>KOREA Multi-purpose Accelerator Complex</td>
</tr>
<tr>
<td>KURRI</td>
<td>Kyoto University Research Reactor Institute</td>
</tr>
<tr>
<td>LBNF</td>
<td>Long-Baseline Neutrino Facility</td>
</tr>
<tr>
<td>LCLS</td>
<td>Linac Coherent Light Source</td>
</tr>
<tr>
<td>LCLS-II</td>
<td>Linac Coherent Light Source II</td>
</tr>
<tr>
<td>LCO</td>
<td>Lead collimators</td>
</tr>
<tr>
<td>LEBT</td>
<td>Low energy beam transport</td>
</tr>
<tr>
<td>LEP</td>
<td>The Large Electron-Positron Collider</td>
</tr>
<tr>
<td>LET</td>
<td>Linear energy transfer</td>
</tr>
<tr>
<td>LHC</td>
<td>Large Hadron Collider</td>
</tr>
<tr>
<td>LIMS</td>
<td>Liquid Metal Ion Source</td>
</tr>
<tr>
<td>LINAC</td>
<td>Linear accelerator</td>
</tr>
<tr>
<td>LN</td>
<td>Low-energy neutrons interactions</td>
</tr>
<tr>
<td>LSS</td>
<td>Long straight section</td>
</tr>
<tr>
<td>Abbreviation</td>
<td>Description</td>
</tr>
<tr>
<td>--------------</td>
<td>-------------</td>
</tr>
<tr>
<td>LUIS</td>
<td>Laser Undulator Illuminating Source</td>
</tr>
<tr>
<td>MCAM</td>
<td>the CAD/Image-based Automatic Modeling Program for Neutronics and Radiation Transport</td>
</tr>
<tr>
<td>MD</td>
<td>Molecular Dynamics simulation method</td>
</tr>
<tr>
<td>MEC</td>
<td>Matter in Extreme Conditions</td>
</tr>
<tr>
<td>MLF</td>
<td>Materials and Life Science Experimental Facility</td>
</tr>
<tr>
<td>MMBLB</td>
<td>MARS-MAD Beamline Builder</td>
</tr>
<tr>
<td>MMBLB</td>
<td>MAD-MARS Beam Line Builder</td>
</tr>
<tr>
<td>MMRBB</td>
<td>MAD-X-MARS15-ROOT Beamline Builder</td>
</tr>
<tr>
<td>MR</td>
<td>Main Ring</td>
</tr>
<tr>
<td>MSU</td>
<td>Michigan state University</td>
</tr>
<tr>
<td>MYRRHA</td>
<td>Multi-purpose hYbrid Research Reactor for High-tech Application</td>
</tr>
<tr>
<td>NAA</td>
<td>Neutron Activation Analysis</td>
</tr>
<tr>
<td>NDPS</td>
<td>Neutron Data Production System</td>
</tr>
<tr>
<td>NEA</td>
<td>Nuclear Energy Agency</td>
</tr>
<tr>
<td>NIPS</td>
<td>NanoParticle Ion Source</td>
</tr>
<tr>
<td>NIST</td>
<td>National Institute of Standards and Technology</td>
</tr>
<tr>
<td>NSLS-II</td>
<td>National Synchrotron Light Source II facility</td>
</tr>
<tr>
<td>NSC</td>
<td>Nuclear Science Committee (NEA)</td>
</tr>
<tr>
<td>NSSC</td>
<td>Nuclear Safety and Security Commission</td>
</tr>
<tr>
<td>NYSERDA</td>
<td>New York State Energy Research and Development Association</td>
</tr>
<tr>
<td>OECD</td>
<td>Organisation for Economic Co-operation and Development</td>
</tr>
<tr>
<td>OFHC</td>
<td>Oxygen Free High Conductivity</td>
</tr>
<tr>
<td>ORP</td>
<td>Outer reflector plug</td>
</tr>
<tr>
<td>P3</td>
<td>Plasma Physics Platform</td>
</tr>
<tr>
<td>PAL-XFEL</td>
<td>Pohang Accelerator Laboratory Free Electron Laser</td>
</tr>
<tr>
<td>PBS</td>
<td>pencil-beam-scanning</td>
</tr>
<tr>
<td>PBT</td>
<td>pink beam transport</td>
</tr>
<tr>
<td>PE</td>
<td>polyethylene</td>
</tr>
<tr>
<td>PET</td>
<td>Positron-emission tomography</td>
</tr>
<tr>
<td>PGM</td>
<td>Plane Grating Monochromator</td>
</tr>
<tr>
<td>PHITS</td>
<td>Particle and Heavy Ion Transport code System</td>
</tr>
<tr>
<td>PIC</td>
<td>Particle-in-cell</td>
</tr>
<tr>
<td>PS</td>
<td>Proton Synchrotron</td>
</tr>
<tr>
<td>pva</td>
<td>Primary collimators above</td>
</tr>
<tr>
<td>pvb</td>
<td>Primary collimators below</td>
</tr>
<tr>
<td>PXS</td>
<td>Plasma X-ray Source</td>
</tr>
<tr>
<td>QDC</td>
<td>Quantum Dot Channel</td>
</tr>
<tr>
<td>RAON</td>
<td>Institute for Basic Science in Daejeon</td>
</tr>
<tr>
<td>RBE</td>
<td>Relative biological effectiveness</td>
</tr>
<tr>
<td>RCA</td>
<td>Radiation Control Area</td>
</tr>
<tr>
<td>RCNP</td>
<td>Research Center for Nuclear Physics</td>
</tr>
<tr>
<td>RCS</td>
<td>Rapid cycling synchrotron</td>
</tr>
<tr>
<td>RFQ</td>
<td>Radio Frequency Quadrupole</td>
</tr>
<tr>
<td>RMS</td>
<td>Radiation monitoring system</td>
</tr>
<tr>
<td>RP</td>
<td>Radiation Protection</td>
</tr>
<tr>
<td>RSICC</td>
<td>Radiation Safety Information Computational Center</td>
</tr>
<tr>
<td>S10C</td>
<td>Carbon steel</td>
</tr>
<tr>
<td>SATIF</td>
<td>Shielding Aspects of Accelerators, Target and Irradiation Facilities</td>
</tr>
<tr>
<td>SBS2</td>
<td>Secondary bremsstrahlung shield</td>
</tr>
</tbody>
</table>
SCW70        Superconducting Wiggler  
SHiP        Search for Hidden Particles  
SIMS        Secondary Ion Mass Spectrometer  
SLAC        National Accelerator Laboratory (US DOE)  
SLT         Slits  
SNS         Spallation Neutron Source  
Spring-8    Super Photon ring-8 GeV  
SPS         Super Proton Synchrotron  
SR          Synchrotron radiation  
SREL        Space Radiation Effect Laboratory  
SRF         Superconducting Radiofrequency  
SRW         Synchrotron Radiation Workshop  
SS          Stainless steel  
SST         Spectroscopy Soft and Tender  
STAC8 code  Synchrotron Radiation Calculations Between Analytical Code  
SUS         Stainless steel  
TENDL       TALYS-based evaluated nuclear data library  
TERESA      TEstbed for high Rereptition rate Source of Accelerated particles  
ThomX       The compact X-ray source  
TMCS        Target–moderator–collimator–shield  
TNSA        Target normal sheathe acceleration  
TOF         Time-of-flight  
TRAM        Target, reflector and moderators  
TRIUMF      Canada’s particle accelerator centre  
TS          Target Station  
UCL         Université Catholique de Louvain  
UITF        Upgraded Injector Test Facility  
WBS         White beam stop  
wwg         Weight window generator  
XRF         X-ray Fluorescence  
XS          Cross-section
Executive summary

The 14th workshop on Shielding Aspects of Accelerators, Targets and Irradiation Facilities (SATIF-14) took place at Hilton Gyeongju in Gyeongju, Korea on 30 October – 2 November 2018. The workshop was chaired by Y.S. Min (KOMAC) and H.S. Lee (PAL) and was attended by 64 participants representing 30 organisations from 10 countries.

Support for the SATIF workshop is now part of the mandated activity of the Expert Group on Radiation Transport and Shielding (EGRTS, chaired by R. Grove from ORNL) of the Working Party on Scientific Issues of Reactor Systems (WPRS) of the Nuclear Energy Agency (NEA) Nuclear Science Committee (NSC). The EGRTS also co-ordinates maintenance and development of the Shielding Integral Benchmark Archive and Database (SINBAD) of reactor shielding, fusion neutronics and accelerator shielding benchmark experiments.

The main objectives of the SATIF workshops are to:

- promote the exchange of information among experts in the field of accelerator shielding and related topics;
- identify areas where international cooperation can be fruitful;
- undertake a programme of work in order to achieve progress in specific priority areas.

SATIF-14 was sponsored by KOMAC and PAL and co-sponsored by the NEA and its NSC. The workshop consisted of six technical sessions and one poster session, and a wrap-up session summarising achievements and defining further work for the next two years. The highlight of the workshop was a trip to the high power proton linac and utilising facilities at KOMAC and Pohang Light Source II (PLS-II) and PAL X-ray Free Electron Laser (PALXFEL). Those facilities are very impressive. The workshop participants also enjoyed exploring ancient capital city, Gyeongju, and a local fish market.

The six technical sessions were as follows:

- Session I: Source terms and related topics
- Session II: Radiation shielding and dosimetry
- Session III: Beam-plasma and laser-plasma interactions and acceleration
- Session IV: Code status and medical and industrial accelerators
- Session V: Induced radioactivity
- Session VI: Code benchmarking and intercomparison

The first session was chaired by H. Nakshima (JAEA) and focused on source term and related topics. Three presentations were made, including various sources of highly-intensive protons, dark current of long linac and photoneutrons. The latter two subjects were not supported by sufficient experimental data until now.
The second session was chaired by I. Popova (ORNL), S. Rokni (SLAC) and V. Mares (Helmholtz Munchen) and was dedicated to shielding and dosimetry. The session contained 13 presentations. CERN CHARM & CSBF mixed field facility where about 150 configurations are available is useful for shielding study and dosimetry of high-energy radiation. Fundamental measurements of Bi(p, xn) radionuclide yields and benchmarking contributes to improve the code development. Such an experimental study is recommended to be performed by SATIF community members. Reflecting the recent trend, the presentations of shielding studies of heavy ion accelerators and high power accelerators became more interesting.

The third session was chaired by R. Qui (Tsinghua Univ.) and targeted beam-plasma and laser-plasma interactions. This topic was introduced to shielding group recently. Three presentations gave experimental results, dose estimation and application method of shielding design.

The fourth session was chaired by H. Iwase (KEK) and focused on code status and medical, industrial accelerators. Three presentations introduced shielding and activation issues of the latest and important types of medical accelerator: particle therapy, BNCT and PET cyclotron. The decommissioning of PET cyclotrons became a big issue because of the number of operating machines in the world; the process is somewhat different from the decommissioning of high-energy accelerators. Also, a new functional idea of uncertainty in PHITS calculations was presented.

The fifth session was chaired by A. Ferrari (Helmholtz Dresden) and targeted induced activity. The session contained five presentations. Wider application of ActWiz II was introduced. A Japanese group lead by K. Kimura presented impressive studies of a “standard” concrete model. A possible collaboration with other countries and facilities was suggested.

The sixth session was chaired by V. Vylet (Jefferson Lab); eight presentations were dedicated to code benchmarking and intercomparison. One of the primary objectives of SATIF meetings has been de facto simulation code benchmarking. Several experimental results gave sufficient results for the code developers and for the entire SATIF community. Serial study of code comparison lead by H. Hirayama revealed the strengths and limitations of each code used by SATIF community members.

Following the previous meeting, a poster session was held, due to the high number of submitted contributions in the oral session. It helped to keep sufficient time for oral presentations and to make detailed discussions possible during the poster presentations. The session attracted nine contributions in the topics of shielding analysis (5), concrete composition (1), benchmarking (1), new calculation module (1) and facility status (1).

The session highlights were summarised during the last session by N. Mokhov and H. Hirayama. The objectives and activities of the SATIF community were emphasised: the idea is to co-ordinate analysis and propose action items, not just to hold workshops. During the last two decades, every SATIF meeting has highlighted the dedicated, well-thought experimental studies, benchmarking simulation results against the data, simulation code intercomparison and those results and future plans on the code developments. It was also suggested to add the following items from the objectives of SATIF workshop:

- information necessary to improve the shielding design, such as new or improved data;
requests of new functionalities to code developers needed for the improvement of shielding design, such as a survey of the participants or a task force created by the time of the next SATIF workshop;

- In particular, the same requests for the “standard” concrete.

Practical action items related to the above were proposed. Gathering the appropriate information will be useful for future activities of the SATIF workshops.

The NEA Secretariat representative, S. Tsuda, explained how the NEA Nuclear Science Committee (NSC) supports SATIF-related activities, and the role of SATIF. With the impressive status of the upgrading plan of SINBAD (Shielding Integral Benchmark Archive and Database), which evaluates the present data and performs the technical review, it is suggested that all SATIF colleagues join the plan.

It was suggested that the next SATIF workshop (SATIF-15) be held in 2020 in the United States following the tradition of rotating the venue between America, Europe and Asia. Dali Georgobiani, on behalf of the Facility for Rare Isotope Beams (FRIB), Michigan State University (MSU), presented a proposal for the SATIF-15 that was planned to take place in 2020. The workshop will take place at FRIB (MSU campus). Due to the COVID-19 pandemic, the date of SATIF-15 was tentatively moved to September 2022.

The members of the Scientific Committee of SATIF-14 were: S. Ban (KEK), F. Cerutti (CERN), A. Ferrari (HZDR), R. Grove (ORNL), J. Gulliford (NEA), H. Hirayama (KEK), G. Hughes (LANL), B. Kirk (honorary), H.-S. Lee (the general co-chairman of SATIF-14, PAL), N. Mokhov (FNAL), G. Muhrer (ESS), T. Nakamura (honorary), H. Nakashima (JAEA), S. Roesler (CERN), S. Rokni (SLAC), M. Silari (CERN), S. Tsuda (NEA), T. Valentine (ORNL), and P. Vaz (IST).

The members of the Local Organising Committee were: N.S. Jung (PAL), M.H. Kim (PAL), H.-S. Lee (PAL), Y.S. Min (the general co-chairman of SATIF-14, KOMAC), and S.G. Park (KOMAC). All members of PAL Radiation Safety Team contributed for the friendly atmosphere and intense scientific discussions.
Session I: Source terms and related topics
Measurement of the energy spectrum of spallation neutron emitted to the most backward angle of 180° produced from thick mercury target irradiated with 3 GeV protons

Hiroki Matsuda1*, Shin-ichiro Meigo1, Hiroki Iwamoto1, Hayato Takeshita2
1J-PARC/JAEA
2Kyushu University
*matsuda.hiroki@jaea.go.jp

In Accelerator-Driven System (ADS) and a high-intensity spallation neutron source such as Materials and Life Science Experimental Facility (MLF) in Japan Proton Accelerator Research Complex (J-PARC), the radiation dose rate is reduced by placing heavy shielding materials around the target. As for neutrons emitted to the most backward angle of 180° towards the proton beam direction, shielding of the neutron is not easy because the neutrons penetrate through the proton beam duct. Although the most backward neutrons are essential for the shield design, the experimental data of energy spectra emitted to the backward direction are scarce. For the validation of source term of radiation in the high-intensity facility, we have measured the energy spectrum of the neutron emitted to the most backward angle of 180° produced from mercury target irradiated with 3 GeV proton at the MLF in J-PARC.

In the experiment, the spectrum was measured by time-of-flight (TOF) method with a liquid scintillator (NE213) placed at the upstream of the bending magnet with flight-path of 126 m to the mercury target. The neutron detection efficiency was calculated with SCINFUL-R code with the correction of experimental result with 252Cf source in low energy region less than 10 MeV. In order to validate calculation code for shielding, the energy range of the measurement was set above 1 MeV, where a spallation neutron is a dominant fraction. As a cross-check of the experiment, the absolute neutron flux was also measured with the activation of In foil using 113In(n,n')115mIn reaction with the threshold of about 1 MeV. It was found that the absolute intensity result obtained by both techniques of the TOF and In foil activation showed good agreement within those uncertainties. Using the present experimental data, the calculation of PHITS (Sato T., et al., 2018) code was validated. It was found that the calculation with INCL4.6 cascade model (Boudard A., et al., 2013) coupled with GEM (Furihata S., 2000) are in good agreement with the experiment. On the contrary, the calculation with Bertini cascade model with GEM overestimates the experiment.
Dark current and radiation of KEK-Linac accelerating structure protons

Hiroshi Iwase¹, Hiroyasu Ego², Shuji Matsumoto², Yoshihito Namito¹, Hideo Hirayama¹
¹Radiation Science Center, KEK, ²Accelerator Laboratory, KEK
*hiroshi.iwase@kek.jp

Secondary radiations induced by dark current from an S-band disc-loaded accelerating structure in the KEK-Linac were measured and calculated. The structure, which has about 2 m length with an accelerating gradient of 20 MV/m, is operated in a test room for conditioning and the structure will be placed back in the KEK-Linac on completion of conditioning. The room has 50 cm-thick concrete walls, but the radiation levels are not satisfactory low at the sides around levels of 20 MV/m. Additional shields are required to reduce the radiations. However, the source terms for the shielding calculation, i.e. the dark current energies, directions and intensities in the structure are unclear. In this study, a model of dark current beam acceleration was developed to predict the induced secondary radiation. The model, in which electrons uniformly generated on surfaces around the beam apertures of the discs along the structure will be accelerated in uniform electric fields, shows good agreement with the radiation measurements performed inside and outside the room.
Preliminary results of re-analysis of photoneutron experiments performed at PAL using MC codes

Hee-Seock Lee\textsuperscript{1,2,*}, Mahdi Bakhtiar\textsuperscript{2}, Yong-uk Kye\textsuperscript{1,2}, Tatsuhiko Sato\textsuperscript{3}, Shuichi Ban\textsuperscript{4}, Toshiya Sanami\textsuperscript{4}

\textsuperscript{1}Pohang Accelerator Laboratory, POSTECH, Pohang, Korea
\textsuperscript{2}Division of Advanced Nuclear Engineering, POSTECH, Pohang, Korea
\textsuperscript{3}Japan Atomic Energy Agency (JAEA), Ibaraki, Japan
\textsuperscript{4}High Energy Accelerator Research Organization (KEK), Ibarak, Japan

\textsuperscript{*}lee@postech.ac.kr

The joint-experiments of PAL, KEK and Kyoto university for measurements of differential photoneutron yields by 2 GeV and 2.5 GeV electron beam had been carried out from 1998 to 2001. Many data were produced for several different kinds of target elements and also compared with calculation results of Monte Carlo codes. The comparison was not proved at that time. Then we have tried the comparison using the latest Monte Carlo codes: FLUKA, PHITS and MCNPX. And when the photoneutron yields were taken using TOF method, the effect of the attenuation Pb plates, which were used to supress strong gamma flash, was compensated using the removal cross-section. The removal cross-section was calculated using two methods. So the removal cross-section has been verified by re-calculating using Monte Carlo codes at this paper. It is confirmed that there is no missing contribution from other structure of those experimental setup. The preliminary results of such re-analysis of old experiments are presented.
Monte Carlo simulations for the quantification of beta skin dose

Thomas Frosio¹, Philippe Bertreix¹, Matteo Magistris¹, Chris Theis¹
¹CERN, Radiation Protection Group
*thomas.frosio@cern.ch

The European Organization for Nuclear Research (CERN) operates particle accelerators and facilities for high energy physics research. CERN experiments can lead to the activation of equipment which may require shipping to external workshops, institutes or repositories for radioactive waste. The IAEA safety standards provide nuclide-specific activity limits and shipping criteria for the transport of radioactive equipment.

However, the literature suffers from a lack of data concerning exotic radionuclides of interest which are produced at CERN’s particle accelerators. Such exotic radionuclides are assigned either a reference, conservative activity limit, or a nuclide-specific activity limit which needs to be evaluated on the basis of radiological quantities like the dose to the skin. In particular, the skin dose originating from beta emitters is difficult to estimate by analytic equations without resorting to strong hypotheses.

In this paper, we propose the determination of these beta doses by two different methods. These methods are based on Monte-Carlo simulations with the FLUKA code. The first method is based on simulating a radionuclide as a source. The second method establishes energy dependent transfer functions from activity to dose, and eventually allows off-line analytic calculations using beta spectra as input data. Both methods give similar results and the differences come from differing data evaluations that are taken into account.

In the context of the radioactive shipping activities of CERN’s Radiation Protection group these methods are used to calculate shipping activity limits for the exotic radionuclides produced at CERN and which are not included in the IAEA safety standards. In addition, the beta skin doses can also be calculated for all radionuclides whose decay data are known.

Introduction

Radioactive shipping follows recommendations established by IAEA safety standards (IAEA, 2014). The classification and packaging of materials to be shipped depends on the samples’ activities:

- For activities below a given threshold (so-called A₁/A₂ limits) determined in the reference (IAEA, 2014), the appropriate package is defined as Type A.
- For activities above this threshold, the appropriate package is defined as Type B.

Type B packages have more constraints associated to the verifications and the tests they have to follow. Consequently, they are related to higher costs and processes that are more complex.

Some radionuclides produced at CERN, especially in CERN-MEDICIS facility (Dos Santos Augusto et al., 2014) do not have any A₁/A₂ limits associated in current legislation. As a consequence two solutions can be envisaged for shipping. The first one is to consider
restrictive $A_1/A_2$ limits as described in (IAEA, 2012). This solution will often lead to a Type B package. Alternatively, one can propose $A_1/A_2$ limits regarding the methodology proposed by IAEA’s so-called Q System.

The Q System is based on a serial of exposure scenarios (Figure 1) of the public in case of a potential shipping accident.

$Q_A$ concerns the external dose due to photons, $Q_B$ is relative to the external dose due to beta emitters, $Q_c$ reflects the internal dose via inhalation, $Q_D$ denotes the skin contamination dose and $Q_E$ is linked to the submersion dose due to gaseous isotopes.

Analytical calculations exist for $Q_A$, $Q_c$ and $Q_E$ are well described in the literature. However, for $Q_B$ and $Q_D$, the literature suffers of a lack of knowledge as they are difficult to calculate analytically since the beta energy deposition in matter is not linear.

Figure 1. IAEA Q-values describing the different exposure scenarios

![Figure 1](image)


Computations of Q-values can be performed with the FLUKA (Battistoni G et al., 2015; Ferrari et al., 2005), Monte Carlo simulation code for each radionuclide of interest not referenced in the IAEA safety standard. However, computation times can be very important to reach satisfying statistical significance, mainly for radionuclides with low-energy emissions for small yields. The purpose of this work is to create transfer functions depending on emission energies of particles. These transfer functions have been built to be compiled with nuclear data libraries in order to get $Q_B$ and $Q_D$ values without additional computation time.

The first paragraph of this document details the calculations done with FLUKA. Then, we describe the transfer functions obtained with these calculations. Finally, we perform two kinds of comparisons and explain the differences.
Description of the FLUKA simulations

**Q\textsubscript{B} calculation**

Q\textsubscript{B} value represents the activity of a radionuclide which gives a skin dose H\textsubscript{p}(0.07) of 0.5 Sv in 0.5 hours at 1 metre from the source due to beta radiation. The main shielding enclosure is assumed to have been lost during the transport accident but a residual shielding medium is taken into account via a proposed reduction factor. The value of Q\textsubscript{B} can be calculated as follow:

\[
Q_B = \frac{10^{-12}.SF}{e_\beta}[TBq]
\]

(1)

Where:

- \textit{SF} represents the residual shielding factor \(SF = e^{0.017 E_{\beta,\text{max}}^{-1.14} d}\). IAEA assumes a conservative value of 3 which will be used in this work for pure mono-energetic electron emitters.
  - \(E_{\beta,\text{max}}\) is the maximal energy of the beta spectrum;
  - \(d = 150\) mg/cm\(^2\) is the absorption coefficient.

- \textit{e}_\beta represents the equivalent skin dose rate coefficient for beta emission at a distance of 1 m from the shielded material \([Sv\cdot Bq^{-1}\cdot h^{-1}]\).

In order to calculate Q\textsubscript{B} a FLUKA geometry has been built which is illustrated in Figure 2 below. A punctual isotropic source of discrete electrons or positrons is located in the centre of a sphere of air. At 1 metre of distance a sphere of skin, modelled according to the ICRU51 (Allisy et al., 1993) composition is placed in air. Secondary photons are discarded. We score the deposited energy in GeV/g/primary in a layer of 50 to 90 micrometres thickness in the skin.

**Figure 2. FLUKA geometry for Q\textsubscript{B} calculations**

This scoring is performed for different emissions energies between 350 keV to 12 MeV. Examples of the results are presented in Figure 3.

**Figure 3. Energy deposition of beta for different primary energies associated to the computation of Q₀ calculation Q₀**

![Figure 3. Energy deposition of beta for different primary energies associated to the computation of Q₀ calculation Q₀](image)


Q₀ value represents the activity of a radionuclide which gives a skin dose $H_p(0.07)$ of 0.5 Sv during a period of 5 hours assuming that a person has been contaminated. It is assumed that 1% of package contents are spread uniformly over an area of 1 m² and results to a contamination of 10% of this level to the hand. Consequently, the fraction of package distributed per unit area of skin is assumed at $10^{-3}$ m⁻². The value of Q₀ can be calculated as follow:

$$Q_0 = \frac{2.8 \times 10^{-2}}{h_{skin}} [TBq]$$

(2)

Where $h_{skin}$ is the equivalent skin dose rate per unit activity per unit area of skin [Sv. s⁻¹. TBq⁻¹. m²]

In this case the FLUKA model is a slab of skin (pink volume) in air, irradiated by an isotropic area source of discrete positrons or electrons at different energies. The source is in contact with the skin. As previously, the skin composition is the one taken from ICRU51 and secondary photons are discarded. The deposited energy is scored in GeV/g/primary in a layer of 50 and 90 micrometres of depth.

**Figure 4. FLUKA geometry for Q₀ calculations**

![Figure 4. FLUKA geometry for Q₀ calculations](image)

Electrons and positrons energies have been simulated in the range 50 keV – 12 MeV giving as examples the deposited energy maps below (see Figure 5).

Figure 5. Energy deposition of beta for different primary energies associated to the computation of $Q_\beta$

![Energy deposition of beta for different primary energies](source: CERN, 2020)

**Transfer functions**

The transfer functions have consequently been created by using a Marquardt non-linear least square algorithm (Marquardt, 1963). They are represented in Figure 6 for $e_\beta$ and $h_{skin}$ computations. Each point represents a FLUKA calculation for a fixed energy emission and the lines are the result of the Mardquardt algorithm. Electrons and positrons give the same results as secondary photons are discarded. A local maximum can be observed at low energies as the probability that a particle deposits an important fraction of its energy increases at low energies.
The transfer functions have been created to be compiled with a decay data library. In this work, we have used decay data from ICRP107 (ICRP, 2008). Eventually the following equations are used to calculate the parameters of interest:

\[
\begin{align*}
    e_{\beta} &= \frac{1}{SF} \int_{E} \mathcal{H}_{e^{\pm},Q_{B}}(E)\lambda(E)dE \\
    h_{\text{skin}} &= \int_{E} \mathcal{H}_{e^{\pm},Q_{D}}(E)\lambda(E)dE
\end{align*}
\]

Where

- \( \mathcal{H}_{e^{\pm},Q_{B}} \) and \( \mathcal{H}_{e^{\pm},Q_{D}} \) represent respectively the transfer functions for establishing \( Q_{B} \) and \( Q_{D} \) for electrons and positrons emitted at an energy \( E \).
- \( \lambda(E) \) is the beta spectrum of the respective radionuclide including mono-energetic electrons (Auger and Internal Conversion).

**Comparison of the results**

In order to have confidence in the calculations performed with transfer functions, two kind of comparisons have been done. First, the computed values from this work have been compared to the ones that can be found in IAEA SSG-26 document. Then, they have been compared to a second calculation method based on FLUKA.

**Comparison with IAEA SSG-26**

In general the comparison between IAEA SSG-26 and the transfer function method has shown very good agreement for \( h_{\text{skin}} \) and globally good agreement for \( e_{\beta} \) (see Table 1). It appears that for \( h_{\text{skin}} \), ratio are always between 0.92 and 1.01, except for Be-7. On the other hand, for \( e_{\beta} \) the ratios show stronger discrepancies as they are between 0.16 for Cs-137 and 1.45 for Cu-67. It should be noted that Be-7 as well has very high discrepancy.
The differences for Be-7 in both cases can be easily explained as IAEA assumes fixed minimum values for $e_\beta$ (1E-15 Sv.Bq$^{-1}$.h$^{-1}$) and $h_{\text{skin}}$ (2.8E-05 Sv.m$^{-2}$.TBq$^{-1}$.S$^{-1}$). However, these fixed values are not physical ones that are derived from appropriate nuclear decay data and over-estimate the dose considerably.

For the other radionuclides with higher discrepancies found in the $e_\beta$ calculation, the differences come from the choice of the model. The decay data evaluation used for establishing $e_\beta$ coefficients by IAEA comes from (ICRP38, 1983; Eckerman et al., 1993) whereas we have used ICRP107. Moreover, the doses are calculated in air (respectively in skin for our method) with approximations based on Continuous Slow Down Approximation in water. These differences in the model could be responsible for the observed discrepancies with respect to $e_\beta$.

Table 1. Comparison of values found in this work with those from IAEA SSG-26

<table>
<thead>
<tr>
<th>Decay Mode</th>
<th>Radio nuclide</th>
<th>$e_\beta$</th>
<th>Ratio IAEA/WORK</th>
<th>$h_{\text{skin}}$</th>
<th>Ratio IAEA/WORK</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Transfer function</td>
<td>SSG-26</td>
<td>SSG-26</td>
<td>Ratio</td>
</tr>
<tr>
<td>EC</td>
<td>Be-7</td>
<td>5.48E-19</td>
<td>1.00E-15</td>
<td>1.00E-15</td>
<td>1.825</td>
</tr>
<tr>
<td>ECB+</td>
<td>Na-22</td>
<td>2.60E-13</td>
<td>2.60E-13</td>
<td>2.60E-13</td>
<td>0.93</td>
</tr>
<tr>
<td>EC</td>
<td>Ti-44</td>
<td>2.20E-12</td>
<td>1.60E-12</td>
<td>1.60E-12</td>
<td>0.73</td>
</tr>
<tr>
<td>ECB+</td>
<td>Fe-52</td>
<td>4.85E-12</td>
<td>3.10E-12</td>
<td>3.10E-12</td>
<td>0.64</td>
</tr>
<tr>
<td>ECB+</td>
<td>Co-58</td>
<td>7.59E-15</td>
<td>1.30E-15</td>
<td>1.30E-15</td>
<td>0.17</td>
</tr>
<tr>
<td>B-</td>
<td>Co-60</td>
<td>3.28E-15</td>
<td>1.40E-15</td>
<td>1.40E-15</td>
<td>0.43</td>
</tr>
<tr>
<td>B-</td>
<td>Cu-67</td>
<td>1.66E-15</td>
<td>2.40E-15</td>
<td>2.40E-15</td>
<td>1.45</td>
</tr>
<tr>
<td>ECB+</td>
<td>As-72</td>
<td>5.43E-12</td>
<td>3.60E-12</td>
<td>3.60E-12</td>
<td>0.66</td>
</tr>
<tr>
<td>ECB+</td>
<td>I-124</td>
<td>9.76E-13</td>
<td>1.70E-13</td>
<td>1.70E-13</td>
<td>0.17</td>
</tr>
<tr>
<td>B-</td>
<td>Co-137</td>
<td>7.54E-13</td>
<td>1.20E-13</td>
<td>1.20E-13</td>
<td>0.16</td>
</tr>
<tr>
<td>B-</td>
<td>Ce-141</td>
<td>3.11E-15</td>
<td>3.10E-15</td>
<td>3.10E-15</td>
<td>1.00</td>
</tr>
<tr>
<td>B-</td>
<td>Pt-197</td>
<td>4.38E-14</td>
<td>4.20E-14</td>
<td>4.20E-14</td>
<td>0.96</td>
</tr>
</tbody>
</table>


Comparison with direct FLUKA calculation

In order to validate the method numerically, the coefficients calculated with the transfer method function have been compared with pure Monte Carlo calculations. The monoenergetic sources described in the paragraph “Description of the FLUKA simulations” have been replaced by a radioactive isotope source with FLUKA’s built-in HI-PROPERTY card. All the other characteristics of the models remain unchanged. Energy deposition has been scored and results are summarised in Table 2.

Some very small discrepancies can be observed, with ratios found between 0.97 and 1.33 (FLUKA with radioactive isotope source results are often below the results of the transfer function method).
Table 2. Comparison of two different methods of calculation for the computation of $e_\beta$

<table>
<thead>
<tr>
<th>Radio nuclide</th>
<th>Ratio Transfer function vs FLUKA source methods</th>
<th>Transfer function method</th>
<th>FLUKA source method</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Sv.Bq$^{-1}$.h$^{-1}$</td>
<td>Sv.Bq$^{-1}$.h$^{-1}$</td>
</tr>
<tr>
<td>Na-22</td>
<td>0.97</td>
<td>2.80E-13</td>
<td>2.90E-13</td>
</tr>
<tr>
<td>Co-58</td>
<td>1.09</td>
<td>7.60E-15</td>
<td>7.00E-15</td>
</tr>
<tr>
<td>Co-60</td>
<td>1.22</td>
<td>3.30E-15</td>
<td>2.70E-15</td>
</tr>
<tr>
<td>Cu-61</td>
<td>0.99</td>
<td>8.60E-13</td>
<td>8.70E-13</td>
</tr>
<tr>
<td>As-71</td>
<td>0.99</td>
<td>8.30E-14</td>
<td>8.40E-14</td>
</tr>
<tr>
<td>Pr-140</td>
<td>1.00</td>
<td>2.30E-12</td>
<td>2.30E-12</td>
</tr>
<tr>
<td>Tb-149</td>
<td>1.03</td>
<td>3.90E-13</td>
<td>3.80E-13</td>
</tr>
<tr>
<td>Tb-152</td>
<td>1.33</td>
<td>1.30E-12</td>
<td>9.80E-13</td>
</tr>
<tr>
<td>Tb-156</td>
<td>1.30</td>
<td>3.00E-14</td>
<td>2.30E-14</td>
</tr>
<tr>
<td>Tb-161</td>
<td>0.98</td>
<td>3.90E-15</td>
<td>4.00E-15</td>
</tr>
<tr>
<td>Tl-209</td>
<td>1.00</td>
<td>3.40E-12</td>
<td>3.40E-12</td>
</tr>
</tbody>
</table>


In order to explain these discrepancies, a beta spectrum based on FLUKA’s decay data has been scored and compared to data found in ICRP107. One of these comparisons is detailed in Figure 7 for a Co-60 spectrum. The differences between both spectra are particularly observable at energies above 300 keV, in the area of definition of $\mathcal{H}_{e^-,Q_B}$.

The spectrum has subsequently been convolved with the transfer function. The results obtained has been the same than the one we got with FLUKA HI-PROP calculation showing that differences observed in results of Table 2 are totally explained by differences of decay data.

Figure 7. Comparison of ICRP107 and FLUKA beta decay data for Co-60

![Figure 7. Comparison of ICRP107 and FLUKA beta decay data for Co-60](image)

Conclusions

We have shown the possibility to compute Q-values for shipping of radioactive goods in the case that data are missing for particular radionuclides created at CERN. Comparisons show generally good agreement and all discrepancies can be explained, mainly due to different nuclear data libraries that have been used for different approaches or codes.

A C++ software has been developed to allow for the computation of shipping limits if necessary. This software has the advantage of performing these calculations in one fraction of a second instead of time consuming Monte Carlo calculations.

References


ICRP (2008), Nuclear Decay Data for Dosimetric Calculations, ICRP Publication 107, Annals of the ICRP, Volume 38, No. 3.


Radiation gradient evaluation at the CERN CHARM mixed-field facility using RPL, RadMON and FLUKA simulation

Melanie Krawina¹, Angelo Infantino¹*, Chiara Cangialosi¹, Rubén García Alía¹, Markus Brugger¹, Matteo Brucoli¹, Francesco Cerutti¹, Salvatore Danzeca¹
European Organization for Nuclear Research (CERN) , CH-1211, Geneva 23, Switzerland
*angelo.infantino@cern.ch

The CERN CHARM facility provides a unique complex radiation environment characterised by particle energy spectra representative of high-energy accelerators, ground and atmospheric conditions and space applications. CHARM is conceived to be an irradiation facility for the qualification of large electronic systems and components in a mixed-field radiation environment, generated from the interaction of a 24 GeV/c proton beam with a copper or aluminium target. A movable shielding made of layers of concrete and iron allows changing the hardness and the particle population (neutron, proton, kaon, pion, muon, electron, positron and photon) in predefined test locations. To ensure a full representativeness and reproducibility of the tests, an accurate dosimetry of the complex mixed irradiation field is mandatory. The significant size of the available test area, the multiple facility configurations as well as the strong radiation gradient present in some of the test locations make the radiation monitoring a challenge. Therefore, an accurate and validated Monte Carlo model of the facility becomes essential to characterise the radiation field for all the possible configurations and for non-standard irradiation tests. A FLUKA Monte Carlo model of the facility has been created and developed over the last years to support the calibration of the facility. This work aims to characterise the CHARM radiation field in test positions affected by a strong radiation gradient: T0, a high dose rate test location close the target revolver; R1, R10, R11 and R13, four standard test location for irradiation tests on electronics equipment and devices. Three different quantities of interest in the radiation effects to electronics community have been assessed: Total Ionizing Dose, High Energy Hadrons fluence (hadrons with energy >20 MeV) and 1-MeV neutron equivalent fluence. Experimental measurements have been conducted with RPL dosimeters and RadMON detectors, the latter equipped with pMOS RadFET sensors, P-I-N diodes and SRAM memories. The measurement campaign produced a large amount of data, which were compared with FLUKA Monte Carlo simulation. The concluding comparison shows different levels of agreements, depending on the quantity of interest, the position and on the detector used: overall, measurements and FLUKA simulations agree within a factor 2 which, considering the strong radiation gradient and the limitation of the different detectors in the mixed-field environment, can be considered satisfactory. A full presentation of the results and the data analysis, as well as of the FLUKA simulations, will be given in the talk.
Evaluation of future employment of boron carbide shielding at CERN High Energy AcceleRator Mixed Field (CHARM) Facility

Chiara Cangialosi*1, Angelo Infantino1, Salvatore Danzeca1, Rubén García Alía1, Markus Brugger1, Alessandro Masi1
1European Organization for Nuclear Research (CERN), CH-1211, Geneva 23, Switzerland
* chiara.cangialosi@cern.ch

The complex radiation environment of CERN Large Hadron Collider (LHC) accelerator consists of several particle types at different energies ranging from GeV down to thermal neutron (0.025 eV). The Radiation Hardness Assurance of electronic components plays a key role in the design and development of electronic devices for the LHC, such as power converter and control systems. In this context the mixed field radiation test facility, CHARM, has been built at CERN to test large electronic systems and components in representative accelerator-like environments, where high radiation levels are expected. However, its complex radiation environment (characterised by neutron, proton, kaon, pion, muon, electron and photon presence) is not only representative of particle accelerators, but also of atmospheric, ground level and space applications. The radiation field of CHARM is unique in terms of the multitude of particles generated from the interaction of a mono-energetic 24 GeV/c proton beam with a copper (Cu) or aluminium (Al) target. The use of different targets and of movable shielding slabs (made of layers of concrete and/or iron) allows for varying the particles spectrum at the different test locations.

The radiation environment and the particle energy distribution vary considerably in the LHC, depending on the location inside the accelerator. In particular, in heavy shielded alcoves for electronics, a relevant thermal neutron (ThN, Ekinetic ≤ 0.5 eV) fluence is expected. The presence of the ThN can have a strong impact on the accelerator equipment’s failure rate and can present a considerable additional risk for the personnel involved in the operation of the LHC. In nuclear power plants, Boron Carbide (B4C) is used for its capability to absorb neutrons without forming long-lived radionuclides. For this reason, the employment of B4C in the LHC alcove is taken into account to protect the accelerator equipment, reducing the failure rate. Our study aims to evaluate the performance of B4C shielding in an accelerator-like environment. The unique field present at the CHARM facility results in a perfect test environment for this study. The tests are carried out at CHARM in positions characterised by high values of ThN fluence. The measurements are performed using SRAM memories, part of the CERN Radiation Monitoring system, sensitive to ThN and further compared with FLUKA Monte Carlo simulations. The reduction of ThN fluence due to the Boron Carbide shielding is evaluated using sheets characterised by different percentages of B content (80% and 25%) and different thickness. Concluding, we observe that in an environment characterised by a high level of ThN, the use of B4C sheets can lead to a ThN fluence reduction up to 80%. Moreover, at our irradiation conditions, the results seem not to be affected by the percentage of B content in the sheets. Finally, a future improvement of the CHARM facility using slabs of a B4C as shielding is currently under investigation using FLUKA Monte Carlo simulations.
Measurement of neutron energy spectra after penetrating concrete shield at CERN/CSBF with 24 GeV/c protons on thick copper target

Eunji Lee1,*, Nobuhiro Shigyo1, Tsuyoshi Kajimoto2, Toshiya Sanami3, Noriaki Nakao4, Masayuki Hagiwara3, Hiroshi Yashima5, Takahiro Oyama3, Robert Froeschl6, Stefan Roesler6, Elpida Iliopoulou6, Angelo Infantino6, Markus Brugger6

1Kyushu University, Motooka, Nishi-ku, Fukuoka 819-0395, Japan
2Hiroshima University, 1-4-1, Kagamiyama, Higashi-Hiroshima 739-8527, Japan
3High Energy Accelerator Research Organization (KEK), Oho, Tsukuba 305-0801, Japan
4Shimizu Corporation, 3-4-17 Etchujima, Koto-ku, Tokyo 135-8530, Japan
5Research Reactor Institute, Kyoto University, 2-1010 Asashiro-nishi, Kumatori, Sennan, Osaka 590-0494, Japan
6CERN, 1211 Geneva 23, Switzerland
*ejlee@kune2a.nucl.kyushu-u.ac.jp

Neutron energy spectra from 24 GeV/c protons on a thick copper target were measured at CERN Shielding Benchmark Facility (CSBF) situated laterally above CERN High Energy Accelerator Mixed Field (CHARM). The data were used to evaluate the transmission of high-energy neutrons in concrete shield.

The experiment was carried out using a 24 GeV/c proton beam from the CERN PS with a maximum intensity of 5 × 1011 protons per pulse and a pulse length of 350 ms. The proton hits the cylindrical copper target of 8 cm in diameter and 50 cm length. In the target, high-energy neutrons were generated as a result of nuclear reaction. The neutrons were measured using a neutron detector, a 12.7 cm thick and 12.7 cm in diameter NE213 liquid organic scintillator, 6.8 m to 7.4 m away from the target at 90° with respect to the proton beam axis. On the pathway from the target to the detector, there were 10 cm marble slab, 40 cm thick cast iron support plate and removable concrete blocks. The thicknesses of the removable concrete blocks were chosen to be 0, 40, 80, 120, 160, 200, 240 and 360 cm for the experiment.

**Introduction**

Study on neutron energy spectrum after penetrating a shield is important in shielding design of high-energy accelerators. In a deep penetration problem of high-energy neutrons (E > 10 MeV), a most shielding design has been generally performed by using a Moyer model, which is based on single exponential attenuation of neutron fluence. The Moyer model includes attenuation length in the exponential equation. The attenuation length depends on the density and shielding material. Concrete is the most common shielding material. Until now, several experiments have been performed to obtain the attenuation lengths for concrete (Ban, 1980; Stevenson, 1982; Ishikawa, 1994; Nakao, 1996; Nunomiya, 2000; Sasaki, 2002). According to expectations, the precise attenuation lengths for concrete would be applicable to shielding design. However, the previous measurements of attenuation lengths for concrete have wide range from 110 to 172 g/cm² (Filges, 2010).

The shielding design has been widely performed with Monte Carlo codes. The codes have been used to simulate transport of neutrons, but it is necessary to validate the simulation by
using the experimental data. At beam energies below 3 GeV, there are sufficient experimental data of neutron energy spectra (Ishibashi, 1997; Amian, 1992; Leray, 2002). On the other hand, at the high-energies beam above 3 GeV, the comparable experimental data are very scarce. The lack of the comparable experimental data in the high-energy region leads to measurement benchmark experimental data.

In this study, we measured neutron energy spectra behind concrete shields by changing the thickness. The neutron spectra according to the thickness were used to obtain neutron yields by integrating the spectra over the energy range from 10 to 350 MeV. The neutron yields were fitted by single exponential to estimate attenuation length for concrete. The neutron energy spectra and attenuation length were compared with the calculated results of PHITS code.

**Experiment**

The CERN High Energy AcceleRator Mixed Field (CHARM) Facility is located in the CERN Proton Synchrotron (PS) East Experimental Area to study radiation effects on electronic components (R2E project) (Froeschl, 2014; Froeschl et al., 2015). In the roof of the CHARM facility, the CERN Shielding Benchmark Facility (CSBF) is prepared for deep shielding penetration studies (Iliopoulou, 2018). Information about experimental conditions are similar in “Neutron energy spectrum measurement using an NE213 scintillator at CHARM” (Kajimoto, 2018a). Only the outline of the experiment is described in this section.

Figure 7.1 shows a vertical cross-section of the CSBF along the beam line in the case of 160 cm thick concrete. The CHARM received pulsed 24 GeV/c proton beam from the CERN PS with intensity of $3 \times 10^9$ to $5 \times 10^{11}$ protons per pulse and a pulse length of 350 ms. The protons hit the cylindrical copper target of 8 cm in diameter and 50 cm length. In the target, high-energy neutrons were generated as a result of nuclear reaction. Neutrons were measured using a neutron detector, a 12.7 cm in diameter and 12.7 cm thick NE213 liquid organic scintillator, placed on the top roof, 6.8 m to 7.4 m away from the target at 90° with respect to the proton beam axis. Two veto detectors were placed to distinguish charged particle events on bottom surfaces (veto1, $15 \times 15 \times 0.6$ cm$^3$) and the upstream (veto2, $15 \times 15 \times 0.3$ cm$^3$) of the neutron detector. On the pathway from the target to the detector, there were 10 cm thick marble slab, 40 cm thick cast iron support plate, and removable concrete blocks. The thicknesses of removable concrete blocks were chosen to be 0, 40, 80, 120, 160, 200, 240 and 360 cm. The chemical composition and density of the materials were the same as in “Measurements and FLUKA simulations of bismuth and aluminium activation at the CERN Shielding Benchmark Facility (CSBF)” (Iliopoulou, 2018).

Electric circuit were used to collect the integral QDC values from detectors; it consisted of standard NIM and VME modules. Data of QDC values of the neutron detector, two veto detectors, and time between the pre-trigger signal of beam pulse and the neutron detector signals were recorded event-by-event. For separation of neutron and $\gamma$-ray events, signals from the neutron detector were integrated with total and slow gates (Zucker, 1990). The two gate lengths were set to be 200 and 160 ns, respectively.
Data analysis

Data analysis is also basically the same as in “Neutron energy spectrum measurement using an NE213 scintillator at CHARM” (Kajimoto, 2018a). Only the outline of data analysis is described in this section. The unfolding technique is commonly used to obtain neutron energy spectrum when time-of-flight method is not applicable by reasons of behind a thick shield. The data taken above conditions were processed by the following procedures in prior to spectrum unfolding: 1) extraction of neutron events through event selections; 2) calibration of channel to light output; 3) spectrum unfolding using an iterative Bayesian algorithm (D’Agostini, 1995) in the RooUnfold package (Adye, 2011).

Before the extraction of neutron events, beam events were selected from counts for each spill corresponding to the time between the pre-trigger signal of beam pulse and the neutron detector signals shown in Figure 2. In order to extract uncharged particle, i.e. neutron and γ-ray events, QDC value of veto signals were used. Figure 3 shows an example of the QDC value of two veto detectors. After applying, neutron and γ-ray events were discriminated each other using the signals with the total and slow gates, i.e. pulse shape discrimination technique. Figure 4 shows a graphical scatter plot of QDC values with total and slow gates. The dashed line depicted in Figure 4 indicates the lower QDC value discrimination level, which corresponds to 4.323 MeVee (MeV electron equivalent). It is clear that neutron events were successfully distinguished from γ-ray ones at the discrimination level.

The QDC value of total gate were converted into a light output value in MeVee using a calibration curve shown in Figure 5. The calibration curve was fitted by calibration points which were acquired by measuring Compton edge of γ-rays with approximately 1.253 MeV (average of 1.173 and 1.333 MeV) from $^{60}$Co, 1.28 MeV for $^{22}$Na, 4.439 MeV for $^{241}$Am-Be sources, and 20.5 MeVee for cosmic-ray muons in natural background (Kajimoto, 2018b). The point around 100 MeVee was obtained from the maximum energy deposition of protons, which was obtained by finding the boundary point in two-dimensional QDC.
values plot of the total gate and veto detector. The maximum energy deposition was calculated to be 124 MeV using the SRIM code (Ziegler, 2010) for NE213 scintillator with a thickness of 12.7 cm.

Figure 2. Extraction of beam events

![Figure 2. Extraction of beam events](image1)

Source: Kyushu University, 2020.

Figure 3. Extraction of uncharged particle events

![Figure 3. Extraction of uncharged particle events](image2)

Source: Kyushu University, 2020.

The response functions were obtained using the SCINFUL-QMD code (Satoh, 2006), which was modified by Kajimoto et al. (Kajimoto, 2011). Source neutrons were vertically and uniformly emitted to the scintillator flat surface. The neutron energy region between 0 and 1 GeV was divided into 200 steps in a linear interval. A scoring region with the same shape and size as the neutron detector was set in the calculation.

After these procedures, the neutron energy spectra were obtained by unfolding method using the iterative Bayesian algorithm in the RooUnfold package.
Monte Carlo simulation

PHITS code (Sato, 2013) (ver. 3.08) was used for Monte Carlo simulations to calculate the neutron energy spectra. The calculation takes into account simplified geometry only the roof shield (top of the target). The side wall, beam dump and the floor were neglected. The same chemical composition and density of the materials reported in “Measurements and FLUKA simulations of bismuth and aluminium activation at the CERN Shielding Benchmark Facility (CSBF)” (Iliopoulou, 2018) were used for calculation. Nuclear reaction for neutron production and transportation adopted in the simulation are JENDL-4.0 data library (Shibata, 2011) below 20 MeV, and INCL cascade model (Boudard, 2013) from 20 MeV up to 3 GeV, and the JAM cascade model (Nara, 2000) above 3 GeV. To shorten the calculation time, we applied variance reduction techniques, referred as cell importance method in PHITS, which can artificially increase the probability of rare event occurrences. The neutron energy spectra were acquired by neutron track to obtain spatial distributions of neutrons. Figure 7.6 represents the neutron track calculated using PHITS code in the case of 160 cm thick concrete at 90 degrees.

Figure 6. Neutron tracks for the case of 160 cm thick concrete in the PHITS calculation

![Figure 6](image)

Source: Kyushu University, 2020.

Results and discussions

Figure 7 shows the experimental neutron energy spectra for each concrete thickness from 10 to 350 MeV range. Solid lines indicate the calculated results by PHITS. In calculated
results of 0 cm thick concrete, there are fluctuation due to the large statistical uncertainties. Black circles denote the results for unfolded spectra. The unfolded spectra show only uncertainty originating in the RooUnfold package during unfolding. The neutron spectra decrease with the thickness. The calculated results are generally in good agreement with experimental results in all figures. However, calculated results of 0 to 80 cm thick concrete have discrepancy with the experimental results between 20 MeV and 200 MeV with the ratio difference 0.69, 0.72 and 0.73, respectively. The calculated result of 200 and 240 cm thick concrete overestimates the experimental result.

Figure 7. Neutron energy spectra by unfolding method

Source: Kyushu University, 2020.

The neutron yields were fitted by single exponential using a simple equation (1) as follows to estimate attenuation length $\lambda_{\text{att}}$ for concrete:

$$I(x) = I_0 \exp^{-\rho x / \lambda_{\text{att}}},$$

where $I$ is the neutron yields, $x$ is the shield thickness, $I_0$ is the magnitude parameter, $\rho$ is the density of shielding material (2.4 g/cm$^3$ for normal concrete), $\lambda_{\text{att}}$ is the attenuation length for the shielding material. The exponential function accounts for the attenuation of the neutron transmission by a concrete shield. The neutron energy spectra according to the thickness were used to obtain neutron yields by integrating the spectra over the energy range from 10 to 350 MeV. Figure 8 shows the neutron attenuation profiles for concrete.
removable blocks from 0 to 360 cm. However, the concrete thickness of 360 cm was considered to be 330 cm. Because the geometry for 360 cm thickness consist of normal concrete blocks (160 cm) and removable activation sample holder concrete (200 cm), which have 3 slots of 10 cm deep hole that are centred along the vertical axis of the block (Iliopoulou, 2018). Black points indicate the experimental neutron yields. Red points indicate the calculated neutron yields using PHITS. The errors were only the statistical errors. Experimental and calculated results from 40 to 240 cm thick concrete were used in the fit of two solid lines using a maximum likelihood estimation. The neutron yields from 330 cm thick concrete was affected by the background, which is due to same thickness of both the removable concrete blocks and surrounding shield, and thus was not included in the fitting procedure.

![Figure 8. Attenuation profiles from neutron yields for concrete](image.jpg)

The neutron attenuation lengths for concrete correspond to those for neutrons above 10 MeV are 111.4 ± 4.1 g/cm² from experimental results and 125.4 ± 8.1 g/cm² from calculated results. These values are just between those given by “Handbook of Spallation Research: Theory, Experiments and Applications” (Filges, 2010). PHITS reproduces the spectral shape well and attenuation length was in good agreement within factor of 1.1; however, its values were overestimated.

**Summary**

A deep penetration experiment, which was performed at CERN Shielding Benchmark Facility (CSBF) situated laterally above CERN High Energy AcceleRator Mixed Field (CHARM). Neutron energy spectra from 24 GeV/c protons on a thick copper target were measured with 0, 40, 80, 120, 160, 200, 240 and 360 cm thick concrete removable blocks. The neutron energy spectra were compared with the calculated results of PHITS code. The PHITS code generally reproduces the neutron energy spectra; however, it overestimates the experimental data. The neutron energy spectra according to the thickness were used to obtain neutron yields by integrating the spectra over the energy range from 10 to 350 MeV. The attenuation lengths for concrete shield were 111.4 g/cm² from experimental results and 125.4 g/cm² from calculated results. The ratio of the experimental results to calculated
results is 1.1. This means physics model and nuclear data should be improved for appropriate shielding design. The measured attenuation length of concrete is significant for investigating the accuracy of deep penetration calculation.

Acknowledgements

We thank to CHARM operation teams for their support of our experiments.

References


Session II: Shielding and dosimetry
Shielding and skyshine of a 10 MeV electron accelerator enclosed in a large hall

Vaclav Vylet*
Thomas Jefferson National Accelerator Facility
*vylet@jlab.org

Upgraded Injector Test Facility (UITF) at Thomas Jefferson National Accelerator Facility (aka JLab) is a new 10 MeV electron accelerator designed to explore and test advanced concepts in the design of accelerator guns, accelerator diagnostic devices, detectors and other equipment to be used in JLab’s CEBAF accelerator, experimental halls and possible future installations. The facility consists of a heavily shielded existing electron gun test area, Cave 1, currently housing an electron linac with a short cryogenic accelerator module. A new area, Cave 2, has been added to provide space for testing experimental equipment. The use of a cryogenic gas, liquid helium, and the requirement of movable roof panels presented challenges for shielding design. A large opening in the side-shielding wall just under the roof shielding is required to keep the facility at the desired Oxygen Deficiency Hazard 0 level. In this work we present a radiation safety assessment for normal operation and accident scenarios using a combination of Monte Carlo and simpler shielding calculation techniques.

Introduction

Upgraded Injector Test Facility (UITF) at Thomas Jefferson National Accelerator Facility (aka JLab) is a recent extension of a well-shielded enclosure used for research and tests of electron accelerator guns. The large building housing this facility, shown in Figure 1, originally contained the Space Radiation Effect Laboratory (SREL) operated by a consortium of universities for NASA from 1966 until 1980. The SREL included a synchrocyclotron with a primary beam of 600 MeV protons and secondary beams of 400 MeV pions and muons produced for studying the effects of radiation on materials planned for use in space. After Jlab acquired the site, the original shielding has been largely removed or reconfigured for other purposes.
The gun test area, Cave 1, is a remainder of the original shielding shown in grey in Figure 2. The shielding wall on its west side is approximately 5.5 m thick. Cave 1 now contains an electron gun that injects electrons to a short accelerator module containing cryogen-cooled RF cavities. The 10 MeV beam will be used in the newly built addition, Cave 2, to test experimental equipment and accelerator diagnostic devices. In the near future the facility programme will focus on testing the cryogenic target for the HDIce experiment (see references by Bass et al. and Lowry et al.). The experiment will use an elevated beam line shown in Figure 3.
Operational parameters and shielding requirements

During the HDIce tests, it is expected that beam current of 100 nA will be used for tuning and current calibration measurements, approximately 20% of the total running time of 900 hours per year. This is necessary because the very low currents equal or below 10 nA needed for HDIce target tests are unsuitable for steering and current calibrations. The HDIce target tests will consume the remaining 80% of time budget.

The Test Lab building is accessible to any Jlab employees without requirement of radiation safety training. The shielding requirements are therefore the same as for public, i.e. not to exceed the effective dose regulatory limit of 1 mSv per year. The shielding design goal for such areas is conservatively lower, 0.1 mSv/y. However, facility use and occupancy factors can be applied. As part of the ALARA approach, Jlab Radiation Safety implemented the concept of Radiation Control Area (RCA), accessible only to radiation workers. RCAs are posted where routinely exceeding dose rates of 0.5 μSv/h may be expected, e.g. around radioactive material storage locations. This RCA dose rate threshold value is derived from the above 1 mSv/y regulatory limit, assuming a 2000 h/y occupancy.

If the shielding were designed so that a loss of 100 nA results in a dose rate outside shielding at the RCA limit, this would already slightly exceed the design goal of 0.1 mSv/y if we assume the above usage and full beam loss. While not all such losses will occur in places without local shielding (e.g. outside Faraday cups or dumps), calculated dose rates consistent with the RCA limit are in this case a useful reference. After two years of HDIce tests, it is envisioned to deliver higher currents, up to 100 μA, to Cave 2. This will require a combination of enhanced external and local shielding, together with a certified beam loss accounting system currently under development.

Initial shielding evaluation

The UITF project collaboration envisioned to assemble the Cave 2 extension mostly by reusing spare interlocking shielding blocks, available in a thickness of ~120 cm (4 ft). Similarly, interlocking roof panels recovered from earlier projects, ~ 53 cm (1.75 ft) thick, were available. Use of thicker roof panels is precluded by the need to periodically remove part of the roof to allow installation of bulky experimental equipment and by the limitations
of the overhead crane. Additional 60 cm (2 ft) thick blocks were used for elements of the entry maze.

The initial shielding evaluation was done mostly using source terms and attenuation tenth-value layers from the NCRP Reports 51 (NCRP, 1977) and 144 (NCRP, 2003). The exception was a forward source term, calculated using GEANT3 (Degtiarenko, 2015) for a narrow electron beam hitting, under a glancing angle, a steel pipe terminated by a thick flange. The source term is averaged within a 2° forward angle and is an order of magnitude lower than the NCRP value of 300 rad at 1 m for a 100 nA beam loss. It is worth noting that at 90° the GEANT3 source term is 3 rem/h at 1 m, almost 60% lower than the NCRP value of 7.2 rem/h. Results are summarised in Table 1 below, assuming that 5% of the 100 nA beam is being lost constantly, with occasional mis-steering loss of up to 100 nA. These results show that, with the wall and roof thickness as specified above, there is potential for exceeding dose rate limits corresponding to Radiation Area and High Radiation Area designation, i.e. 0.05 and 1.0 mSv in an hour, respectively.

Table 1. Expected dose rates from routine and accidental beam losses

<table>
<thead>
<tr>
<th>Location</th>
<th>Beam loss</th>
<th>E [Sv/h]</th>
<th>Note</th>
</tr>
</thead>
<tbody>
<tr>
<td>Side wall</td>
<td>5%</td>
<td>2.21E-02</td>
<td>below RCA</td>
</tr>
<tr>
<td></td>
<td>100%</td>
<td>4.42E-01</td>
<td>below RCA</td>
</tr>
<tr>
<td>Forward</td>
<td>5%</td>
<td>2.8</td>
<td>RCA</td>
</tr>
<tr>
<td></td>
<td>100%</td>
<td>56.0</td>
<td>Radiation Area</td>
</tr>
<tr>
<td>Roof</td>
<td>5%</td>
<td>17.1</td>
<td>Radiation Area</td>
</tr>
<tr>
<td></td>
<td>100%</td>
<td>341</td>
<td>High Radiation Area</td>
</tr>
</tbody>
</table>


Since there is no need for frequent access to the roof and additional shielding is not practical for logistical reasons, the roof will be inaccessible during accelerator operation. One additional complication in this area is a large opening on the north-east side that is necessary to allow He gas to vent out of the UITF Cave 2 enclosure in case of accidental cryogen loss from the large Dewar container near the HDIce target. The forward shielding on the north side of Cave 2 borders an infrequently occupied passage towards an exit door from the building. However, adjacent to this passage is the North Annex building (see Figure 1 above). Its second floor, approximately at the roof level of UITF, contains a storage area permanently occupied by three custodians. This area could be affected by both shine from the UITF roof and radiation leakage in the forward direction. In the next phase, we studied these issues and their solutions using the FLUKA particle transport code (Ferrari, 2005) with FLAIR graphical interface (Vlachoudis, 2009).

Monte Carlo shielding and skyshine calculations

Helium Vent

The helium vent mentioned above creates a rectangular opening, 564 cm long and 105.6 cm high, in the east wall just under the roof towards the north-east corner. In case of an accidental release helium flows through this aperture into a trench between the end of the roof panels and the massive shielding wall on the eastern side. A large concrete block
plugs the north end of the trench between the roof and the east wall. As the helium flows upwards, it passes through a limiting aperture of of same length as the wall opening, 564 cm, but only 60 cm wide. The XY cross-section of this opening is shown in Figure 4a. The vent location is in the area of the HDIce experiment, with an elevated beamline, and levels of radiation streaming through the vent will be most sensitive to beam losses in this area. The beamline runs parallel to this aperture, at a distance of 127 cm from the east wall and 30 cm below the lower aperture limit. Initial FLUKA simulations indicated that in the absence of the He vent radiation levels on the roof remain below 1 mSv/h in case of 100 nA beam loss, almost a factor of 10 lower than predicted by the semi-empirical method. However, with the vent in its original configuration levels exceeding 10 mSv/h, High Radiation Area may be reached near the vent exhaust on the roof. This will be mitigated by addition of a second concrete block at the south end of the vent. The two blocks in the trench will then support a roof plank spanning the space above. Figure 4b shows a 3D view of a section through such configuration. The flow of helium is restricted by the 60 cm wide horizontal aperture between roof blocks and the adjacent east wall. Adding a slab of iron (section 15cm x 7.5 cm) at the bottom of the trench, as shown in Figure 4 b, will therefore not impede the gas flow at that point and helps to restrict the scatter into the trench. An aluminium grid covering the vent orifice will prevent access to a potential High Radiation Area. This arrangement also limits the spread of the skyshine to the east and south sides of the building. A comparison of FLUKA simulations for these two configurations is presented in Figures 5 and 6. In these (and most of the following figures) white areas indicate dose rates below RCA level.

Figure 4: a) XY Cross-section of the Helium vent as built; b) 3D view of the same vent with added iron plate (in green) and concrete plank on the top

Figure 5. Radiation levels [mSv/h] for 100 nA loss in the He vent area per as-built design


Figure 6. Radiation levels [mSv/h] for 100 nA loss in the He vent with added roof plank

Entrance maze

The beamline emerges from Cave 1 at a height of 107 cm above ground. As it enters Cave 2 space, it bends upwards, as shown in Figure 3, to bring beam to the HDIce experimental area. This inflection point faces the entrance to the exit maze from Cave 2. Another side beamline will be added in the future to bring the beam towards the west sidewall as shown in Figure 2. Since beam losses due to mis-steering may occur in this area, we simulated this scenario shown below in Figure 7 for a full loss of 100 nA. It appears that for the operation at 100 nA no local shielding will be needed.

Figure 7. Radiation levels from beam loss facing the maze

Forward shielding and north annex storage area

Both early simple estimates and FLUKA simulations confirmed that the original forward shielding is not entirely sufficient to keep radiation levels below the RCA limit in the 1st floor access corridor and the 2nd floor storage area in the North Annex. Furthermore, using “notched” concrete blocks lying on their side created 7.5 cm thick and 60 cm deep gaps in the shielding at a height of 2.4 m above ground in both the north and west side. These gaps were covered with a hollow square-profile steel tube and painted, which delayed discovering this before it became impractical to disassemble the structure and fill the gaps. Insufficient overlaps of roof planks over the north and west sidewalls also introduce additional weak points. Considering these aspects and future plans to operate with higher beam currents above 100 nA, it was decided to add a 66 cm-thick layer of steel blocks to the north wall that are available on site. Since the steel block wall will not quite reach the top of the second roof panel layer, several options are being investigated. One
option consists of erecting vertical steel plates attached to the end of concrete roof panels on the north side and protruding over the top of the roof to limit roof shine into the North Annex storage area. FLUKA simulation for this option is presented in Figure 8 below. While it appears adequate, early results for another option, a custom concrete block spanning the iron wall and the north end of the roof, seem to provide a more effective protection.

Figure 8. Radiation levels in a YZ slice along beamline with source at HDIce

Building effect on scatter and skyshine
The results and discussion above indicate that radiation levels on the floor level surrounding UITF are not likely to exceed RCA levels during UITF operations. Most radiation leakage will occur through the thin roof and helium vent. The perimeter around the facility will remain under RCA levels, as illustrated in Figures 7 and 8.
White areas are below RCA levels.

The XY dose rate binning presented in Figure 9 again confirms levels below RCA on the work floor. A ray streaking from the western edge of the roof indicates leakage from insufficient roof overlap over the sidewall. Slightly below, the effect of the gap under the notched block is apparent. A coarser XY binning in Figure 10 shows the same slice extended over and beyond the boundaries of the building.

**Figure 10.** XY slice at HDIce target level – 1 m³ binning. White areas are below RCA levels

Decreasing the binning threshold to dose rates well below RCA level in Figure 11, it is apparent that there is a slight penumbra at the west wall, with isodoses curving first downwards and then lifting up, as one would expect.

**Figure 11. XY slice at HDeIce target level – 1 m3 binning. Range extended below RCA**

![Image of XY slice](image-url)


It appears that radiation on the work floor results from skyshine, scatter from the building walls and ceiling, and to leakage through side shielding. To evaluate the effect of scatter in the building, we performed two almost identical simulations, where in one of them the materials of the Test Lab walls and ceiling were changed to air. The results are presented in Figure 12.
While the comparison would benefit from a greater number of events scored, a slight enhancement of the radiation levels due to backscatter from the building is apparent in the right side of the figure. The white penumbra in lower right corner of the picture on the left exhibits higher levels when walls are present.

**Conclusions**

An unusual aspect of this project was that at times construction preceded design, due to reuse of an earlier facility and need to quickly use or reuse available shielding material. First shielding estimates by semi-empirical methods were typically conservative compared to Monte Carlo, within a factor of 3 to 5 for side or roof shielding, and even more so in the forward direction. FLUKA Monte Carlo simulations were useful in estimating radiation levels for complex geometries. Facility commissioning is likely to occur by mid-2019, when we expect to have opportunities for benchmark measurements to verify shielding calculations done so far.

**Acknowledgements**

This work was sponsored by the Department of Energy under contract number DE-AC05-06OR23177. The author wishes to thank colleagues Pavel Degtiarenko, Matt Poelker and Tom Renzo for helpful input and discussions.
References


Degtiarenko, P. (2015), Geant-3 source terms, private communication not available to the general public.


Radiological aspects of LCLS-II injector and early injector commissioning

J. Blaha*, S. Rokni, M. Santana
Radiation Protection Department, SLAC National Accelerator Laboratory, Menlo Park, CA, United States
*jblaha@slac.stanford.edu

The LCLS-II (Linac Coherent Light Source II) facility at SLAC National Accelerator Laboratory will further expand the X-ray Free Electron Laser science programme in the US and worldwide. The facility will use a new superconducting electron accelerator that will be operated at up to 4 GeV and 100 kHz (with capability up to 1 MHz). The construction of LCLS-II has begun and the first light is planned for 2020. In parallel with the construction, the injector gun source will be commissioned and tested independently prior to the rest of the machine. This project, called early injector commissioning (EIC), is one of the milestones of LCLS-II.

The injector gun is designed to provide a continuous wave electron beam for the superconducting RF linac. It consists of the RF gun, drive laser, RF buncher, low-energy beam transport and Faraday cup. The entire system will be operated remotely from the SLAC accelerator control room, for ease of testing and validation of the control system for LCLS-II.

The gun and associated services has been installed inside the existing accelerator enclosure, originally built for the SLAC 2-mile linac. The complete radiological analysis has been conducted to upgrade the shielding for the injector vault. This analysis considers multiple beam loss scenarios, including high energy electrons (GeV range), due to field emission of superconducting cavities, that might be transported back to the vault. In addition, temporary shielding has been designed and built to separate the EIC from the rest of the accelerator enclosure. This allows for the installation of the cryomodules in an area downstream of the gun during EIC operation, which will run for almost one year. Moreover, the accuracy of modelled events has been validated experimentally (radiation monitors installed inside and outside of the injector vault, and radiological surveys). Finally, the project allows for characterisation of radiation detector prototypes, which must operate in radiation fields caused by 1 MHz beams, and which were especially developed for the LCLS-II beam containment system.

This work was supported by US Department of Energy contract DE-AC02-76SF00515.
Preliminary shielding design of the NSLS-II High-Energy Engineering X-ray Scattering (HEX) beamline enclosures

M. Benmerrouche* and C. Sunil
National Synchrotron Light Source-II, Brookhaven National Lab, Upton, NY 11973-5000, US
*benmerrouche@bnl.gov

The HEX beamline is a new beamline currently being designed at NSLS-II that will provide a suite of most advanced hard X-ray capabilities required for engineering-scale material studies using hard X-ray diffraction, scattering and imaging. HEX beamline will be approximately 100 metres long and consist of three independently-operating branches: (i) side branch, (ii) centre branch consisting of two experimental enclosures, and (iii) white-beam branch. The centre and white-beam branches are white-beam compatible. The HEX facility will include six beamline enclosures of which one concrete end station will be located in a satellite building. The five remaining shielding enclosures will be installed inside the NSLS-II experimental hall and will be made of lead. Currently the highest critical energy at NSLS-II is 11 keV provided by a damping wiggler of 1.85T. A superconducting wiggler with 70mm period (SCW70) and a nominal magnetic field of 4.3T with a critical energy of 25.7 keV will be used as the source of synchrotron radiation for the HEX beamline. The radiation shielding design of the white beam enclosures must consider (a) the primary gas bremsstrahlung (GB) created in the straight section of the SCW70 due to electron beam interactions with the residual gas molecules inside the storage ring vacuum chamber; (b) scattered bremsstrahlung generated due to the interaction of the primary GB with the beamline components such as masks, collimators and mirrors; (c) photoneutrons generated due to the interaction of primary GB with the beamline components; and (d) scattered synchrotron radiation (SR) from beamline optical components. The FLUKA Monte Carlo code is used to simulate GB interactions with beamline components while the STAC8 code is used to estimate the dose rates due to SR. Since the HEX beamline components are still in preliminary design stage and have not been finalised, the preliminary shielding analyses are based on standard scatter targets to mimic components that are typical in a beamline such as masks, mirrors, monochromators and beam stops. Copper and silicon targets with different dimensions placed at various locations inside the beamline enclosure were used for GB simulations. For the SR calculations, an optimum scatter target was used to maximise scattering and build-up factors in the shielding material were considered.

Introduction

The National Synchrotron Light Source II (NSLS-II) facility is a 3-GeV 500 mA third generation synchrotron radiation light source of high spectral brightness and low horizontal and vertical emittances, capable of accommodating approximately 60 beamlines. NSLS-II began routine operation in 2014 with seven beamlines that included a Damping Wiggler, Elliptically Polarized Undulator and In-Vacuum Undulators that were part of the NSLS-II construction project. Currently there are 28 beamlines operating in the top-off mode. The High Energy Engineering X-ray (HEX) beamline is a new beamline funded by the New
York State Energy Research and Development Association (NYSERDA) and is currently in the preliminary design stage. This beamline will be used for in situ and in-operando studies of sustainable energy materials and devices. To penetrate thick metallic samples, the X-rays will be of high energy. The scope of the project includes a superconducting wiggler optimised for high energies, any necessary modifications to the NSLS-II storage ring and the Front End inside the storage ring tunnel. Outside the tunnel, the beamline comprising of the optics, diagnostics, safety components and sections of shielded transport pipe will pass through a series of lead enclosures to an external satellite building housing an experimental station within a concrete bunker (approx. 20m x 5m wide). The initial scope will include the single central branch running to the satellite building. The beamline, in its mature configuration will have three separate branches.

The radiation shielding analysis of the beamline is carried out to determine the minimum required thickness of the beamlines enclosures. Figure 1 shows the layout of the 27ID-HEX beamline enclosures.

Figure 1. Layout of the HEX beamline enclosures in the horizontal plane


**Radiation shielding methodology**

Under normal operating conditions the radiation shielding design of the HEX enclosures must consider the following as sources of radiation.

- **Bremsstrahlung** generated when the electrons in the stored beam interact with residual gas molecules in their path. This is usually termed as gas bremsstrahlung (GB) and in the case of HEX beamline it is generated in the straight section of 27ID.

- **Scattered bremsstrahlung** generated by the interaction of the primary GB with the beamline components such as masks, collimators and mirrors.

- **Photoneutrons** generated by the interaction of primary GB with the beamline components.

- **Scattered SR** from beamline optical components.

The enclosures must be designed to reduce dose rates during normal operation to less than 0.05 mrem/h (0.5 μSv/h). Assuming occupancy of 2000 hours per year, this will keep annual dose to staff and users to less than 100 mrem (1 mSv). For any individual not involved in the operation of the NSLS-II facility, the annual dose must be kept below 25 mrem (250 μSv) considering the occupancy factor. The latter dose requirement affects areas outside the satellite building where personnel not involved in the operation of NSLS-II may occupy. The FLUKA Monte Carlo code (Böhlen, 2014; Ferrari, 2005) is used to simulate spatial distribution of dose rates due to GB interactions with beamline components while the analytical code STAC8 (Asano and Sasamoto, 1994) is used to estimate the dose...
rates due to SR interactions using optimal scatter targets. Since the HEX beamline components layout and ray tracing drawings have not yet been finalised, the shielding analysis is carried out using a generic approach with standard scatter targets to mimic the components that are typical in a beamline such as masks, mirrors, monochromators and beam stops. This paper consists of two parts, one dealing with the GB as the source and the other with the scattering of the SR. Both are constant sources of radiation and the dose rates that are present represent the normal operating conditions.

Gas bremsstrahlung shielding analysis

The shielding analysis for the GB has been carried out using the FLUKA Monte Carlo code. The source of the primary GB radiation is $1/E$ energy spectrum that extends up to 3 GeV and sampled with the FLUKA source routine. The full GB power is assumed to be incident on target. The total ambient dose equivalent rates are estimated in a three-dimensional mesh (USRBIN in FLUKA) by scoring the fluence and folding them with the fluence to ambient dose equivalent conversion coefficients. All the dose rates results given herein are on contact with the enclosure walls (and roof) in units of mrem/h. The storage ring parameters used for the GB shielding analysis are given in Table 1.

Table 1. Storage ring parameters used for the GB shielding analysis

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Electron Beam Energy</td>
<td>3 GeV</td>
</tr>
<tr>
<td>Stored beam Current</td>
<td>500 mA</td>
</tr>
<tr>
<td>Average pressure in the straight section</td>
<td>$1 \times 10^{-9}$ torr</td>
</tr>
<tr>
<td>Length of the straight section</td>
<td>15.5 m</td>
</tr>
</tbody>
</table>


The 27ID-HEX beamline shielding enclosures are rectangular except for the First Optical Enclosure (FOE) where a slanted section of its inboard wall is defined by a storage ring accelerator enclosure. The length of the enclosures varies between 3 m and 19.5 m (inner dimensions not including the shielding thickness). All white beam enclosures have their lateral walls located at least 1.5 m from the beam centerline except for 27ID-C outboard lateral wall, which is located at 0.5 m from beam centreline. The roof of the lead enclosures (27ID-A to 27ID-E) is positioned at 2.09 m from beam centreline and at 1.8 m for the concrete enclosure. The top and side views of the FLUKA model for the 27ID-A enclosure is shown in Figure 10.2 whose length is approximately 18 m. A white beam stop (WBS R=2 cm, L=3cm) made of copper followed by a lead bremsstrahlung stop (BRS R=10cm, L=30cm) were placed 50 cm upstream from the downstream wall. In this configuration, the WBS/BRS is of little use in stopping the scattered bremsstrahlung from upstream targets. The scatter targets considered in the GB simulation include copper block and a silicon crystal. In Figure 10.2 the copper target (R=2cm, L=3cm) is positioned at 10 cm from the upstream wall of the enclosure and the silicon target (4cm x 4cm, 20 cm long) positioned in the middle of the enclosure. The results of GB simulation shown in this report correspond to this configuration because it gives the highest ambient dose equivalent rate on the downstream wall of the enclosure. For the lateral walls and the roof, the highest ambient dose rates were obtained with a Cu target that is approximately 20 cm long as well as when GB directly strikes the WBS. Various configurations were simulated that included Cu and Si targets placed at different locations inside each beamline enclosure, varying the length of the Cu target and using either short or long Si targets with different orientations to the beam direction. The thickness of the downstream wall and combination of additional local shielding were also varied. Figure 2 inset shows a downstream wall 10 cm thick reinforced
with additional shielding comprising of a 2 cm Pb guillotine (40 cm x 40 cm) on the inside of the enclosure and on the outside 3 cm Pb plate (140 cm x 140 cm) followed by 10 cm polyethylene plate (140 cm x 140 cm).

**Figure 2. FLUKA geometry for the 27ID-A (FOE) enclosure (top and side views)**

In this configuration, the total ambient dose rates in and around the 27ID-A enclosure are given in Figure 3 and the corresponding neutron ambient dose rates are given in Figure 4. The total ambient dose rates on the lateral wall and roof are found to be less than 0.05 mrem/h with neutrons contributing more to the dose rates. Figure 5 (a) and (b) gives the top and side view of the total ambient dose rate away from the downstream wall. Figure 5 (c) and (d) shows the total ambient dose rates on contact with the lead plate and polyethylene plate respectively. The left most plot shows the shielding configuration on downstream wall. As seen from Figure 5 (c) and (d), the ambient dose rates in areas accessible to personnel are below 0.05 mrem/h. Figure 6 shows the total ambient dose rates outside the downstream wall of 27ID-A (FOE) when the primary GB strikes individual components WBS/BRS, Si and Cu. The ambient dose rates are below 0.05 mrem/h with Cu as the worst-case target.
Figure 3. Total ambient dose equivalent rates in and around the FOE when the primary GB strikes the Cu target located at 10cm from the upstream wall with a Si target located in the middle of the enclosure.

The top figure shows the horizontal view \((y=0.0)\) and the bottom figure shows the vertical view \((x=0.0)\).


Figure 4. Same as Figure 3 showing only the neutron ambient dose equivalent rates in and around the FOE.

Figure 5. Total ambient dose equivalent rates on the exterior of the downstream wall of 27ID-A (FOE)

The additional shielding configuration on the downstream wall is shown in the leftmost plot.

Figure 6. Total ambient dose equivalent rates on the exterior of the downstream wall of 27ID-A (FOE) with primary GB directly striking WBS/BRS (top figure), Si target (middle figure) and Cu target (bottom figure)

The GB and SR also enters the white beam enclosure 27ID-C. The shielding configuration for the GB analysis uses a 3 m long enclosure with 5 cm thick Pb downstream wall (no apertures), 25 mm thick Pb inboard lateral wall located at 1.5m from beam axis, 25 mm thick Pb outboard lateral wall (reinforced with 6 mm Pb belly band at ± 50 cm around beam centerline) located at 0.5 m from beam axis and 22 mm thick Pb roof at 2.09 m above beam axis. The downstream wall is reinforced with a 7 cm Pb guillotine (80 cm x 80 cm) on the inside of the enclosure and on the outside 3 cm Pb plate (120 cm x 120 cm) followed by 10 cm polyethylene plate (120 cm x 120 cm). In this configuration, the total ambient dose rates in and around the 27ID-C enclosure are given in Figure 7. It shows that the ambient dose rates exceed the criteria for upstream wall, outboard wall and in a small area near the exterior surface of the downstream wall.

Figure 7. Total ambient dose equivalent rates in and around the 27ID-C enclosure when the primary GB strikes the Cu target located at 10cm from the upstream wall with a Si target located in the middle of the enclosure.

The top figure shows the horizontal view (y=0.0) and the bottom figure shows the vertical view (x=0.0).

Figure 8. Total ambient dose equivalent rates on the exterior of the downstream wall of 27ID-C

The additional shielding configuration on the downstream wall is shown in the leftmost plot.


Figure 9. Total ambient dose equivalent rates on the exterior of the outboard lateral wall without/with additional polyethylene shield (a and b) inboard lateral wall (c) and roof (d) of 27ID-C

Figure 8 (a) and (b) give the top and side view of the total ambient dose rate outside the downstream wall. Figure 8 (c) and (d) show the total ambient dose rates on contact with the lead plate and polyethylene plate respectively. The left most plot shows the shielding configuration on downstream wall. As seen from Figure 8 (c), the dose rates outside the lead is approximately 0.08 mrem/h which slightly exceeds the dose criteria. However, spatially this is very localised and drops to below 0.05 mrem/h at the surface of the polyethylene plate as seen in Figure 8 (d). The dose rates on the downstream wall will be re-evaluated once the beamline components are finalised. The total ambient dose rates on the lateral walls and roof are shown in Figure 9. The ambient dose rates are less than 0.05 mrem/h for the inboard lateral wall and the roof. However, it exceeds 0.05 mrem/h on contact with outboard lateral wall due to its shorter distance to the beam centreline of 0.5 m compared to inboard wall which is at 1.5 m. A 10 cm polyethylene belly band (±50 cm around beam centreline) reduces the ambient dose rate to below 0.05 mrem/h within ±50 cm around beam centreline but no so much outside this area. The polyethylene belly band will need to be extended beyond ± 50 cm. Instead of a belly band, it is more effective to place the shield near the scatter target. Once all the beamline components dimension and position inside 27ID-C enclosure are finalised, the appropriate local shields will be designed to keep ambient dose rates below 0.05 mrem/h on outboard lateral wall and upstream wall of the enclosure.

The white beam enclosures 27ID-D and 27ID-E also have both GB and SR entering the enclosure. The shielding configuration for the GB analysis uses a 6 m long enclosure with 5 cm thick Pb downstream wall (no apertures), 25 mm thick Pb lateral walls located at 1.5 m from beam axis and 22 mm thick Pb roof at 2.09 m above beam axis. The downstream wall is reinforced with a 7 cm Pb guillotine (80 cm x 80 cm) on the inside of the enclosure and on the outside 3 cm Pb plate (140 cm x 140 cm) followed by 10 cm polyethylene plate (140 cm x 140 cm). In this configuration, the total ambient dose rates in and around the 27ID-D/E enclosure are given in Figure 10. It shows that the ambient dose rates criteria are slightly exceeded for upstream wall and in a small area near the exterior surface of the downstream wall. However, the upstream wall of 27ID-D is shared with 27ID-C (see Figure 1) which will be unoccupied by personnel when beam is allowed into the enclosure. In addition, the upstream wall of 27ID-E is shared with 27ID-D (see Figure 1). Figure 11 (a) and (b) give the top and side view of the total ambient dose rate away from the downstream wall. Figure 11 (c) and (d) show the total ambient dose rates on contact with the lead plate and polyethylene plate respectively. The left most figure shows the shielding configuration on downstream wall. As seen from Figure 11 (c), there is small amount of radiation leakage just outside the lead plate that amounts to approximately 0.09 mrem/h which slightly exceeds the dose criteria. However, the spatial extent of this dose rate is localised and drops to below 0.05 mrem/h at the surface of the polyethylene plate as seen in Figure 11 (d). Once all the beamline components are finalised the dose rates on the downstream wall will be re-evaluated.
Figure 10. Total ambient dose equivalent rates in and around the 27ID-D and E enclosures when the primary GB strikes a Cu target with a Si target located in the middle of the enclosure.

The top figure shows the horizontal view (y=0.0) and the bottom figure shows the vertical view (x=0.0).


Figure 11. Total ambient dose equivalent rates on the exterior of the downstream wall of 27ID-D and E enclosures.

The additional shielding configuration on the downstream wall is shown in the leftmost figure.

The white beam enclosure 27ID-F has both GB and SR entering the enclosure. The shielding configuration for the GB analysis used a 19 m long enclosure with walls and roof made of Portland concrete (standard construction material) at 2.3 g/cm³ density. The downstream wall is 100 cm thick with no apertures and the upstream wall is 70 cm thick. The inboard and outboard lateral walls are 70 cm thick and are located at 3.5 m and 1.5 m from beam axis respectively. The roof is 60 cm thick at 1.8 m above beam centreline. The downstream wall is reinforced with a 5 cm Pb plate (130 cm x 130 cm) on the inside of the enclosure. In this configuration, the total ambient dose rates in and around the 27ID-F enclosure are given in Figure 12. Clearly the ambient dose rate on the lateral walls, upstream wall and roof are below 0.01 mrem/h. Figure 13 (a) and (b) give the top and side view of the total ambient dose rate away from the downstream wall. Figure 13 (c) and (d) show the total ambient dose rates on contact with the exterior surface of the wall and 1 m away respectively. They are approximately 0.02 mrem/h on contact and below 0.01 mrem/h at 1 m away from the wall.

Figure 12. Total ambient dose equivalent rates in and around the 27ID-F enclosure when the primary GB strikes a Cu target with a Si target located in the middle of the enclosure

The top figure shows the horizontal view (y=0.0) and the bottom figure shows the vertical view (x=0.0).

Figure 13. Total ambient dose equivalent rates on the exterior of the downstream wall of 27ID-F enclosure

Synchrotron radiation analysis

The 27-ID (HEX) beamline is a superconducting wiggler (SCW70) source and the calculations presented here are based on the following accelerator and insertion device parameters.

- Electron Energy: 3 GeV;
- Stored beam Current: 500 mA;
- SCW70:
  - Period Length: 70 mm;
  - Number of Poles: 29;
  - Max magnetic Field: 4.5 T.
- Horizontal Fan entering the enclosures:
  - 27ID-A, 27ID-C, 27ID-D and 27ID-E: 1.4 mradH;
  - 27ID-B: 0.2 mradH;
  - 27ID-F: 1.0 mradH.

The SCW70 will operate at a nominal field of 4.3T and assuming a 4.5T field is conservative in nature allowing for an over-field condition. The SCW70 spectrum calculated with STAC8 using the above parameters is given in Figure 14, which agrees well with the calculated spectrum using a separate code SRW (Chubar, 1998).

Figure 14. SCW70 photon spectrum as calculated by STAC8 and SRW codes

The STAC8 code was used to calculate the ambient dose equivalent rates in the occupied areas outside of beamline enclosures. The build-up factor in shield was included in the calculation. The results for the shielding analysis for white beam lead enclosures (27ID-A, C, D, E) is summarised in Figure 15 for (a) 25 mm lead lateral wall at 1.5 m from beam centreline (b) 31 mm lead lateral wall at 0.5 m from beam centreline (c) 22 mm lead thick roof at 2.1 m from beam centreline and (d) 50 mm lead downstream wall 1 m from scatter.
target. It shows that a lateral wall that is at least 1.5 m from the beam centreline require a thickness of 25 mm Pb to keep the ambient dose rates below 0.05 mrem/h, shown as purple line in Figure 10.15. For the lateral wall that is 0.5 m from the beam centreline as in the case of 27ID-C enclosure, the Pb thickness increases to 31 mm. Figure 15 (c) shows that a minimum lead thickness of 22 mm is required for the roof of the white beam enclosures to keep the ambient dose rates below 0.05 mrem/h. The results for the downstream wall are shown in Figure 15 (d). A thickness of 5 cm of lead will keep the ambient dose rates below 0.05 mrem/h, except for forward scattering angles below approximately 20 degrees. Below 20 degrees, the thicker 10 cm Pb downstream wall of 27ID-A enclosure and additional shielding on the downstream wall of 27ID-C, D, E enclosures which consist of a 3 cm Pb plate and 7 cm Pb guillotine will reduce ambient dose rates below 0.05 mrem/h in the forward direction.

For the concrete enclosure 27ID-F, the SR calculation is carried out using an optimum scatter target assumed at 150 cm from the lateral wall and 1 m from the downstream wall. The results of the calculations are shown in Figure 16 for (a) lateral wall ignoring build-up in the 700 mm thick concrete (red trace) and (b) downstream wall – the purple plot with the (+) sign corresponds to 100 cm concrete and 10 mm Pb. The results indicate that at least 70 cm of concrete is required for the outboard lateral wall and 100 cm of concrete for the downstream wall with a local lead plate to cover small angle scattering. Figure 16 (a) shows that the ambient dose rates are underestimated by over two orders of magnitude without build-up factors in the concrete stressing the importance of build-up in the concrete.

**Figure 15. Photon dose rates on contact with the white beam enclosure walls and roof as a function of the scattering angle**

This shielding should keep ambient dose rates in accordance with NSLS-II shielding policy for staff occupying areas inside the experimental floor and to any individual not involved in the operation of the facility (e.g. maintenance workers) occupying areas outside the satellite building. The requirement for the roof is 60 cm thick concrete assuming a 10% occupancy on the roof of the satellite building (lower dose rates for building maintenance staff) and 20% occupancy on the roof of the 27ID-F enclosure for NSLS-II staff. It is planned to restrict access to the enclosure roof since any equipment requiring routine maintenance will not be located on the roof. The shielding requirements for the inboard lateral wall and upstream wall are 64 cm and 65 cm respectively but it is planned to construct these with the same thickness as the outboard lateral wall.

The monochromatic beam enclosure 27ID-B has only SR entering the enclosure. The dose rates outside the 27ID-B enclosure when the monochromatic beams are scattered from an optimal target are discussed here. The scatter target is assumed to be located at 1 m as the closest distance from the lateral wall, 1m from the downstream wall, and 2 m from the roof. The ambient dose rates were calculated for the 4 operational energies of the 27-ID-HEX beamline corresponding to 39, 64, 75 and 117 keV, including higher harmonics and assuming any mirrors downstream from the side bounce monochromator are out of the beam, which is more conservative. As an example, results for the radiation shielding analysis for the lateral and downstream wall are given in Figure 17(a) for 10 mm Pb thick...
lateral wall at 1.0 m from beam centreline and Figure 17(b) for 30 mm Pb thick downstream wall at 1.0 m from scatter target. It shows that the lateral wall that is at least 1.0 m from the beam centreline requires a thickness of 10 mm Pb. Similarly, the minimum required thickness for the roof was calculated to be 10 mm Pb. For the downstream wall, a thickness of 30 mm of lead will keep the ambient dose rates below 0.05 mrem/h except for forward scattering angles below approximately 10 degrees. Below 10 degrees the monochromatic beam (30 cm x 30 cm) that is 35 mm thick will keep ambient dose rates below 0.05 mrem/h in the forward direction.

**Summary**

A detailed preliminary shielding analysis has been carried out for the new NSLS-II 27ID-HEX beamline that is currently in the design stage. Since the beamline components layout and ray tracing drawings have not yet been finalised, the shielding analysis followed a generic approach using standard scatter targets. The shielding requirement specified should keep the radiation dose to personnel as low as reasonable achievable and in compliance with the NSLS-II shielding policy. The following points are important for the final design of the HEX beamline.

- Shielding requirements for the downstream wall of both concrete and Lead white beam enclosures is primarily determined by GB scattering off beamline components.
- Shielding requirements for the lateral wall, upstream wall and roof is dominated by white beam synchrotron radiation scattering off beamline components.
- Additional neutron shielding may be required for thick scatter targets located near upstream wall or lateral wall.
- Build-up factors in the concrete shielding is very important and should be considered.
- The preliminary shielding requirements should be reevaluated once the beamline components and penetrations through the walls and roof are finalised – additional localised shielding may be required.
- Special attention should be given to penetrations needed for beam transport pipes and associated interface to the enclosure walls to minimise radiation leakage.

**Acknowledgements**

This research used resources of the National Synchrotron Light Source II, a US Department of Energy (DOE) Office of Science User Facility operated for the DOE Office of Science by Brookhaven National Laboratory under Contract No. DE-SC0012704. The authors wish to thank A. Broadbent, M. Lucas and Dr Z. Zhong for useful discussions and support during the shielding analysis simulations and calculations.

**References**


Radiation shielding analysis of the SST beamline at NSLS-II

Sunil C1* and M. Benmerrouche1
1National Synchrotron Light Source-II, Brookhaven National Lab, Upton, NY 11973-5000, US
*schitra@bnl.gov

The Spectroscopy Soft and Tender (SST) is a National Institute of Standards and Technology (NIST) partner beamline at NSLS-II, capable of carrying out X-ray Photoelectron Spectroscopy and Near Edge X-ray Absorption Fine Structure Spectroscopy. It has two branches that are powered by an out of vacuum Elliptically Polarized Undulator (EPU60, SST-1 branch) and an out of vacuum Undulator (U42, SST-2 or Tender branch). The layout consists of the Front End (FE) components (slits, masks, collimators etc.) inside the ring, First Optical Enclosure (FOE), two lead shielded pink beam transport (PBT) pipes, 15 mirrors (3 inside FOE) and 7 experimental end stations (EES). The pink beams are assayed outside the FOE by a Plane Grating Monochromator (PGM in SST-1) and a Double Crystal Monochromator (DCM in SST-2). By a set of mirrors and transfer branches, the monochromatic beam from SST-1 can also be brought to the two EES of SST-2 either separately or simultaneously. The radiation safety aspects of this beamline are discussed for the top off radiation safety as well as for gas bremsstrahlung (GB) and the synchrotron radiation (SR) sources. The FLUKA Monte Carlo code is used to simulate the top off radiation safety accidents and the GB source while the STAC8 code is used to estimate the dose rates due to SR. For top off radiation safety, the beam trajectories and the safe end points for the injected electrons (45 nC/m) are established by particle tracking and are reproduced in the FLUKA simulations. With the FE and FOE components in the geometry, 18 scenarios are analysed. The ray traces are used to identify the beam incident points for GB and SR. For the GB simulations, the spectrum (17µW for 15.5 m long straight section and Intorr pressure) is transported through the FOE and the PBT pipes complete with all important components for 17 incident points. The scattering of the SR from various components starting from the FOE up to the EES are analysed with the white beam spectra from EPU60 and U42 calculated by the SRW code as inputs. The reflections from the mirrors are considered at their lowest possible angles. Mis-steered beams are identified from ray traces and are also analysed. For monochromatic beams, the highest fundamental energies and their harmonics with bandwidths are used after the DCM/PGM.

Introduction

The National Synchrotron Light Source -II (NSLS-II) is a 3 GeV 500 mA third generation synchrotron radiation light source of high spectral brightness and low horizontal and vertical emittances with the capability to accommodate about 60 beamlines. The 07-ID Spectroscopy Soft and Tender (SST) is a National Institute of Standards and Technology (NIST) partner beamline with 17 mirrors and 7 experimental end stations (EES) capable of carrying out photoelectron spectroscopy, absorption spectroscopy including microprobes, nanoprobeS, and X-ray absorption near edge structure X-ray spectroscopy, imaging and scattering. The beamline is powered by two Undulators (U42 and EPU60) for the two beamlines; Soft (also known as the M branch) and Tender (L branch). Figure 1 shows the
layout of the beamline with the blue line depicting the L branch (tender) and the green showing the M (soft) branch.

Figure 1. The SST beamline layouts

![Diagram of SST beamline](image)


The L branch is powered by the U42 out of vacuum undulator while the EPU60 (Elliptically Polarized Undulator) caters to the M branch. The beamline branches inside the FOE and the pink beams are transported up to the Double Crystal Monochromator (DCM) in the L branch and the Plane Grating Monochromator (PGM) in the M branch, both outside the FOE on the experimental floor. Downstream of the PGM, the beam can be diverted to the M branch end stations or, by a set of mirrors and transfer branches, to the two EES of the L branch (tender) also. SST-1 can deliver X-rays from 100 eV to 2.2 keV with a tunable focus spot from 2cm to 10µm. SST-2 can deliver X-rays ranging from 2.0 keV to 7.5 keV and a fully tunable focus spot from 10-50µm. The dual beam arrangement lets the experimental end stations in SST-2 receive X-rays from 100 eV to 7.5 keV at a single spot in a single experiment.

Figure 2 shows the Front End (FE) of the beamline that is situated inside the storage ring. The components marked in red are the radiation safety components that are modelled for the Top Off radiation safety analysis using the FLUKA Monte Carlo radiation transport code (Böhlen et al., 2014; Ferrari et al., 2005). These are the fixed mask (FM), lead collimators (LCO), two sets of slits (SLT) each set containing a horizontal and vertical slant that intercept the synchrotron fan when closed, the two lead safety shutters and the ratchet wall complete with collimator containing lead and polyethylene and concrete. Figure 3
shows the First Optical Enclosure (FOE) and the optical and the radiation safety components that are also used in the FLUKA simulations. The major components inside the FOE include the dual aperture burn-through (BT) device and the fixed mask (FM1) to let the beams from the two undulators into the FOE. These are followed by L1 (first mirror in the L branch), a single aperture secondary bremsstrahlung lead shield (SBS-1), M1 (the first mirror in the M branch), white beam stop (WBS) a lead secondary bremsstrahlung shield without any aperture (SBS2), a dual aperture lead secondary bremsstrahlung shield (SBS2), FM4, FM2, a tungsten bremsstrahlung stop(BST), the photon shutters PSH1 and PSH4 and the lead guillotine(GULL). The white beam stop has two inclined copper blocks that stops the white beams when the mirrors (M1 or L1) are retracted with an intermediate region that allow the beams to pass through when they are inserted. Inside the FOE, the L branch has two mirrors (L1 and L2, both at 0.53° minimum angles) while the M branch has one (M1 at a minimum angle of 1.5°). The FOE hutch has lead shields 18 mm on the side wall, 50 mm on the downstream wall and 10 mm on the roof. The lead guillotine on the downstream wall is 10 cm thick and has dual apertures to let the 2 beamlines. Outside the FOE, the 8 mm lead shielded on 2 mm SS beam pipe stops before the PGM and the DCM with bellows and gate valves in between. The L branch has two and the M branch has five experimental end stations (ESS).

**Figure 2. Layout of 07ID-SST front end showing major components**

![Figure 2. Layout of 07ID-SST front end showing major components](source: Brookhaven National Laboratory, 2020)

Every beam line in NSLS-II undergoes a comprehensive Instrument Readiness Review (IRR) which is a multipronged safety analysis covering not just radiation safety but other
types of safety analysis also such as electrical, vacuum, laser, cryogenic etc. The radiation safety analysis of the beamline is carried out for accidental and normal operating conditions. Under the accidental condition, the top off accident is considered for the simulation while gas bremsstrahlung (GB) and synchrotron radiation (SR) sources are considered for normal operating conditions. Some accidental SR conditions are also considered such as mis-steered beams.

The NSLS-II shielding policy (BNL (2014a) dictates that the maximum ambient dose equivalent rate under normal operating conditions shall be less than 0.5 mrem/h (5.0 μSv/h) (based on an occupancy of 2000 hour per year) for areas continuously occupied by an individual involved in the operation of the facility. However, NSLS-strives to keep the dose rates below 0.05 mrem/h (0.5 μSv/h) based on the shielding guidelines (BNL, 2014b) so that the users can access the experimental floors without a personal monitoring device such as a TLD. All beamline shielding calculations are carried out accordingly. The policy further stipulates that the dose rates shall not exceed 100 mrem/h (1.0 mSv/h) in the case of an abnormal condition. If the shielding calculations show that the dose rates in an area exceed 100 mrem/h, then an interlocked Area Radiation Monitor (ARM) is used as a credited control to keep dose rates below 100 mrem/h. If the dose rates estimated are higher than 2000 mrem/h (2.0 Sv/h), a second independent system is required to mitigate the resulting radiological impacts. This could be a second ARM or an Accumulated Charge Monitor Interlock (ACMI) that limits the integrated charge.

The primary radiological safety concern for Top-Off injection, with the FE safety shutters (SS1 and SS2 in Figure 2) open, is the scenario where the injected electrons are transported down to the Front End (FE) of beamline due to an erroneous combination of lattice magnetic field settings and beam energy. The front end is the part of the beam line that is situated inside the storage ring. These primary electrons will interact with the FE components resulting in secondary radiation, which will stream in to the FOE and interact with the components that may lead to potentially high dose rates outside the FOE walls.

The Top-Off and GB analyses are carried out using the FLUKA Monte Carlo radiation transport code while the analysis for SR is carried out using the STAC8 code (Asano and Sasamoto, 1994).

**Top-off radiation safety analysis**

At NSLS-II, the Top-Off Safety System ensures that the injected electron beam does not channel down the FE and into the beamline FOE, by ensuring their capture at the designated safe end points in the FE. Backward particle tracking by the accelerator physicists establishes the safe end points in the FE and the beam trajectory. In the case of 07-ID beamlines, these are deemed to be ±2 mm away from the apertures of the LCO2. The end points could be different if a mirror is present in the FE. The electron beam starting position is assumed to be ±2 mm from the aperture of the Fixed Mask in the Front End. The FLUKA model of the beamline is shown as two figures for clarity. Figure 4 shows the geometry of the FE while Figure 5 shows the rest of the beamline (FOE and shielded beam transport pipes). The beam directions could also be from outboard of FM to inboard/outboard of the LCO2 giving rise to four scenarios, of which one is shown in Figure 6. The close-up views of the FM and the LCO2 are shown in insets. The radiation environments were estimated with three configurations of the slits. They are (a) all slits open (b) slits 1 and 2 open while slits 3 and 4 closed and (c) slits 1 and 2 closed while slits 3 and 4 open. Figure 7 shows one such arrangement.
Several scenarios were considered such as with and without the mirrors in the FOE, inserting and retracing all three lead secondary bremsstrahlung shielding (SBS-1, BSS and SBS-2) sequentially. However, all results shown here are without these lead components in the model and may seem counterintuitive. This is done so for practical reasons of separating the accelerator start up from the beamline operation. By excluding the lead shields in the FOE, the accelerator configuration control checklist that needs to be executed prior to start up need not include the beamline components making it a self-contained task. If the assessment of the top off radiation safety shows that the lead shields inside the FOE are required, they form a part of this checklist and requires its execution also before start-up of the accelerator.

Figure 4. Horizontal (at Y=0) and elevation view (in line with the ratchet wall collimator aperture) of the front end FLUKA geometry used in Top-off simulation

Figure 5. The horizontal (at Y=0, top figure) and the elevation view (bottom figure) of the FOE part of the geometry used in Top-Off simulation of the 07ID-SST beamline


Figure 6. Beam direction defined in FLUKA for the inboard-inboard case

Figures 8 shows results from two configurations. The results are normalised to a booster to storage ring injection charge rate of 45 nC/min and the total ambient dose equivalent rates are reported in units of mrem/h. In the figures, the top plot is the horizontal view while the bottom plot is the vertical view. For both the plots, the beam direction is the same, from the outboard side of the FM to the outboard side of the LCO2. In the plot on the left, the first two slits are closed and the mirrors in the FOE are inserted in the path of the SR beam. As can be seen, the dose rates outside the FOE and the storage ring bulk shield are all less than 100 mrem/h (1 mSv/h). On the right, slits 1 and 2 are now open while slits 3 and 4 are closed with the mirrors now retracted from the SR beam. The dose rates outside the FOE is more than 1 mSv/h by as much as 4-5 times. This scenario requires that an interlocked Area Radiation Monitor (ARM) be positioned such that it intercepts the streaming radiation and trips the machine to comply with the NSLS-II shielding policy.

Figure 8. Total ambient dose equivalent rates as estimated for two scenarios of top-off accident
Gas bremsstrahlung source

The GB source is modelled using the source subroutine. The spectrum was established by an earlier simulation for the long straight section and the GB power is assumed to be 17μW from 15.5 m straight section length, 10^{-9} torr pressure and 3 GeV electron energy. The beam loss points are established through a bremsstrahlung ray trace carried out by the beamline engineer. Figure 9 is a typical ray trace diagram showing the horizontal bremsstrahlung rays. The figure is anamorphic with the horizontal scale representing several tens of metres and the vertical scale showing about 0.5 m.

Figure 9. The horizontal bremsstrahlung ray trace showing the beam loss points are identified as white dots


In the analysis, the GB is assumed to be lost at a point at the various components in the FOE as numbered in the figure. While the GB is a fan with certain divergence, the assumption of a point loss provides some additional safety factor. In all 17 beam loss points were identified. Some are, inboard side of the burn through device, inboard side of the aperture of the FM1, upstream edge of L1 mirror, centre of M1 mirror and the outboard side of WBS. Figure 10 shows results from two such points. The top plots are horizontal views while the bottom ones are vertical views.
Figure 10. Total ambient dose equivalent rates for two different GB beam incident scenarios


On the left plot, the beam was incident on the inboard side of the burn through device, which is the first safety critical component on the beamline. On the right, the beam hits the FM2. The dose rates on the left are all well below 0.05 mrem/h and is acceptable as per the NSLS-II shielding guidelines while the right plot shows streaming radiation through the shielded beam pipes requiring additional shield. Since the streaming was limited to the inboard side as ascertained by a separate angular USRBN card, the shielding was limited to just that side of the beam pipe. In addition, some streaming can also be seen towards the inboard side of the downstream wall of the FOE, which was also noticed for another scenario where the beam was hitting the WBS. This required a 30cm (H) x 40 cm (V) lead shielding outside the FOE.

**SR scatter analysis**

The STAC8 code was used to analyse the dose rates due to the scattering of the SR beam. The SRW code was used to estimate the photon spectra that is generated in the undulators and trimmed through the FE masks before entering the FOE. This is used as the input spectra for all further calculations. Figure 11 shows the input as used for the STAC8 calculations.

Figure 11. The SR spectra from U42 and EPU60 entering the FOE

The lower cut off is kept at 1 keV to match the energy range considered by STAC8. The spectra are propagated through the mirrors that are upstream of the scatter targets. Mirrors can also be scatter targets and when they are considered so, their reflective properties are neglected. When they are considered as mirrors for reflection, the loss of reflectivity due to the inherent roughness is not considered. The angles of the mirrors are ascertained from the synchrotron ray traces including the nominal and minimum angles as well as all possible mis-steers. For example, in Figure 12 the beam can be seen to reflect from the M2 mirror in the PGM chamber of the M (Soft) branch. The mirror is mounted on a hexapod, which in principle can position it at any possible angle but is prevented from doing so by hard stops that are then kept under configuration control.

Figure 12. Possible reflections from M2 in the PGM chamber


The various scattering targets that were considered inside the FOE are white beams on mirrors and white beam stop, pink beams on the fixed masks and photon shutters. Outside the FOE on the experimental floor, pink beams can reach up to the pink beam stops and could scatter from components before it is stopped by the PBS. An important consideration is the loss of vacuum condition on the floor. While the beam transport pipes are shielded by 8 mm lead wrappings on the SS beam pipe, there are several locations where lead shielding is not present such as the bellows, which are very thin (typically 0.2 mm) SS sheets, pink beam stop chambers, and the PGM and DCM chambers. Downstream of the PBS, the beam is monochromatic, and the analysis is performed for all chambers and targets they encounter. Figure 13 shows the various scatter points considered for the SR analysis.

In the L branch, the beam is reflected by two mirrors (L1 and L2) before the DCM, both at minimum angle of 0.53°. In the M branch, the beam exits the FOE after reflection from before entering the PGM chamber. The PGM consists of a mirror M2 and the grating. If the grating is out of the beam path, the zero-order beam can in an accidental condition be technically transported out of the PGM chamber. All possible nominal and accidental scenarios are considered in both the beamlines individually before the end stations where both the beams are utilised and hence had dual source terms. From the loss of vacuum condition, for the beamlines on the experimental floor, additional SS shields are recommended for the 2 mm SS pipe before the PGM and the bellows up to the precision
slits in the M and transfer branches. For the L branch, additional SS shields are recommended for all the bellows before the pink beam stop.

**Figure 13. Scatter points considered for the SR analysis**


**Summary**

The engineering support for an efficient and accurate radiation transport analysis is important for geometry, beam conditions and to anticipate accidental scenarios. In this analysis, extensive use of such methods has been used starting from the component design, particle tracking in the storage ring and ray traces for both gas bremsstrahlung and synchrotron radiation. From the top off radiation safety analysis, the maximum radiation dose rate estimated at the exterior of the FOE is estimated at about 400 mrem/h (4 mSv/h) outside the downstream wall and 100-150 mrem/h (1.0-1.5 mSv/h) on the sidewall requiring an interlocked ARM on the sidewall of the FOE as an effective control point. From the GB analysis, additional shielding was recommended on the downstream wall of the FOE and on the shielded beampipe on the tender branch extending up to about the DCM chamber. From the SR scatter analysis, all bellows in the pink beam transport section were shielded with additional SS sheets while the monochromatic beam transport section was found to be comply with the NSLS-II shielding policy.

**Acknowledgements**

This research used resources of the National Synchrotron Light Source II, a US Department of Energy (DOE) Office of Science User Facility operated for the DOE Office of Science by Brookhaven National Laboratory under Contract No. DE-SC0012704. The authors wish to thank Dr Quackenbush for providing the subway style figure of the SST beamline, the beamline scientists Dr Fischer, Dr Jaye and Dr Weiland for useful discussions and support in preparation of the manuscript.

**References**


BNL (2014a), Photon Sciences Shielding Policy, BNL Document PS-C-ASD-POL-005.


Radiation environment studies for the Facility for Rare Isotope Beams at MSU

Dali Georgobiani1*, Mikhail Kostin1, Georg Bollen1
1Facility for Rare Isotope Beams (FRIB), Michigan State University, East Lansing, MI 48824 US
*georgobi@frib.msu.edu

The Facility for Rare Isotope Beams (FRIB), a project supported by the US DOE Office of Science, is in the final stage of construction at Michigan State University and beam commissioning has started. The project will use projectile fragmentation and induced in-flight fission of heavy ion beams from oxygen to uranium, at energies of 200 MeV/u and higher, and at a beam power of up to 400 kW, to produce rare isotope beams for nuclear physics experiments. The production of rare isotope beams during FRIB operations creates a high radiation environment. Radiation hazards at FRIB include prompt doses to workers and general public, production of radioactive nuclides, induced radioactivity, and radiation damage. The design and future operations of FRIB needs radiation transport calculations in many areas. Some examples of calculations and analyses are presented and discussed.

Introduction

The Facility for Rare Isotope Beams (FRIB) will be one of the world leading nuclear science facility based on the high power heavy ion accelerator in the world. Its key features are 200 MeV/u beam energy for uranium and higher energies for lighter beams, and 400 kW beam power for all ions. The facility will provide the means of separation of isotopes in-flight, with fast development time for any isotope; it will be suited for most elements and short half-lives, with fast, stopped and reaccelerated beams. The facility will be serving at least 1 400 users. The project is on track for early completion in 2021. Figure 1. shows a conceptual sketch of the facility.

The FRIB beam line starts with the double-folded linear accelerator (linac) where the beams are accelerated and led to the target hall, which accommodates the first section of the fragment separator including the rare isotope production target and the primary beam dump. There about 25% of the total beam power is lost in the target and another 75% is absorbed by the beam dump, creating a high radiation environment. Figure 2 shows the engineering sketch of the facility where the linear accelerator and the target facility are emphasised. The target facility accommodates the pre-target beam delivery system and the three-stage magnetic fragment separator for production and delivery of rare isotopes with high rates and high purities to maximise FRIB science reach (Figure 3). The front end of the fragment separator which includes the rare isotope production target and the primary beam dump is located in the target hall where all beam line components are installed in three vacuum vessels. The separator front end includes the rotating target for in-flight production of rare isotopes that are then magnetically separated and delivered to user experiments; and the rotating beam dump to intercept unreacted primary beam and fragments that are not needed. The target and the beam dump are the strongest radiation sources because heavy ion primary beams interact with them directly.
Figure 1. Facility for Rare Isotope Beams


Figure 2. Technical sketch of FRIB

Red frame: linear accelerator; blue frame: the target facility and last fragment separator section; green frame – experimental areas.
The red circle shows the location of the target hall, the area with the highest radiation environment.
Figure 3. Rare isotope production facilities

The fragment separator front end located in the target hall accommodates target and beam dump assemblies and magnets for beam focusing and separation. Beam line components are placed in the three vacuum vessels. Top shielding (multi-colour pieces) consists of cast iron blocks.


Radiation transport studies: Inputs and assumptions

Detailed radiation transport studies have been performed to guide shielding design and support safety of future operations. Adequate shielding is placed around the beam line components to ensure personnel and public protection. Radiologically bounding beams were used for design optimisation. Radiation transport model of the target hall was developed based on mechanical design of the facility. Figure 4 shows the radiation transport model of the preseparator front end. The model was created using the Visual Editor software (Schwarz et al., 2018) and it contains all the major components of the engineering design model. Note the removable cast iron shielding on top of the vessels. Some parts of the model were produced with the conversion software MCAM (Wu et al., 2009) that provides an interface between an engineering design model and the radiation transport model.

To evaluate radiation environment, we use Monte Carlo radiation transport codes PHITS, Version 2.76 (Sato et al., 2018) and MCNPX (Pelowitz, 2011). To calculate induced radioactivity, the PHITS module DCHAIN-SP is used. Capability of the models to transport ions in magnetic fields is important for our purposes: the magnet models are appropriately realistic, with magnetic field settings based on detailed ion optical calculations.
Figure 4. Radiation transport model: An elevation view of the preseparator front end along the beam line, showing beam line components, with cast iron shielding on top of the vacuum vessels.


Examples of radiation transport studies

We present three examples of radiation transport analysis. We discuss recent studies of soil and ground water activation around the target facility and show that drinking water safety standards are maintained. We present studies of component hands-on accessibility on top of the vacuum vessel shielding in the high radiation areas in the target hall. Finally, we address the tritium production in the target hall and its release at the facility, and show that, under current operation assumptions, the release levels of tritium during operations are below the project goals.

Groundwater activation

Beam losses during operations will result in radionuclide production in the soil surrounding the facility. This was studied earlier in sufficient detail to support design and civil construction of the facility. It was shown that radionuclide concentrations in drinking water in the soil surrounding the facility are well below the safety standards. Groundwater activation studies were performed by Kostin et al. (2014) in the soil next to the FRIB linac tunnel for a conservative, bounding proton beam of 200 MeV, 611 MeV, and 1 GeV in the subsequent linac segments, assuming 1 W/m losses in the linac tunnel. Groundwater activation calculations were also performed for the soil below the target facility for a conservative $^{18}$O beam at 637 MeV/u (upgrade energy assumption) and power of 400 kW. In the work presented here, we study the groundwater activation on the west side of the target facility, in the areas closest to the target and beam dump where most radiation energy losses occur and where the soil surrounding the facility is closest to the radiation sources (Figure 5). The goal of the analysis is to assess shielding adequacy for maintaining drinking water activation levels below the limits on the West side of the target hall, where the soil is located relatively close to the major radiation sources, i.e. the target and the beam dump. Additional side concrete shielding was put between the vacuum vessels containing these sources and the target hall outer wall. The use of concrete only in the calculation is conservative, as in the final design and realisation of this side shielding cast iron blocks were incorporated. This increases the shielding effectiveness, particularly against high-energy neutrons that are responsible for star production. Again, a conservative $^{18}$O beam at
637 MeV/u and at a full operational power of 400 kW was used to calculate the high-energy neutron flux in the soil.

The soil activation is assessed by means of determining the star density production rates in soil. A star is one inelastic nuclear interaction between high-energy hadrons and the nuclei of the materials. Energy threshold of a star is somewhat ambiguous. We use 20 MeV in this work. Hadrons having lower energies do not produce a significant amount of activation products because reaction threshold energies of important radionuclides (see below) are 20 MeV or greater. In this study, PHITS is used to calculate high-energy neutron fluxes in soil. These fluxes are converted to star densities (for detailed formalism, see, e.g. Cossairt, 2014). The obtained star density distribution is averaged over some volume and normalised to an operational year. These star density rates are then converted to specific concentrations of the dominant radionuclides $^3$H and $^{22}$Na using the Fermilab Radionuclide Concentration Model (Wehmann et al., 1997). It is worth mentioning that produced dominant nuclides depend on soil content and production mechanism. We use Fermilab soil composition for FRIB studies; the dominant nuclides in this soil, $^3$H and $^{22}$Na, are produced through spallation mechanism.

**Figure 5. Radiation transport model views across the beam line at the two worst radiation locations:**

A) Target module; B) Beam dump module

Beam direction is into the page. Yellow components are cast iron shielding blocks. Soil region (light blue) on the left of the West wall is studied. Note that in the actual design, the concrete side shielding on the West side of the vacuum vessels is also filled with cast iron shielding blocks. This makes the analysis conservative.


High-energy neutron fluxes are calculated inside the 2.2 m thick concrete target building wall with PHITS (Figure 6) shows a high energy neutron flux map at the beam dump location). Using the calculated neutron energy spectra inside the wall and a simple representative wall model, the transmission coefficient through the wall is found to be $10^{-3}$ in the worst case. This factor of 1 000 reduction is then used to estimate the high energy neutron flux in the soil outside the target hall wall. The approach is conservative; after passing through think concrete, neutrons in the soil have lower energies. The worst case situation is observed at the beam dump location. Assuming an interaction length of 34 g/cm$^2$ (see Cossairt, 2014), a soil density of 2.24 g/cc, and one operational year of 5 500 h, we arrive at a star density rate of $dS/dt = 10^8$ stars/cc/year in the soil at this location,
and $dS/dt = 10^6$ stars/cc/year at the target location. In both cases, this is the maximum star density next to the wall.

**Figure 6. High-energy neutron flux map at the beam dump location, an elevation view across the beam line**

![Neutron flux map](image)


We assess specific activities of the dominant nuclides: $C(^3\text{H}) = 0.24$ pCi/ml and $C(^{22}\text{Na}) = 0.005$ pCi/ml. These values are significantly lower than the maximum allowed concentrations in drinking water, that are, according to Environmental Protection Agency (EPA), $C_{\text{max}}(^3\text{H}) = 20$ pCi/ml and $C_{\text{max}}(^{22}\text{Na}) = 0.4$ pCi/ml, respectively. The regulatory requirements (EPA) stipulate that the sum of concentrations of all radionuclides of importance divided by the maximum allowed concentrations in drinking water, $\Sigma_i C_i/C_{i,\text{max}}$ must be less than or equal to 1. In the current analysis $\Sigma_i C_i/C_{i,\text{max}} = 0.024$ is obtained.

According to Kostin et al. (2014), average star densities in the soil next to the linac tunnel reach $3\cdot10^8$ stars/cc/year; this number represents star density rates averaged over the volume containing 99% of the stars. This is a common way of star density estimates to properly account for the mixing of the water in the soil. Similar analysis of soil activation under the beam dump location produced average star densities of $1\cdot10^7$ stars/cc/year.

For the linac tunnel (Kostin et al., 2014), estimates show that soil concentrations of the predominant radionuclides, $^3\text{H}$ and $^{22}\text{Na}$, are below the drinking water limits. In the work presented here, the calculated star densities are lower than the estimates for the linac soil. We evaluate concentrations of the major radionuclides and show that they are below the limits, as expected from comparing star density results in the linac soil and in the soil on the West side of the target facility.

One should point out that specific activities of $^3\text{H}$ and $^{22}\text{Na}$ in soil under the target facility below the beam dump were estimated at $C(^3\text{H}) = 0.59$ pCi/ml and $C(^{22}\text{Na}) = 0.012$ pCi/ml. In the linac soil, the worst case numbers (99% of activity; at saturation, similarly to this
work) were $C(\text{H}) = 9.54$ pCi/ml and $C(\text{Na}) = 0.22$ pCi/ml. All numbers are below the EPA limits.

The analysis shows that the vacuum vessel side shielding configuration on the West side, where the high radiation sources are closest to the soil outside the facility, provides adequate groundwater and soil protection. Major radionuclide concentrations in the soil are below drinking water limits.

**Component hands-on access during beam-off times**

We study hands-on accessibility of the components located above the vacuum vessel cast iron shielding during beam-off times to support future operations. Target and beam dump modules are the most activated, frequently moved components. Target will be replaced as often as every 2 weeks (a duration of an experiment), while the beam dump is expected to be replaced annually when full-power operation conditions are reached. We evaluate residual doses on top of the cast iron shielding to assess a possibility of hands-on access to utilities for disconnects and reconnects and component maintenance during exchange or replacement of these modules. Based on calculated dose rates together with estimates for the time needed for each operation, one can evaluate the total dose to a worker performing a certain operation. Note that the target hall is not accessible by working personnel during the beam-on times. It was shown that, once a beam is stopped, the target hall can be accessed by personnel after 4 hours of cooling (beam-off) time.

To analyse residual doses on top of the vacuum vessel shielding due to activated components, we use a two-step approach. First, PHITS and DCHAIN-SP codes are used to generate photon sources produced by radionuclides in the cast iron shielding and other beam line components in the vicinity of the areas of interest. Then the activated components in the model are populated with these photon sources, and MCNPX code is used to calculate residual doses for various beam-on/beam-off times.

We present dose rate estimates around the top of the beam dump assembly utility chase. The utility chase is located in a high radiation environment, in the vicinity of the beam dump drum where up to 300 kW of beam power will be dissipated during full-power operations. The utility chase represents a doglegged penetration in the vessel top shielding. The top part of the chase that is located above the vessel shielding needs to be accessed annually during beam-off time to disconnect the components, change out the beam dump module, then reconnect the utilities. Activation of the shielding and the beam dump module are the main sources of exposure to personnel during these operations. Figure 7 shows the mechanical engineering model of the utility chase as well as the radiation transport model of the top part of the beam dump utility chase with the surrounding cast iron shielding.
We use a conservative beam/energy combination, $^{58}\text{Ca}$ at 261 MeV/u, with full operational power 400 kW. We assume an irradiation time of 1 year, the planned time of the beam dump module changeout. We consider various cooling times, particularly 4 hours (minimum planned access time to the target hall) and 24 hours (assumed time when the beam dump utility chase will be accessed). The activated beam dump drum location is chosen to be in its topmost position below the cast iron shielding to produce the most amount of residual activity above the shielding. Local dose rates are shown in Figure 8 after 24 hours of cooling time. Average dose rates in this area are 5 mrem/h after 4 hours and 1.3 mrem/h after 24 hours of cooling.

The estimated time for removal and installation of the beam dump module is up to 16 hours. Conservatively assuming an average exposure of 3 mrem/h would lead to a total dose to personnel less than 50 mrem annually which is 10% of the MSU ALARA (As Low As Reasonably Achievable) goal for workers. The analysis confirms that hands-on access to the beam dump utility chase top parts is possible after the planned cool-down (beam-off) time of 24 hours.
Figure 8. Residual dose rate map in the vicinity of the beam dump utility chase top

Local numbers are in mrem/h. View along the beam line is shown, with the beam direction from right to left. One operational year of irradiation and 24 hours of cooling (beam-off) are assumed. The 1 m wide box shows the area of dose rate averaging.


Tritium production and release

Beam operations with high power heavy ions ranging from oxygen to uranium will create a high radiation environment for the beam line elements located in the target hall. Significant amounts of tritium will be produced in the beam line elements and shielding, then released into vacuum, and subsequently into the air and emitted from the facility.

Earlier facility release studies addressed air activation in the linac tunnel and the target hall. In this work, additional tritium production and release is estimated for beam-on and beam-off scenarios. Doses from air emissions containing this additional tritium release are assessed and compared with the project design goals. PHITS code is used for the tritium production calculations. Using the same conservative beam/energy combination as for the hands-on studies, $^{48}$Ca at 261 MeV/u with full operational power 400 kW, tritium production is assessed in various beam line and shielding components of the target hall. To assess different scenarios, components that can release tritium are sorted into various groups: shielding (vacuum vessel steel, cast iron vessel shielding, concrete shielding, walls); beam line components (target module, post-target shield, beam dump module, wedge module, magnets); cells filled with air, water and helium. The tritium production is further sorted according to component material, location and operational scenario.

The total rate of tritium produced in all system components, without the beam dump water where a beam stops, is $5 \cdot 10^{13}$ particles/s.
Diffusion coefficients $D$ are found in literature for a multitude of materials at various temperatures. Tritium diffusion coefficients are used when available. Otherwise, hydrogen diffusion coefficients are used with a proper scaling. Diffusion coefficients and diffusion lengths are used to estimate effective diffusion constants. For tritium produced by neutrons, the attenuation lengths should be close to the neutron attenuation lengths in the given materials. Diffusion length is conservatively estimated as half of the tritium production attenuation length for a thick component. For example, diffusion lengths of 7 cm for iron or steel, or 12 cm for concrete are obtained based on estimates.

Several viable operational scenarios are considered. During beam-on scenario, tritium from the beam line components is released via the vacuum exhaust into the target hall air, then outside of the facility. Tritium is also produced in the shielding outside the vessels and released to the target hall air directly. No personnel will be present in the target hall when a beam is on. During beam-off scenarios, in a similar way, tritium will continue to be released from the irradiated beam line components via the vacuum exhaust into the target hall, and from the target hall shielding outside of the vacuum vessels directly into the target hall air. Also, vessel sections will be open when components such as the target module or the beam dump module are removed from the vessels for repair and exchange; tritium will be released from the removed components directly into the target hall air. During beam-off scenarios, personnel will be allowed to enter the target hall after 4 hours of cooling time.
Tritium concentrations, as well as the resulting dose rates, are estimated in the target hall and at the most exposed (according to earlier air activation estimates) public receptor location for all these scenarios. Resulting worst-case doses for workers at the facility and public outside of the facility are compared with the facility goals. The annual dose to workers due to the tritium released to the target hall air is 75 mrem. As mentioned above, MSU ALARA goal for radiation workers is 500 mrem/year. Dose rate at the maximally exposed public receptor location from the tritium releases is 4 µrem/h. Total estimated dose rates from tritium and other released isotopes at that receptor are 5 µrem/h. For the annualised doses, we obtain 0.53 mrem/year from all sources (air activation + tritium releases). The result is well below the EPA public annual dose limit of 10 mrem from all nuclides specifically in air.

The detailed study of the tritium production in target hall components and its release shows that concentrations in the target hall, as well as hourly and annual doses at the most exposed public receptor, are below the FRIB design goals.

**Summary and conclusions**

FRIB is being built at MSU as a national user facility to provide fast, stopped and reaccelerated beams of rare isotopes for Nuclear Science. The facility will represent the high power heavy ion accelerator. High radiation environment during future beam operations and maintenance demands detailed radiation transport analysis.

Several examples of radiation transport studies at FRIB are presented. Groundwater activation analysis is revisited and updated as part of design completion; it is shown that levels of concentration of the most prevalent radionuclides are below the EPA limits for drinking water. Analysis of the vacuum vessel shielding penetrations is performed to assess hands-on accessibility during beam-off times; results confirm that components above the vessel top shielding will be hands-on accessible. Tritium production and release is studied to augment air activation analysis; results show that tritium releases for different operational scenarios are below the facility design goals. Radiation transport studies support current design and future operations of FRIB.

**Acknowledgements**

This material is based on work supported by the US Department of Energy Office of Science under Cooperative Agreement DE-SC0000661, the State of Michigan and Michigan State University. Michigan State University designs and establishes FRIB as a DOE Office of Science National User Facility in support of the mission of the Office of Nuclear Physics.

**References**


High-energy internal beam dump system for the Super Proton Synchrotron (SPS)

Daniel Björkman*, Helmut Vincke
CERN European Organisation for Nuclear Research
*daniel.bjorkman@cern.ch

The Super Proton Synchrotron (SPS) at CERN is receiving several major upgrades in order to provide higher beam intensities to CERN’s experimental areas and to the Large Hadron Collider (LHC) during its high luminosity era. With this upgrade, the SPS will receive a complete revamp of its beam dump system with it being relocated and upgraded to suit the new requirements and to provide a much safer work environment than the current dump system. Following beam optics constraints, the new beam dump system had to be designed as an internal dump encased in a massive multi-layered shielding to protect equipment and personnel against radiation and to limit the production of airborne radioactivity. The work presented encompasses the main efforts, which were carried out considering the ALARA principle to strongly reduce dose received by personnel and public.

Introduction

The Super Proton Synchrotron (SPS) at CERN is currently operating as the last injector in the Large Hadron Collider (LHC) injector chain as well as beam provider for the various fixed target experiments. New luminosity goals for the LHC and beam intensity demands for the proposed Beam Dump Facility (BDF) requires that sectors of the SPS have to be upgraded to meet the requested beam intensity demands. One of these upgrades can be found in the integration of a new beam dump.

The current beam dump system in the long straight section 1 (LSS1) of the SPS is the most radioactive part of the circular machine with residual dose rates measuring up to 25 mSv/h at 1-metre distance from the beam dump 30 hours after discarded beams. In addition to the severe radiation exposure risk to intervening personnel, are electronics and power cables subject to harmful levels of radiation damage. This puts the availability of the machine at risk due to the long hazardous interventions that would be needed if any exposed component would fail. The planned solution for mitigating these problems will be to install a safer and more reliable beam dump system in the long straight section 5 (LSS5) of the SPS in the ECX5 cavern during the long operation shutdown, scheduled in the years 2019 and 2020. Following beam optics constraints, the beam dump had to be designed as an internal beam dump, located around the beam line. It will be encased in a massive multi-layer shielding, which will allow it to operate while protecting equipment and personnel against radiation and limit the production of airborne radioactivity in the tunnel.

Beam dump design

The circulating beam will pass through the five-metre long beam dump during operations and it will be diverted and transversally dispersed onto its graphite absorber block (Figure 1) by a vertical and a horizontal kicker system when the remaining beam needs to
be discarded. The beam dump core will be encapsulated by an outer shielding according to Figure 2.

**Figure 1. Cross-section of the inner dump core**

The coloured line presents the impact location of the dispersed beam.


**Figure 2. Beam dump design consisting of core, an outer shielding and two masks**

The outer shielding will surround the dump core and will be constructed from interior to exterior in three functional layers:

- 40 cm inner concrete encasement;
- 1-metre iron composite shielding;
- Exterior 40 cm concrete or marble layer.

The inner concrete encasement attenuates the radiation field emerging from the dump by about a factor of three. It provides an inner environment that is, in favour over iron, less radioactive close to the dump core. The 1-metre thick iron composite shielding and the exterior concrete/marble layer will further attenuate the particle showers by another factor.
of about 1,000 to reduce effectively activation of air and material outside the shielding. After beam operation, the shielding will provide attenuation of the residual radiation field emitted by radioactive isotopes that are produced during beam operation inside the dump core and the shielding itself. The exterior layer of the beam dump consisting of marble has the advantage over concrete of reduced activation production when being exposed to hadronic radiation fields. This results in low dose rates in the proximity of the beam dump at locations shielded by marble. Therefore, marble was chosen as exterior material at all points where personnel might intervene after beam operation.

Two identical masks will be located downstream of the dump which consist of an iron core encased by 20 cm of marble. These masks shield sensitive accelerator elements from forward scattered high energetic particles escaping the beam dump. Their design is optimised to minimise the residual dose rate at accessible locations, of exterior material and of the activation of the air within the ECX5 dump cavern.

Radiological aspects

Hazards related to radioactive isotope build-up throughout the geometry were calculated with the Monte Carlo FLUKA (Böhlen, et al., 2014; Fassò, et al., 2005) code, version 2011 2x.2. These estimations considered $2 \times 10^{18}$ protons dumped per year for a period over 20 years followed by a single day with an averaged beam intensity of $5 \times 10^{12}$ protons/s, which represents the maximum capacity that the SPS can deliver averaged over one day. All dumped protons were conservatively considered to be of SPS maximum momentum of 450 GeV/c.

**Induced residual activity**

The effectiveness of the outer shielding is made evident when comparing the dose rate maps of Figure 3. This image compares the residual dose rate emitted by the unshielded dump core after the aforementioned operation scheme plus 1 week of cooling (left picture, shielding set to vacuum in decay step of simulation) with the fully shielded situation (right picture). The comparison proves that the attenuation capability of the shielding is at least three orders of magnitude for any personnel working in close vicinity of the beam dump shielding.

The weak points of the dump system show themselves upstream and downstream of the outer shielding in the right image of Figure 3. The requirement of transporting water and air coolant through the upstream beampipe opening meant that the opening had to be enlarged, and is therefore leaving part of the dump core unshielded. Activation downstream of the beam dump originate from hadronic particles escaping the dump core and leaving the beam dump through the downstream beampipe opening to activate the downstream area. Methods for mitigating these residual radiation hazards outside the outer shielding are further discussed under the heading “Residual radiation mitigation methods”. 
Air activation

The presence of the beam dump shielding structure results in a strong reduction of the airborne radioactivity production outside the shielding when being compared to the unshielded situation. Due to this effect, the air inside the permanently vented ECX5 cavern is predicted to stay at very moderate levels, allowing a fast access to LSS5 after beam operation.

All air volumes between the dump core and the outer shielding will activate significantly and are therefore kept under-pressurised with a dedicated air extraction system. This prevents its uncontrolled release to the ECX5 region outside the dump shielding. The air is routed safely to a measurement location where it is analysed and mixed with the major part of the air volume coming from ECX5 to be released safely to the environment. The annual committed dose to the reference group of the public is predicted to be less than 1 μSv.

Water-cooling

A closed demineralised water-cooling circuit will extract the beam-induced heat deposited inside the dump core through the outer shielding down onto a water-cooling system located next to the dump platform. There the extracted water will be cooled and filtered for radioactive ions, which will remain within dedicated ion filter. The isotope build-up in the water was estimated with ActiWiz 3 (Theis and Vincke, 2016) using proton, neutron and pion fluence spectra being scored inside the water pipe of the dump core using FLUKA simulation. A wide assortment of short- and long-lived isotopes will be produced within the water-cooling circuit as hadronic showers caused by the beam impact activate the nuclei of the water molecules. Most of the short-lived isotopes produced will decay within two hours, which will serve as a guideline for the minimum cool down time before access to the water-cooling skid is granted. The remaining long-lived isotopes will accumulate within the ion filter, which will be placed behind a concrete shielding to protect any personnel working on or close by the water-cooling skid.

Dump alignment manipulations

The massive outer shielding will naturally make up a barrier between the dump core and the personnel that needs to access the core to align it to the rest of the circular machine.
Access to the core will be made possible through a series of horizontal and vertical plugs that can be taken out of the outer shielding after operation has ceased. The alignment of the dump core can then be performed from a distance with the use of long alignment rods that are inserted through the holes revealed when removing the plugs. Figure 4 shows the residual radiation field emitted by the exposed core in the absence of the alignment plugs after 1 week of cool down. Safety guidelines for limiting radiation exposure to personnel during alignment operations will therefore include:

- limit personnel occupancy directly in front of removed plug cavities;
- utilisation of mirrors or remote cameras to guide alignment rods to their intended destinations.

Figure 4. Residual dose rate in the area of the dump, considering removed alignment plugs after 1 week of cool down


Residual radiation mitigation methods

The section discusses further ALARA methods for reducing residual radiation exposure of personnel.

Upstream collimation mask

A marble collimation mask will be placed between the upstream beampipe opening and the upstream beampipe elements according to Figure 5. The purpose of this mask will be to create a safer work environment upstream of the shielding by limiting the direct line of sight to the dump core. The mask will affectively attenuate the residual radiation field escaping the upstream beampipe opening of the outer shielding. Heavy hadronic backscattering from the beam impacts through the beampipe opening made marble the favourable choice of mask material. The potential choice of a steel mask was ruled out due to its higher level of activation when being irradiated by hadronic particles.
Figure 5. Upstream collimation mask and beampipe elements


**Residual dose rate reduction by choosing an adequate floor material structure**

Figure 6 shows the mitigation plan by choosing different materials to reduce the activation of the lateral and downstream floor outside of the dump shielding. A 10 cm thick iron composite slab will replace the concrete top layer of the walking floor laterally of the dump to limit the sodium-24 build-up. This build-up, typically caused by low energy neutrons in concrete, is relevant for heavy short-term beam operations followed by short cooling times. This In contrast to the lateral location, the high-energy hadronic particle fluence being present downstream of the beam dump shielding made metals unsuitable to serve as uppermost floor layer, as they would become considerably radioactive. The top layer of the platform just downstream of the beam dump is subject to high radiation exposure leading to significant levels of activation. Therefore, in order to support easier separation of future radioactive waste into different waste categories, a modular system of the floor is proposed. The possibility of exchanging these parts also means that they can be replaced with non-radioactive replicas to reduce the residual radiation hazard in the area. Finally, some parts of the exchangeable downstream area will be made out of marble at locations where personnel will frequently pass by to further reduce radiation exposure to personnel.
Dose rate in accessible areas as a function of the beam impact position on dump core

Figure 7 (left hand side) presents the vertical location of the beam impact position on the dump. Currently, it is planned to intercept the beam 2.5 cm on average below the edge of the graphite absorber block. During a dedicated study, it was found that a lower impact position significantly reduces the residual dose rate outside the beam dump shielding. As it is shown on the right hand side of Figure 7, a lowering of the vertical dumped beam axis by 1 cm will result in a reduction of the integrated dose rate of up to 50% at accessible areas up- and downstream of the dump. The integrated dose rate in this context is a summation of every single scoring bin for every step in the longitudinal co-ordinate \( z \) in the 3D xyz-mesh. This result will motivate further studies to reduce residual dose rate hazards of accessible areas outside of the SPS beam dump shielding.

Figure 7. Vertical impact position of the beam on the dump graphite absorber (left) and the dose rate comparison (right) of the two different beam impact scenarios

The vertical location of the two beam impacts differs by 1 cm.

Conclusions

In view of future new challenges for the SPS accelerator, a new beam dump system will be implemented in the near future. The aim of this system, when being compared to the current dump installation, is to strongly reduce radiation hazards, like, prompt dose to equipment, material activation in accessible areas and airborne radioactivity production. In order to comply with these constraints a five-metre long beam dump core surrounded by a massive shielding structure including two masks, absorbing high energy particles will be installed. The shielding will be capable of attenuating the secondary radiation field emitted by the dump core by at least three orders of magnitude. This results in a strongly reduced residual dose rate environment in accessible areas around the beam dump, allowing fast accessibility of the area after beam operation. Dedicated ventilation and cooling systems are designed to strongly reduce radiological hazards for personnel and environment. Currently additional ALARA related improvements are investigated to mitigate some remaining minor radiological hazards at the extremities of the dump system.

Acknowledgements

We are indebted to Antonio Marcone, Stefano Pianese, Damien Grenier, Jerome Humbert, Michael Lazzaroni, Frederic Galleazzi, Jose Antonio Briz, Mathieu Baudin for their valuable input and discussions throughout the project.

References


Cross-sections measurement and benchmark studies on radionuclide yields of Bi(p,xn) reactions

Leila Mokhtari Oranj1*, Mahdi Bakhtiari2, Nam-Suk Jung1, Arim Lee1, Hee-Seock Lee1
1Pohang Accelerator Laboratory, POSTECH, Pohang 37673, Korea
2Division of Advanced Nuclear Engineering, POSTECH, Pohang 37673, Korea
*leila@postech.ac.kr

This work has two purposes as measuring Bi(p,xn)207,206,205,204Po reactions cross sections as well as evaluating the capability of Monte Carlo codes, FLUKA, PHITS/DCHAIN-SP, and MARS to predict experimental data of product yields. Reactions cross sections were measured to complete the excitation functions up to 100 MeV. Excitation functions of mentioned reactions were calculated by using the theoretical model based on the latest version of the TALYS code and compared to the new data and other data in the literature. Three rounds of experiment were done to measure yields of products by using 100 MeV and 70 MeV proton beam. The targets were arranged in stacks consisting of Bi, Al, Au foils and Pb and Bi plates. The proton beam intensity was determined by the activation analysis method using 27Al(p,3p1n)24Na, 197Au(p,pn)196Au, and 197Au(p,p3n)194Au monitor reactions. A satisfactory agreement was observed between the present experimental data and the previously published data. Additionally, the radionuclide yields at different depths of target produced by 100 and 70 MeV protons were calculated using the Monte Carlo codes, and compared with our measurements. The ratios of the results of calculations by the codes to the measured yields of Po radio-nuclei did not exceed 1.5 over the whole beam range. In conclusion, a good agreement was observed between the present experimental data and the simulations.
Shielding and activation analyses of the MYRRHA accelerator injector up to 5.9 MeV

Yurdunaz Celik1*, Alexey Stankovisky1, Gert Van den Eynde1

1SCK•CEN
*ycelik@sckcen.be

The low energy part of the MYRRHA accelerator up to 5.9 MeV is currently being installed for testing at the Université Catholique de Louvain (UCL) in Belgium. A shielded vault door and a local shielding around the beam dump must be constructed in order to cope with radiation produced by the interactions of the proton beam (5.9 MeV, 10 mA) with the low energy accelerator components (LEBT, RFQ, CH cavities, test bench) and a beam dump. The radiation field and the induced activity/residual field are mainly determined by the proton beam interactions in the beam dump. Therefore to define the best beam dump material from a point of view of neutron production and activation, a comparison was performed for five candidate beam dump materials, which are OFHC copper, aluminium alloy-6061, graphite, tantalum and tungsten. Large differences between the proton-induced reaction cross sections taken from the evaluated nuclear data libraries TENDL-2017, LA150, JENDL-4/HE, PADF-2007 and calculated with the CEM03.01 model were found. Adjustments of the evaluated cross-sections were made to correct for observed deficiencies. The prompt neutron and gamma dose rates were calculated with MCNPX Monte Carlo radiation transport code using different variance reduction techniques. Residual dose rates were calculated with MCNPX using the delayed radiation spectra for gamma-rays determined with the ALEPH2 depletion code. The final design of the shielded vault door and local shielding that meet the dose rate limits imposed by the UCL Security and Radioprotection Service are presented for the investigated materials.

Introduction

The MYRRHA facility will consist of a 600 MeV linear proton accelerator and sub-critical core at SCK•CEN in Mol, Belgium (Aït Abderrahim et al., 2003). The linear proton accelerator will have 17 MeV and 100 MeV beam dumps for the target stations. The first injector part of the accelerator is currently in construction phase at the CRC/UCL (Cyclotron Resource Centre of Université Catholique de Louvain) for the commissioning and accelerator tests. The injector will accelerate protons up to the energy of 5.9 MeV. It will consist of the following components: ion source, low energy beam transport (LEBT), radiofrequency quadruple (RFQ) cavity and Cross-bar H-type cavities (CH-cavities). Additionally, diagnostic elements (test bench) and a beam dump will be also installed.

The radiation field along the beam line will be governed by the beam dump that will stop 10 mA proton beam of 5.9 MeV. Contribution of regular beam losses on the accelerator components to the total radiation is negligible compared with the radiation produced in the beam dump. Selection of the beam dump material is important to minimise the prompt doses and radioactivity at the surrounding materials of the beam dump, the beam dump itself and accelerator components. In this respect possible candidate beam dump materials, which are copper (OFHC Cu), aluminium alloy (Al-6061), graphite (C), tantalum (Ta),...
tungsten (W) were tested in this work to determine appropriate material in terms of neutronic properties.

The prompt dose rates outside the vault for the considered beam dump materials vary up to several orders of magnitude. A local shielding around the beam dump was designed for each candidate material to meet the dose rate requirement defined by the radioprotection team of Université Catholique de Louvain (UCL) as 0.5 µSv/h (Rodeghiero, 2017) behind the vault. To avoid challenges due to extremely heavy shielding, the total weight of the designed local shielding should be taken into account when material selection for the beam dump is made.

The beam dump material must also allow personnel access to the vault for maintenance during beam-off phases. Residual dose rates due to the induced activity were calculated to determine minimum waiting time for the manual maintenance without restrictions.

The vault housing the beam line was built without a maze at the vault entry, which is located at the opposite direction of the beam, due to limited space. Therefore, a very heavy shielded door consisting of a steel frame filled with polyethylene was also designed for a mazeless entry in this work. However, to obtain dose rates at the vault entry was challenging since low neutron yield imply significant computing time in order to generate statistically converged results of sufficient precision or even non-zero results, even though well-known variance reduction techniques embedded in MCNPX Monte Carlo radiation transport code were applied. Those techniques were compared with the results MCNPX calculations with weight widows produced by the ADVANTG code (Mosher et al. 2015) that employs deterministic neutron transport. The weight windows produced by ADVANTG code give the dose rates with low uncertainties in a reasonable computing time.

**Calculation model**

The calculations reported here were performed with the MCNPX-2.7.0 general purpose Monte Carlo radiation transport code (Pelowitz, 2011). The nuclear data library used for all the present dose rate calculations is based on the JEFF 3.3 for neutron-induced reactions (Santamarina et al. 2009) and TENDL-2017 for proton-induced reactions (Koning et al., 2017) which is also distributed under the JEFF-3.3 umbrella.

In MCNPX, ambient dose equivalent dose rates were calculated using flux-to-ambient dose conversion factors for neutrons, gammas and protons (ICRP, 1997). Various variance reduction techniques implemented in MCNPX (Pelowitz, 2011) can be invoked: (forward) weight windows, DXTRAN, adjoint weight windows and also ADVANTG code (Mosher et al. 2015). The weight windows variance reduction technique can be described as a space-energy-dependent splitting and Russian roulette technique. The MCNPX user specifies a lower weight bound for the particles. If a particle is below the specified lower weight Russian roulette is played. As a result, the particle's weight will either be increased to a certain value within the window or the particle will be terminated. The upper bound is a constant multiple of the user-specified lower bound. When a particle is above the upper weight bound, it is split. This ensures that all the particles are within the window. The weight window generator of MCNPX was used to produce superimposed mesh-based or cell-based weight windows for entire geometry. The adjoint weight window generator of MCNPX can only be used in the multigroup mode. Only neutrons and photons can be transported in backward (from the detector region to the source region) in adjoint calculations, which is preferable when the source region is large and detector region is small. Once weight windows are produced with adjoint weight window generator it can be
used in forward calculation. Since probability of scattering towards the region is often very small, a DXTRAN sphere, which stands for deterministic transport, is also used to increase the sampling in a small region of interest that would otherwise be difficult to sample. Depending on sampling a collision or source density function, DXTRAN estimates the correct weight fraction that should scatter towards, and arrive without collision at, the surface of the sphere. The DXTRAN method then puts this correct weight on the sphere. The source or collision event is sampled in the usual manner, except that the particle is killed if it tries to enter the sphere because all particles entering the sphere have already been accounted for deterministically (Radiation Transport Group, 1993). ADVANTG is an automated tool for generating variance reduction parameters for fixed-source continuous energy Monte Carlo simulations with MCNP5 (X-5 Monte Carlo Team, 2003) based on approximate 3-D multigroup discrete ordinates adjoint transport solutions generated by Denovo deterministic code. The ADVANTG code can be used only with neutron, photon, and coupled neutron-photon simulations (Mosher et al., 2015). Weight windows are used to obtain the results if not mentioned otherwise.

To evaluate the total activity in various beam dump core materials over the operational life time, the irradiation of the beam dump during 30 days with 10 mA beam current in continuous mode was modelled. After 30 days of operation a cooling time of 10 years is modelled. Activation calculations are performed with the ALEPH2 depletion code (Stankovskiy and Van den Eynde, 2012). It uses MCNPX to calculate the steady state particle flux and spectrum to determine reaction rates for subsequent depletion calculation.

The ALEPH2 code is capable of producing delayed radiation spectra (photons per second per unit volume) for gamma, alpha and beta decay of radioisotopes for the defined irradiation and cooling times. It can also produce an MCNPX input file with spatial distribution of gamma decay source among the volumetric cells of the geometry. They may be used separately in MCNPX calculations as source energy distributions in a simplified geometry model to predict the residual dose rates at any cooling time. In this way the photon source distribution of the considered materials was used in separate MCNPX calculations to simulate residual doses.

**Structure of the beam line**

The initial part of the MYRRHA injector accelerating protons up to the energy of 5.9 MeV will consist of six components:

- Electron Cyclotron Resonance (ECR) Ion Source (IS) – 0 to 30 keV;
- Low Energy Beam Transport (LEBT) – 30 keV;
- Radio Frequency Quadrupole (RFQ) – 30 keV to 1.5 MeV;
- 7 Cross-bar H-type (CH) cavities – 1.5 MeV to 5.9 MeV;
- Test Bench (TB) – 5.9 MeV;
- Beam Dump (BD) – 5.9 MeV.

A simplified design of the beam line for the 5.9 MeV is shown in Figure 15.1. The ion source (IS) and the Low Energy Beam Trasport (LEBT) are not included for the calculations since the beam energy (30 keV) is lower than the neutron production threshold energy for Cu (2.167 MeV for $^{65}$Cu and 4.215 MeV for $^{63}$Cu). The simplified cylindrical geometry of the RFQ consists of a Cu tube (2.7 cm thick) placed at the centre of an Al box (4.5 cm thick). Seven CH cavities were modelled as two concentric stainless steel (SS)
cylinders (the inner diameters of 3 and 64.7 cm and wall thicknesses of 0.8 and 5 cm, respectively). The geometry of TB consists of a 0.25 cm thick SS cylinder with an inner diameter of 5.5 cm. A drift tube consisting of a SS cylinder with 10 cm diameter and 0.25 cm thickness is located between TB and BD. The length of the drift tube is adjustable according to the required local shielding thickness at the entrance of the beam dump.

A V-shaped beam dump is used to stop 5.9 MeV proton beam with 10 mA beam current. The design of the beam dump is shown in Figure 1. The beam dump core consists of two plates made by Cu. The angle between the two beam dump plates is 10 degrees. The dimensions of the beam dump is 50 cm (in length) x 2.3 cm (in thickness) x 25 cm (in width). The cooling water is circulated within the Cu plates. The vacuum chamber and two plates closing the open window from left and right side of the Cu beam dump core is made of SS 304.

Prompt dose rate calculations were performed in the scoring areas (100 cm x 100 cm x 10 cm) placed in the concrete wall (80 cm thick) and behind it (see Figure 1). The scoring mesh cell size has been chosen from optimising computational time point of view taking into account degree of uniformity of the flux inside each mesh cell.

Figure 1. MCNPX design of the bunker from top view

Concrete walls are represented with grey colour, air is light blue, vacuum is white.

Figure 2. 3D technical drawing of the V-shape beam dump (orange) cooled with water (blue) in a stainless steel casing (grey)

Since the purpose of this work is to investigate and determine the best beam dump material in terms of low activation/residual and prompt dose field, the same conceptual beam dump design and dimensions were used for all the candidate beam dump materials. The thickness of the beam dump plates (2.5 cm) is enough to stop protons for all the considered beam dump materials as it is shown in Table 1.

### Table 1. Stopping ranges of investigated beam dump materials calculated with the SRIM code (Ziegler, 2010)

<table>
<thead>
<tr>
<th>Materials</th>
<th>Stopping range (mm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Copper</td>
<td>0.10</td>
</tr>
<tr>
<td>Graphite</td>
<td>0.25</td>
</tr>
<tr>
<td>Aluminium</td>
<td>0.26</td>
</tr>
<tr>
<td>Tantalum</td>
<td>0.089</td>
</tr>
<tr>
<td>Tungsten</td>
<td>0.075</td>
</tr>
</tbody>
</table>


#### Neutron yield

First, the effect of the proton-induced nuclear data libraries and nuclear models are tested to determine the most suitable one for the MCNPX calculations separately for all beam dump candidate materials (Cu, C, Al, Ta, W). Threshold proton energies for neutron production in these materials are indicated in Table 2. Even though isotopic abundance of $^{13}$C is 1.07%, it is responsible for all the neutron production in carbon below 19 MeV due to extremely high threshold energy for neutron production on $^{13}$C.

### Table 2. Threshold proton energies for neutron production

<table>
<thead>
<tr>
<th>Nuclide</th>
<th>Threshold (MeV)</th>
<th>Nuclide</th>
<th>Threshold (MeV)</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{12}$C</td>
<td>19.64</td>
<td>$^{181}$Ta</td>
<td>0.99</td>
</tr>
<tr>
<td>$^{13}$C</td>
<td>3.24</td>
<td>$^{183}$W</td>
<td>4.61</td>
</tr>
<tr>
<td>$^{27}$Al</td>
<td>5.8</td>
<td>$^{185}$W</td>
<td>3.6</td>
</tr>
<tr>
<td>$^{63}$Cu</td>
<td>4.22</td>
<td>$^{183}$W</td>
<td>1.35</td>
</tr>
<tr>
<td>$^{65}$Cu</td>
<td>2.17</td>
<td>$^{184}$W</td>
<td>2.28</td>
</tr>
<tr>
<td>$^{180}$Ta</td>
<td>0.08</td>
<td>$^{186}$W</td>
<td>1.37</td>
</tr>
</tbody>
</table>


Figure 3 (left) shows the neutron yield per primary proton interaction with the beam dump materials at 5.9 MeV incident proton energy. The calculations were performed using TENDL-2017 (Koning et al., 2017), LA150 (Young and Chadwick, 1998) and JENDL-4/HE (Kunieda., et al., 2016) proton-induced libraries and, intranuclear cascade model CEM03.01 (Mashnik and Sierk, 2012) implemented in MCNPX. However, there is no data in LA150 proton library for $^{13}$C and Ta.

The CEM03.01 model in MCNPX underestimates the neutron production and gives the lowest results in Cu, Al and C. The model results for Ta and W are not shown due to extremely high statistical uncertainties, which require significant computing time to reach
acceptable values. The neutron yield is in a good agreement for Cu and Ta in the TENDL-2017, LA150 and JENDL-4/HE libraries. However, for some, such as Al, $^{13}$C and W the neutron yields show discrepancies. A large discrepancy is observed in the nuclear data libraries of Al. TENDL-2017 does not represent the experimental data near the threshold energy (see Figure 3, right). The only library that closely represents the experimental data near the threshold is PADF-2007 (Proton Activation Data File) (Konobeyev et al., 2007). It should be noted that PADF-2007 data are not suitable for transport calculations. Therefore, three modifications were made in the nuclear data libraries used in the MCNPX simulation. In a first one, the TENDL-2017 proton-induced neutron production data for Al were replaced with PADF-2007 data (denoted as $^{27}$Al(p,x)n, PADF 2007). Second modified TENDL-2017 file contains correction made for the threshold energy of $^{13}$C, (denoted as Corrected threshold for $^{13}$C) while the third run has been performed with TENDL-2017 but the data for $^{13}$C and $^{15}$C were taken from JENDL-4/HE (denoted as $^{12}$C and $^{13}$C, JENDL-4/HE). From now on the production calculations for Al and $^{13}$C were performed with the corrected libraries, which are $^{27}$Al(p,x)n, PADF 2007 and corrected threshold for $^{13}$C, respectively. TENDL-2017 is used for the final calculations of Cu, Ta and W.

Among the investigated materials, the highest number of neutrons is produced in Cu, while the lowest ones are produced in Ta and W. With the selected data libraries neutron yield in C is bigger by a factor of 1.5 with respect to Al.

**Figure 3.** Neutron yield per proton for the 5.9 MeV calculated with different proton-induced nuclear data libraries and CEM03.01 model (left) and evaluated neutron production cross-sections and experimental data (EXFOR, 2018) of $^{27}$Al (right)


**Prompt dose rates**

Neutron and gamma dose rates obtained in the scoring areas (see Figure 1) with the investigated candidate beam dumps are presented in Figure 4. There is no local shield around the beam dump. The neutron dose rates along the shielding reflect the same discrepancy with the neutron yields in Figure 3. Only the neutron dose rates obtained with the Ta and W beam dump are close to the dose rate limit at the back end of the shield layer.
The highest neutron dose rates are obtained due to Cu beam dump as expected. Carbon and Al-6061 beam dumps give similar dose rates along the shielding layer.

As it is obvious a local shielding around the beam dump is necessary to obtain lower dose rates behind the concrete wall. A shielding optimisation study was performed using different shield materials such as polyethylene (PE), 30% wt. borated PE (BPE) with natural boron (20% $^{10}$B, 80% $^{11}$B) for neutron and lead (Pb) for gamma attenuation. A cylindrical geometry is used for local shielding surrounding the beam dump with partially open window with 10 cm diameter as it is shown in Figure 5.

**Figure 4.** Neutron (left) and gamma (right) dose rates in the scoring area without local shielding

![Neutron and gamma dose rates](image)


An example of dose rates with local shielding is given for the Cu beam dump in Figure 6. The obtained neutron dose rates are just above the dose rate limit if local shielding is designed with 40 cm PE or 30% BPE surrounded by 10 cm Pb (see Figure 6, left). If Pb is used as a first shielding layer the dose rates decrease by a factor of 2 compared to the geometry with Pb used as a second shield layer. Boron content in PE decreases the gamma dose rates by a factor of 8 along the concrete wall (see Figure 6, right). Borated PE is more efficient than PE because the yield of gammas at 2.2 MeV originating from neutron capture in hydrogen is suppressed and the neutron field is attenuated thanks to the large $^{10}$B(n, alpha) capture cross section. The produced alpha particles are immediately stopped and the produced 0.48 MeV gamma-ray has much shorter attenuation length than the 2.2 MeV gamma-ray. To use Pb as a first shield material followed by 40 cm 30% BPE gives dose rates higher by the factor of 1.4 compared to the case where Pb is used as last shield material after 40 cm of 30% BPE. However Pb (density 11.342 g/cm$^3$) or 30% BPE (density 1.19 g/cm$^3$) as first shield material will give significant difference in the weight of the whole local shielding. The geometry with Pb as a first shield layer will be 8 tonnes lighter compared to the case with Pb in a second shield layer behind 30% BPE. In order to obtain dose rates lower than the dose rate limit, the final shielding design is determined using 15 cm Pb surrounded by 40 cm 30% BPE.

Table 3 shows local shielding designs and their total weight for each investigated beam dump material. Use of Ta and W will give advantage in the weight of the local shielding compared to Al, C and especially Cu. Considering the limited space around beam dump to design beam dump with high Z materials will decrease the prompt dose rates and thus thickness of the local shielding.
Figure 5. The geometry of local shielding around the beam dump

Pb shield is represented with orange colour, yellow is BPE, vacuum is white, green is SS vessel around the beam dump.

Figure 6. Neutron (left) and gamma (right) dose rates in the scoring area with local shielding

Cu used as the beam dump material.

Table 3. Local shielding design in order to obtain dose rate lower than the dose limit behind the beam dump score area for all different beam dumps

<table>
<thead>
<tr>
<th>Shielding design</th>
<th>OFHC Cu</th>
<th>Al-6061</th>
<th>C</th>
<th>Ta</th>
<th>W</th>
</tr>
</thead>
<tbody>
<tr>
<td>15 cm Pb + 40 cm 30% BPE</td>
<td>7 cm Pb + 30 cm 30% BPE</td>
<td>7 cm Pb + 20 cm 30% BPE</td>
<td>10 cm 30 % BPE</td>
<td>10 cm 30 % BPE</td>
<td></td>
</tr>
<tr>
<td>Shielding weight (tonnes)</td>
<td>9.64</td>
<td>4.11</td>
<td>3.16</td>
<td>0.33</td>
<td>0.33</td>
</tr>
</tbody>
</table>


**Activation**

The activities were calculated considering continuous operation of 30 days and 10 years of cooling.

The largest activity right after the shutdown is obtained with the Cu beam dump, the smallest with the Ta, see first graph in Figure 7. While the graphite becomes the least active.
material already after the second hour of cooling. Cu has largest activities during whole cooling period. Activity profiles of other materials are quite similar until 1 year of cooling. After 10 years of cooling least activations are produced in C, Ta, W, respectively, though the activities of Al and Cu are high and similar.

During cooling time the total activity of each of the beam dump materials evolves in a different way, see Figure. The residual activity of Cu is higher than those of the other materials imainly due to $^{63}$Zn ($t_{1/2} = 38$ min) and $^{65}$Zn ($t_{1/2} = 244$ d). Total activity of Al alloy during cooling is mainly due to its impurities except $^{27}$Si produced via $^{27}$Al(p,n) and has dominant activity right after the irradiation. Activation of C is determined in the short term by $^{13}$N ($t_{1/2} = 10$ m) and in the long term by $^{7}$Be ($t_{1/2} = 53.22$ d). After 5 years of cooling $^{14}$C ($t_{1/2} = 5700$ y) is the most dominant radioisotope. Activation of Ta is determined by dominant radioisotopes $^{181}$W ($t_{1/2} = 121.2$ d) and $^{182}$Ta ($t_{1/2} = 114.74$ d) during 10 years of cooling. Dominant contributors to the total activity of W are radioisotopes of Re, which are produced from $^{182,183,184,186}$W via (p,xn) reactions.

Figure 7. A comparison of total specific activities and detail activity profiles of different beam dump materials after continuous operation of 30 days with a proton beam of 5.9 MeV and current intensity of 10 mA.
Residual doses

Residual dose rates are important for assessing the cooling time before allowing maintenance personal to interact with the activated material after the beam is turned off. To calculate the residual dose rates, several detectors were placed around all components. The detectors have a radius of 5 cm. Residual gamma dose rates were obtained in the detectors using gamma ambient equivalent dose conversion factors in MCNPX calculation. The numbering (or ID) of the detectors start from left side of the accelerator beam line and increases clock wise, see Figure 8. For this calculation only OFHC Cu and Ta are considered (the materials giving the highest and lowest neutron yields). Local shielding around the Cu beam dump is 10 cm Pb followed by 40 cm 30% BPE, while it is 10 cm 5% BPE.

Residual gamma dose rates in the detector cells for 1 hour, 1 day and 1 week of cooling are presented in Figure 9. Dose rates include only contribution of the beam dump due to the negligible contributions from other components. The results around the BD and TB are higher than the ones obtained around the CH and RFQ in both Cu and Ta cases. However, Ta gives the lowest dose rates after 1 day of cooling. Contribution of the activated local shielding to the residual gamma dose rates is lower than the activated material of the components due to direct primary proton interactions.

It should be noted that the geometry model is simplified. In reality additional structural materials will act as a shield and thus the presented values could be lower. Right after the beam-off the hand-on maintenance will not be allowed. Manual maintenance is possible in
the area without restrictions (dose rates lower than the 0.5 µSv/h limit for workers) only after 1 day of cooling with Ta as beam dump material. If intervention is needed before required cooling time, an occupancy factor of minimum 10 hours per year allows meeting the constraint of 2 mSv per person and per intervention, the criterion used in CERN for the design of the nuclear facilities (Sentis and et. al. 2006).

**Figure 8. Detector positions with local shield around the beam dump**

![Detector positions with local shield around the beam dump](source)


**Figure 9. Total gamma dose rates after 1 hour, 1 day and 1 week of cooling in the detectors around the all components**

![Total gamma dose rates](source)


**Vault shielding**

To deal with the prompt doses due to beam losses in the accelerator components and high power distribution on the beam dump, the dose rates in backward direction should meet the dose rate requirements. The Cu beam dump is used for the vault shielding since it gives the highest prompt doses behind the concrete wall (see Figure 4). Contributions of other accelerator components to total prompt dose rates in the vault are negligible compared to that of beam dump. To calculate the dose rates, a detector with 5 cm radius was placed at the exit of the vault. Obtained total dose rate (neutron + photon) at the exit of the vault is about 70 µSv/h, which is 140 times higher than the dose rate limit without any shielded door.

Therefore a shielded door was placed at the exit of the opening vault (see Figure 1). Local shield around the beam dump has completely open window in backward direction as a
A conservative approach. Several shield thicknesses of steel (for gamma shielding) and PE (for neutron shielding) were tested to find the minimum door thickness to reach the dose rate limit behind the door. Neutron and gamma dose rates at the detector placed behind the door are shown in Table 4. It can be seen that the door with 5 cm steel frame filled by 20 cm PE will decrease the dose rates much below the dose rate limit of 0.5 µSv/h.

Table 4. Dose rates (µSv/h) behind the door

<table>
<thead>
<tr>
<th>Door geometry</th>
<th>Particle</th>
<th>Dose rate (µSv/h)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Steel(2.5 cm)+PE (10 cm)</td>
<td>neutron</td>
<td>1.07 ± 0.13</td>
</tr>
<tr>
<td></td>
<td>photon</td>
<td>0.9 ± 0.28</td>
</tr>
<tr>
<td>Steel(2.5 cm)+PE (10 cm)+Steel(2.5 cm)</td>
<td>neutron</td>
<td>0.81 ± 0.29</td>
</tr>
<tr>
<td></td>
<td>photon</td>
<td>0.49 ± 0.19</td>
</tr>
<tr>
<td>Steel(2.5 cm)+PE (20 cm)</td>
<td>neutron</td>
<td>0.22 ± 0.06</td>
</tr>
<tr>
<td></td>
<td>photon</td>
<td>0.68 ± 0.21</td>
</tr>
<tr>
<td>Steel(5 cm)+PE (20 cm)</td>
<td>neutron</td>
<td>0.16 ± 0.08</td>
</tr>
<tr>
<td></td>
<td>photon</td>
<td>0.31 ± 0.10</td>
</tr>
<tr>
<td>Steel(5 cm)+PE (20 cm)+Steel (5 cm)</td>
<td>neutron</td>
<td>0.06 ± 0.02</td>
</tr>
<tr>
<td></td>
<td>photon</td>
<td>0.06 ± 0.03</td>
</tr>
</tbody>
</table>


The statistical uncertainties of the results obtained with the weight window generator (wwg) variance reduction technique vary between 2% and 30%. A direct simulation (without variance reduction technique) of MCNPX was also performed (3 days of run time with 96 2.7-GHz hyper-threaded CPUs), however zero results were obtained for this problem. Low neutron yield per low-energy proton and large distance (17 m) between the proton source and the detector bring challenges to obtain the dose rates at the detector. This motivated a validation of the results obtained with weight windows with other variance reduction techniques (VRTs) like DXTRAN, adjoint weight window, and ADVANTG code. For this comparison very heavy shielded door made of 5 cm steel frame filled with 20 cm PE, is used.

A comparison of the results obtained in the farthest detector is presented in Table 5. In the table calculation time and source particle number (NPS) are also shown.

Calculations employing DXTRAN method were performed with neutron source distribution obtained from primary beam interaction with the beam dump materials. The same neutron source is used for adjoint weight window and ADVANTG code calculations. Relative statistical uncertainty at the detector with DXTRAN is 31% after 25 hours of running. Similar dose rate with even lower uncertainty (15%) was obtained using wwg after 10 hours. Importance functions obtained with adjoint mode calculations were then used in forward calculation with continuous energy cross-section libraries. Relative uncertainty of adjoint mode calculation is 90% with the same running time of DXTRAN. To obtain importance functions with ADVANTG code takes only 13 seconds. These importance functions are also used in forward calculation with continuous cross section libraries in MCNPX. The highest efficiency is observed with the ADVANTG code resulting 4% of uncertainty after 16 hours of running.
Table 5. Dose rates obtained behind the shielded door with different variance reduction techniques

<table>
<thead>
<tr>
<th>Methods</th>
<th>Source</th>
<th>Dose rate [µSv/h]</th>
<th>Std Dev.</th>
<th>NPS</th>
<th>Comp. time [h]</th>
</tr>
</thead>
<tbody>
<tr>
<td>ADVANTG</td>
<td>neutron</td>
<td>0.06</td>
<td>4%</td>
<td>2E+08</td>
<td>16</td>
</tr>
<tr>
<td>DXTRAN</td>
<td>neutron</td>
<td>0.04</td>
<td>31%</td>
<td>5E+08</td>
<td>25</td>
</tr>
<tr>
<td>WWG (forward)</td>
<td>proton</td>
<td>0.05</td>
<td>15%</td>
<td>5E+08</td>
<td>10</td>
</tr>
<tr>
<td>ADJOINT mode</td>
<td>neutron</td>
<td>0.03</td>
<td>90%</td>
<td>4E+10</td>
<td>25</td>
</tr>
</tbody>
</table>


The only difficulty experienced with ADVANTG code when applied for the accelerator shielding geometry is a representative definition of the neutron source distribution from primary proton interactions since it does not deal with protons.

Conclusion

MYRRHA 5.9 MeV injector is going to be located at CRC/UCL hall. The considered design for the 5.9 MeV beam dump assumes V-shape with 50 cm length, 10° half opening angle, and 2.3 cm thickness. The choice of material is important to mitigate the prompt doses and induced activity. Five candidate beam dump materials, namely Cu, C, Al, Ta and W, were studied from the shielding and resistance to neutron-induced activation point of view. Local shielding is proposed on the basis of results obtained with MCNPX calculations for the prompt neutron and gamma dose rates. The residual activities after 30 days of operation time were obtained using the ALEPH2 depletion code.

A local shielding was estimated for all considered beam dump materials to obtain the dose rates behind the concrete bunker lower than the dose rate limit at UCL (0.5 µSv/h). The drawbacks of Cu as a beam dump material are its high prompt doses and induced activity, while Ta shows better profile from all these points. Residual doses obtained with Ta indicate that the dump area will be accessible after a short cooling time (1 day).

For the vault shielding only Cu was used in calculations as a beam dump since it gives the highest dose rates. Due to the difficulty of obtaining dose rates with acceptable statistical uncertainties at the entrance of the vault, various variance reduction techniques implemented in MCNPX, as well as the ADVANTG code were tested to determine the most efficient technique. It was shown that ADVANTG is very efficient to obtain results in short time with very low statistical error in neutron shielding problems. Even though it is only applicable for neutron and photon penetration problems, it can be successfully applied for the accelerator applications provided that the secondary neutron and photon distributions obtained from primary proton interactions are properly determined.

References


Session III: Beam-plasma and laser-plasma interactions and acceleration
Radiological protection studies on high intensity laser facilities

Rui Qiu\textsuperscript{1,2}, Bo Yang\textsuperscript{1,2}, Shuoyang Wei\textsuperscript{1,2}, Minghai Yu\textsuperscript{3}, Jinlong Jiao\textsuperscript{3}, Wei Lu\textsuperscript{1,2,4}, Yonghong Yan\textsuperscript{3}, Zhimeng Zhang\textsuperscript{3}, Weimin Zhou\textsuperscript{3,5} and Junli Li\textsuperscript{1,2}

\textsuperscript{1}Department of Engineering Physics, Tsinghua University, Beijing 100084, China
\textsuperscript{2}Key Laboratory of Particle and Radiation Imaging (Tsinghua University), Ministry of Education, Beijing, China
\textsuperscript{3}Science and Technology on Plasma Physics Laboratory, Laser Fusion Research Center, China Academy of Engineering Physics, Mianyang 621900, China
\textsuperscript{4}Institute of Disease Control and Prevention, Academy of Military Medical Sciences, Beijing 100071, China
\textsuperscript{5}IFSA, Collaborative Innovation Center, Shanghai Jiao Tong University, Shanghai 200240, China

*qiurui@tsinghua.edu.cn

When a strong laser beam irradiates a solid target, a hot plasma is produced and electrons and protons are accelerated. These hot electrons and protons subsequently generate hard X-rays in the solid target. To date, limited studies have been conducted on this radiological protection issue. In order to systematically characterise the source term of electrons and protons, extensive literature reviews on the physics and properties of electrons and protons have been conducted. The photon dose generated by the interaction between hot electrons and solid target was simulated with FLUKA for the laser intensity ranging from 1E19 to 1E21 W/cm\textsuperscript{2}. Furthermore, an equation to estimate the photon dose generated from ultra-intense laser-solid interactions based on the laser intensity is given. It is found that the results in this work are in better agreement with the experimental ones when compared with previous model in literature. In addition to the dose estimation, the shielding effects of common materials including concrete and lead were also studied for laser-driven X-ray source. The transmission curves and TVLs in concrete and lead were calculated through Monte Carlo simulations. In experimental aspects, the temperature of hot electrons, X-ray doses, angle distribution and energy spectrum for laser intensity of 7E18~4E19 W/cm\textsuperscript{2} were measured on “XG-III” 300TW laser devices. The results show that the hot electron temperature is in good agreement with the Wilks scaling. The maximum dose of X-ray generated by a single laser pulse of ~153 J on a 1mm Ta target is about 16.8 mSv at 50cm and is located near the laser incident direction. The X-ray dose at laser incident direction increases significantly with the laser intensity, and agrees well with the calculated results based on the built equation. The dose at 90° changes relatively little with the laser intensity. The measured X-ray spectrum can be described by an exponential distribution with X-ray temperature, where the X-ray temperature at 0° is of 0.4–1.15MeV and the X-ray temperature at 90° is of 0.25–0.54MeV. Based on these information, dosimetric evaluation of laser-driven X-ray and neutron sources for XG-III laser facility was performed. For the laser-driven bremsstrahlung X-ray source, the photon dose reaches up to 53mSv/shot at 1m. For the laser-driven proton-induced neutron source, the neutron dose is ~159μSv/shot at 1m while the photon dose due to hot electrons is ~ 7.8mSv/shot. These results provide valuable information for the radiological protection of the X-ray produced in ultra-intense laser-solid interaction.
Radiation protection and shielding at current and planned laser facilities at SLAC

Taiee T. Liang*, Johannes M. Bauer¹, James C. Liu¹, Sayed H. Rokni¹
¹SLAC National Accelerator Laboratory
*tliang6@slac.stanford.edu

Experimenters at the Matter in Extreme Conditions (MEC) instrument at the SLAC National Accelerator Laboratory focus a 25 TW optical laser beam to spot sizes of few micrometres to achieve laser intensities up to \(10^{21}\) W/cm\(^2\). The interaction between the high-intensity laser and its target generates a plasma and is used to explore a wide variety of physics. Interactions with solid foils are used to study warm dense matter physics, pressure-driven shockwaves in materials, and conditions inside giant planets like Jupiter. In addition, high-intensity laser interactions with solid foils are also being studied as a potential source of MeV-range protons for medical applications.

High-intensity laser facilities such as MEC at SLAC and others worldwide present a unique radiological hazard: ionising radiation from optical laser light. The ionising radiation hazard is often composed of “hot” electrons from high-intensity laser-plasma interactions and the subsequent production of bremsstrahlung. The radiological protection group at SLAC has utilised the particle-in-cell code EPOCH to characterise the hot electron source term (energy spectrum, angular distribution, etc.) as a function of laser intensity between \(10^{17}\) and \(10^{22}\) W/cm\(^2\). This source term is then implemented in FLUKA Monte Carlo code to calculate the bremsstrahlung dose yield (mSv normalised to joules of laser hitting a solid target) and the tenth-value layer shielding thicknesses. This paper presents characterisation of the ionising radiation source term from high-intensity laser-solid interactions along with measurements from SLAC’s MEC and how these studies have guided radiation safety analysis and shielding design for the current terawatt-level and future petawatt MEC-U laser facility at SLAC.

Introduction

High power terawatt and petawatt lasers are known to generate a radiological hazard as a result of interactions between intense laser light and matter. The number of facilities worldwide that house such lasers continue to grow, and many laser systems at pre-existing facilities are being upgraded to achieve even higher laser powers. One such facility is the Matter in Extreme Conditions (MEC) instrument Linac Coherent Light Source (LCLS), which is depicted in Figure 1 and located at SLAC National Accelerator Laboratory (Nagler et al., 2015). At this laser facility, experimenters can focus the high power laser up to 25 TW (1 J in 40 fs) onto solid foils to study warm dense matter physics (Fletcher et al., 2015), pressure-driven shock waves in solid materials (Kraus et al., 2016), and conditions inside giant planets like Jupiter and Neptune. These laser systems can also be used to produce energetic beams of protons (Gauthier et al., 2016) and electrons and betatron X-rays (Albert et al., 2013) with the use of gas jets or cells as the laser’s targets.
Figure 1. The Matter in Extreme Conditions (MEC) instrument with its 25 TW (1 J in 40 fs) short-pulse laser system, which is located at SLAC’s LCLS


High intensity laser facilities like the MEC instrument at SLAC pose a unique radiation safety challenge of being able to generate ionising radiation hazards from optical laser light. When a high power laser is focused in vacuum onto a solid target, a plasma layer will be formed on the surface of the target. Additional interactions between the laser pulse and the plasma can accelerate plasma electrons to tens and even hundreds of MeV in energy (Tajima and Dawson, 1979; Wilks and Krueer, 1997), which are commonly referred to as “hot” electrons. The interaction of these “hot” electrons with both the target material and the vacuum chamber’s walls will generated bremsstrahlung photons (Chen et al., 2004), which can present an ionising radiation hazard to personnel.

The Radiation Protection (RP) group at SLAC has developed models to estimate the ionising radiation hazard and to design shielding from laser facilities like MEC as a function of laser-optic-target parameters (Liang et al., 2017b; Liang et al., 2018). These models were developed from calculations using EPOCH particle-in-cell (PIC) code (Arber et al., 2015) coupled with FLUKA Monte Carlo code (Ferrari et al. 2005; Böhlen et al., 2014), which were benchmarked with a series of measurements at SLAC’s MEC instrument (Liang et al., 2017a).

This paper details SLAC RP’s characterisation and mitigation of ionising radiation hazards generated from laser-matter interactions at MEC and also presents the preliminary shielding design for a petawatt-class laser facility at SLAC.

**Characterisation of hot electron source term with EPOCH**

The hot electron source term for laser interactions with solid targets has been characterised in previous studies at SLAC (Liang et al., 2017b) for the purpose of radiological protection. The studies utilised the EPOCH (Arber et al., 2005), which is a particle-in-cell (PIC) method code designed to study high energy density physics and laser-plasma interactions. The code is suitable for femtosecond timescale and micrometre size-scale interactions. Figure 2 plots a snapshot during a 2-dimensional EPOCH simulation of the electron density of a simulated plasma at 120 fs after a laser pulse with intensity of $10^{20}$ W/cm² (travelling left-to-right) interacts with the plasma. The incident laser pulse has created an indentation in the simulated plasma, and “hot” electrons are observed to be accelerated in both the
upstream and downstream directions in x. Similar simulations were performed for lasers with intensities between $10^{17}$ and $10^{22}$ W/cm$^2$ and were used to characterise the energy distribution, angular distribution and laser-to-electron conversion efficiency of hot electrons.

Figure 2. Snapshot of the electron density inside a 2D EPOCH simulation at 120 fs

The laser with intensity of $10^{20}$ W/cm$^2$ is emitted from the left boundary, propagates in x, and interactions with the simulated plasma.

The hot electrons in EPOCH simulations such as the snapshot shown in Figure 1 were found to have energies characterised by a Maxwellian distribution given as:

$$f(E) \sim E^2 \, e^{-\frac{E}{T_h}}$$

(1)

where E is the electron energy and $T_h$ is the hot electron temperature that characterises the slope the energy distribution. Figure 3 plots energy spectra of hot electrons that are accelerated by the laser pulse from 0-400 fs during the 2D EPOCH simulation ($10^{20}$ W/cm$^2$) in Figure 2. A Maxwellian distribution can be fitted and gives a $T_h$ of 2.1 MeV.

Figure 3. Hot electron energy spectra integrated from 0-400 fs during a 2D EPOCH simulation for $10^{20}$ W/cm$^2$

A Maxwellian fit of the hot electrons gives a $T_h$ of 2.1 MeV (Liang et al., 2017b).

Figure 4 plots the $T_h$ as a function of laser intensity between $10^{17}$ and $10^{22}$ W/cm$^2$, which were derived from fitting the Maxwellian distribution to the hot electron energy spectra. The $T_h$ calculated from EPOCH simulations were found to be in good agreement with ponderomotive scaling formulas given in literature (Wilks et al., 1992; Wilks et al., 1997; Kluge et al., 2011). For a laser wavelength of 0.8 μm, the hot electron temperature $T_h$ (MeV) scales with laser intensity $I$ (W/cm$^2$) as:

$$T_h = 1.05 \times 10^{-10} \times I^{0.514}. \quad (2)$$

Figure 4. Hot electron temperature as a function of laser intensity as calculated from 2D EPOCH simulations (Wilks et al., 1992; Wilks et al., 1997; Kluge et al., 2011)

As the laser pulse interacts with the plasma in Figure 2, a population of the hot electrons generated from laser-plasma interactions will be accelerated in the forward and backward directions relative to the laser direction. The trajectories of the hot electrons can be tracked in EPOCH, and the angular distribution of the hot electrons can be approximated with a Gaussian distribution as:

$$f(\theta) \sim e^{-\frac{\theta^2}{2\sigma^2}}. \quad (3)$$

where $\theta$ is the polar angle of the hot electron emission with respect to the laser direction and $\sigma$ is the standard deviation, which was found to be about 45°. Figure 5 plots an example of the angular distribution of hot electrons accelerated in the backward and forward directions from a 1020 W/cm$^2$ laser incident on a plasma (see Figure 2). The ratio of hot electrons emitted in the laser’s forward direction versus backward direction was calculated using EPOCH and scales with laser intensity as:

$$R(I) = 2.8 \times 10^{-9} \times I^{0.46} \quad (4)$$

where $I$ is the laser intensity in units of W/cm$^2$. As laser intensity increases, the hot electrons were found to be significantly more forward-peaked.
Figure 5. The angular distribution of hot electrons accelerated in the backward (left) and forward (right) directions from a $10^{20}$ W/cm$^2$ laser can be fitted with a Gaussian distribution with a $\sigma$ of 49° and 47°, respectively (Liang et al., 2017b).

The fraction of the laser pulse energy converted into hot electron energy was also calculated using EPOCH simulations. This laser-to-electron conversion efficiency is plotted in Figure 6 and scales with laser intensity from about 10% at $10^{17}$ W/cm$^2$ up to a maximum of about 60% starting at $10^{21}$ W/cm$^2$.

Figure 6. Laser-to-electron conversion efficiency scales as a function of laser intensity up to a maximum of about 60% (Liang et al., 2017b)

Calculation of bremsstrahlung dose yield with FLUKA

The bremsstrahlung dose yield from laser-solid interactions can be calculated using the hot electron source term characterised in the previous section: electron energies sampled according to a Maxwellian distribution (Equation 1) with exponential slope $T_h$ (Equation 2), angular distribution of electrons sampled according to a Gaussian distribution (Equation 3) with $\sigma$ of about 45°, forward-to-backward emission ratio (Equation 4), and laser-to-electron conversion efficiency (see Figure 6). The dose yield parameter (mSv/J) is the
ambient dose equivalent (mSv) of bremsstrahlung photons generated from hot electrons and normalised to the laser pulse energy on target (J).

FLUKA Monte Carlo code (Ferrari et al., 2005; Böhlen et al., 2014) was used to calculate the optimal bremsstrahlung yield generated from hot electrons interacting with 2 cm by 2 cm copper foils of thickness equal to one continuous-slowing-down approximation (CSDA) range for an electron with energy of 1.5Th (the mean energy of the Maxwellian distribution from Equation 1). Copper was chosen as the target material due to it being a common target of choice at MEC for laser-solid experiments. For other materials, the bremsstrahlung dose yield was found to scale with the atomic number: \( \sqrt{Z_{\text{solid}}} / \sqrt{Z_{\text{Cu}}} \).

Figure 7 plots the FLUKA-calculated bremsstrahlung dose yield at 1 metre distance as a function of laser intensity for laser-solid interactions. The attenuation effect of 2.54-cm-thick Al target chamber has already been accounted for, and the dose yield is provided for several angles relative to the laser’s direction. For laser intensities greater than about 1019 W/cm², the bremsstrahlung dose yield becomes increasingly forward-peaked. For lower intensities, the dose yield in the forward and backward directions are more 1:1.

**Figure 7. Estimated bremsstrahlung dose yield at 1 metre distance outside a simple 2.54 cm Al target vacuum chamber and for several angles relative to the laser direction (Liang et al., 2017b)**


**Measurements at SLAC’s MEC**

A series of measurements were performed by SLAC’s RP group to benchmark the bremsstrahlung dose yield model, which was developed from coupling EPOCH and FLUKA. These measurements were performed at SLAC’s MEC 25 TW short-pulse laser facility (Liang et al., 2017a), and one other measurement was performed at Lawrence Livermore National Laboratory’s Titan laser facility (Bauer et al., 2011). Ion chambers and passive dosimeters were used to characterise the bremsstrahlung dose yield generated from the laser-solid experiments at these laser facilities. Details of these measurements can be found in the references (Bauer et al., 2011; Liang et al., 2017a; Liang et al., 2018). Figure 8 plots a comparison of the measurement data with the bremsstrahlung dose yields developed from coupling EPOCH and FLUKA calculations. The spread in the measurement data is due to several factors such as different measurement angles, uneven target chamber attenuation, target thickness and uncertainties in the laser beam stability.
from shot-to-shot. Overall, the curves predict the dose yield well as a function of laser intensity. The FLUKA-calculated bremsstrahlung dose yield model combined with EPOCH’s hot electron source term characterisation provide guidelines for radiation hazard analysis for laser-solid interactions between $10^{17}$ and $10^{22}$ W/cm².

**Figure 8. Comparison of bremsstrahlung dose yield measurements with model developed from coupling EPOCH and FLUKA calculations (Liang et al. 2017b)**

![Comparison of bremsstrahlung dose yield measurements with model developed from coupling EPOCH and FLUKA calculations](image)


**Tenth-value layers for laser-solid interactions**

The bremsstrahlung radiation generated from high-intensity laser-solid interactions presents an ionising radiation hazard to personnel in the vicinity. Thus, radiation shielding is required to mitigate the bremsstrahlung dose generated from laser-solid experiments. FLUKA was used to calculate the transmission of bremsstrahlung dose (originating from a hot electron source) through common shielding materials used at SLAC’s LCLS experimental halls. The tenth-value layer (TVL) thickness is the thickness of material needed to reduce the ambient dose equivalent of the incident radiation (bremsstrahlung from hot electrons) by a factor of 10 and can be derived from the different materials’ transmission curves. The shielding materials evaluated were Pyrex glass (2.23 g/cm³), Portland concrete (2.3 g/cm³), aluminium (2.7 g/cm³), MagnaDense heavy concrete (4.0 g/cm³), iron (7.87 g/cm³), lead (11.34 g/cm³), and tungsten (19.25 g/cm³). The composition of MagnaDense heavy concrete is provided in Table 1 as its composition is not as commonly known.
Table 1. Material composition of MagnaDense heavy concrete

<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Cement</td>
<td>3.15</td>
<td>300</td>
<td>7.51</td>
<td>95.2</td>
</tr>
<tr>
<td>Water</td>
<td>1.0</td>
<td>167</td>
<td>4.18</td>
<td>167</td>
</tr>
<tr>
<td>Glenium</td>
<td>1.1</td>
<td>1.311</td>
<td>0.03</td>
<td>1.2</td>
</tr>
<tr>
<td>MagnaDense 8S</td>
<td>4.85</td>
<td>1.424</td>
<td>35.62</td>
<td>296.1</td>
</tr>
<tr>
<td>MagnaDense 20S</td>
<td>4.85</td>
<td>2.105</td>
<td>52.66</td>
<td>437.5</td>
</tr>
<tr>
<td>Heavy concrete</td>
<td>4.01</td>
<td>3,997.311</td>
<td>100</td>
<td>997</td>
</tr>
</tbody>
</table>


Figures 9 and 10 plot the TVL₁ and TVLₑ thicknesses for the different materials as a function of laser intensity from $10^{17}$ to $10^{22}$ W/cm². TVL₁ is the material thickness needed to reduce the incident bremsstrahlung dose by a factor of 10. TVLₑ is equilibrium tenth value layer and is the thickness needed to reduce the dose by each additional factor of 10. As observed in Figures 9 and 10, materials with higher Z and density are most effective for shielding the bremsstrahlung from hot electrons. Shielding with materials such as lead and tungsten require less material to attenuate the incident radiation compared with other materials at a given laser intensity. Pyrex glass, concrete and aluminium have similar TVL thicknesses due to having similar densities. When designing radiation shielding using Figures 9 and 10, it is necessary to use TVL₁ for the first TVL and TVLₑ for each subsequent TVL.

The TVL thicknesses for a variety of materials have been calculated to give flexibility in designing radiation shielding for a high-intensity laser facility. Pre-existing facilities that desire to upgrade their laser systems to achieve higher laser powers and intensities may not have the financial means or time to demolish and/or rebuild the facility housing when the wall shielding is no longer adequate. Therefore, these facilities may choose to use local shielding to mitigate the bremsstrahlung from hot electrons. When space is limited, lead is a common shielding material, but it comes with additional hazards such as chemical toxicity, potential for activation by high-energy bremsstrahlung, and material outgassing (if used inside vacuum). Therefore, tungsten and iron shielding may be chosen instead. Furthermore, as lasers continue to reach higher and higher intensities, neutrons generated from high-energy bremsstrahlung via photonuclear (γ,n) interactions may become significant. It is well known that hydrogenous material like concrete (both normal and heavy) are more effective in attenuating neutrons than higher Z materials like lead and tungsten. Depending on a high-intensity laser facilities layout and shielding needs, a combination of shielding configurations may be used to mitigate the bremsstrahlung dose from hot electrons.
Upgrade of MEC to petawatt-class laser facility

Many of these laser facilities have plans in the upcoming years to upgrade their laser systems to achieve even higher laser powers. For example, laser scientists at SLAC’s MEC instrument have plans to upgrade the current 25 TW (1 J in 40 fs) laser system to 3 PW (450 J in 150 fs) and even 20 PW (2 kJ in 100 fs) in the future. Table 17.2 provides the laser parameters for the petawatt laser upgrades planned for MEC. The 20 PW upgrade would require a completely new facility to house its laser system, and MEC plans to have the capability to interchangeably run either the 3 PW laser or 20 PW laser to the same target interaction chamber.
Table 2. Laser parameters for upgrade of MEC to petawatt-class laser

<table>
<thead>
<tr>
<th>Solid Targets</th>
<th>3 PW</th>
<th>20 PW</th>
</tr>
</thead>
<tbody>
<tr>
<td>Laser intensity</td>
<td>Up to $3 \times 10^{22}$ W/cm²</td>
<td>Up to $2 \times 10^{23}$ W/cm²</td>
</tr>
<tr>
<td>Pulse energy</td>
<td>150-450 J</td>
<td>2 kJ</td>
</tr>
<tr>
<td>Pulse duration</td>
<td>150 fs</td>
<td>100 fs</td>
</tr>
<tr>
<td>Spot size ($1/e^2$ radius)</td>
<td>3 μm</td>
<td>3 μm</td>
</tr>
<tr>
<td>Shots in 1 hour</td>
<td>60 (1 shot/minute)</td>
<td>60 (1 shot/minute)</td>
</tr>
<tr>
<td>Shots per experiment</td>
<td>100</td>
<td>100</td>
</tr>
<tr>
<td>Experiments per year</td>
<td>10</td>
<td>10</td>
</tr>
</tbody>
</table>


Figure 11 depicts a conceptual drawing of the MEC laser facility but reconfigured as a petawatt-class laser facility. The new facility will have a thick radiation shield wall that separates the target area room from the laser room where personnel may be during high-intensity laser shots on target materials. The radiation shielding required for the petawatt-class lasers is based on the maximum parameters of the 20 PW laser system ($2 \times 10^{23}$ W/cm²; 2 kJ; 100 shots per experiment; 10 experiments per year). At this laser intensity, 20 cm of heavy concrete (Magnadense in Table 1) is one TVL compared with about 50 cm of normal Portland concrete. Given the space constraints of installing a petawatt-class laser in the pre-existing MEC facility, heavy concrete was chosen as the shielding material.

The shielding design proposed by SLAC’s RP group is 1 meter of heavy concrete (Magnadense in Table 1) to meet a design criterion of 1 mSv in 1 year to personnel outside the facility (“Radiological Control Manual”, 2018). For each additional 20 cm of heavy concrete, then the petawatt laser facility can operate with 10x more laser shots on target (i.e. 1 000 shots per experiment and 10 experiments per year for 20 PW operation mode).

Figure 11. Conceptual layout for MEC petawatt-class laser with its radiation shield wall that separates the target area room from the laser room

Figure 12 plots an example of a FLUKA simulation of the radiation dose per laser shot, with a shot-on-demand 20 PW of 2 kJ focused to 2×10^{23} W/cm^2. This figure demonstrates the efficacy of a 1-m-thick heavy concrete wall in shielding the 20 PW laser. The hot electrons are emitted corresponding to a laser shooting from “northeast to southwest”, and the resulting forward-directed bremsstrahlung is directly challenging the radiation shield wall. This is a conservative scenario as administrative controls for high-intensity laser operations at SLAC will require the laser direction to be directed away from where personnel may be. During typical operation, the laser direction would instead be pointed towards the “right” of the figure, which is the end of the experiment hall where no personnel will be.

**Figure 12.** FLUKA simulation of radiation dose per laser shot for a 20 PW laser (2×10^{23} W/cm^2)


In addition to bulk wall shielding, the MEC petawatt laser facility will have a motorised radiation shield door, composed of a combination of lead (attenuates bremsstrahlung and X-rays) and borated polyethylene (attenuates neutrons). As seen from Figure 12, the thickness of the door may require further optimisation to prevent leakage of radiation from the doorway. Shielding design of penetrations in the shield wall (such as laser beam transport pipe, cabling, ventilation, etc.) will be designed by SLAC’s RP group in the next phases of the facility design plan.

**Other source terms from laser-matter experiments**

This section briefly covers two other source terms from laser-matter experiments at SLAC’s MEC. These two types of experiments comprise about one out of ten experiments at MEC per year, and although only a small fraction of MEC’s overall short-pulse laser experiments, they pose radiation hazards quite different from those described earlier in this paper.

**Electron acceleration experiments with gas targets**

Laser-matter experiments with millimetre-thick gas targets or cells can generate very forward-directed beams of electrons (with laser direction). The mechanism laser wakefield acceleration (LWFA) can easily accelerate electrons to a few-hundred MeV (Albert et al., 2013). For such an experiment at MEC, SLAC’s RP group deployed local 8-cm-thick
tungsten and 10-cm-thick lead shielding inside the MEC target chamber as depicted in Figure 13. The local shielding acted as beam dumps to intercept the hundred MeV electrons. Passive dosimeters that were deployed around the target chamber and outside the experimental hall measured 60 mSv (at about 15° relative to laser direction) and 85 μSv, respectively, integrated over the course of the experiment. This 60 mSv integrated dose from the experiment was expected by the RP group and had been estimated by FLUKA calculations. In addition, no personnel can be in the MEC experimental hall during laser shots on target. This gas target experiment at MEC demonstrates the need to properly identify the radiation source term for non-standard high-intensity laser experiments and to shield accordingly to mitigate ionising radiation hazards to personnel.

**Figure 13. Local tungsten and lead shielding deployed inside the MEC target chamber to intercept the hundred MeV electron beam generated by a LWFA gas target experiment at SLAC**


**Proton and ion acceleration experiments**

Acceleration of protons and ions is also achievable with a short-pulse high-intensity laser by target normal sheath acceleration (TNSA). As hot electrons are accelerated in the forward direction of the laser, charge separation created by the electrons also accelerate the ions in the plasma. This acceleration mechanism can be utilised with both liquid H₂ jet targets (Gauthier et al., 2016) or solid targets (MacKinnon et al., 2002). Figure 14 plots the proton energy spectra calculated from EPOCH simulations of a laser with intensity $10^{20}$ W/cm² interacting with a plasma consisting of electrons and hydrogen ions (or protons). These results were benchmarked against the study by MacKinnon et al. (2002), and as can be observed, the energies of protons can reach a maximum of about 20 MeV. For such laser-matter experiments, RP must evaluate whether there is need for a beam dump to intercept the generated proton beam and also if neutron shielding is required around the dump to mitigate neutrons from (p,n) interactions.
Figure 14. Proton energy spectra from 2D EPOCH calculation at increasing time increments during the simulation

The laser intensity was $10^{20}$ W/cm$^2$, and interacted with a plasma consisting of electrons and hydrogen ions (or protons).


**Summary**

SLAC’s RP group has developed models for performing hazard analysis and for developing radiological controls for high-intensity laser-solid experiments at SLAC for intensities between $10^{17}$ and $10^{22}$ W/cm$^2$. The models were based on characterisation of the hot electron source term with the PIC code EPOCH and the Monte Carlo code FLUKA and were benchmarked with several measurements at SLAC’s MEC laser facility. The models are also used to design the radiation shielding for a petawatt laser upgrade to MEC, which would require on the order of 1 metre of heavy concrete. Other experiment and target types at MEC (electron beam acceleration with gas targets; proton and ion beam acceleration) have been analysed case-by-case, and SLAC’s RP group has developed radiological controls for them to mitigate the ionising radiation hazard generated by such laser-matter experiments.

**Acknowledgements**

This material is based upon work supported by the US Department of Energy, Office of Science, Office of Basic Energy Sciences, under Contract No. DE-AC02-76SF00515.

**References**


A novel active detector for the Bremsstrahlung source term measurement in ultra-intense laser-plasma experiments

Anna Ferrari\(^1\),*, Maria Molodtsova\(^1\),\(^2\), A. Laso\(^1\), D. Stach\(^1\), D. Weinberger\(^1\),\(^2\), Tom Cowan\(^1\)

\(^1\)Helmholtz-Zentrum Dresden-Rossendorf, Bautzner Landstraße 400, 01328 Dresden (Germany)
\(^2\)Technische Universität Dresden, Dresden (Germany)

*a.ferrari@hzdr.de

Ultra-intense laser-matter interaction physics is of growing interest worldwide, because of its ability to create new extreme states of matter and to explore technologically interesting processes such as new concepts for particle acceleration, material science and fusion energy.

A key aspect in laser-solid interaction is the acceleration of relativistic electrons with a Maxwellian-like energy spectrum and a temperature strongly coupled to the laser intensity. Their transport and interaction in the target and in the surroundings materials generates in a sub-ps timescale an ultra-intense bremsstrahlung radiation (gamma flash) with a very high intensity (up to \(~10^{10}\) photons). The measurement of this source term is crucial both for understanding plasma dynamics and then fully explain the electron acceleration mechanism inside the plasma and to assess a proper shielding.

Due to the high instantaneous gamma fluxes, associated with EMP pulses typical of ultra-intense laser-plasma environment, usual spectrometry techniques using pulse height analysis cannot be used.

The concept of a novel active bremsstrahlung detector, based on a multi-layered scintillator structure and SiPM read-out, is here presented. It allows the characterisation of the longitudinal development of the radiation, and by measuring the deposited energy in each layer the photon spectrum can be reconstructed by using an unfolding technique. Optimising a structure crystals/absorbers with variable density and depth appeared to be the key point to perform a successful unfolding. Via extensive FLUKA Monte Carlo simulations, the detector was optimised to resolve the photon spectrum in the dynamic range between 50 keV and 20 MeV, and the most promising model was chosen to realise the first calorimeter prototype. This prototype was tested in 2017 and 2018 with the well-characterised bremsstrahlung spectra provided by the gELBE beamline at the Helmholtz-Zentrum Dresden-Rossendorf (HZDR): the reconstruction of the 11 MeV, 13 MeV and 15 MeV endpoints, together with the demonstrated ability to discriminate between more and less “hardened” spectra, served as proof of concept of the detector. In a second moment, the multi-layer bremsstrahlung calorimeter has been working for the first time in the ultra-intense gamma fields provided by the DRACO laser facility at HZDR.

The optimisation process, together with the gELBE measurements results and the first preliminary measurements at DRACO, are here presented and discussed.
Session IV: Code status and medical and industrial accelerators
New analytical method to estimate systematic uncertainty in PHITS

Shintaro Hashimoto*, Tatsuhiko Sato
1Japan Atomic Energy Agency
*hashimoto.shintaro@jaea.go.jp

Particle and Heavy Ion Transport code System (PHITS) (Sato T. et al., 2018) can deal with various particles such as neutrons, protons, and heavy ions on the basis of the Monte Carlo method, and simulate their behavior in radiation facilities. PHITS has been used for many purposes such as shielding calculation in accelerator facilities. The number of the PHITS users is over 3 000 in Japan. PHITS is available via Data Bank of the Nuclear Energy Agency (NEA DB) and Radiation Safety Information Computational Center (RSICC) for users in some other countries.

Reliability of the Monte Carlo particle transport simulation code is usually evaluated by calculating statistical uncertainties related to the number of trials. However, the systematic uncertainty caused by uncertainty of input parameters such as water content of concrete material is also very important to confirm accuracy of calculated results quantitatively. Because accurate values of the water content or compositions of the used concrete in facilities is often unknown, the systematic uncertainty in the calculation might be large as in Ref. (Aalto E.J., 1965).

To estimate the systematic uncertainty of the calculated result, we plan to introduce ANalysis Of VAriance (ANOVA) (Casella, 2008) as a standard analytical method. ANOVA was proposed to analyse experimental data of biology. This method can estimate both systematic and statistical uncertainties; this is an important feature in analysing calculated results of the Monte Carlo simulation, where the mean value and the two uncertainties vary and then finally converge as the number of trials increases. To use the ANOVA method, we must repeat the PHITS calculation with a certain number of trials varying a value of an input parameter such as the water content. From their results, ANOVA gives estimated values of the mean value as well as those of the systematic and statistical uncertainties.

This presentation shows the current status of PHITS including recent improvements and introduce the ANOVA method and its result for neutron shielding calculation.

Figure 1 shows the neutron fluence in the concrete material of 50 cm thickness induced by the 20 MeV neutron. Error bars represent systematic uncertainties of the calculated fluence, where the statistical one becomes negligible by setting a large number of trials. These results were calculated by PHITS with assuming that the uncertainty of the water content is 30% of a default value. The uncertainties become gradually larger as the depth increases, because the number of neutron collisions changes by the influence of varying the water content.
Figure 1. The neutron fluence in the concrete material induced by the 20 MeV neutron


References


Stray neutron measurements in scanning proton and carbon ion therapy

Vladimir Mares1, Sebastian Trinkl2, Martin Dommert3, Thomas Tessonnier4, Marek Wielunski1, Werner Rühm1, Katia Parodi4,5

1Helmholtz Zentrum München, Institute of Radiation Protection, 85764 Neuherberg, Germany
2Federal Office for Radiation Protection (BfS), 85764 Neuherberg, Germany
3Physikalisch-Technische Bundesanstalt, Bundesallee 100, 38116 Braunschweig, Germany
4Heidelberg University Hospital, Department of Radiation Oncology, 69120 Heidelberg, Germany
5Ludwig-Maximilians-Universität München, Department of Medical Physics, 85748 Garching, Germany

*mares@helmholtz-muenchen.de

Ion beam therapy is rapidly growing around the world, with 56 proton beam facilities and 10 carbon ion facilities in operation on January 2017. Further 61 facilities are either planned or under construction. While proton therapy is the most commonly form of particle therapy, use of carbon ions is also growing and being actively researched.

Particle therapy is of particular benefit in treating cancers that are difficult to treat with surgery and where conventional radiotherapy would damage surrounding tissue to an unacceptable level. Furthermore, protons in particular are becoming increasingly accepted for the treatment of selected paediatric cancers, where the need to avoid secondary radiation-induced tumours is evident due to the potential long-term survival of the young patients. Further, there is some evidence that heavy ions, including carbon ions, due to their higher relative biological effectiveness (RBE), may be more effective in treating radio-resistant tumours than low-LET radiation.

Long-term side effects following radiation therapy are related to the dose deposited in the healthy tissue surrounding the tumour. In ion beam therapy the major contribution to the dose in out-of-field regions is delivered by neutrons, which are produced by nuclear interactions of the primary protons or carbon ions with the beam line components, the patient’s body and the components in the treatment room. In view of this fact, the measurement of stray radiation (mainly secondary neutrons) in the treatment room around phantoms irradiated with protons or carbon ions is a very important first step for estimating health risks of developing second cancers among patients.

In this study we report on the neutron spectrometry and dosimetry performed at the Heidelberg Ion-Beam Therapy Center (HIT) in Germany, using state-of-the-art pencil-beam-scanning (PBS) delivery with proton and carbon ions. Two different energy layers with a field size of 11 x 11 cm² within 30 x 30 x 30 cm³ PMMA phantom were irradiated with mono-energetic protons (75 MeV/u; 140 MeV/u) and carbon ions (138 MeV/u; 264 MeV/u) simulating similar Bragg-peak depths. The secondary neutron energy distributions were measured with extended range Bonner sphere spectrometer (ERBSS) at four positions around the phantom: at 0° and a distance of 1.5 m from the isocenter, and at 45°, 90°, 135° at a distance of 2 m from the isocenter.

It was found, that the neutron fluence and ambient dose equivalent (H*(10)) values for both, carbon ions and protons, have a strong energy and angular dependence. The high-energy (E > 19.6 MeV) neutrons contribute between 1% up to around 70% to the neutron fluence, while fast (100 keV ≤ E < 19.6 MeV) neutrons contribute between 20% and 50%. In terms of neutron H*(10), fast and high-energy neutrons contribute over 90% to the neutron dose for both protons and carbon ions, at all measurement positions concerned in this study.
Acknowledgements

We would like to acknowledge staff of the Heidelberg Ion-Beam Therapy Center (HIT) for giving us the opportunity to perform spectrometry measurements. We would like to give special thanks to Stephan Brons from HIT for his professional on-site support.
Radiological estimation and validation for the accelerator-based boron neutron capture therapy facility at the Ibaraki Neutron Medical Research Center

Hiroshi Nakashima, Takemi Nakamura¹, Hitoshi Kobayashi², Susumu Tanaka, Hiroaki Kumada³
¹Japan Atomic Energy Agency, ²High Energy Accelerator Research Organization
³University of Tsukuba, Japan
* nakashima.hiroshi@jaea.go.jp

Aiming of development of facilities for boron neutron capture therapy (BNCT) that can be installed in hospitals, an accelerator-based BNCT facility is being developed at the Ibaraki Neutron Medical Research Center under a collaboration among the Japan Atomic Energy Agency, the High Energy Accelerator Research Organization, the University of Tsukuba, and other institutions. It consists of a proton accelerator, having a maximum beam power of 80 kW with a proton energy of 8 MeV and a maximum average beam current of 10 mA, and a target–moderator–collimator–shield (TMCS) system. For the design concept, to satisfy the BNCT beam conditions and achieve a low activation, the radiation behaviour in the TMCS system was simulated by the Monte Carlo method and this system configuration was optimised accordingly. In addition, the radiation estimation of the TMCS system was verified via several experiments and its applicability for BNCT was proved. This report reviews the estimation and validation studies for the development of the accelerator-based BNCT facility.

Introduction

Boron Neutron Capture Therapy (BNCT) is a cancer treatment method, in medical terms, a noninvasive therapeutic modality for treating locally invasive malignant tumours. Its first step consists in the administration of drugs containing stable boron isotopes ($^{10}$B) to the tumour; after being irradiated with neutrons, $^{10}$B releases high-linear energy transfer (high-LET) α and $^7$Li particles through the boron neutron capture reaction, $^{10}$B(n,α)$^7$Li. The effectiveness of this method has been demonstrated all over the world; for example, clinical trials for malignant brain tumours have been conducted at the University of Tsukuba using the Japan Research Reactor 4 (JRR-4) of the Japan Atomic Energy Agency ( JA EA). However, it is still not widely adopted because of the many limitations associated with the use of nuclear reactors (Nakagawa, et al., 2003; Horiguchi, et al., 2001).

In recent years, the remarkable progress of accelerator technology has enabled the installation of accelerator-based BNCT facilities in hospitals, and now, various accelerator-based BNCT related initiatives are being promoted all over the world (Green, 1998, Kreiner, et al., 2014). Among them, a project for an accelerator-based BNCT facility at the Ibaraki Prefecture of Japan (iBNCT) is being conducted under the collaboration among the University of Tsukuba, the High Energy Accelerator Research Organization, JAEA, the Hokkaido University, and the Ibaraki Prefecture. The radiological estimation for the facility was carried out via Monte Carlo simulation based on the recommendations by the International Atomic Energy Agency (IAEA) for the sophisticated configuration of accelerators and target–moderator–collimator–shield (TMCS) systems. The facility has
been constructed at the Ibaraki Neutron Medical Research Center and already passed the licensing examination (Kumada et al., 2014; Yoshioka et al., 2014, Kobayashi et al., 2012). In this paper, the radiological estimation and experimental validation studies for this iBNCT facility are reviewed.

Radiological estimation and design

The iBNCT facility consists of a proton accelerator, having a maximum beam power of 80 kW with an 8 MeV proton energy and a maximum average beam current of 10 mA, and a TMCS system. Figure 1 shows its schematic view and Table 1 summarises the main parameters.

![Figure 1. Schematic view of the Ibaraki boron neutron capture therapy facility](source: JAEA, 2020.)

Table 1. Main parameters of the Ibaraki boron neutron capture therapy facility

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Proton energy</td>
<td>8 MeV</td>
</tr>
<tr>
<td>Average current</td>
<td>&gt; 5 mA (Maximum 10 mA)</td>
</tr>
<tr>
<td>Peak current</td>
<td>50 mA</td>
</tr>
<tr>
<td>Beam width</td>
<td>1 ms</td>
</tr>
<tr>
<td>Beam duty</td>
<td>20 %</td>
</tr>
</tbody>
</table>


Concept

The most important concept is realising an accelerator-based BNCT facility that has a neutron beam performance equal to or higher than that of a reactor-based one. To obtain a sufficient neutron beam intensity, the system must accelerate a high-intensity proton beam. Additionally, to be installed in hospitals, its operation and maintenance handling should be easy; for high-power proton accelerators, the most serious issue in this viewpoint is to reduce excess radiation exposure of patients and workers during the system activation.
Thus, the second main concept of this project is to achieve activation of the devices and the room as low as possible.

**Accelerators**

As high-power proton accelerators, a Radio Frequency Quadrupole (RFQ) and a drift-tube linear accelerator (linac) (DTL) are selected. These two technologies are applied as front-end linac of the Japan Proton Accelerator Research Complex (J-PARC), which has a track record of high-power continuous operation for over 10 years. To improve the proton beam stability, the high duty factor by 20%, and the maximum average beam current over 5 mA, accelerators can be modified, for example, by using permanent magnets. The linac has a length of about 8 m and a maximum width of 1.5 m; the RFQ output energy is 3 MeV and the further energy gain from the DTL is 5 MeV.

For the low activation, the acceleration energy is set to 8 MeV, although the Be neutron yield increases with the proton energy. The maximum neutron energy, achieved via the Be(p,n) reaction with the 8 MeV protons, is about 6 MeV; it is below the threshold energy of the main activation reactions of structural elements such as Al, Fe and Pb.

**Design criteria and calculation method**

The TMCS system mainly consists of a target, a moderator, a collimator and surrounding shields. Due to patent constraints, only its conceptual diagram is shown in Figure 2. The TMCS system has been optimised with design criteria based on the recommendations in the IAEA Technical Document 1223 (TECDOC-1223) (IAEA, 2011), as summarised in Table 2, to achieve a neutron beam performance applicable for BNCT. The recommendations specify effective neutron energy range, intensity, spread, and size as well as the permissible contamination ratios of fast and thermal neutrons and gamma-rays for BNCT. Moreover, based on the JRR-4 experience, some additional criteria such as the neutron beam flatness and the external dose rate per patient have also been considered for the optimisation.

*Figure 2. Conceptual diagram of the target–moderator–collimator–shield system*

Table 2. Neutron beam design criteria for the target–moderator–collimator–shield system of the Ibaraki boron neutron capture therapy facility, based on the International Atomic Energy Agency Technical Document 1223

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Required values</th>
</tr>
</thead>
<tbody>
<tr>
<td>Energy range</td>
<td>0.5 eV–10 keV</td>
</tr>
<tr>
<td>Epithermal beam intensity</td>
<td>$1 \times 10^9 \ (n\ cm^{-2}\ s^{-1})$ desirable minimum</td>
</tr>
<tr>
<td></td>
<td>$&gt; 5 \times 10^8 \ (n\ cm^{-2}\ s^{-1})$ useable</td>
</tr>
<tr>
<td>Fast neutron component</td>
<td>$&lt; 2 \times 10^{-13} \ (Gy\ cm^2)\ per\ epithermal\ neutron$</td>
</tr>
<tr>
<td>Gamma-rays component</td>
<td>$&lt; 2 \times 10^{-13} \ (Gy\ cm^2)\ per\ epithermal\ neutron$</td>
</tr>
<tr>
<td>Ratio of thermal and epithermal neutron fluxes</td>
<td>$&lt; 0.05$</td>
</tr>
<tr>
<td>Ratio of total neutron current and flux</td>
<td>$&gt; 0.7$</td>
</tr>
<tr>
<td>Beam size</td>
<td>12–14 cm (17 cm for glioblastoma)</td>
</tr>
</tbody>
</table>


For the configuration optimisation, the parametric study has been carried out by Monte Carlo simulation using the PHITS2 (Sato et al., 2013) and MCNP5 (X-5 Monte Carlo Team, 2003) codes with the JENDL–4 (Shibata et al., 2011) and ENDF/B–VII (Chadwick et al., 2006) nuclear data libraries. The angular dependent neutron spectra shown in Figure 3 were used as neutron sources in the simulations; the spectra were obtained by the interpolation of the incident proton energy based on the neutron energy spectra measured using a thick Be target and an 11 MeV proton energy (Kamata et al., 2011). The total neutron yield was normalised to $1.03 \times 10^{10} \ n/\mu C$ by the reaction cross-section at 8 MeV.

**Figure 3. Angular dependent neutron spectra used as neutron sources in the simulations**

**Target system**

The neutron production target also seriously affects the activation aspect. At first, light elements such as Li and Be were nominated as target candidates in this project. Li has an advantage on the incident proton energy because the cross section of the $^7\text{Li}(p,n)^7\text{Be}$ reaction has a peak around the energy of 2.5 MeV and lower acceleration energy is enough compared with Be. However, tritium (T) is produced when using Li and, due to its chemical characteristics, it is difficult to handle it in hospitals. Hence, Be has been finally selected to realise the neutron production target.

However, a Be target also presents a serious problem that could overcome its favourable mechanical and chemical characteristics. This problem is its blistering; in the case of high-power accelerators, a lot of protons hit the target and are absorbed in it, forming hydrogen blisters that can deteriorate the target. This phenomenon may decide the target life. A new technique has been used to solve this problem; it consists in loading a blistering mitigation material between the metal Be target and a Cu heat sink by diffusion bonding with hot isostatic pressing. The blistering mitigation material absorbs most of the hydrogen by using a 0.5 mm thick Be, although the neutron yield decreases compared with a thick target. The thickness is less than the Bragg peak position in the proton range of Be. The three-layer structure neutron target system is outlined in Figure 4 (Kumada et al., 2015).

The three-layered Be target system is cooled by the water flowing in the Cu heat sink because the deposited power energy in the system reaches a maximum of about 80 kW.

**Moderators**

In the moderator design for BNCT applications, the materials, sizes and locations of the components, that is, the moderators and the neutron and gamma-rays filters, have been
optimised to effectively generate epithermal neutrons in the energy range between 0.5 eV and 10 keV.

As the main moderator, fluoride or a mixture thereof is usually used to generate epithermal neutrons because fluorine has a unique energy dependency on the capture cross section. Some F resonance peaks are just above 10 keV and the cross section increase towards lower energies is less than a few keV, making fluorine an ideal element to produce epithermal neutrons. Based on these parametric studies on various fluorides, magnesium fluoride has been selected.

As regards fast neutron filtering, a W filter located just in front of the Be target effectively reduces fast neutrons by inelastic scattering in the energy region above 100 keV. A Fe filter is also included to further reduce them because of its small absorption cross-section compared with the W one. Thus, a combination of W and Fe filters is adopted in this design.

Since reducing the external exposure dose per patient is a crucial issue, a Bi plate is located near the outlet of the TMCS system to reduce the secondary gamma-rays generated in it, mainly Fe and Mg. Compared with a Pb plate, a Bi one also slightly increases the epithermal neutrons.

The space around the target system increases the number of stray neutrons through the proton beam duct, increasing the activation of the accelerator devices, the room wall and air, and also the moderators. Therefore, it should be filled with low-activation materials as much as possible. In the tested design, this space has been filled using Al block; Al also increases the epithermal neutrons by acting as a neutron reflector, although it slightly increases the fast neutrons as well.

**Collimator**

The collimator is key to preventing the of the neutron beam divergence, which affects the thermal neutron distribution in patients. In addition, the collimator contributes to reducing the secondary gamma-rays emitted from the outlet. In the tested design, LiF-loaded polyethylene is used as collimator material to satisfy these conditions.

**Shielding**

The moderators and collimator are surrounded by shields to reduce leakage neutrons and gamma-rays so that the prevention of excessive patient exposure is ensured. Moreover, the doses at the controlled area and site boundaries should satisfy the licensing regulations. The shields act also as neutron reflectors, increasing the epithermal neutrons. Considering the nuclear performance and ease of use at hospitals, Pb has been selected as the inner shield material, while a second neutron shield surrounding the inner one is made of an ordinary concrete.

**Calculated neutron beam**

The neutron energy spectrum calculated at the outlet of the TMCS system is compared with the JRR-4 one in Figure 5. It satisfies almost all the design criteria and, in addition, its absolute intensity is higher than that of the JRR-4 one, even at 5 mA. Furthermore, since it is shifted towards the high-energy region, the neutrons reach higher penetration depths into patients, suggesting an increase in the therapeutic effect.
Figure 5. Comparison of calculated neutron spectra of the Ibaraki boron neutron capture therapy (iBNCT) facility and the Japan Research Reactor 4 (JRR-4) at the outlet of the target–moderator–collimator–shield system


To estimate the thermal neutron flux in patients for the iBNCT facility case, the thermal neutron flux distribution in a water phantom installed at the TMCS system outlet was calculated. Figure 6 relatively compares this calculated distribution along the beam axial direction with that of JRR-4 (Nakamura et al., 2011). The peak position of the iBNCT thermal neutrons is shifted towards a deeper position due to the harder neutron spectrum of iBNCT than of JRR-4. Thus, the therapeutic range in tumours would be also expanded.

Figure 6. Comparison of calculated thermal neutron flux distributions of the Ibaraki boron neutron capture therapy (iBNCT) and the Japan Research Reactor 4 (JRR-4) along the beam direction in a water phantom

Experimental validation

To validate the calculated neutron beam performance, some experiments have been carried out. In the first experiment (Masuda et al., 2017), the neutron energy spectra at 1 and 2 m from the TMCS system outlet were measured using a Bonner ball detector having a maximum diameter of 0.127 m. Figure 7 compares the measured and calculated spectra at these positions; they are in very good agreement in both cases, validating the calculations and demonstrating that the iBNCT facility generates neutrons suitable for BNCT.

In the second experiment, the thermal neutron distribution in the water phantom was measured using an activation method with gold wires. A comparison between the measured and calculated distributions is shown in Figure 8. In this figure, experimental distribution is normalised to the proton beam intensity of 1mA, and calculation is normalised to the experiment. The relative shapes are in very good agreement. Thus, calculated neutron spectrum is validated by the experiment.
The gamma-ray dose resulting from residual radioactivity has also been measured. The water phantom was placed at the patient irradiation position and irradiated with the same amounts of neutrons used for actual treatments; seven minutes after the irradiation, the dose was measured with a survey metre. The measured value was 40 μSv/h, which is significantly lower than the calculated one.

**Summary**

The radiological estimation for the iBNCT facility was performed via Monte Carlo simulation under the design criteria based on the IAEA recommendations and the JRR-4 experience. The iBNCT has been fabricated based on this estimation, passed the licensing examination, and has been completed at the end of 2016. Some experiments validated the estimation and demonstrated that the system can generate epithermal neutron beams with a sufficient intensity for BNCT. In addition, the low residual gamma-ray dose indicated that the low activation condition has been achieved.

As future plans, a nonclinical study (cells and mice irradiations) requiring from clinical trials will be carried out in 2018. After that, a clinical study phase I about malignant melanoma was planned for 2019.

**Acknowledgements**

This work was supported in part by JSPS KAKENHI Grant Number 26460738.

**References**


Activation assessment of self-shielded and non-self-shielded PET cyclotrons

Nam-Suk Jung1*, Arim Lee1, Leila Mokhtari Oranj1, Hee-Seock Lee1, Da Yeong Gwon2, Yongmin Kim2, June-Ho Cho3, Ssang-Tae Kim3
1Pohang Accelerator Laboratory, POSTECH, Korea
2Catholic University of Daegu, Korea
3Carecamp Inc., Korea
*nsjung@postech.ac.kr

As of the end of 2017, there are 41 cyclotrons in operation in Korea. These cyclotrons are mainly used for the production of $[^{18}\text{F}]$ FDG. Because these cyclotron facilities are expected to dismantle faster than the design life due to upgrade, changes of purpose and place of use, etc., it is necessary to understand the activation level of the cyclotron facility to prepare for decommissioning.

To understand the difference of activation characteristics between the self-shielded and the non-self-shielded cyclotron, we performed activation experiments on same cyclotron, GE PETtrace. The results from the self-shielded cyclotron and the non-self-shielded cyclotron were compared. The thermal neutron flux related to the activation was measured using Au foils. With respect to metal and concrete activation, specimens were used because we were not able to sample from active cyclotron facilities. Copper, aluminum alloy, carbon steel and stainless steel, which constitute the cyclotron, were attached to the outer surface of the cyclotron, and after being took out for a certain irradiation time. The major radionuclides such as Co-60, Mn-54, were identified and quantitated by gamma spectroscopy. Especially in case of non-self-shielded cyclotron, we made a stacking structure of concrete at 3 cm intervals in the vault, and the depth profile of Eu-152 was measured.

Measured thermal neutron flux was $10^6$ n/cm$^2$’s near the FDG target. The difference of the metal activation between the type was discussed. For the concrete activation for non-self-shielded cyclotron, maximum location of the thermal neutron flux and the specific activity of interested radionuclide was found at the depth of 9 cm. The FLUKA results were 2 to 4 times (non-self-shielded) larger than measured results. This is due to the difference of thermal neutron flux between calculation and measurement.

Introduction

The medical cyclotrons in Korea produce mainly radiopharmaceuticals. The production of $[^{18}\text{F}]$ FDG, which is used in PET, is largest among the radiopharmaceuticals (Gwon, 2018). The upcoming issue of the PET cyclotron is the decommissioning. Large number of decommissioning are expected due to the upgrade or the change of the purpose rather than the machine life. When the PET cyclotron can no longer make a profit any more at the medical centre, it will be dismantled. Until now, two PET cyclotrons in Korea have moved to different place in Korea and Philippine (Woo, 2013; Hwang, 2017). Since July 2015, the ministry of food and drug safety in Korea has applied GMP (Good Manufacturing Practice), a system for ensuring that products are consistently produced and controlled according to quality standards, to radiopharmaceuticals including $[^{18}\text{F}]$ FDG, as decreed by the Korean Ministry of Food and Drug Safety. Therefore, in order to produce the $[^{18}\text{F}]$ FDG in Korea,
the medical centre should create the system suitable for GMP including the installation of additional facilities such as a white room, and obtain the GMP certification. This change is the big burden for small medical centre, which is one of the factor to accelerate the decommissioning of PET cyclotrons.

The purpose of this study is the establishment of the activation evaluation techniques. To prepare for decommissioning, it is necessary to understand the activation level of PET cyclotron facilities. Especially, the understanding of the difference of activation characteristics between the self-shielded and the non-self-shielded cyclotron is required. We performed activation experiments using the same cyclotron, GE PETtrace.

**Method**

The GE PETtrace has typically been used for the production of \[^{18}\text{F}\] FDG by 16.5 MeV, 40 to 55 \(\mu\)A proton beam. Production targets are located at the beam port 1 and 4. The main shielding material of the self-shielded GE PETtrace is boron-added (3.5\%) water, and the polyethylene and the lead are installed around the small space around the target as shown in Figure 1. The non-self-shielded GE PETtrace has been installed in the concrete vault as shown in Figure 2, and the internal dimension of the concrete vault is 4.7 m(W) x 4.6 m(D) x 3.2 m(H).

For the evaluation of the radioactive products, it is best to use a core boring to take samples from the facility. However, it was forbidden because both PET cyclotron facilities were in operation. Hence, we made several specimens and irradiated them for a certain period. Metals had chosen the materials such as copper, aluminium alloy, carbon steel and stainless steel that make up the cyclotron itself. Positions of the metal sample were in front of the production target 1 as shown in Figure 3.

In the case of concrete for the non-self-shielded cyclotron, we made a stacking structure to get the depth profile of interested radionuclide such as Eu-152 along the vault inside as shown in Figure 4. The outer size of the structure was 60 cm(W) x 40 cm(D) x 60 cm(H). At the centre of the structure, we installed 13 concrete samples to get the depth profile. The size of each concrete sample was \(\Phi 10 \times 3\) cm. In addition, we measured thermal neutron flux at the sample position to understand the production of radionuclide by the Au and Cd-covered Au foils. After the irradiation, the activity concentration and the thermal neutron flux were obtained using an HPGe detector.
Figure 1. Input geometry of the self-shielded GE PETtrace cyclotron for FLUKA

Source: Pohang Accelerator Laboratory, POSTECH, 2020.

Figure 2. Input geometry of the self-shielded GE PETtrace cyclotron for FLUKA

Source: Pohang Accelerator Laboratory, POSTECH, 2020.

Figure 3. Metal sample position for type-comparison, Left is for self-shielded cyclotron and right is for non-self-shielded cyclotron

Source: Pohang Accelerator Laboratory, POSTECH, 2020.
Figure 4. Concrete sample structure for non-self-shielded cyclotron to get the depth profile along the vault inside

Source: Pohang Accelerator Laboratory, POSTECH, 2020.

**Result and discussion**

**Metal activation**

Figure 5 and Figure 6 show the comparison results of metal activation for the self-shielded type and the non-self-shielded type. We compared the results with the experiments using mainly the same target of port 1. The used time of the first target was 70% for the self-shielded type and 99% for the non-self-shielded type. In the OFHC (Oxygen Free High Conductivity) Cu, Cu-64 and Co-60 were measured, and in the aluminium alloy, Na-24 was measured. In the stainless steel (SUS-304) and carbon steel (S10C), we commonly measured Mn-54 and Fe-59 from Fe atom, and Cr-51 from Cr atom. Especially in the SUS-304, Co-60 and Co-58 were measured because the number of Co and Ni atoms in SUS-304 was higher than the number of atoms in carbon steel. Among the measured radionuclides, Cu-64 produced in OFHC Cu had the highest saturated activity concentration, but Cu-64 has a half-life of 12.7 hr, so it will not be issued during the decommissioning. Likewise, since the half-life of Na-24 is short (15 hr), aluminium alloys will not be issued. Considering the half-life and saturated activity concentration, the radionuclides that can be a problem in the metal during the decommissioning were Co-60 and Mn-54.
The comparison of the saturated activity concentration of radionuclides by the type and the position according to the neutron energy could be summarised as follows:

(Termal neutron-induced reaction)
Position 1 for self-shielded > Position 1 for non-self-shielded
\[ \cong \] Position 2 for non-self-shielded > Position 2 for self-shielded

(Fast neutron-induced reaction)
Position 1 for self-shielded $\equiv$ Position 1 for non-self-shielded

> Position 2 for self-shielded $\equiv$ Position 2 for non-self-shielded

This difference is due to the neutron spectrum difference. Figure 7 shows the neutron spectrum at each position by FLUKA code (Ferrari, 2005). The tendency of the calculated thermal neutron is similar with the measured concentration of radionuclides. For the fast neutron-induced reactions which are \((n,p)\) reaction in two steels and \((n,\alpha)\) reaction in aluminium, the reaction cross-section is highest near the neutron energy of 6 to 13 MeV. In this neutron energy region, calculated neutron flux at each position were similar regardless of the type.

Hence, the FLUKA results and the experimental results tend to match qualitatively. However, the FLUKA results in the quantitative comparison of the thermal neutron flux are 2 to 4 times larger than the measured ones.

**Figure 7. Calculated neutron spectrum by FLUKA and reaction cross-section for the fast neutron-induced reactions in steels and aluminium alloy**

Source: Pohang Accelerator Laboratory, POSTECH, 2020.

**Concrete activation for non-self-shielded PET cyclotron**

The stacking structure as shown in Figure 4 is not great because it located at the bottom, and there is 8.5 cm gap at the top of backside. Therefore, the credible depth, which is the depth similar at real profile of the vault wall, was determined in the stacking structure. From the comparison of calculated neutron depth profile between the vault wall and the stacking structure by FLUKA code, the credible depth was determined as from surface to near 20 cm. As the depth increases in real wall, the neutron flux continued to decrease, however, it increased by neutrons scattered in our structure.

The depth profile of the most important radionuclide, Eu-152, and thermal neutron flux are shown in Figure 8. Measured thermal neutron flux was highest at the depth of 9 cm, but it was not significant difference with the surface. The calculated neutron flux by FLUKA was...
2 to 4 times larger than the measured one by Au foils within the credible depth. The measured specific activity of Eu-152 followed well with the measured thermal neutron flux. Maximum depth from measurement were 3 to 9 cm. To compare with the measured results, we calculated specific activity using the thermal neutron flux by FLUKA (USRTRACK at each sample). For the important parameter, atom number, we used measured Eu contents in the ordinary concrete by the ICP-MS (Lee et al., forthcoming). The red-line was calculated using the mean value of Eu atoms, and the maximum and minimum values were used to represent the yellow region. The ratio of specific activity of Eu-152 and the thermal neutron match well. It means that the real Eu contents in the concrete sample were within the measured contents.

Figure 8. Depth profile of specific activity of Eu-152 and thermal neutron flux in the concrete stacking structure

Source: Pohang Accelerator Laboratory, POSTECH, 2020.

Conclusion

To estimate the activation of cyclotron itself and to understand the difference of activation characteristics between the self-shielded and non-self-shielded PET cyclotron, remarkable radionuclides and their saturated activity concentration from metal was measured and compared. The thermal neutron flux was measured successfully by Au activation. Measured thermal neutron flux was $10^6$ n/cm² s near the FDG target. With respect to the concrete activation of non-self-shielded PET cyclotron, activation depth profile was measured about 20 cm depth from the surface despite the limitation of the stacking structure. The maximum specific activity was founded at the 3 to 9 cm depth. Experimental
results were compared with the calculation data by the FLUKA code. The results of FLUKA were matched well but slightly overestimated. These results are helpful to understand the amount of the radioactive waste from PET cyclotron facility.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety (KOFONS), granted financial resource from the Nuclear Safety and Security Commission (NSSC), and Korea (Grants No. 1603005).

References


Hwang, S.Y. et al. (2017), Transaction of Korean Association for Radiation Protection Spring Meeting, April 13, Gunsan, Korea.


Session V: Induced radioactivity
Activation zoning of the experimental areas of the CERN antiproton decelerator using ActiWiz 3

Chris Theis\textsuperscript{1*}, Robert Froeschl\textsuperscript{1}, Helmut Vincke\textsuperscript{1}
\textsuperscript{1}CERN, Switzerland
\textsuperscript{*}Christian.Theis@cern.ch

Beginning with January 2018 the Swiss clearance limits have become considerably more restrictive for most of the radionuclides that are radiologically relevant at CERN. This paper describes the activation assessments of typical infrastructure material in the experimental areas of the CERN antiproton decelerator. For this purpose FLUKA has been used to determine the source spectrum originating from antiproton annihilation and subsequently ActiWiz 3 to determine the radionuclide inventories and radiotoxicity, applying already the new clearance limits. Consequently, a clearance perimeter around the beam lines has been determined which is also appropriate in the case of unknown or highly complex material compositions like electronic circuit boards. Different scenarios in terms of irradiation and cooling periods have been studied to ensure that this boundary is applicable to the clearance of operational equipment as well as waste in view of dismantling of the areas.

Introduction

Beginning with January 2018 the Swiss clearance limits have become considerably more restrictive for most of the radionuclides that are radiologically relevant at CERN (BAG 2018). In order to remove material that has previously been used in a radiation area from regulatory control a number of criteria have to be fulfilled (Roesler, 2018). On one hand the net ambient dose equivalent rate at 10 cm distance must not exceed 0.1 \textmu Sv/h, surface contamination levels have to be below certain thresholds but also activity levels of radioisotopes found within the material must not exceed their respective exemption limits. As described in (Roesler, 2018) for material containing a mixture of radionuclides of artificial origin, the following sum rule should be applied to remove it from any further regulatory control:

\begin{equation}
\left( LL = \sum_{i=1}^{n} \frac{a_i}{LL_i} \right) < 1 \tag{1}
\end{equation}

where \( a_i \) is the specific activity (Bq/kg) or the total activity (Bq) of the \( i \)-th radionuclide of artificial origin in the material, \( LL_i \) is the respective clearance limit for the radionuclide \( i \) in the material and \( n \) is the number of radionuclides present. In accordance with current Swiss legislation CERN’s clearance limits are taken directly from (Roesler, 2018).

For the experimental areas of the Antiproton Decelerator (AD) an activation study has been carried out using version 2011.2c.5 of the FLUKA Monte Carlo code (Ferrari, 2005; Böhlen et al., 2014) as well as ActiWiz version 3.2 (Vincke and Theis, 2018). Using FLUKA the particle fluence spectra originating from antiproton annihilation has been calculated which subsequently were used in ActiWiz to evaluate the nuclide inventories and the corresponding radiotoxicity given by Eq (1) for 85 chemical elements (= full periodic system of stable elements + 5 isotopes). In this context also the minimum distances from
the antiproton annihilation points have been determined beyond which those types of materials would become either candidates for clearance or could be cleared immediately as the radiation fields would not be sufficiently strong to render the material radioactive in the legal sense. The application to the complete set of stable elements as well as the determination of such a perimeter eventually resolves the problem of unknown or complex material compositions like electronic circuit boards and allows for devising a simple and efficient clearance method that is based on verification measurements. These comprise a dose-rate evaluation of the object to be cleared as well as regular gamma spectroscopic control of material samples that should be distributed in the area for so-called “global control”.

**Activation studies**

Using the FLUKA Monte Carlo code (version 2011.2c.5) antiproton annihilation has been simulated by the interaction of antiprotons of 5.3 MeV with a copper sphere of 1 cm radius (see Figure 1). The choice of the dimension stems from the fact that it is considerably larger than the penetration depth of the antiprotons, while at the same time being smaller than the typical interaction length of the emitted secondary particles. While the interaction of the antiproton with iron might seem most probable due to the fact that beam pipes are made of stainless steel, copper was conservatively chosen as an annihilation partner as it results in slightly higher neutron and photon yields. In the energy region of < 100 MeV high Z materials like tungsten show a yield that is a factor of 2-3 higher than the one originating from copper. Yet, this scenario can be considered as over-conservative as the annihilation is most likely to occur with accelerator infrastructure like beam pipes. It should be noted that the differences of secondary particle spectra obtained from the annihilation of antiprotons at 5.3 MeV or at energies of 100 keV, as provided in the facility in the future, are negligible.

**Figure 1. FLUKA geometry of the Copper target used to enforce the antiproton-proton annihilation**

The ensuing secondary radiation field of neutrons, protons, charged pions and photons was recorded in 1 m distance (Froeschl, 2013). 
In a distance of 1 metre the fluence spectra of neutrons, protons, charged pions as well as photons originating from the annihilation process have been recorded and can be seen in Figure 2.

**Figure 2. Particle fluence spectra per impacting antiproton, originating from the annihilation process in a copper sphere of 1 cm radius**

![Fluence spectra graph](image)

Statistical errors are smaller than the depicted symbols.


**Activation of compound materials not directly exposed to the beam**

Evaluating equation (1) the radiotoxicity of 85 chemical elements has been determined for assuming 30 years of irradiation with $5 \times 10^5$ antiprotons/second to ensure near-saturation of potentially critical radioisotopes like Co-60. For each material several cooling times have been studied ranging from 12 hours up to 30 years, which reflects scenarios of operational clearance of functional equipment up to potentially radioactive waste.

Based on the fluence spectra depicted in Figure 2 a prompt effective dose of 0.062 pSv per annihilation or 0.023 pGy per annihilation has been calculated with ActiWiz for a distance of 1 metre. Considering a beam intensity of $5 \times 10^5$ antiprotons/second this translates into 112 uSv/h or 41 uGy/h in 1 m distance from the annihilation point. Calculating the ratio of the prompt dose rate with respect to the radiotoxicity level eventually yields a prompt dose rate level, which has to be exceeded in order to reach activation levels that are sufficiently high to still classify the material as legally radioactive after 12 hours of cooling.

In summary, it was shown that in all areas where the prompt effective dose exceeds ~270 uSv/h at least one chemical element (Manganese) would still show radiotoxicity levels of LL > 1 after 12 hours of cool-down and 30 years of irradiation. Using again the $1/r^2$ law one can determine the distance from the annihilation point beyond which the radiation field’s strength is not sufficient anymore to activate material in the legal sense. It should be stressed that the subsequently reported distances are valid only for a beam intensity of $5 \times 10^5$ antiprotons/second, whereas the prompt dose rate limits given in (uSv/h)/LL are beam intensity independent and can be used in a more general way to establish zoning boundaries. The list of the 10 most critical elements can be found in Table 1.
Table 1. Listing of prompt effective dose rates for a radiotoxicity level of LL = 1, rendering the respective chemical element legally radioactive, after 12 hours of cool-down

<table>
<thead>
<tr>
<th>Element</th>
<th>Dose rate per LL ((uSv/h) / LL)</th>
<th>min. distance for clearance where LL &lt; 1 (cm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>EUROPIUM</td>
<td>4.21E+02</td>
<td>5.15E+01</td>
</tr>
<tr>
<td>MANGANESE</td>
<td>2.68E+02</td>
<td>6.46E+01</td>
</tr>
<tr>
<td>NICKEL</td>
<td>4.34E+02</td>
<td>5.07E+01</td>
</tr>
<tr>
<td>RHODIUM</td>
<td>5.64E+02</td>
<td>4.45E+01</td>
</tr>
<tr>
<td>SODIUM</td>
<td>5.00E+02</td>
<td>4.72E+01</td>
</tr>
<tr>
<td>YTTRIUM</td>
<td>4.36E+02</td>
<td>5.06E+01</td>
</tr>
<tr>
<td>ZINC</td>
<td>6.68E+02</td>
<td>4.09E+01</td>
</tr>
<tr>
<td>ZIRCONIUM</td>
<td>5.10E+02</td>
<td>4.68E+01</td>
</tr>
<tr>
<td>THORIUM-232</td>
<td>4.15E+02</td>
<td>5.19E+01</td>
</tr>
</tbody>
</table>

In addition the minimum distance from the annihilation point is calculated beyond which the radiation field is not sufficiently strong anymore to reach LL =1, based on a beam intensity of 5 \times 10^5 antiprotons/second. For reasons of conciseness only the 10 most critical elements are listed.


In the case of Manganese the clearance perimeter at 5 \times 10^5 antiprotons/second translates into > 65 cm. It might be striking at first that cobalt is not among the most critical chemical elements, which is often found to be highly problematic at high-energy proton accelerators. This can be understood from the fact that this radiation field is dominated by high-energy neutrons whereas cobalt is highly susceptible to thermal neutron capture resulting in Co-60. At the same time it should be recalled that the cooling period is only 12 hours, whereas Co-60 due to its long half-life becomes dominant only for longer decay times. Eventually one can conclude that at a distance of > 65 cm from an annihilation point a beam intensity of 5 \times 10^5 antiprotons/second is not sufficient to render any of the 80 studied chemical elements legally radioactive considering 12 hours of cooling.

The same exercise has been carried out considering a decay time of 30 years, which is usually applicable to radioactive waste. Considering this scenario one can conclude that at locations where prompt effective dose rates of > 2 mSv/h are found at least one chemical element (Europium) would still be classified as legally radioactive, even after 30 years of cool-down. Applying the 1/r^2 law this translates into 23 cm distance from the annihilation point.

Combining the results obtained for a short cooling time of 12 hours (operational material clearance) as well as the one for 30 years (radioactive waste clearance) one can formalise the following guideline summarised in Table 2.
Table 2. Classification for any studied chemical element that is not in direct contact with the beam

<table>
<thead>
<tr>
<th></th>
<th>loss-point</th>
<th>non-loss point</th>
</tr>
</thead>
<tbody>
<tr>
<td>Immediate free release</td>
<td>&gt;65 cm</td>
<td>&gt;25 cm</td>
</tr>
<tr>
<td>Candidate for free release</td>
<td>&gt; 25 cm</td>
<td>&lt; 25 cm</td>
</tr>
<tr>
<td>Radioactive waste</td>
<td>&lt; 25 cm</td>
<td>-</td>
</tr>
</tbody>
</table>

The radii give the minimum distance from the annihilation/loss point necessary for the respective classification. For column 3 they refer to the nearest directly exposed component. A “non-loss point” is defined as a point where on average less than 10% of the beam intensity is lost. It should be noted that for practicality the exact numerical values have been rounded up to the nearest multiple of five.


It should be noted that these calculations only include statistical errors, which have been omitted as they are < 1%. In order to account for uncertainties in the physics models of the simulations a pragmatic safety factor of ~2 would be prudent. Consequently, one would consider any material, even such with unknown composition, with a distance of more than 1 meter from the annihilation point as non-radioactive. For “non-loss points” where on average less than 10% of the beam is lost this would reduce to 0.5 metres.

Summary and conclusions

Beginning with January 2018 the Swiss clearance limits have become considerably more restrictive. In order to establish a clearance method that on one hand is compliant with the regulatory requirements as well as efficient FLUKA and ActiWiz calculations have been carried out to determine a clearance perimeter around the beam lines. These studies comprised the full set of stable elements + some radioactive ones (U-234, U-235, U-238, Th-232, Sr-90) which makes it applicable also in case of unknown or highly complex material compositions like electronic circuit boards. It was found that any material from areas where the prompt effective dose is < 270 uSv/h can be cleared immediately after only 12 hours of cooling. For waste clearance, which considers a cooling period of 30 years, this limit increases to 2 mSv/h.

Assuming an average beam intensity of $5 \times 10^5$ antiprotons/second these dose rate values can be translated into distances from the annihilation point. Including a pragmatic safety factor of about 2 one can say any material (including waste!) which is > 1 metre from the annihilation point can be cleared right after 12 hours of cool-down, whereas in a distance between 0.5 m and 1 m one can find candidates for clearance. For areas like transfer lines where on average less than 10% of the beam is lost this translates into > 50 cm (including the safety factor of 2) and between 25 cm and 50 cm respectively. It should be stressed that the perimeter expressed in uSv/h is beam intensity independent whereas the stated distances are only valid for an average intensity of $5 \times 10^5$ antiprotons/second. If beam intensities will be increased in the future, the distances could be recalculated from the dose rate limits, using the fact that one finds a dose rate of $6.2 \times 10^2$ pSv/antiproton in a distance of 1 meter from the annihilation point.

References

BAG (2018), Strahlenschutzverordnung (StSV) 2017 (Radiological Protection Ordinance 2017), as of 1 January 2018, 814.501, Bundesamt für Gesundheit (Federal Office of Public Health).


Experience with inner reflector plug exchange in SNS

I.I. Popova1, F.X, Gallmeier, S. Schwahn, M. Dayton, Ch. Elam

1ORNL, Oak Ridge, TN, US

* popovai@ornl.gov

The inner reflector plug (IRP) is a central component of the Spallation Neutron Source target monolith, which houses the mercury target and four litre-sized neutron moderator units. It is exposed to high-level radiation fields during routine operation and builds up significant activity. The IRP needs to be replaced due to moderator neutron poison and decoupler burn-out, which is used for shaping neutron pulses. The first IRP exchange took place in March 2018. The old IRP was extracted from the target monolith, providing space for the new one. It was split into three segments, each of which was handled separately. The lowest section of the IRP is the largest segment in size and in activity and is temporarily stored on site for cool down before conduction post irradiation examination. In support of planning the replacement activities, a wide range of activation and transport analyses were performed. This included calculating isotope inventories and the radiation fields for each segment as it is extracted in storage casks, and the radiation field from the empty IRP pit in the target monolith. While the replacement was taking place, measurements were performed and later on compared to the calculations. During these studies, it was discovered that a significant contributor to the radiation field from the lower IRP segment is from photoneutrons. Photo-nuclear physics was added to the analyses and calculated results compared well with measured dose rates.

Introduction

The Spallation Neutron Source (SNS) is an accelerator-driven neutron scattering facility for materials research and fundamental physics applications, which is presently capable to operate at 1.4 MW proton beam power incident on a mercury target with proton beam energy of 1 GeV and with a pulse structure of 60 Hz repetition rate. The high-energy proton beam intercepts the liquid mercury target housed in a double walled steel core vessel. Fast neutrons, produced due spallation reactions in mercury, are moderated by four moderators located directly above and below the target vessel housed by the inner reflector plug (IRP), which is a geometrically complex and composite part of the SNS target station. Being a central component of the target monolith, IRP is exposed to high-level radiation fields during routine facility operation and over time builds up significant radioactivity. The IRP assembly contains besides the four moderators, a beryllium reflector and a steel shielding. Replacement is necessary due to burnup of gadolinium and cadmium poison and

1. This manuscript has been authored by UT-Battelle, LLC under Contract No. DE-AC05-00OR22725 with the U.S. Department of Energy. The United States Government retains and the publisher, by accepting the article for publication, acknowledges that the United States Government retains a non-exclusive, paid-up, irrevocable, world-wide license to publish or reproduce the published form of this manuscript, or allow others to do so, for United States Government purposes. The Department of Energy will provide public access to these results of federally sponsored research in accordance with the DOE Public Access Plan.
decoupler, used to shape the time structure of the neutron pulses emitted from the moderators. Degradation of the poison and decoupler determines the lifetime of IRP, which initially was estimated to be at about 32 000 MWh proton beam energy delivered to the target. The IRP exchange took place in March 2018, when the IRP had about 41 000 MWh. It was kept in service for longer than initially planned, as the manufacturing process for the replacement IRP was delayed, and considerable degradation of the neutron pulse structure was experienced.

The old IRP was extracted from the target monolith, providing space for the new one. It was split into three segments, each of which was handled separately. The lowest section of the IRP is the largest segment in size and in activity. It is temporarily stored on site in the target building for cool down before conduction post irradiation examination.

In support of careful planning the replacement activities, a wide range of activation and transport analyses were performed. This included calculating the isotope inventories and the radiation fields for each segment as it was extracted in storage casks, the radiation fields from the empty IRP shaft in the target monolith and the radiation fields during different stages of IRP segments extraction. While the replacement was taking place, measurements were performed and later were compared to the calculations. During these studies, it was discovered that a significant contributor to the radiation field from the lower IRP segment was from photoneutrons. Photo-nuclear physics was added to the analyses and calculated results compared well with measured dose rates.

This paper addresses work, performed to calculate dose rates for the lower IRP segment inside the temporary storage cask, from the opened monolith pit after IRP extraction and the dose rates, and during different stages of IRP segment extraction.

Methods and codes

Monte Carlo radiation transport calculations using the MCNPX version 2.7.0 code (Pelowitz, 2011; Gallmeier F. X., 2006) were performed to obtain neutron fluxes and spallation-induced radionuclide production rates to feed transmutation calculations and to assess the residual dose rates distribution with the resulting decay gamma sources for the three IRP segments and the IRP shaft.

A refined spatial resolution of the decay gamma sources distributions was obtained by subdividing the IRP segments and IRP shaft into smaller cells for each of which activation calculations were performed based on cell specific isotope production rates and neutron fluxes. Transmutation calculations were done by applying the standardised ACTIVATION_SCRIPT (Gallmeier, 2008), with usage of CINDER90 code (Wilson W.B., 2007, Ferguson P. D., 2006). The irradiation history divided the 40 443 MWh of proton energy delivered to the target over 11 years in 30 steps, as illustrated in Table 1.
Table 1. Irradiation history of IRP

<table>
<thead>
<tr>
<th>Time step</th>
<th>Period, days</th>
<th>Delivered energy, MWh</th>
<th>Time step</th>
<th>Period, days</th>
<th>Delivered energy, MWh</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>730</td>
<td>3749.28</td>
<td>16</td>
<td>124</td>
<td>2410.56</td>
</tr>
<tr>
<td>2</td>
<td>248</td>
<td>1964.16</td>
<td>17</td>
<td>57</td>
<td>0</td>
</tr>
<tr>
<td>3</td>
<td>117</td>
<td>0</td>
<td>18</td>
<td>124</td>
<td>2470.08</td>
</tr>
<tr>
<td>4</td>
<td>248</td>
<td>2380.8</td>
<td>19</td>
<td>60</td>
<td>0</td>
</tr>
<tr>
<td>5</td>
<td>117</td>
<td>0</td>
<td>20</td>
<td>124</td>
<td>2470.08</td>
</tr>
<tr>
<td>6</td>
<td>248</td>
<td>3571.2</td>
<td>21</td>
<td>57</td>
<td>0</td>
</tr>
<tr>
<td>7</td>
<td>117</td>
<td>0</td>
<td>22</td>
<td>120</td>
<td>2499.84</td>
</tr>
<tr>
<td>8</td>
<td>248</td>
<td>4047.36</td>
<td>23</td>
<td>63</td>
<td>0</td>
</tr>
<tr>
<td>9</td>
<td>117</td>
<td>0</td>
<td>24</td>
<td>120</td>
<td>2499.84</td>
</tr>
<tr>
<td>10</td>
<td>124</td>
<td>2232</td>
<td>25</td>
<td>62</td>
<td>0</td>
</tr>
<tr>
<td>11</td>
<td>60</td>
<td>0</td>
<td>26</td>
<td>120</td>
<td>2498.84</td>
</tr>
<tr>
<td>12</td>
<td>124</td>
<td>2232</td>
<td>27</td>
<td>62</td>
<td>0</td>
</tr>
<tr>
<td>13</td>
<td>57</td>
<td>0</td>
<td>28</td>
<td>90</td>
<td>1998</td>
</tr>
<tr>
<td>14</td>
<td>124</td>
<td>2410.56</td>
<td>29</td>
<td>2</td>
<td>0</td>
</tr>
<tr>
<td>15</td>
<td>60</td>
<td>0</td>
<td>30</td>
<td>30</td>
<td>1008</td>
</tr>
</tbody>
</table>


Decay gamma sources for a defined history of build-up are extracted from each cell and compiled to a source term in MCNPX language by running the GAMMA_SOURCE_SCRIPT (Wohlmuther, 2008). The source was assumed to be homogenously distributed inside each cell and has energy structure for each cell. Due to the complexity of the lower segment geometry and large amount of segment cells, transport analyses for the residual dose rate required elaborate work for the definition of decay source terms. The cells that contribute less than a fraction of 10^{-4} times compared to the total source strength were omitted. The gamma sources were distributed homogeneously inside each cell. The source description in the moderators used lattice structure due to moderators geometry modelling.

Decay gamma sources for IRP lower segment in the cask were prepared starting 1.33 MeV and the photonuclear production was switched on in transport analyses.

Because of limitations of the MCNPX code, the source terms for IRP lower segment and empty IRP shaft were not possible be described in one complete representation in one analysis. Thus, the analyses were conducted in multiple steps considering the various parts of the source terms. The results of the various analyses were finally added up, to give the full representation of the radiation field.

For scoring the neutron and gamma dose rates two types of tallies were used: mesh and volume detectors tallies. The volume tallies were set at requested key locations. Dose rate were obtained by folding fluxes with flux to dose conversion coefficients (Popova, 2012).

Geometry

The IRP model with segmentation for waste disposal is shown in Figure 1. The IRP is a cylindrical component, which is approximately 454-cm tall and 100-cm in diameter and is housed in the target monolith.
The SNS as-built target station model is a CAD type model that details the target module: the cylindrical IRP including four moderators, the beryllium and steel reflectors, the cylindrical outer reflector plug (ORP), and the proton beam window assembly. A full three-
A dimensional model was developed for transport analyses to simulate proton interactions on the proton beam window and target, and secondary radiation fields in the IRP environment. The IRP and ORP were modelled in an as-built representation, while the core vessel (CV) and monolith shielding were modelled in as much detail as neutronically necessary. A vertical cut of the target station transport analyses model containing the IRP and used to obtain residual gamma source terms is shown in Figure 2.

Figure 3. IRP lower segment transport analyses model inside on-site storage cask

The IRP model for residual gamma transport was extracted from target monolith model and was divided, according to the drawings, into three segments, each of which was going to be disposed separately. The extracted model of lower IRP segment was put into temporary on-site storage cask. The model is presented in Figure 3.

The IRP lower segment consists of the reflector and has an attached shielding on top. Reflector has an opening for the target vessel and houses four moderators, two moderators are located above the opening for the target and two moderators are located below opening for the target. Moderator vessels are comprised from aluminium, two are wrapped partially with cadmium, and have a gadolinium plate for pulse shaping in their centre. The moderator vessels are fed by liquid water and hydrogen through invar piping. The plug material is the beryllium (red colour at Figure 3), extending radially to 32 cm, and then followed by stainless steel 304, extending radially to 47.625 cm. Reflector is enclosed into aluminium (light orange colour at Figure 3), extending radially to 49.53 cm. The shielding on top is made from stainless steel (blue colour). The extracted IRP lower section is placed in to a cylindrical container for the on-site storage and cooling down shown in Figure 3. The storage container is made from the lead (yellow colour in Figure 3), enclosed into steel with thickness of 1.25 cm. The lead thickness in the cylindrical wall vary along the container height from 30.32 cm on the bottom to 9.398 cm on the top. The bottom of the cask is a carbon steel gate with thickness of 18.415 cm.

Figure 4. Vertical cross-section through the target monolith including the IRP lower section as pulled to the elevation of the core vessel flange and including the opened transport cask

The model for the dose rate calculations during IRP extraction procedure included details of opened monolith as the core vessel flange, a steel platform including a shielding sleeve (lead filled steel ring) reaching about 50 cm into open shaft and a fairly open pipe trench on the downstream side of the IRP shaft covered by high density concrete T-beams. The vertical cut of IRP shaft MCNPX model including the IRP lower section as pulled to the elevation of the core vessel flange including the transport cask opened at the bottom are shown in Figure 4.

Results

The residual dose rates were calculated for 28 days of cool down, it is about the time when IRP lower section was extracted from the target monolith and was placed into on-site storage cask. Then for the IRP lower section in the storage case the dose rates were calculated for 76 days of cool down, when measurements were performed.

**Dose rates estimate during the IRP replacement**

The IRP replacement represented a critical operation demanding significant planning. A work plan was established that divided the operation into subtasks for which dose were provided, and worker dose was assessed involving potential exposure times. In order to support extraction of IRP a set of neutronics analyses simulating dose rates during different extraction stages were performed (Gallmeier, 2017).

As a preparation for the IRP extraction, target vessel is retracted, and the shielding on the top of core vessel was removed. The shielding sleeve, including working platform, was added. The dose rates are shown in Figure 5. Dose rates at the core vessel flange area are below 10 mrem/h, except at pipe chase cutouts and shielding gaps, where dose rates of the 10 mrem/h level were expected. Measured dose rate was about 50 mrem/h on the top of pipe chase and 5 mrem/h in general area.

![Source: ORNL, 2020.](image)

The IRP was removed as three separate segments. Pipe cutting and removal operations were required prior to the removal of each segment. Tie rods that structurally connect the three segments were loosened to allow separation of each segment from the one below.
After top and middle segments were removed, the extraction of lower segment, which was the most irradiated of the three segments, was started by placing of the temporary on-site storage cask on the CV opening. Then the retraction of the lower segment from the CV shaft into cask followed.

As the shielding sleeve size does only barely overlap with the cask outer dimensions, radiation was leaking through the cover plate with peak dose rates of 10 Rem/h predicted producing a dose rate halo of 100 mrem/h around the transport cask. Also, the pipe trench was well lit up as seen in the contour lines of dose rate plots shown in Figure 6.

The geometry and the dose rates around the CV with lower IRP segment partially extracted from the shaft is shown in Figure 6. Measured dose rate was about 20 Rem/h on east side of cask.

Figure 6. Vertical cross-section through the target monolith axis including the IRP lower section as pulled to the elevation of the core vessel flange, and including the transport cask opened at the bottom, (left), horizontal dose rate maps at core vessel sleeve (middle), mrem/h and vertical dose rate map (right), mrem/h


The geometry and the dose rates around the CV with IRP extracted is shown in Figure 7. Right column shows the configuration for the open shaft of about 4.4-metre depth and 1.4-metre diameter. The predicted dose rates on top of IRP shaft peaked at about 50 Rem/hr. The dose rates at the bottom of the shaft rised to 10 000 Rem/hr. Measured dose rate around the shaft opening was about 42 Rem/h.
To reduce the radiation exposure in the target building, while the IRP was removed, the IRP shaft was covered with a steel lid of 25 cm thickness as shown in Figure 8. Dose rates on top of the covered IRP shaft were reduced to the mrem/h level as shown in Figure 8 except for the perimeter of the cover lid, which exposed dose rates at 10 mrem/h because of insufficient overlap of the cover lid and the lead shielding sleeve of the steel platform. Measured dose rate was about 2 mrem/h.

Lower IRP segment in transport case – calculations vs measurements

The extracted IRP lower segment stays on site in the temporary cask for the cool down. Dose rates for the time, when measurements were taking place, were analysed (Gallmeier, 2017). Source spectrum from the IRP lower segments structural materials after 76 days of cool down, which was used to calculate dose rates, is shown in Figure 9. The most overall residual gammas were produced in gadolinium, the main sources of the residual gamma with high energy were stainless steel components.
Source spectrum from IRP lower segment materials

![Source spectrum from IRP lower segment materials](image)


Calculated dose rate, neutrons and gammas through beam centerline in vertical and horizontal cross-sections are shown in the Figure 10. Initially storage contained was designed to shield against the gammas, it is why only lead was used as for the cask construction.

Dose rates inside and outside of the IRP in the on-site storage cask

![Dose rates inside and outside of the IRP in the on-site storage cask](image)
The gamma dose rates plots are shown in two scales in order to reflect source density of the IRP segment, top row in Figure 10, then scaling was changed to be consistent with the neutron dose rates scaling.

The measurements were taken about 76 days after the beam termination around the cask at contact and at about 30 cm from the cask surface at 2 elevations – a lower elevation, which was near by the opening for the target vessel between the moderators, and at about 180 cm from the floor, and at a higher elevation which was about where the IRP shielding starts. Ludlum detector model 9-4 was used for gamma measurements and Thermo/Eberline ASP-2e Remball detector was used for neutron measurements. Remball was calibrated to unmoderated Cf252.

The IRP geometry is not homogeneous, so the dose rate in azimuthal direction vary. When measurements were taken, it was impossible to define the IRP positioning inside the cask. Therefore, the calculated dose rates shown in the Table 2 were averaged over the full azimuthal range as were the measurements.
Table 2. Dose rates outside the IRP on-site storage container

<table>
<thead>
<tr>
<th>Position</th>
<th>Distance from the surface, cm</th>
<th>Calculated dose rate, mrem/h</th>
<th>Measured dose rate, mrem/h</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Neutrons</td>
<td>Gammas</td>
</tr>
<tr>
<td>Lower</td>
<td>30 cm</td>
<td>31.8</td>
<td>4.6</td>
</tr>
<tr>
<td>Higher</td>
<td>30 cm</td>
<td>7</td>
<td>7.7</td>
</tr>
<tr>
<td>Lower</td>
<td>Contact</td>
<td>60</td>
<td>8</td>
</tr>
<tr>
<td>Higher</td>
<td>Contact</td>
<td>6</td>
<td>12</td>
</tr>
</tbody>
</table>


Dose rate attenuation outside of the IRP in the on-site storage cask starting from the container surface is shown in Figure 5. For so-called higher elevation two set of curves were plotted at about 180 cm from the floor and at 160 cm from the floor (so-called on the plot 20 cm down). Dashed line is about 30 cm from the container. Calculated values were in a good agreement with measured ones, considering the uncertainty of the detector positioning and the complexity of the analyses.

Figure 11. Dose rate attenuation outside of the IRP in the on-site storage cask

Dashed line is about 30 cm from the container.

Conclusions

The residual dose rates were calculated for 28 days of cool down to support IRP extraction for the stages of IRP replacement, some of which are presented in this paper. Measured dose rates during the extraction were presented also and compare well with the predictions. IRP lower section in the storage case the dose rates were calculated for 76 days of cool down and compared with the measurements. Calculated dose rates were in a good agreement accounting the uncertainty in positioning of the detector and the complexity of analyses.

Acknowledgements

This material is based upon work supported by the US Department of Energy, Office of Science under contract number DE-AC05-00OR22725 96OR22725 with UT-Battelle Corporation for ORNL.

References

Ferguson, P.D. (2006), CINDER’90 for SNS Activation Studies, SNS-106100200-TR0142-R00, Oak Ridge National Laboratory.


Popova, I. (October 2012), Flux to Dose Conversion Factors, SNS-NFDD-NSD-TR-0001-R02, Oak Ridge National Laboratory.


A Beam Dump Facility (BDF) at CERN: The concept and a first radiological assessment

C. Ahdida¹, M. Casolino¹, P. Avigni¹, M. Calviani¹, J. Busom¹, J. P. Canhoto Espadal¹, J-L. Grenard¹, R. Jacobsson¹, K. Kershaw¹, M. Lamont¹, E. Lopez Sola¹, S. Roesler¹, V. Vlachoudis¹ and H. Vincke¹
¹CERN, Switzerland
*claudia.ahdida@cern.ch

The Beam Dump Facility (BDF) Project, currently in its design phase, is a proposed general-purpose fixed target facility at CERN, dedicated to the Search for Hidden Particles (SHiP) experiment in its initial phase and whose aim is to fully absorb the high intensity 400 GeV/c SPS beam. The BDF target complex would be located underground at a depth of about 10 m and is designed to contain most of the cascade generated by the primary beam interaction. Due to the high beam intensity delivered on target, the high density and high-Z composition of the target/dump, high activation of the material is expected, therefore evaluation of radiological protection risks is a crucial aspect for the design of this facility. In particular, high prompt and residual dose rates call for considerable shielding and remote interventions in the target area. Also the risk and environmental impact from air, water and soil activation heavily influence the design. In order to respect the applicable CERN radiological protection legislation regarding doses to personnel as well as the environmental impact, a preliminary radiological study was carried out. In order to validate the design of the BDF target, a scaled prototype was tested during 2018 in the SPS North Area at CERN. Preliminary results of radiological aspects for this test will be presented as well. To assess the above-mentioned radiological protection aspects, extensive simulations were performed with the FLUKA Monte Carlo particle transport code.

Introduction

The Beam Dump Facility (BDF), which is currently in its feasibility study phase, is a general-purpose fixed target facility at CERN in which a 400 GeV proton beam shall be directed from the Super Proton Synchrotron (SPS) towards a fixed target complex in the North Area of CERN’s Prévessin site (Figure 1). The extracted beam is transferred along the existing transfer line to the North Area up to a switch in an existing cavern (TDC2/Junction cavern). This switch sends the beam into a ∼350 m long new beam line.
The BDF target complex is located underground with the production target at the heart of the facility (Kershaw et al., 2018). The target is designed to contain most of the cascade generated by the primary beam interaction. It is composed of a hybrid solution of (solid) molybdenum alloy (TZM) and pure tungsten (W) cladded in tantalum alloy, for a total target length of 1.5 m. The target is embedded in the so-called hadron absorber absorbing the remaining primary protons and produced hadrons emerging from the target.

In its initial phase, BDF will be dedicated to the Search for Hidden Particles (SHiP) experiment (Alekhin et al., 2016; Anelli et al., 2015). For this experiment, the maximum SPS beam intensity (Table 1) shall be fully absorbed in the dense target/hadron absorber, while the produced muons shall be swept away by an active muon shield (Akmete et al., 2017). In theory, very weakly interacting particles could be produced, which shall be investigated by a suite of particle detectors located downstream of the target complex in the experimental hall.

The main radiological protection challenges arise from the high beam power, the proximity to the surface, other experimental facilities and the CERN fence, but also from keeping the flexibility for future installations. In order to respect the applicable CERN radiological protection legislation regarding doses to personnel as well as the environmental impact, a first radiological assessment was carried out for the design of the BDF facility.
Table 1. Key BDF beam parameters

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Momentum [GeV/c]</td>
<td>400</td>
</tr>
<tr>
<td>SPS beam intensity per cycle</td>
<td>4×10^{19}</td>
</tr>
<tr>
<td>Cycle length [s]</td>
<td>7.2</td>
</tr>
<tr>
<td>Spill duration [s]</td>
<td>1</td>
</tr>
<tr>
<td>Avg. beam power on target [kW]</td>
<td>355</td>
</tr>
<tr>
<td>Protons on target (POT)/year</td>
<td>4×10^{19}</td>
</tr>
<tr>
<td>Total POT in 5 years data taking</td>
<td>2×10^{20}</td>
</tr>
</tbody>
</table>


Radiological protection evaluation

The design of the facility was optimised based on general radiological protection guidelines and specific studies on prompt and residual dose rates, air and helium activation, ground and water activation as well as radioactive waste production since they heavily influence the design. To assess the above-mentioned radiological protection aspects, extensive simulations were performed with the FLUKA Monte Carlo particle transport code (Ferrari et al., 2005; Böhlen et al., 2014).

The target complex was designed under the condition that the target hall can be accessed during beam operation and classified as a Supervised Radiation Area with low occupancy (< 15 μSv/h) (see Figure 2). On the contrary, no access during beam operation will be permitted to the underground target bunker or the experimental hall. Due to the proximity of BDF/SHiP to the ground level (~15 m), other experimental facilities (~20 m) and public areas (~70 m), massive shielding is required to keep the prompt radiation in the various accessible areas of the facility and the surrounding reasonably low.

In addition to personnel protection regarding prompt dose rates, considerable shielding is indispensable to reduce the residual dose rates and the environmental impact from activated air, helium and soil as well as to relax radiation levels on electronics equipment. The shielding was consequently designed with the objective to keep the various radiological hazards originating from the operation of the BDF/SHiP facility as low as reasonably possible, while taking the constraints from the different stages of the experiment, that is the construction, operation, maintenance and dismantling, into account. The envisaged configuration is such as to avoid activation of the fixed concrete civil engineering structures simplifying not only the dismantling, but also possible changes of scope of the installation.

Figure 2 illustrates the shielding configuration with the hadron absorber composed out of approximately 3 700 tonnes of cast iron and steel shielding further surrounded by concrete shielding. Several 5 cm wide gaps within the shielding were included in the FLUKA model of the facility such as to conservatively account for imperfect alignment, ducts for cooling, electronics etc. Part of the downstream shielding is magnetised. Together with the active muon shield magnets it will sweep away high-energy muons produced in the target for reducing experimental backgrounds (Anelli et al., 2015). The shielding blocks were specially designed and optimised for remote handling, since they, as well as the target, will become highly activated. Two different handling concepts have been developed, the “crane
concept” and the “trolley” concept allowing for fully remote handling and manipulation of the target and the surrounding shielding (Kershaw et al., 2018).

Figure 2. Side (top) and cross-sectional (bottom) view of BDF as implemented in FLUKA


The air volumes of the facility were minimised to reduce the production of airborne radioactivity. In the most critical area, that is the central region around the target and hadron absorber, the air was further replaced by a helium environment. This is motivated by the fact that pure helium gives only rise to the formation of tritium, which has a significantly lower radiological impact than the radionuclides arising from air. The service pit on top of the helium vessel, in which air activation is expected, is further separated by an airtight concrete block from the target hall to avoid unjustified exposure to personnel. In practice, however, perfect confinement by physical barriers is not feasible as some openings in the containment are necessary to allow for access, transfer of equipment, etc. A pressure cascade between the various compartments of the BDF is therefore foreseen, at least during beam operation, in order to compensate for the defects of the static confinement.

Prompt and residual dose rates

The prompt ambient dose equivalent rates were evaluated with FLUKA by convoluting the fluence of neutrons, protons, charged pions and muons with the respective energy-dependent fluence-to-effective dose conversion coefficients (Roesler et al., 2006). The
contribution from photons, electrons and positrons was neglected for saving computing time. According to former studies, a dose contribution from electromagnetic processes of less than 10% of the hadronic contribution can however be expected behind the shielding (Arduini et al., 2015; Strabel et al., 2015). The prompt dose rates were evaluated for the average beam intensity of $4 \times 10^{13}$ protons per 7.2 s.

The expected prompt dose rates in the target complex are depicted in Figure 3. As expected, the highest dose rates can be found in the region of the target reaching a few $10^6$ Sv/h. They are reduced by a few orders of magnitude in the surrounding iron and steel shielding. Above the helium vessel the prompt dose rates reach approximately 100 mSv/h. The prompt dose rates are further reduced by the above concrete shielding, such that they drop down to below 1 μSv/h in the target hall allowing for access to the target hall during operation. At the bottom and the sides of the target bunker, the dose rates drop down to below 1 mSv/h in the surrounding soil such that levels of soil activation are considered acceptable (Strabel et al., 2015).

Figure 3. Prompt (left) and residual (right) dose rates (in [μSv/h]) in the target complex

The residual ambient dose equivalent rates were estimated assuming the average beam intensity of $4 \times 10^{13}$ protons per 7.2 s for an 81 days long operation time per year for a total of five consecutive years (i.e. a total of $2 \times 10^{20}$ POT). They were obtained by convoluting the fluence of photons, electrons and positrons from $\gamma$- and $\beta$-decays with the respective energy-dependent fluence-to-effective dose conversion coefficients like for the prompt dose rate calculation. Note that the air- and helium-filled regions were selectively set to vacuum when producing and transporting the radioactive decay products. In that way, radioactive decay products originating from the activated air and helium were ignored. This is useful since the activated air and helium are released into the environment in case of access and will therefore no longer contribute to the respective residual dose rates. The air and helium activation will be discussed in a separate section.

Figure 3 further shows the expected residual dose rates in the target complex for a cooling time of 1 week. The highest dose rates can be found in the region of the target. They are in the order of a few $10^8$ μSv/h (100 Sv/h). The closest area where human intervention is required is above and next to the helium vessel enclosing the shielding. Here maximum residual dose rates of a few μSv/h after 1 week of cooling are reached considering that the helium vessel is closed and all shielding elements are in place. The residual dose rates in the target hall can further be considered negligible.
Figure 4 illustrates the expected prompt dose rates and residual dose rates for 1 week of cooling in the experimental area. The prompt dose rates, which are mainly due to muons, reach a maximum of approximately 100 mSv/h at the beginning of the cavern and drop down to below 1 mSv/h in the surrounding soil. The level of soil activation is considered acceptable (Strabel et al., 2015), particularly due to the fact that the dose rates are dominated by muons. The highest residual dose rates can be observed at the first part of the muon shield magnets reaching up to about 10 μSv/h after 1 week of cooling, thus allowing for access to this area after operation.

**Figure 4.** Prompt (left) and residual (right) dose rates (in [μSv/h]) in the experimental area


The expected prompt dose rates from muons in the surrounding areas are shown in Figure 5. The dose rates on the ground level behind the experimental hall reach up to about 50 μSv/h. It is therefore planned to cover the area with at least 3 m of soil. This allows to reduce the muon dose to below 0.5 μSv/h (non-designated area), since the muons pass the soil at a small angle. The area shall furthermore be fenced off. The muon dose rates in the existing beam lines TT81/2/3 and the EHN1 experimental hall lie below 0.5 μSv/h. This demonstrates that the SHiP operation does not have a negative impact on the surrounding installations.

**Figure 5.** Prompt muon dose rates (in [μSv/h]) above the experimental hall (top) and at the level of the existing beam lines TT81/2/3 (bottom left) and the EHN1 experimental area (bottom right)
Air and helium activation

The air and helium activation in the target complex was evaluated assuming the maximum beam power for five operational years. To evaluate the production of radionuclides in air a total of 39 isotopes were considered, including the radiologically most relevant short-lived isotopes $^{11}$C, $^{13}$N, $^{15}$O, $^{18}$O and $^{39}$Ar as well as $^{3}$H, $^{7}$Be, $^{14}$C, $^{32}$P, $^{33}$P and $^{35}$S among those with long half-lives. The highest air activation resulted in the air surrounding the helium vessel with $1.9 \times 10^7$ Bq after 60 s of cooling. When comparing it to the CA$^2$ values of the Swiss legislation (The Swiss Federal Council, 2017) it results in 0.7 CA.

For all helium-filled regions, a realistic purity of 99.9 % helium and 0.1 % air contamination was considered. In the most critical helium region, that is the innermost region of the helium vessel surrounding the target, a total activity of $2.8 \times 10^9$ Bq for helium and $6.1 \times 10^7$ Bq for air after 60 s of cooling was evaluated. It results in 0.4 CA for helium and $7.5 \times 10^5$ CA for air. This demonstrates the effectiveness of helium, which only gives rise to the formation of tritium that has a significantly lower radiological impact than the radionuclides arising from air.

When considering the accident of a helium vessel breakdown with a complete mixing of the activated air and helium, it would result in 2.7 CA and a committed effective dose per hour of stay of 8 μSv. This was used to define the classification of the BDF ventilation system, for which the ISO norm 17873:2004 for nuclear installations was taken as guideline for BDF.

In addition, the radiological impact from an accidental release of the activated air and helium into the environment without any cooling time was estimated. Therefore, six reference population groups around the BDF facility were identified. They include an existing reference group, which are waste-dump workers (WDW) as well as four hypothetical houses (W, NW, NE and SE of the facility) and one farmer group (A) exposed due to agriculture. To conservatively estimate the effective doses in an easy age-independent way, the maximum dose coefficients from different age groups were used for the residential groups. Since the ventilation stack parameters are not yet defined, a ground

2. Person working 40 hours per week, 50 weeks per year with standard breathing rate in activated air with CA = 1 receives 20 mSv.
3. Design and operational criteria of ventilation systems for nuclear installations other than nuclear reactors.
release was further conservatively assumed. The highest annual dose received is expected for the housing groups north-west and west of the facility. It amounts to 0.5 nSv per year, which is sufficiently below the dose objective for the public, which is 10 μSv/year from all facilities at CERN.

One thing, which was not taken into account in the above-given calculations, is the contribution from tritium out-diffusion. To estimate this contribution, the tritium production of all the critical regions inside of the Helium vessel was calculated. The highest contribution namely with 95% comes from the target itself. It amounts to a total of 18 TBq. In the cast iron and concrete shielding the tritium production amounts to 1 TBq and 2 MBq, respectively. The tritium can then be absorbed by the target cooling water in form of HTO and can outgas from the iron and concrete shielding into the helium and air environment. Assuming an immediate release efficiency of 100 % would be over-conservative. However, calculating the out-diffusion is deficient due to the fact that diffusion constants for tritium are available only for few materials and not in the full temperature range. Therefore, tritium out-diffusion experiments for all of the materials relevant for BDF are currently performed (see Section “BDF target prototype test”).

Radioactive waste production

A waste study was performed to predict the amount and the characteristics of the radioactive waste that will be produced during BDF operation. The objectives of such a study are to improve the management of radioactive waste and to eventually reduce the overall radioactive waste production. To distinguish areas of radioactive waste from conventional ones the liberation limits $LL$ from the Swiss legislation (The Swiss Federal Council, 2017) were used. To exempt a material containing a mixture of radionuclides of artificial origin from any further regulatory control, the following sum rule should be respected:

$$\sum_{i=1}^{n} \frac{a_i}{LL_i} < 1$$  \hspace{1cm} (1)

where $a_i$ is the specific activity (Bq/kg) or the total activity (Bq) of the $i^{th}$ radionuclide of artificial origin in the material, $LL_i$ is the respective CERN exemption limit for the radionuclide $i$ in the material and $n$ is the number of radionuclides present. If the sum rule is not fulfilled, the material is radioactive according to the Swiss legislation. Twelve different cooling times ranging from 15 minutes to 30 years were assessed. Figure 6 shows the waste zoning of the BDF target area after one year of cooling time. Note that the values in air- and helium-filled regions are insofar not representative, as the air and helium will be diluted by leakage and/or extraction into the environment. The zoning plots show that the most activated parts of BDF are the target and the cast iron and steel shielding elements. The radioactive waste production was minimised by making activated parts of the shielding removable such that they can easily be separated from the non-radioactive parts.
BDF target prototype test

The reliability of the target is a critical aspect for radiological protection, but also for physics, availability of the facility, etc. Therefore, a target prototype was tested with a representative beam scenario to be able to evaluate the material response and instrumentation under unprecedented temperature and stresses. The North Area target zone, called TCC2, was considered as ideal location for such a test, where beam conditions similar to BDF (400 GeV protons, 320 kW, $5 \times 10^{12}$ p/pulse) could be tested. Due to the high beam load, density and Z composition of the target, considerable material activation was expected. Therefore, the shielding and residual dose rates were studied with FLUKA. Standard concrete blocks of 80 cm thickness were used for shielding the target resulting in the expected residual dose rates for the envisaged $3 \times 10^{16}$ POT as shown in Figure 7. Two beam periods were foreseen and for the first $0.9 \times 10^{16}$ POT were reached. The dose rates at a distance of 40 cm from the target were measured after 1 week of cooling reaching a maximum of 43 mSv/h. The simulations result in 30 mSv/h for the same number of POT and therefore show relatively good agreement with the measurements. Discrepancies can be explained by the fact that the simulations do not account for indirect irradiation of the setup from the secondary particles originating from standard operation in TCC2.

Figure 6. Waste zoning (in multiple of LL) of the BDF target area after 1 year of cooling time


Figure 7. Residual dose rates (in $[\mu$Sv/h]) for $3 \times 10^{16}$ POT after 1 week of cooling at the BDF target prototype test

A dedicated water-cooling circuit for the target cooling was installed, in order to avoid contamination of the TCC2 cooling circuits in case of a target break. The cooling circuit is supplied with demineralised water and includes mechanical filters and an ion exchanger to catch impurities and activated ions. The expected dose rate from water activation itself was estimated using Actiwiz3 (Theis and Vincke, 2016). Assuming that all the isotopes will be caught in the cartridge of the ion exchanger, the residual dose rate after 1 h of cooling at 40 cm from the cartridge will be around 14.5 mSv/h. When comparing it to the measurements by a PMI, which was placed 40 cm away from the cartridge, the dose rate with 15.2 mSv/h after 1 hour of cooling showed good agreement. A sampling valve further allowed to take water samples and to investigate the target integrity. The analysis of the gamma radiation emitted from water samples showed the presence of Ta/W spallation products in the water. Ongoing studies will characterise the dust found in the water.

The air activation due to the BDF prototype target test can furthermore be considered negligible, since its total number of POT represents only 1 % of the total POTs received in this area during the whole year.

The BDF prototype test was also used to conduct tritium out-diffusion experiments in order to measure the tritium out-diffusion for the materials most relevant for BDF, which is tungsten, tantalum, TZM, cast iron and concrete. The samples were supported on the outer target tank of the BDF prototype target. The samples had dose rates of the order of mSv/h at contact and O(1-10) μSv/h at 40 cm distance. The tritium outgassing is measured in air with the help of a bubbler and the outgassing in water by immersing the samples. The first measurements with the concrete samples showed tritium presence both in air and in water. Further measurements and analysis are on-going.

Summary and conclusions

An in-depth study of the proposed BDF at CERN’s SPS is underway. The main RP challenges arise from the high beam power, the proximity to the surface, other experimental facilities and the CERN fence, but also from keeping the flexibility for future installations. The design of the facility is optimised based on in-depth studies of the relevant RP aspects in order to fully comply with the personnel and environmental protection requirements. The BDF project team aims to produce a Comprehensive Design Study by the end of 2018 to accompany the SHiP proposal to be considered in the next update of the European Strategy for Particle Physics (ESPP) expected in 2020.

Acknowledgements

The authors would like to thank all the members of the BDF project team and the SHiP collaboration team for fruitful discussions and input for the paper.

References


Standardisation of concrete composite for radiation shielding

Ken-ichi Kimura¹, Mikihiro Nakata², Koichi Okuno³, Yoshihiro Hirao⁴, Satoshi Ishikawa⁵, and Yukio Sakamoto⁶
¹Fujita Corporation, ²MHI NSE, ³HAZAMA-Ando, ⁴National Maritime Research Institute, ⁵ITOCHU Techno-Solutions Corporation, and ⁶ATOX Corp.
kkimura@fujita.co.jp

Concrete is widely used as radiation shield in nuclear reactors and irradiation facilities, because of its flexibility and sufficient supply. However, there are few discussions on the content of the shielding concrete composition, and very old data (which have uncertain information regarding material properties beside of elemental data) are still used in the shielding calculation. So, we organised “Radiation Shielding Material Standardization Working Group under Atomic Energy Society of Japan (AESJ) to proceed the standardisation of shielding concrete. We have been targeting to propose the final draft of a standard of concrete composite for radiation shielding within 2020, and to establish the standard at 2021. Outline of the present draft, aim of the standard, and road map of the standardisation are introduced including the concept of the standard concrete for radiation shield based on the building material aspects, such as mixture proportion, ratio of water to cement, cement hydration and so on. Furthermore, some of the calculation for dose attenuation in simple concrete geometry with potential candidates of the standard concrete composite are carried out with the comparison of those with old data mentioned the above, in order to confirm the effectiveness proposed standard. The comment and discussion on our activity in SATIF14 will be help our further standardisation activity.
Evaluation of the effect of the airborne natural radioactivity near KOMAC

Sung-Kyun Park1*, Yi-Sub Min1, and Jeong-Min Park1
1Korea Multi-purpose Accelerator Complex, Korea Atomic Energy Research Institute
181, Mirae-ro, Geoncheon-eup, Gyeongju-si, 38180, South Korea
*skpark4309@kaeri.re.kr

Korea Multi-purpose Accelerator Complex (KOMAC) is branched off from Korea Atomic Energy Research Institute (KAERI). KOMAC has developed and operated a 100-MeV high-power proton accelerator to offer an optimum proton beam. This proton accelerator has been installed in the radiation controlled areas of the accelerator building. The radiation monitoring system (RMS) is monitoring the ionising radiation near the proton accelerator. In addition, the contamination measurement activities near the proton accelerator of KOMAC should be performed periodically for the radiation safety of KOMAC. Since generally the natural radioactivity affect the background of the contamination measurement result, the effect of the natural radioactivity near the facility should be evaluated. Especially, the radon and the radon progeny of the airborne natural radioactivity is one of the most influential factors in the background formation of the alpha contamination measurement. As the radiation safety activity of KOMAC, the radon concentrations near KOMAC are detected periodically and the radon progeny can be calculated from the amount of the radon detected and measured by the alpha spectrometer. The radon concentration in the general area of KOMAC is also important for the health of the workers and user. In this research, the radon concentration near KOMAC will be described and the effect of the background formation of the radon and radon progeny will be evaluated.
Session VI: Code benchmarking, and intercomparison
Improvement of a high-energy fission model for spallation reactions

Hiroki Iwamoto*, Shin-ichiro Meigo¹
¹J-PARC Center, Japan Atomic Energy Agency, Japan
*iwamoto.hiroki@jaea.go.jp

In the research and development of spallation neutron source facilities such as Accelerator-Driven Systems (ADSs), it is necessary to accurately predict the yields of fission fragments produced from the spallation reactions, which include various volatile materials such as xenon, iodine and krypton important to their analysis of the gas treatment system for the target. To ameliorate the accuracy of the yield prediction, we have been improving the high-energy fission model of Generalized Evaporation Model (GEM) implemented in the Particle and Heavy Ion Transport code System, PHITS. In this study, the description of the fission probability implemented in GEM was modified to account for the fission cross-sections for subactinide nuclei. The fission probability was deduced from Prokofiev’s phenomenological systematics of the proton-induced fission cross-sections and the Liège intranuclear cascade model version 4.6, INCL4.6. Comparing with experimental data and different nuclear reaction models (i.e. original GEM model, CEM03.03 and INCL++/ABLA07) shows that the modified model can predict the fission cross-sections for a wide range of target nuclei from the threshold energies to the GeV range, and its reproducibility is relatively good.

Introduction

In the research and development of spallation neutron source facilities such as Accelerator-Driven Systems (ADSs), it is necessary to accurately predict the yields of fission fragments produced from the spallation reactions, which include various volatile materials such as xenon, iodine and krypton important to analyses of the gas treatment system for the spallation target. To ameliorate the accuracy of the yield prediction, we have been improving the high-energy fission model of Generalized Evaporation Model (GEM) (Furihata, 2001) implemented in the Particle and Heavy Ion Transport code System (PHITS) (Sato et al., 2018). As a first step of the improvement, we proposed a new model to describe the fission probability in the high-energy fission model (Atchison, 1998) embedded in the GEM in the previous study (Iwamoto, 2018), and demonstrated that the proposed model coupled with the Liège intranuclear cascade model (INCL) version 4.6 can predict the proton-, neutron- and deuteron-induced fission cross-sections for a wide range of target nuclei from the threshold energies to the GeV range with markedly improved accuracy.

In this paper, we review comparisons of the fission cross-sections with different spallation models for proton- and neutron-induced fission cross-sections, and shows their predictive accuracy; for the comparison, two models are adopted for the spallation models: a combination of INCL++ (Leray et al., 2013; Boudard., et al., 2013) and ABLA07 (Kelic et al., 2008) and CEM03.03 (Mashnik, 2012) incorporated in MCNP6 (Goorley et al., 2012).
The modified models

Because details of our model are shown in “Unified description of the fission probability for highly excited nuclei” (Iwamoto and Meigo, 2018), a brief description is provided here. Our model assumes that the fission probability $P$ is expressed as the following form:

$$ P = g \cdot P_{\text{total}} $$

(1)

where $P_{\text{total}}$ is the total fission probability defined as the ratio of the fission cross-section and the non-elastic cross-section:

$$ P_{\text{total}} = \frac{\sigma_f}{\sigma_{\text{nonel}}} $$

(2)

For $\sigma_{\text{nonel}}$ and $\sigma_f$, we employ Pearlstein–Niita’s proton-induced non-elastic cross-section systematics (Niita et al., 2001) and Prokofiev’s proton-induced fission cross-section systematics (Prokofiev, 2001). Note that these systematics are functions of incident proton energy of a projectile, charge $Z$ and mass $A$ of a target. To apply the Prokofiev systematics to the Monte Carlo spallation simulation, we introduced corresponding mean incident proton energy $\langle E_p \rangle$, mean target charge $\langle Z_t \rangle$, and mean target mass $\langle A_t \rangle$, which are calculated from the calculation of INCL4.6; each of them is expressed as follows:

$$ \langle E_p \rangle = 0.141 \times \langle E \rangle^\alpha $$

(3)

$$ \langle Z_t \rangle = Z + (0.0216E - 1.037) $$

(4)

$$ \langle A_t \rangle = A + (0.0457E - 1.049) $$

(5)

where $\alpha = 1.964 - 0.000884\langle A_t \rangle$ and $E$ is the excitation energy of the residual nucleus. The factor $g$ is an adjustable factor determined so as to fit the Prokofiev systematics and experimental fission cross-sections, which is of the form:

$$ g = \max \{ \gamma_0 \left( \frac{1}{1 + e^{-\gamma_1(E-E_2)}} - \gamma_3 \right), 0 \} $$

(6)

where $\gamma_{0,1,2,3}$ are adjustable parameters.

The isotopic distributions of the fission fragments are calculated in the same manner as “The GEM code: a simulation program for the evaporation and the fission process of an excited nucleus” (Furihata, 2001).

Results and discussions

Comparison with original model

Figure 1 shows comparisons of proton-induced fission cross-sections for $^{209}\text{Bi}$, $^{208}\text{Pb}$, $^{197}\text{Au}$ and $^{181}\text{Ta}$ calculated with the original GEM model implemented in PHITS version 3.02 and the modified GEM model, together with the Prokofiev systematics, where both models are coupled to INCL4.6. Figure 2 shows the same as Figure 1 but of neutron-induced fission cross-sections. As shown in “Unified description of the fission probability for highly excited nuclei” (Iwamoto and Meigo, 2018), it can be seen that the original model calculations are in agreement with experimental values for over a wide range of incident proton and neutron energy and of target mass, whereas the original GEM model underestimates the fission cross-sections. Figure 3 shows a mass number distribution of the
production cross-section of spallation products for the p (1 GeV) + $^{208}$Pb reaction calculated with the original and modified GEM models, in comparison with experimental data (Enqvist et al., 2001). Due to the underestimation of the fission cross-sections for the original GEM model, the calculated production cross-sections of the fission fragments are also underestimated; on the other hand, the modified GEM model is generally in good agreement with experimental data of the fission fragments. Discrepancies seen in the mass region of the evaporation residue (140 < A < 208) is attributable to the description of other processes (e.g. the evaporation process and the intranuclear cascade process).

Figure 1. Comparison of proton-induced fission cross-sections for $^{209}$Bi, $^{208}$Pb, $^{197}$Au and $^{181}$Ta calculated with the original and modified GEM models with experimental data.

Figure 2. Comparison of neutron-induced fission cross-sections for $^{209}$Bi, $^{208}$Pb, $^{197}$Au and $^{181}$Ta calculated with the original and modified GEM models with experimental data.


Figure 3. Mass number distribution of production cross-sections of spallation products for p(1000 MeV) + $^{208}$Pb reaction calculated with the original and modified GEM model compared with experimental data.


**Comparison with different models**

Figure 4 shows comparisons of the proton-induced fission cross-sections of $^{209}$Bi, $^{208}$Pb, $^{197}$Au and $^{181}$Ta calculated with the modified GEM model, INCL++/ABLA07 and CEM03.03, with the experimental data. Figure 5 shows the same as Figure 4 but of the neutron-induced fission cross-sections. Here, note that the Cascade-Exciton Model (CEM) does not calculate the neutron-induced reactions with energies below 150 MeV due to the specification of the MCNP code; they are calculated with the high-energy nuclear data.
implemented in MCNP. Comparing among three different models shows that they have similar trends, and discrepancies are observed for each model. Among them, the modified GEM model reproduces the experimental data relatively well for both proton- and neutron-induced reactions.

**Figure 4.** Comparison of proton-induced fission cross-sections for $^{209}$Bi, $^{208}$Pb, $^{197}$Au and $^{181}$Ta calculated with CEM03.03 and INCL++/ABLA07 with experimental data

![Graphs comparing proton-induced fission cross-sections](image)


**Figure 5.** Comparison of proton-induced fission cross-sections for $^{209}$Bi, $^{208}$Pb, $^{197}$Au and $^{181}$Ta calculated with CEM03.03 and INCL++/ABLA07 with experimental data

![Graphs comparing proton-induced fission cross-sections](image)

Conclusion

To ameliorate the accuracy of the yield prediction, we have been improving the high-energy fission model of GEM implemented in PHITS. In this study, the description of the fission probability implemented in GEM was modified to account for the fission cross-sections for subactinide nuclei. The fission probability was deduced from Prokofiev’s phenomenological systematics of the proton-induced fission cross-sections and the INCL4.6. Comparing with experimental data and other nuclear reaction models showed that the modified model can predict the fission cross-sections for a wide range of the target nuclei from the threshold energies to the GeV range, and its reproducibility is relatively good.

Acknowledgements

This work was supported by Japan Society for the Promotion of Science (JSPS) Grants-in-Aids for Young Scientists B [grant number 17K14916].

References


High-Energy Intra-Nuclear Cascade Liège-based Residual (HEIR) nuclear data for activation-transmutation studies

M. Fleming\textsuperscript{1,2}, J.-C. David\textsuperscript{3}, D. Mancusi\textsuperscript{4}, J.-L. Rodríguez-Sánchez\textsuperscript{5}, S. Juan\textsuperscript{2,6}, M. Gilbert\textsuperscript{2}

\textsuperscript{1}OECD Nuclear Energy Agency, 92100 Boulogne-Billancourt, France
\textsuperscript{2}UK Atomic Energy Authority, Culham Science Centre, Abingdon, OX143 DB, UK
\textsuperscript{3}Institut de Recherche sur les lois Fondamentales de l’Univers, CEA, Université Paris-Saclay, 91191 Gif-sur-Yvette, France
\textsuperscript{4}Den-Service d’étude des réacteurs et de mathématiques appliquées (SERMA), CEA, Université Paris-Saclay, 91191 Gif-sur-Yvette, France
\textsuperscript{5}University of Santiago de Compostela, Department of Particle Physics, 15782 Santiago de Compostela, Spain
\textsuperscript{6}École nationale supérieure de physique, électronique et Matériaux (Phelma), 38016 Grenoble, France

\*Michael.FLEMING@oecd-nea.org

The use of tabulated nuclear data files is the standard practice for the simulation of most nuclear systems with particle energies up to 20 MeV. The formats for these data are built on the assumption that there are a relatively small number of reaction channels that can be distinguished by the number and species of emitted particles. Fissionable isotopes are treated as a special case, with one integrated fission channel and supplemental data. Above these energies, these assumptions break down, as this compartmentalisation becomes infeasible and the cascade, evaporation, fission, fragmentation and other de-excitation processes must be jointly considered. As a result, stochastic event generators are utilised in the simulation of systems with particle energies of hundreds of MeVs and above. The use of event generators within Monte Carlo calculations offers robust simulation of the model physics, but introduces challenges for some applications, including activation-transmutation. The need to converge the reaction rates for all product radioisotopes may require impractical computational resources in every calculation. By storing previously converged cross sections in tabulated files, a simulation must converge only the energy spectra of incident particles in order to calculate accurate reaction rates for time-dependent inventory studies.

The INCL++5.3 and ABLAv10.3 intra-nuclear cascade and de-excitation models within Geant4 were used in the calculation of over $10^{12}$ simulations of incident protons on $2 \times 10^5$ targets with energies up to 1 GeV. These were collated into data tables in the international-standard ENDF-6 format and then processed using the PREPRO-2017 nuclear data code. The resulting files were provided as group-wise evaluated nuclear data files and were distributed as the HEIR-0.1 library with FISPACT-II version 4.0, which has been extended to splice such data files with lower-energy ENDF-6 data.

The most recent release of Geant4 includes the new INCL++6.0 and a C++ translation of the ABLA07 code. As part of the effort to generate an upgrade version of the HEIR library, new data has been calculated using these models and compared against mass distribution data, as well as cumulative cross sections for residual production of radioisotopes from the EXFOR database. Both the HEIR-0.1 and HEIR-0.2\textbeta data show many improvements over other high-energy nuclear data libraries, such as HEAD-2009 and JENDL-2007/HE, for activation-transmutation applications.
MARS15 code developments and applications to the LBNF/DUNE project

Nikolai V. Mokhov1*, Yury I. Eidelman2, Vitaly S. Pronskikh1, Igor L. Rakhno1, Diane Reitzner1, Sergei I. Striganov1, Igor S. Tropin1
1Fermi National Accelerator Laboratory, Batavia, Illinois 60510, US
2RadiaSoft, LLC, Boulder, Colorado 80301, US
* mokhov@fnal.gov

Recent developments to the MARS15 multi-purpose Monte Carlo code (version of 2018) are described. These include a nuclear event generator at intermediate energies with switchable inclusive, semi-inclusive and exclusive modes, low-energy neutron transport that utilises a new library of ACE files based on the ENDF/B-VIII.0, the improved DPA model libraries of NJOY-generated cross-sections, improved particle decay algorithms with kaon form-factors and muon polarisation taken into account, as well as a refined particle splitting and Russian roulette algorithm. The particle transport modules were further refined for tracking in arbitrary magnetic and electrical fields with possible acceleration and deceleration. Incorporating the ROOT system allows for arbitrary geometry description, debugging and checking for unwanted region overlaps, 2D and 3D visualisation, as well as geometry model sharing between MARS15 and GEANT4. The Graphical-User Interface capabilities have been extended. Integration of MARS15 with MAD-X/PTC allows a cross-talk between the two codes for precise multi-turn modelling of beam loss and induced impact on accelerator components. New modules allow parallel scalable MARS15 execution in an MPI environment on a single computer, supercomputers like ALCF/Theta or grid systems. With all these updates, the code capabilities and predictive power in various accelerator applications were noticeably improved. This is illustrated by recent design studies: justification and optimisation of the neutrino beamline complex for the flagship LBNF/DUNE experiment with benchmarking LBNF approaches and solutions in the existing NuMI experiment.

Introduction

Predictive power and reliability of particle transport simulation tools and physics models for both Energy Frontier and Intensity Frontier applications should be well-understood and justified to allow for viable designs of future accelerators and colliders with a minimal risk and a reasonable safety margin. Simulations with corresponding quality are only possible with a few well-established Monte Carlo codes. One of them, MARS15, is a multi-purpose Monte Carlo code developed since 1974 for detailed simulation of hadronic and electromagnetic cascades in an arbitrary 3-D geometry of shielding, accelerator, detector and spacecraft components in a broad energy range (Mokhov, 1995; MARS15 homepage, 2019; Mokhov, 2014). Driven by the above applications, the code is under continuous development. The most recent enhancements, extensions and benchmarking are described in this paper along with illustrations of the code use for the Long-Baseline Neutrino Facility (LBNF) and Deep Underground Neutrino Experiment (DUNE) design optimisation.
MARS15 main features

MARS15 is a set of Fortran 77 and C++ modules for Monte Carlo simulation of coupled hadronic and electromagnetic cascades, with heavy ion, muon and neutrino production and interactions. It covers a wide energy range: 1 keV to 100 TeV for muons, hadrons, heavy ions and electromagnetic showers, and $10^{-5}$ eV to 100 TeV for neutrons.

All strong, weak and electromagnetic interactions in the entire energy range can be simulated either inclusively or exclusively – in a biased mode or in a fully or partially analogue mode. Nuclide production, decay, transmutation and calculation of the activity distribution is done with the built-in DeTra code (Aarnio, 1998). MARS15 uses ENDF/B-VIII.0(2018) nuclear data library (ACE Libraries, 2018) to handle interactions of neutrons with energies below 14 MeV as well as derive – using the NJOY2016 processing [6]–DPA x-sections below 200 MeV. The elemental distributions are automatically unpacked into isotope distributions for both user-defined and those from the 172 built-in materials.

A tagging module allows one to tag the origin of a given signal for source term or sensitivity analyses. Several variance reduction techniques, such as weight windows, particle splitting and Russian roulette are possible. Geometry can be described in six ways, which include a basic solid-body representation and a ROOT-based engine. The MAD-MARS Beam Line Builder (MMMBLB) and a recent active merge with MAD-X-PTC allows for a convenient creation of accelerator models and multi-turn tracking and cascade simulation in accelerator and beamline lattices. MARS15 is routinely used with ANSYS for iterative studies of thermo-mechanical problems and can be interfaced to a hydrodynamic code to study phase transition and “hydrodynamic tunneling” – first done in 1993 at the SSC Laboratory for a 20-TeV proton beam (Wilson, 1993).

Nuclear event generator

Depending on a projectile type and its kinetic energy $E_0$, the following models – extended and improved in 2017-2018 – are used in MARS15:

- $E_0 < 0.12$ GeV down to 1 MeV (charged particles and gammas) and 14 MeV (neutrons): a combination of the TALYS-based evaluated nuclear data library (TENDL-2015 version (Koning A. J., 2015) in its inclusive, semi-inclusive or exclusive mode (user’s choice) (Rakhno, 2015) and Los-Alamos Quark-Gluon String Model (LAQGSM) (Mashnik, 2008). Note that we use a stable TENDL-2015 release, because some inconsistencies have been found in the latest TENDL-2017.
- $0.12 < E_0 < 0.5$ GeV: a combination of Cascade-Exciton Model (CEM) (Mashnik, 2008) and LAQGSM.
- $0.5 < E_0 < 10$ GeV: LAQGSM.
- $10$ GeV $< E_0 < 100$ TeV: LAQGSM or native MARS inclusive model (user’s choice).

TENDL-calculated and measured results for the p+$^{65}$Cu reaction are shown in Figures 1 and 2, taken from “Modelling proton-induced reactions at low-energies in the MARS15 code” (Rakhno et al., 2015).
Figure 1. Calculated (FDG, TENDL) and measured single neutron production x-section on $^{65}$Cu (Rakhno et al., 2015)


Figure 2. Calculated (ALICE, TENDL) and measured angular distributions of secondary neutrons from $^{65}$Cu (Rakhno et al., 2015)


The performance of MARS15 with CEM and TENDL as a nuclear event generator for 50 and 10 MeV protons on aluminium cylindrical target in Air is shown in Figures 3 and 4. While the neutron flux density distributions calculated with these two models are very similar for $E_P = 50$ MeV, one can observe a drastic improvement by switching from CEM to TENDL for all projectiles and materials at projectile energies 1 to 30 MeV. The latter is crucial for radiological applications as well as for DPA and H/He gas production, especially in microstructures. The CPU performance is similar for CEM and TENDL.
Figure 3. Neutron flux density for 50-MeV protons on Aluminium target in Air MARS15-calculated with CEM (left) and TENDL (right)


Figure 4. Neutron flux density for 10-MeV protons on Aluminium target in Air MARS15-calculated with CEM (left) and TENDL (right)

Three-body decays

The accurate description of tiny effects in unstable particle decays is crucial in numerous muon and neutrino physics applications. Improvements have been made to the three-body decay module in MARS15. A matrix element of a V-A (vector minus axial vector) Lagrangian for the weak interactions and polarisation was added to the basic decay kinematics of kaons and muons. Electron neutrino spectra calculated via the basic decay kinematics, and with a matrix element and polarisation added are shown in Figure 5 for a π⁺ decay chain and for the Far Detector of the LBNF/DUNE experiment.

Figure 5. MARS15 calculated electron neutrino spectra in the \( \pi^+ \rightarrow \mu^+ \rightarrow e^+\nu_e\bar{\nu}_\mu \) decay chain for π⁺ energies 100 MeV, 1 GeV, 10 GeV and 100 GeV (left) and at a DUNE Far Detector (right).


Atomic displacements (DPA)

In MARS15, the energy of recoil fragments and newly created charged particles generated from elastic and inelastic nuclear interactions is used to calculate atomic displacement cross sections \( \sigma_{\text{DPA}} \). The Norgett-Robinson-Torrens (NRT) model (Norgett, 1975) with and w/o damage efficiency \( \xi(T) \) (Stoller, 2000; Nordlund K. et al., 2018) is used then for the number of stable defects. Atomic screening parameters are calculated using the Hartree-Fock formfactors and recently suggested corrections to the Born approximation. The effective threshold displacement energy values have been updated for all materials according to the recent evaluations. NJOY2016+ENDF/B-VIII.0(2018) was used to generate a database of \( \sigma_{\text{DPA}} \) for 490 nuclides for neutrons from \( 10^{-5} \) eV to 150 MeV. As an example, neutron \( \sigma_{\text{DPA}} \)
in the latter energy range are presented for several materials in Figure 6. DPA in neutron-nuclear interactions above 150 MeV are derived the same way as described in the first sentence. In the same run/output, atomic displacement x-sections and resulting DPA in regions of interest are calculated in three ways: pure NRT and with Stoller (Stoller, 2000) and Nordlund (Nordlund et al., 2018) efficiency functions $\xi(T)$. Note that in the Stoller's case, $\xi(T)=1$ is used if it becomes greater than 1.

Figure 6. Excerpts from the MARS15 ENDF/B-VIII.0(2018)+NJOY2016 database of neutron displacement cross-sections $\sigma_{DPA}$ at $10^{-11} < E_n < 150$ MeV for $^7$Be, $^{12}$C, $^{27}$Al, $^{63}$Cu, $^{182}$W and $^{208}$Pb isotopes.

Figure 7 presents the comparison of results from the MARS15 DPA model to data for protons and neutrons on Aluminium and Copper. A very good agreement is found over a broad energy range. The effect of the $\xi(T) < 1$ efficiency for the number of stable effects is clearly seen.

Figure 7. MARS15-calculated - with and w/o Stoller/Nordlund efficiency $\xi(T)$ - displacement cross-sections $\sigma_{DPA}$ for $p^{27}\text{Al}$ (left) and $p^{\text{nat}}\text{Cu}$ and $n^{63}\text{Cu}$ (right) reactions in comparison with other theoretical and experimental data


**Geometry description and particle tracking**

There are six ways to build a geometry model in the current MARS15:

- user-friendly solid-body representation geometry ASCII text files;
- user-generated ROOT (ROOT homepage, 2019) C++ files;
- GDML and ROOT binary files (import/export for two-way user-friendly exchange with other MARS15 and Geant4 (GEANT4 homepage, 2019);
- G4beamline (G4beamline, 2019) BruitDeFond generates MARS’s input files MARS.INP, GEOM.INP and FIELD.INP;
- STEP files from project CAD models used to generate ROOT geometry modules;
- lattice and beamline components such as dipole and quadrupole magnets, correctors, accelerating cavities, cryomodules and tunnel with all the details available on geometry, materials and electromagnetic fields by means of the advanced ROOT-based Beamline Builder.

Customised steppers (with the 8th order Runge-Kutta solver) are used in the generated geometry models for the accurate and optimal particle tracking in Superconducting Radiofrequency (SRF) cavities (with time-dependent electromagnetic fields and dark current production), quadrupole/dipole/solenoid magnets and thick shielding. The Graphical-User Interface (GUI) capabilities have been substantially extended providing a view of the geometry model, particle tracks and histograms. Views of tracks and histograms
are overlapped with the geometry outline. The display characteristics are set automatically with a possibility to adjust almost everything interactively as needed.

Recently, MARS15 has been coupled to the MAD-X (MAD, 2018) which is used in the integrated system to build the beamline and accelerator geometry models and perform precise and fast multi-turn tracking inside the beamline aperture with the MAD-X PTC module. This merge has already demonstrated a high performance and a convenience of use in creation of accelerator models and multi-turn tracking and cascade simulation in accelerator and beamline lattices for beam loss and collimation modelling.

A library containing functions and C++ classes which interfaces MARS with MAD-X is now packed with the MARS15 distribution. This library allows to (1) create a 3-D TGeo ROOT geometry model for the sequence described in a MAD-X-PTC input file with alignment of elements performed by means of the MAD-X survey table; (2) define transformation for each point in the phase space used in the PTC module to the phase space used in MARS15 and vice-versa; (3) pass on particles transported by MARS15 to MAD-X-PTC module using a formulated acceptance for the accelerator code model; (4) for particles transported in MAD-X-PTC, perform check of boundary crossing against the ROOT geometry in MARS15 with the particle is forwarded to the MARS15 stack.

The MARS15-MAD-X integrated system has been recently successfully used to design the Fermilab 8-GeV proton Recycler collimation system. Figure 8 illustrates typical simulation results.

**Figure 8.** Loss probability for 8-GeV beam halo protons passed through the primary collimator vs the number of turns (left) and prompt dose isocontours in the collimation region (right)


**Platform, compilers and MPI**

The following are system requirements for the current MARS15 version: (1) Linux ≥ gcc-4.8 on 64-bit architectures with the compiler policy in general to use the latest stable release, e.g. currently, gcc-8.2, which is ideal for modern C++ codes; (2) C++11 standard;
(3) gsl-2.4 or 2.5; (4) ROOT-6.14; (5) ISO standard Fortran to C interface; (6) many-core job mode – standard for decade – is still supported and used with 10 to 10⁵ cores routinely, with the mode included improved submission scripts and built-in optional averaging of output files after run completion; (7) genuine MPI mode; to eliminate discovered scalability bottleneck, common physics data (x-sections etc.) are accessed via a shared memory window (MPI-3 feature).

**Recent benchmarking**

One of the primary objectives of SATIF meetings has been de facto simulation code benchmarking. This was the primary value of these meetings for the code developers and for the entire community. Results of two impressive benchmarking campaigns were published in 2018. The first one was on thermal neutron fields induced by the 24 GeV/c proton beam on a 50-cm thick copper target at the CERN High Energy AcceleRator Mixed Field (CHARM) Facility (Oyama et al., 2018). Its schematic view is shown in Figure 9. Experimental data were compared to results of calculations by the PHITS (Sato et al., 2018), FLUKA (Ferrari et al., 2005) and MARS15 codes. As concluded in “Measurement and calculation of thermal neutrons induced by the 24 GeV/c proton bombardment of a thick copper target” (Oyama et al., 2018) and seen in Figure 10, PHITS results agree with data within 50% while FLUKA, MARS15 and PHITS results agree with each other within 30%.

**Figure 9. A cross-sectional view taken along the Cu target plane of the CHARM facility**

![Diagram of CHARM facility with experimental locations of gold foils indicated by numbers 1–13.](source)

The numbers 1–13 indicate the experimental locations of the gold foils.

Figure 10. Experimental (black symbols) and PHITS calculation (blue and red symbols) results for the thermal neutron flux (left) and ratios of the FLUKA (circles) and MARS (squares) results to PHITS results for the thermal neutron flux calculated at 3 heights vs longitudinal position in the CHARM facility (right).


To get more confidence in the MARS15-based LBNF target station design (see next section), a benchmarking campaign on air activation has been recently undertaken at the Fermilab NuMI target station for a 120-GeV beam on target (Rakhno, 2018). The results of comparison are shown in Table 1. MARS15 underestimates the $^{41}$Ar production rate by 50% (which is not that bad for this very difficult dynamic benchmarking) and agrees with data for other nuclides within 10-30%.

Table 1. Measured and calculated production rates ($\text{cm}^3 \text{ POT}^{-1} \text{ s}^{-1}$) for the most important radionuclides generated in the air in the beam enclosure of the NuMI target chase

<table>
<thead>
<tr>
<th></th>
<th>$^{41}$Ar</th>
<th>$^{11}$C</th>
<th>$^{15}$N</th>
<th>$^{15}$O</th>
</tr>
</thead>
<tbody>
<tr>
<td>Exp. data</td>
<td>$1.98\times10^{-12}$</td>
<td>$6.38\times10^{-11}$</td>
<td>$4.07\times10^{-11}$</td>
<td>$3.50\times10^{-11}$</td>
</tr>
<tr>
<td>Fermilab ES&amp;H methodology</td>
<td>$6.85\times10^{-12}$</td>
<td>$2.22\times10^{-10}$</td>
<td>$5.22\times10^{-11}$</td>
<td>$9.16\times10^{-11}$</td>
</tr>
<tr>
<td>MARS15</td>
<td>$1.08\times10^{-12}$</td>
<td>$4.44\times10^{-11}$</td>
<td>$3.71\times10^{-11}$</td>
<td>$4.16\times10^{-11}$</td>
</tr>
<tr>
<td>MARS15/data</td>
<td>0.55</td>
<td>0.70</td>
<td>0.91</td>
<td>1.19</td>
</tr>
</tbody>
</table>


**LBNF/DUNE**

The Long-Baseline Neutrino Facility (LBNF) at Fermilab which provides neutrino beams to the Deep Underground Neutrino Experiment (DUNE, the flagship of the US high-energy physics) with its detector in South Dakota 1300 km from Fermilab. LBNF schematic view at the Fermilab site is shown in Figure 11 with various views of the MARS-LBNF geometry model presented in Figure 12. Results of MARS15 simulations for the target station are
shown in Figure 13 and for the decay channel – in Figure 14, with the groundwater activation design limit reached after 5.6 metres of ordinary concrete shielding.

Figure 11. LBNF Beamline schematic


Figure 12. Various views of the MARS-LBNF geometry model

Figure 13. Target station simulations

Particle tracks (E>20 MeV) for 5 protons on target (top); prompt dose isocontours (mSv/hr) (bottom-left) and energy deposition in decay pipe upstream window for nominal operation (black squares), mis-steered beam (blue diamonds) and no-target accident (red crosses) (bottom-right).


Figure 14. Decay channel simulations: power deposition in double-walled steel pipe (left); star density in decay channel concrete shielding (right)

The hadron absorber complex – at 220 m downstream from the target and about 25 m below the Earth surface – is one of the most challenging LBNF subsystems. Its geometry model is shown in Figure 15. Results of MARS15 simulations are shown in Figure 16 for energy deposition in the absorber core for no-target beam accident (with temperature and stress below the limits) and for prompt dose isocontours in the entire complex for normal operation. Carefully designed shielding guarantees prompt and residual dose, air and cooling water activation as well as radiation load on ground water to be below the administrative limits with corresponding safety margins.

**Figure 15. Hadron absorber complex CAD (left) and MARS (right) models**

![Hadron absorber complex CAD and MARS models](source)

**Figure 16. Hadron absorber simulations: energy deposition (mJ/cm³/pulse) for no-target beam accident (left); prompt dose (mSv/hr) for normal 2.4-MW operation (right)**

![Hadron absorber simulations](source)
Changes to the LBNF design – justified in MARS15 simulations – improve its performance and provide substantial mitigation of energy deposition and radiological problems in one of the two most critical systems, the hadron absorber. These also helped mitigate requirements to the muon monitor design.

Thorough search and elimination of differences in MARS15LBNF and G4LBNF models were performed on the optimised neutrino-flux: geometry, materials, magnetic fields etc. Neutrino fluxes at the Far Detector (1 300 km) calculated with MARS15LBNF and G4LBNF now agree in the region of oscillation maxima within 10% as shown in Figure 17. The code related uncertainties were reduced to the differences in the event generators, especially for $K^-$ and $K^0$ mesons.

Figure 17. Neutrino fluxes at the FAR Detector (1 300 km from Fermilab) as calculated with MARS15LBNF (in two event generator modes) and G4LBNF

Acknowledgements

This document was prepared using the resources of the Fermi National Accelerator Laboratory (Fermilab), a US Department of Energy, Office of Science, HEP User Facility. Fermilab is managed by Fermi Research Alliance, LLC (FRA), acting under Contract No. DE-AC02- 07CH11359. It also used a Director’s Discretionary allocation at the Argonne Leadership Computing Facility, which is a DOE Office of Science User Facility supported under Contract DE-AC02-06CH11357.

References


MARS15-MADX integration and its application for design of the Fermilab Booster collimation system

Igor S. Tropin*, Nikolai V. Mokhov

1Fermi National Accelerator Laboratory, Batavia, Illinois 60510, United States
*tropin@fnal.gov

Design of a high-efficiency collimation system with adequate shielding for personnel, environment and equipment in the collimation region is an essential part in development and modernisation of accelerators. An approach for the solution of this class of tasks based on the integration of the MARS15 and MAD-X codes is presented. The system allows scalable runs on either a single host or a supercomputer in the MPI environment. A new collimation system for the Fermilab 8-GeV proton Booster has been designed using this new tool. The optimisation studies have been performed on the ALCF THETA supercomputer. A noticeable improvement of the beam cleaning efficiency has been achieved in the simulations. The proposed shielding configuration and parameters guarantee the prompt dose levels on the berm, residual dose levels in the tunnel and radiation loads on the sump water will be below the administrative limits with required safety margins.

Introduction

The tools described in this paper were created and iteratively developed for solving problems where the primary beam plays the role of the source term and the effect of the beam halo particles lost on an aperture boundary is under investigation. A subject of a study can be, for example, collimation efficiency, radioactivation and radiation damage of equipment installed in the tunnel, ground water activation around the tunnel, dose on top of the berm, and skyshine.

A challenge with simulation of such effects in circular accelerators and storage rings is that beam halo particles can leave the aperture after passage of many turns in a ring. Precise multi-turn beam tracking usually relies on the accelerator physics methods and algorithms. It would be natural to use these specific methods inside the machine aperture, while simulating trajectories and particle-matter interactions outside the aperture with well-established multi-purpose Monte Carlo codes. Based on the experience over the last decade, we have found that coupling of the MARS15 code (Mokhov, 1995; MARS 2019; Mokhov, 2014) in the ROOT (ROOT, 2019) mode for particle-matter stage with the MAD-X (MAD, 2018) modules for tracking in the machine aperture provides all one needs in such applications.

The STREG library from the MARS distribution includes the C++ tools which – among other things – allow creation of user-defined transport classes, called hereafter “steppers”. This feature of the library is described in the first section. The stepper based on the MAD-X PTC module is created. This stepper is a part of the MARS MADX-Beamline library. The library implements interface between MAD-X – particle accelerator design and simulation tool – and the MARS15 code system. Features of the library and example of use in development of the MARS based application for the Fermilab Booster is presented in
the second section. The Fermilab Booster, the proton synchrotron with circumference of 474 m, is a part of the Fermilab accelerator complex. In the current design, the Booster gets protons with kinetic energy of 0.4 GeV from the transfer line coming from the Linac and accelerates the beam to 8 GeV for the Main Injector accelerator.

There is a plan to substantially increase the beam power for the needs of the neutrino experiments. It turns out that the efficiency of the existing collimation system (Mokhov, 2003a) is not enough to keep the radiation levels in and around the Booster at the current (or lower level) with the beam power being increased (Kapin, 2017a). Based on the analysis of the Booster physical aperture, a proposal (Kapin, 2017b; Kapin, 2017c; Sidorov, 2017) called for a replacement of the current classical two-stage collimation system (with very thin primary collimators followed – at the appropriate phase advances – by a meter-long secondary collimators) with one or two “collimation units” made of a thick primary collimator followed practically immediately by a secondary collimator for the quasi-local absorption of the beam halo in the Booster realm. The last section describes the application of the newly-developed MARS15-MAD-X system for the optimisation modelling of the collimation efficiency and radiation environment around the proposed “collimation units”.

**MARS STREG library**

The library implements interface between the MARS15 general-purpose particle transport and interaction code (Mokhov, 1995; MARS, 2019; Mokhov, 2014) and the ROOT Geom library (ROOT, 2019). Corresponding subroutines from the Geom library are called from the basic MARS15 geometry module. The library contains two subroutines mandatory for any Monte Carlo particle transport code providing the answer to the question “where am I” and performing a “make-a-step” action. These subroutines are named \texttt{tgeo\_find} and \texttt{tgeo\_step1}, correspondingly. These and other service functions callable from the FORTRAN code are implemented in C++ and have “C” binding. FORTRAN interfaces for all such functions are combined in the FORTRAN90 module named STREG, where the binding of the FORTRAN and C functions is described according to the ISO standard.

With this approach, the number and type of arguments passed from FORTRAN code to the C functions checked by the compiler, i.e. errors caused by mismatch in parameters of functions can be detected at an early stage of the geometry model building. The usage of the ISO standard method to call C functions from the FORTRAN code, instead of the method specific for the GCC compiler, improves portability of the MARS code. Besides that, it is no longer necessary to add underscore to the name of C functions to make it possible to call from FORTRAN. An overhead related to the use the ISO standard method is that the FORTRAN statement \textit{USE STREG} should be added at the beginning of the declarative part of a FORTRAN code, where subroutines of the STREG library are called. Namely, in the current MARS15 version, the use-statement needs to be added to the following user’s call-back subroutines, contained in m1519.f file:

- \texttt{VFAN} – volumetric procedure, calls subroutine \texttt{tgeo\_blvol} described in the STREG module.
- \texttt{FIELD} – calculates the magnetic field components at the given point, dispatches call to function \texttt{tgeo\_field}, if the ROOT geometry is active.
- \texttt{REG1} – finds a region number for the given point in MARS non-standard, aka “user defined”, geometry, to perform the task it calls function \texttt{tgeo\_find} from STREG module. The subroutine is also used to import or create a ROOT geometry model.
The subroutines \textit{VFAN} and \textit{FIELD}, supplied with the MARS15 distribution in the m1519.f file, do not usually require any modification for use with the ROOT geometry model. However, content of the \textit{REG1} procedure may require modification in the dependence on the way the geometry model is built – imported from the ROOT file or created “in situ” inside the MARS15 application. Both use cases are supported by the subroutines available in the STREG FORTRAN module.

If a developer is fluent with C/C++ and ROOT, then the best way is to follow “in situ” scenario, which implies that the geometry model is built by means of the ROOT tools inside the function named \textit{tgeo\_init}, called from the \textit{REG1} subroutine. This use case is implemented in the \textit{REG1} subroutine supplied with the MARS15 distribution in the m1519.f file. The following set of rules should be followed by a developer of the \textit{tgeo\_init} function:

- The function needs to be implemented in a separate file located in an application directory. Usually, it is called \texttt{tgeo\_init.cc}.
- The file \texttt{tgeo\_init.cc} must be added to the list referenced by a variable \texttt{SRCS} of the GNUmakefile file.
- The function must have the “C” linkage, i.e. a definition of the function needs to be enclosed in the \texttt{extern “C”} block.
- \texttt{gGeoManager} object already exists when the function is called and can be immediately used inside the function.
- \texttt{TGeoMedia} objects used to fill the volumes are not constructed inside the function, but retrieved from the \texttt{gGeoManager} object by means of the \texttt{TGeoManager::GetMedium} method either by the number or name specified in the MARS input file (usually \texttt{MATER.INP}).

For the rest, the rules of the geometry model creation are defined by the C++ and ROOT syntax and semantics. Based on the typical needs which came from application developments, the two complementary extensions were added to the library based on the hooks provided by the classes from the ROOT libGeom library.

One of the extensions is aimed to provide the possibility to change the particle tracking algorithm used in the MARS15 code, for example, to refine the transport in magnetic field or implement transport in the RF cavity.

For this purpose, the function \textit{tgeo\_step1} implements a polymorphic code, which is based on calls to the virtual functions \texttt{IsApplicable} and \texttt{propagate} for objects belonging to a user-defined class derived from the \texttt{GenericStepper} which in turn inherits from the \texttt{TGeoExtension} class defined in the ROOT libGeom library, as shown in Figure 1. The inheritance relationship allows association between objects of classes derived from the \texttt{GenericStepper} class with the ROOT geometry objects – volumes and nodes.
The `tgeo_step1` ("make-a-step") function checks if the custom stepper is associated with the current node or volume. If it is, and call to a virtual member function `IsApplicable` for that stepper returns `true`, then the `propagate` method of the stepper attached to the node/volume is used to make a step in a node. If the custom stepper is not detected, then default steppers provided by the library are used. In the presence of the magnetic field the static object of the class `M15HelixStepper` is used to make the particle step, otherwise object of the class `StraightStepper` is used to propagate a particle.

For implementation of a new stepper class the following actions need to be performed:

- define a class derived from the abstract class `GenericStepper`;
- implement two functions `IsApplicable` and `propagate`.

The `IsApplicable` function should return `true`, if the algorithm implemented in the function `propagate` is applicable for a transport of the particle identified by `PartID` (see Figure 1) and having phase co-ordinates `vin={x, y, z, \Omega_x, \Omega_y, \Omega_z, p}` and time-of-flight `toff` (s); otherwise the function returns `false`. Components of the vector `vin` are `x,y,z` – the particle position in the global Cartesian application reference system (cm); `\Omega_x, \Omega_y, \Omega_z` – components of the unit vector, indicating the motion direction of the particle; `p` – the particle momentum (GeV/c).

The function `propagate` should make a step `s` for the particle with the given `charge` and `mass` starting at the phase point `vin` and the time-of-flight `toff`. From the diagram shown in Figure 1 one can see that the parameters `s` and `toff` are used for input and output. At the input, parameter `s` contains the step size requested by the MARS particle tracking module, a pilot step. If boundary crossing is detected by the function, the value of `s` should be replaced by the pathlength passed by the particle to the volume boundary. Array `vout` should be filled with the particle phase co-ordinates at the end of step `s`, a value referenced by `toff` should be replaced by the time-of-flight to that point. The function should return zero upon successful completion, otherwise it should return -1. The special return code -9050898 tells MARS, that the trajectory is supposed to be continued by the `propagate` function of the custom stepper and should be interrupted in the MARS15 code. That is the case when the particle, moving in the beam aperture, is passed to the MAD-X PTC module.
Object of the developed class can be declared then in the `tgeo_init` function and associated with a volume or node of the ROOT geometry model by calling `SetUserExtension` member-function of the `TGeoVolume` or `TGeoNode` class. After that, at runtime, when the particle moves in a volume with the associated custom stepper, the `propagate` method of that stepper will be used to make a step in the volume, but only if `IsApplicable` method of the stepper returns `true` for the given input parameters. Otherwise, the built-in MARS15 steppes are used to make a step.

The technique described allows setting up special propagation rules in the dedicated regions without affecting underlining the MARS15 code. In general, it can be used to simulate arbitrary quasi-continuous physical interactions which can be experienced by the particle on the path between discrete strong, weak and electromagnetic interactions modelled by MARS15. Examples include modelling coherent interactions in a bent crystal and impact of a gravitational force. The following stepper classes were developed:

- **MagFieldRK4Stepper** – a stepper in magnetic field, 4-order Runge-Kutta-Nyström solver;
- **BFieldRKStepper** – a stepper in magnetic field, 8-order Runge-Kutta solver;
- **CavityStepper** – a stepper in time-dependent electromagnetic field, 8-order Runge-Kutta solver.

**Figure 2. Class diagram for custom steppers and fields**

![Class diagram for custom steppers and fields](source)

Note that the applicability rules for the steppers designed to transport particle in the magnetic field require the presence of a magnetic field in the given volume. In Figure 2 this fact is indicated by the UML “dependency” (dashed line) connections between the stepper classes and the ROOT TVirtualMagField class. To satisfy the dependency, an object of a class derived from TVirtualMagField - FMapField, for example, must be associated with the volume, i.e. the object of the TGeoVolume class, using the `TGeoVolume::SetField` method. The developer needs to pay attention to the following facts: (1) ROOT allows the field association only with the TGeoVolume objects, there is no way to attach field to the geometry node (positioned volume); (2) the same TGeoVolume object can be referenced by multiple nodes; (3) all field classes defined in the STREG and other MARS15 libraries, contain the reference to the TGeoMatrix object, which is the argument of the field constructor function. The matrix defines transformation from the
global reference frame to the local frame used for the field definition. If the same volume
with the associated field is used in several nodes, the field inside all the nodes is calculated
in the frame specified during the field object construction. In other circumstances, this
behaviour can be considered either as a feature or a bug. The developer needs to be aware
of this.

The current STGEO library, initially developed for integration of the ROOT TGeo package
to MARS15, has the polymorphic implementation of the “make-a-step” task which makes
it possible to use a custom propagation algorithm for the certain particle types of a certain
phase volumes. Thereby, the library provides a straightforward way for coupling the
MARS15 code with other simulation codes.

**MARS–MAD-X coupling**

The tools described here are provided in the MARS MADX-BEAMLINE library and
include the stepper class based on the MAD-X PTC module and a geometry generator for
the sequence of elements described in the MAD-X file, the MAD-X-MARS15-ROOT
Beamline Builder (MRRBB). It is assumed that, ideally, the MAD-X input file is prepared
by an accelerator physicist and – besides a description of the beam and the element
sequence the beam is supposed to propagate through – contains the statements to create the
survey and Twiss tables. It is expected that the elements which have the same design are
declared in the MAD-X input file as belonging to the same MAD-X class. For example,
two classes of the Booster combined-function magnets are declared with the cross sections
shown in Figure 3.

**Figure 3. Geometry of the Booster combined-function magnets as implemented in the MARS15 model:**

- DMAG class, defocusing dipole (left) and FMAG class, focusing dipole (right)


There are 96 such dipoles in the Booster lattice which may have different magnetic fields
so the implementation of the geometry model would take a considerable time even for a
relatively small machine, like the Booster. The solution was found via MAD-MARS Beam
Line Builder (MMBLB) which builds a longitudinal beam-aligned structure of the MARS
geometry model using a MAD-generated optics file (Mokhov, 2000; Krivosheev, 2001;
The MMMLB was successfully used by MARS users worldwide for 15 years. Recently, its capabilities have been substantially extended by switching to the MAD-X-MARS15 ROOT Beamline Builder (MMRBB). The general idea is to construct for each of the MAD-X class from the input file a generator of the three-dimensional representation of elements of that class, aka “factory” of the elements. In terms of the ROOT Geom library, the generator returns the unique instance of the \textit{TGeoVolume} or \textit{TGeoVolumeAssembly} objects filled with the geometry models of the MAD-X element. The generated object is placed then into the parent volume using a corresponding transformation matrix from the MAD-X survey module for the given element. It means that the creation of a survey table is mandatory in the MAD-X input file used in concert with the MARS15 application.

A development of a generator for the three-dimensional representation of elements of the particular MAD-X class element is the application specific task. For the geometry generator implementation, a class derived from the C++ abstract class \textit{NodeGenerator} needs to be implemented by the application developer. The main operations of the class are presented in a class diagram shown in Figure 4.

\begin{figure}[h]
\centering
\includegraphics[width=\textwidth]{fig4}
\caption{Base abstract class for implementation of geometry generator}
\end{figure}

The core function \textit{Assemble} builds the element geometry model. This function is purely virtual in the base class, thus its definition in a derived class is mandatory. The meaning of the formal parameters is the following: \textit{Elinfo} is a reference to the \textit{Section} structure object containing complete information about the MAD-X element; \textit{SecVol} is the beamline section volume. It is bounded by the shapes of the \textit{TGeoTube} or \textit{TGeoCylinder} classes by...
default; FieldMtx – a transformation matrix which maps a global frame of the geometry model to a local frame used for definition of magnetic field in the element.

When implementing the Assemble method for the magnets, two objects of the TMADX_BField and FMapField classes (see Figure 2) are created by means of the C++ new operator and used then as input parameters of the TGeoVolume::SetField method to specify magnetic field in the magnet aperture and its body, respectively. The TMADX_BField class is based on an analytical representation of the field using the multipole coefficients defined in the magnet description in the MAD-X file. The parameter Elinfo from the argument list of the Assemble is passed to the constructor on creation of the TMADX_BField class object. The FMapField class is based on a tabular representation of the field. Building the class object, the developer needs to specify a reference to a table, often called the field map, which defines mapping between co-ordinates in a local reference system of a magnet to the components of the magnetic field vector. One more parameter passed to the constructor is a matrix which defines transformation of the local co-ordinates to the global frame of the model. A reverse matrix defines reverse transformation. The field map instance needs to belong to the class derived from the FMapField, and ReadFile and GetFile functions, which are pure virtual in the ancestor. Generally, these functions could be implemented for three-dimensional magnetic field distributions, but in most cases, a 2D hard-edge approximation is quite adequate. The class TFiledMap2D provides the ReadFile function which reads an ASCII file generated by the OPERA code and implements the quadratic interpolation in the GetField function. Creation of the MAD-X/PTC stepper can also be done in the Assemble function. This is as easy in implementation as allocation of an object by means of the C++ new operator. A constructor of the MADXStepper class has two parameters. The first one is the same as that used for building the TMADX_BField class object. The second one is an array of six values, which has the same meaning as MAXAPER parameter in the PTC_TRACK command and is used for the same purpose. In the MAD-X, it is an “array defining upper limits for particle co-ordinates, essentially defining the aperture to trigger particle loss”. It is also used in MARS15 as an acceptance window for the stepper. Association of the stepper object with the aperture volume is performed the same way as for the steppers described in the above sectionMARS STREG library”. The following needs to be implement to the tgeo_init function to build a beamline model:

- Call NodeGenerator::DefineWorld static method. The function sets up the volume specified in the argument as the container for entire beamline.

- Call NodeGenerator::SetRsec static method. The beamline is built of the cylindrical sections – the volumes bounded by the objects of the TGeoTube and TGeoCub classes. These sections supposed to be daughters of the volume used in the argument of the NodeGenerator::DefineWorld method. Each section contains the element geometry model created by the Assemble method. This is used to specify the outer radii for all the sections. The length of each the section corresponds to the length of the element. Reference to the section created by the beamline builder is passed to the Assemble method via the SecVol parameter. The number of sections created by the beamline builder is equal to the number of the non-zero length elements in the MAD-X file.

- Create objects of the developed classes derived from NodeGenerator in memory. As it follows from the diagram shown in Figure 4, a string referenced as Name must be passed to the constructor of the base class. The Name is the name of the MAD-X class. Every time the object of the derived class is created, the base class constructor automatically saves its pointer to the static map Vol indexed by Name. The Vol is used then to find the appropriate
geometry builder for the element of the MAD-X sequence. The memory allocation for the object of the class derived from NodeGenerator implicitly adds it to the map.

- Call int madx(char* fname) function provided by the MARS-MADX-Beamline library. The function initiates the MAD-X session which is started from the execution of the MAD-X file given as the argument. The geometry construction starts after completion of the file processing by the MAD-X interpreter. For each element of the MAD-X sequence, the function tries to retrieve the geometry generator object from the map generated in step 3, using the MAD-X class name of elements as the key. On success, the virtual functions of the object are used to construct the section and geometry model of the element. If the geometry generator specific to the MAD-X class of the element is not found, the function tries to find a geometry generator mapped to the “default” keyword. As a last resort, the function creates just a cylindrical section with a length of the MAD-X element with the outer radius from step 2 and placement retrieved from the survey table. In the case that the geometry generator mapped to the keyword “Tunnel” is provided by the model developer, the tunnel becomes the daughter volume for the section volume, and the deepest node of the tunnel serves as a container for the element geometry model.

The MAD-X session initiated at step 4 continues until completion of the run. It is correct to think that MAD-X and MARS run concurrently with the synchronous execution of the MAD-X statements given as arguments of the madx_stmt function, called from the MARS15 code. In particular, if objects of the MADXStepper class were associated with the aperture volumes in the Assemble methods of the geometry builder objects created at step 3, the MAD-X commands PTC_CREATE_UNIVERSE and PTC_CREATE_LAYOUT are issued from tgeo_init routine followed by the call of the PTCTrackingActive service function. The latter unlocks the steppers and provides the automatic execution of the MAD-X commands needed for the normal termination of the MAD-X PTC tracking and session itself upon completion of the MARS run. The library tools allow rapidly obtaining the working MARS15 model for the entire or any part of the beamline described in the MAD-X file and then gradually complicate the geometry and tracking models by means of the features provided by MARS15-MAD-X and ROOT. As a result, the same geometry model is used to transport particles in both MARS15 and MAD-X PTC parts. Beam particles assumed as lost in the MAD-X PTC module are taken care of by the MARS modules. If a particle is inside the acceptance of the MAD-X PTC tracking module, then the trajectory is passed for simulation to the MAD-X modules. An example of the application where cross-talk between two codes was used is given in the next section.

**New Booster collimation unit optimisation studies**

As described in the introduction, a proposal (Kapin, 2017b; Kapin, 2017c; Sidorov, 2017) – driven by limitations of the Booster physical aperture – calls for a replacement of the current two-stage collimation system with one or two “collimation units” made of a thick primary collimator followed at a short distance by a secondary collimator for the quasi-local absorption of the beam halo (Kapin, 2018). The newly-developed MARS15-MAD-X system was used for the optimisation modelling of the collimation efficiency and radiation environment around the proposed “collimation units” with preliminary results described in “Update on MARS studies of new Booster collimators” (Tropin, 2018).

The Booster lattice consists of 24 sections, also referred as periods. The period is represented in the MAD-X file as the line (Booster MADX files, 2016). To build the MARS model shown in Figure 5 (left), three lines (sequences) were included in the resulting sequence generated by the Booster MAD-X file: BCELO8, BCELO9, BCELO10. The total
length of the resulting sequence is 59.3 m. The entire beamline in the sequence is wrapped in a uniformly shaped tunnel shown in Figure 5 (right). The origin of the co-ordinate system in the model is at the entrance to the BCEL08 line. Each of the periods consists of four dipole magnets. Corrector magnets shown in Figure 6 are located at the upstream ends of the first and third dipoles in each of the periods. The long drift spaces between the second dipole and corrector in each of the periods are supposed to be used for installation of the new collimation unit.

Figure 5. MARS15 model of the Booster periods 8-10 with the collimator unit in the long straight section 8 (left) and tunnel cross-section (right)


A new drift class was added to the MAD-X file and the element of this class was inserted in the BCEL08 line with a corresponding geometry generator created as a placeholder for the collimation unit installed in other period(s). The model can easily be extended to the full ring by adding the period lines in the resulting sequence of the MAD-X file.
The MAD-X PTC steppers are attached to the aperture volumes. Beam halo trajectories generated by means of the steppers are shown in Figure 6. The trajectories are shown in the co-ordinate system defined with respect to the design orbit for the Booster periods 7 and 8. The opportunity to rapidly change the range of elements included in the MARS model via MAD-X file configuration was used here. Vertical lines show boundaries of the elements from the MAD-X file. Yellow regions are the dipole magnet poles. Purple regions are metal parts representing the beam pipe and collimation unit installed in the Booster period 8. The copper primary collimators are followed by the massive secondary ones made of stainless steel and assembled in a single module (unit). In this study, the primary collimator thickness is varied in the range \( T_0 \leq t \leq 16T_0 \), where \( T_0 = 1.016 \) cm. The jaws of the secondary movable collimator have a total length of 60.96 cm with the length of the flat part being 40.64 cm. The aperture of stationary stainless steel masks is 7.62 x 7.62 cm, with a total length of 45.72 cm and a flat part length of 25.4 cm. The unit is encapsulated in the steel shielding with overall size 60x60x400 cm (see details in Figure 9 below).

**Figure 6. Proton trajectories of the 3.0 to 3.9 \( \sigma \) range through the two Booster periods with the collimation unit implemented (beam co-ordinate system, side view)**


Contrary to the canonical two-stage approach used nowadays in multi-turn collimation systems (Mokhov, 2003b), the new Booster collimation unit is aimed at the single-pass beam halo shaving done at the proton injection energy of 400 MeV. The unit includes a relatively thick copper primary collimator, aimed to substantially increase a momentum spread of the scattered particles, and a massive secondary collimator(s), designed to intercept these particles in a local manner.

In this study, the proton spectra \( dN_{pa}(E, t) / dE \) inside the aperture at the exit from the collimation unit (see Figure 7) were simulated with the integrated MARS15–MAD-X system for the optimisation of the primary collimator thickness \( t \) with respect to the highest collimation efficiency. The beam halo hits the aisle-side jaw of the horizontal primary collimator. The opposite jaw is in the garage position. The vertical co-ordinates \( y_{mad} = \alpha_y y_{mad} + \beta_y p_y \) are sampled from the restricted Gauss distribution with zero mean value and standard deviation \( \sigma_y = 0.635 \) cm in such a way that \( \text{abs}(y_{mad}) < h \), where \( h \) is the half-height of the horizontal jaw. The horizontal co-ordinates are sampled from the uniform
distribution \( w(x_{mad}) = (b - a)^{-1} \), where \( a = 3\sigma_x = 1.126 \text{ cm} \) is the jaw opening, \( b = a + 1\text{ mm} \). For this study the conservative scraping rate \( N_a = 3.89 \times 10^{12} \text{ s}^{-1} \) was used, which is 5% of the total beam intensity after the upgrade planned for the Booster. Jaws of secondary collimators in the considered collimator unit – both vertical and horizontal – were aligned with the \( 3\sigma_{x,y} + 2\text{ mm} \) beam envelope. Figure 7 shows the calculated \( dN_{pa}(E, t)/dE \) spectra. For the primary collimator reference thickness \( t = T_0 \), the spectrum has a pronounced maximum at 380 MeV. For larger \( t \), the peak is shifted to lower energies of a decreased height. The peak disappears if the thickness exceeds a proton interaction length of 15 cm in copper.

One can now calculate the number of protons above a certain kinetic energy threshold \( E_{th} \) in the aperture at the collimation unit exit relative to the scraping rate \( N_a \), i.e. to the number of beam halo protons hitting the primary collimator jaw per second:

\[
N_{pa}(E_{th}, t) = \frac{1}{N_a} \int_{E_{th}}^{E_0=400 \text{ MeV}} dE \frac{dN_{pa}(E, t)}{dE}
\]  

This quantity can be referenced as a **collimation inefficiency**. Then the **collimation (absorption) efficiency** is \( \varepsilon = 1 - N_{pa} \). The collimation inefficiency is shown in Figure 8 as a function of the energy threshold \( E_{th} \) for the primary copper collimator thickness in the range \( 1 \leq t / T_0 \leq 16 \). For the relevant to the system performance threshold energies \( 0.32 \leq E_{th} \leq 0.38 \) GeV, the collimation (absorption) efficiency could be as high as \( \varepsilon = 0.9 \) at \( t = T_0 \), \( \varepsilon = 0.95 \) at \( t = 2T_0 \), and \( \varepsilon = 0.995 \) at \( t = 8T_0 \).
Figure 8. Collimation inefficiency as function of energy threshold for several thicknesses of the primary collimator jaws

Figure 9. Plan view of the MARS15 model of the collimation unit in the beam co-ordinate system with primary collimators (red), as well as tapered masks and movable secondary collimators (purple). The unit is surrounded by steel shielding (grey) encapsulated in marble (light grey).

A MARS15 model of the collimation unit is shown in Figure 9. The unit includes two horizontal primary collimators right (phr) and left (phl) of the beam axis, two vertical primary collimators above (pva) and below (pvb) of the beam axis, two fixed aperture masks (mask1 and mask2) along with movable horizontal (shmr and smhl) and vertical (svm) secondary collimators. The fractions of 250-W beam halo power deposited in the
The power deposition in the primary collimator active jaw (phr in this simulation) grows linearly with t in the range $1 \leq t/T_0 \leq 4$, saturating at larger thicknesses. That quantity at the exit of the collimation unit (mask2) decreases linearly with t in the entire range studied $1 \leq t/T_0 \leq 16$, consistent with the shown above increase of the collimation efficiency $\varepsilon = 1 - N_{phr}$. This effect is also clearly seen in Figure 10 for evolution of the hadron flux density above $E_{th} = 1$ MeV along the collimation unit and its substantial reduction at the downstream end of the unit for a thicker primary collimator. One can conclude here that the primary collimator with a thickness in the range $4 \leq t/T_0 \leq 8$, or roughly 4 to 8 cm of copper, provides the desirable increase of the collimation efficiency, with $t = 8T_0$ being a thickness of choice.

**Table 1. Fractions (%) of beam halo power 250 W deposited in the collimation unit jaws**

<table>
<thead>
<tr>
<th>$t/T_0$</th>
<th>phr</th>
<th>phl</th>
<th>mask1</th>
<th>shmr</th>
<th>shml</th>
<th>svm</th>
<th>mask2</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>5.53</td>
<td>0.014</td>
<td>20.04</td>
<td>13.2</td>
<td>20.1</td>
<td>8.72</td>
<td>0.3</td>
</tr>
<tr>
<td>4</td>
<td>21.6</td>
<td>0.21</td>
<td>20.8</td>
<td>4.48</td>
<td>5.95</td>
<td>1.03</td>
<td>0.07</td>
</tr>
<tr>
<td>16</td>
<td>54.0</td>
<td>2.0</td>
<td>9.27</td>
<td>0.15</td>
<td>3.71</td>
<td>0.65</td>
<td>0.02</td>
</tr>
</tbody>
</table>


**Figure 10. Hadron flux density above $E_{th} = 1$ MeV in the collimation unit for two thicknesses of the primary collimator $t = T_0$ (left) and $t = 8T_0$ (right)**


The hadron flux density above $E_{th} = 30$ MeV is often used for quick estimations of air activation in accelerator tunnels as well as ground and sump water activation of the soil outside the tunnel walls. The map of this quantity is shown in Figure 11 (left) in the collimation region and nearby downstream region. The total prompt dose map is shown in Figure 11 (right). The analysis has shown that with the optimised steel-marble shielding around the unit described above, the radiation field is adequately contained in the region with the prompt dose levels on the berm, residual dose rates on the shielding outside and radiation loads on the sump water being below the administrative limits with required
safety margin. Residual dose rates on the upstream ends of the hottest machine components (see Figure 12) are high – although quite typical for the Booster magnet faces – and may require implementation of steel masks in those regions.

Figure 11. Hadron flux density distribution above $E_{th} = 30$ MeV (plan view, left) and total prompt dose distribution (side view, right) in the tunnel and outside the walls for the collimation unit region and the first 40 m downstream it, $t = 8 T_o$ in both cases


Figure 12. Residual dose isocontours on contact at the first dipole (left) and corrector magnet (right) downstream of the collimation unit after 100-day irradiation and 4-hours cooling

Conclusions

The functionality of the STREG library of the MARS15 code which interfaced the ROOT geometry package with the MARS tracking engine has been extended through the polymorphic implementation of a “make-a-step” procedure. The polymorphism provides an opportunity to substitute the built-in MARS15 steppers by a user-defined algorithm without any changes in the main code. The approach was comprehensively tested and refined in implementation of the steppers for particle transport in magnetic and time-dependent electromagnetic fields. A stepper based on the MAD-X PTC module was implemented to the MARS15-MADX system making it even more powerful in accelerator and beamline applications.

The system described in this paper was developed, thoroughly tested and refined in the MARS15 code applications to the needs of the ESS, ILC, MAP and FCC projects. It has been successfully applied to the design optimisation of the new Fermilab Booster collimation system. The parameters of the introduced collimation unit have been optimised with respect to the highest collimation efficiency. It was found that the primary collimator thickness should be $4 \leq t/T_0 \leq 8$, or roughly 4 to 8 cm of copper, with 8 cm being preferable. The final decision here is to be made based on thermomechanical analyses. The collimation unit shielding was optimised to provide the tolerable radiation fields in the unit itself, downstream magnets, in and around the tunnel.

Acknowledgements

The authors are thankful to C. Bhat, J. Johnstone, V. Kapin, W. Pellico and V. Sidorov for the valuable input and discussions. This document was prepared using the resources of the Fermi National Accelerator Laboratory (Fermilab), a US Department of Energy, Office of Science, HEP User Facility. Fermilab is managed by Fermi Research Alliance, LLC (FRA), acting under Contract No. DE-AC02-07CH11359. It also used a Director’s Discretionary allocation at the Argonne Leadership Computing Facility, which is a DOE Office of Science User Facility supported under Contract DE-AC02-06CH11357.

References


Intercomparison of particle production

Hideo Hirayama\(^1\)\(^2\) and Toshiya Sanami\(^1\)

\(^1\)KEK, High Energy Accelerator Research Organization
\(^2\)Nuclear Regulation Authority

*hideo.hirayama@kek.jp

In accordance with the discussion at SATIF13, we propose the same intercomparison problems of neutron production from thick targets. There were large differences than our expectation in the intercomparison at SATIF-13 between Geant-4, MARS, MCNPX and PHITS. In this paper, we will present comparison between major Monte Carlo codes including FLUKA concerning neutron production with high-energy protons as desired at SATIF-13.

Introduction

Based on the discussion at SATIF-11, the problem of neutron production from thick targets of Al, Cu or Au for 1, 10 and 100 GeV protons induced reactions were prepared as the intercomparison for SATIF-12. There were large differences between the results than our expectation in the simple geometry calculations (Hirayama, 2015). At SATIF-13, the problem revised slightly including 0 and 180 degrees for spectral, angular integral neutron spectrum above 20 MeV and energy integral neutron fluence to find the reason of the differences. Relatively large differences existed in the comparison of angular spectrum at 0 degree especially for 1 GeV proton (Hirayama, 2016).

For SATIF-14, the same problems were sent to all contributors to summarise this intercomparison with included FLUKA and MCNP6.

Problems for an intercomparison at SATIF-14

Incident particle

Pencil beam of protons used with the following energies:

- (a) 1 GeV;
- (b) 10 GeV;
- (c) 100 GeV.

Target materials and their sizes:

The target geometry was a cylinder.

Source protons were incident on the centre of the cylinder bottom.

The target detector distance from the centre of the cylinder was 500 cm.

- (a) Al: length = 40 cm, diameter = 4.0 cm and density = 2.7 g/cm\(^3\);
- (b) Cu: length = 16 cm, diameter = 1.6 cm and density = 8.63 g/cm\(^3\);
(c) Au: length = 10 cm, diameter = 1.0 cm and density = 19.3 g/cm³.

**Quantities to be calculated**

As the quantities to be calculated, 0 degree was added together with angular integral neutron spectrum and energy integral neutron fluence above 20 MeV as follows:

- (1) Neutron spectrum above 20 MeV in n/MeV/sr/proton at 0, 15, 30, 45, 60, 90, 120, 150 degrees with angular width ±0.5 degrees;
- (2) Angular integral neutron spectrum above 20 MeV in n/MeV/cm² proton;
- (3) Energy integral neutron fluence for (1) and (2).

**Summary of contributors**

The results with FLUKA were sent to the organiser. Table 1 lists the participants, the names of the computer codes and the database and physical model used.

**Table 1. Summary of contributors**

<table>
<thead>
<tr>
<th>Name of participants and organizations</th>
<th>Code</th>
<th>Data Base</th>
<th>Physical model</th>
<th>Contributed at</th>
</tr>
</thead>
<tbody>
<tr>
<td>Norihiro MATSUDA (JAEA) and PHITS development team</td>
<td>PHITS (Sato T. et al. (2013), Niita K. et al (2006))</td>
<td>Original (PHITS)</td>
<td>INCL4.6(γ&lt;3 GeV)+JAM(γ&gt;3 GeV)+GEM</td>
<td>SATIF-13</td>
</tr>
<tr>
<td>Koi Tatsumi (SLAC)</td>
<td>Geant4 v10.02.p02 (use &quot;Shielding&quot; physics list) (S. Agostinelli et al. (2003), J. Allison et al., (2006, 2016))</td>
<td></td>
<td>BERT style cascade up to 50 GeV and from 4 GeV Fritof (FTF) string model</td>
<td>SATIF-13</td>
</tr>
<tr>
<td>Toshiya Sanami (KEK)</td>
<td>MARS 15, 16 (MARS (2019))</td>
<td>Original MARS</td>
<td>Default</td>
<td>SATIF-13</td>
</tr>
</tbody>
</table>


**Results and discussions**

**Comparison of neutron fluence above 20 MeV**

Comparison between codes are presented in the form of tables to show differences clearly. Table 2 shows total neutron fluence above 20 MeV. Maximum and minimum value in each case together with the ratio as (maximum value/minimum value). Difference of total neutron fluence above 20 MeV emitted from the target is within less than factor 2 between
Al, Cu and Au for protons from 1 GeV to 100 GeV. Differences for each target are as follows:

- Al: within 42 %, difference decreases with increase of proton energy;
- Cu: within 76 %, difference is almost same;
- Au: within 103 %, difference increases with increase of proton energy.

**Comparison of neutron energy fluence above 20 MeV**

Table 3 shows the comparison of total neutron energy fluence above 20 MeV for Al, Cu and Au, respectively. Difference of total neutron energy above 20 MeV emitted from the target is less than factor 2 between Al, Cu and Au for protons from 1 GeV to 100 GeV. Differences for each target are as follows:

- Al: within 46 %, difference is almost same with fluence;
- Cu: within 76 %, difference is almost same with fluence;
- Au: within 39 %, difference is less than half than that of fluence.

**Table 2. Total neutron fluence above 20 MeV (neutrons/proton)**

<table>
<thead>
<tr>
<th>Energy(GeV)</th>
<th>Geant4</th>
<th>PHITS</th>
<th>MCNPX</th>
<th>MARS</th>
<th>FLUKA</th>
<th>max/min</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Al target</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1</td>
<td>0.966</td>
<td>0.809</td>
<td>1.02</td>
<td>0.903</td>
<td>0.717</td>
<td>1.42</td>
</tr>
<tr>
<td>10</td>
<td>2.56</td>
<td>2.67</td>
<td>3.07</td>
<td>2.52</td>
<td>2.20</td>
<td>1.40</td>
</tr>
<tr>
<td>100</td>
<td>6.38</td>
<td>11.5</td>
<td>5.89</td>
<td>6.14</td>
<td>5.50</td>
<td>1.16</td>
</tr>
<tr>
<td></td>
<td>Cu target</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1</td>
<td>1.27</td>
<td>0.980</td>
<td>1.18</td>
<td>1.49</td>
<td>0.849</td>
<td>1.76</td>
</tr>
<tr>
<td>10</td>
<td>4.37</td>
<td>3.50</td>
<td>3.97</td>
<td>5.03</td>
<td>2.99</td>
<td>1.68</td>
</tr>
<tr>
<td>100</td>
<td>12.1</td>
<td>13.8</td>
<td>8.63</td>
<td>13.3</td>
<td>8.4</td>
<td>1.64</td>
</tr>
<tr>
<td></td>
<td>Au target</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1</td>
<td>1.47</td>
<td>1.33</td>
<td>1.49</td>
<td>1.36</td>
<td>1.21</td>
<td>1.24</td>
</tr>
<tr>
<td>10</td>
<td>9.93</td>
<td>6.06</td>
<td>5.39</td>
<td>5.65</td>
<td>5.28</td>
<td>1.88</td>
</tr>
<tr>
<td>100</td>
<td>27.9</td>
<td>21.6</td>
<td>13.7</td>
<td>15.3</td>
<td>17.1</td>
<td>2.03</td>
</tr>
</tbody>
</table>


Table 3. Total neutron energy fluence above 20 MeV (MeV/proton)

<table>
<thead>
<tr>
<th>Energy (GeV)</th>
<th>Geant4</th>
<th>PHITS</th>
<th>MCNPX</th>
<th>MARS</th>
<th>FLUKA</th>
<th>max/min</th>
</tr>
</thead>
<tbody>
<tr>
<td>Al target</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1</td>
<td>163</td>
<td>141</td>
<td>168</td>
<td>157</td>
<td>136</td>
<td>1.24</td>
</tr>
<tr>
<td>10</td>
<td>112</td>
<td>153</td>
<td>105</td>
<td>139</td>
<td>126</td>
<td>1.46</td>
</tr>
<tr>
<td>100</td>
<td>73</td>
<td>103</td>
<td>72</td>
<td>82</td>
<td>82</td>
<td>1.44</td>
</tr>
<tr>
<td>Cu target</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1</td>
<td>178</td>
<td>148</td>
<td>163</td>
<td>222</td>
<td>139</td>
<td>1.60</td>
</tr>
<tr>
<td>10</td>
<td>125</td>
<td>153</td>
<td>109</td>
<td>192</td>
<td>125</td>
<td>1.76</td>
</tr>
<tr>
<td>100</td>
<td>77</td>
<td>97</td>
<td>72</td>
<td>101</td>
<td>84</td>
<td>1.41</td>
</tr>
<tr>
<td>Au target</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1</td>
<td>172</td>
<td>158</td>
<td>161</td>
<td>162</td>
<td>149</td>
<td>1.16</td>
</tr>
<tr>
<td>10</td>
<td>155</td>
<td>157</td>
<td>118</td>
<td>147</td>
<td>128</td>
<td>1.33</td>
</tr>
<tr>
<td>100</td>
<td>84</td>
<td>92</td>
<td>73</td>
<td>66</td>
<td>87</td>
<td>1.39</td>
</tr>
</tbody>
</table>

Red: maximum value, Blue: minimum value.


Comparison of angular neutron fluence above 20 MeV

Tables 4-6 show the comparison of neutron angular fluence above 20 MeV for Al, Cu and Au, respectively. Differences at 0 degree for 1 GeV proton are larger than other cases. For 10 GeV and 100 GeV protons, differences at 0 degree are same or less compared with those for other angles. For 100 GeV, differences at 0 degree are smaller than other angles.
### Table 4. Neutron angular fluence above 20 MeV, Al target (neutrons/sr per proton)

<table>
<thead>
<tr>
<th>Angle</th>
<th>Geant4</th>
<th>PHITS</th>
<th>MCNPX</th>
<th>MARS</th>
<th>FLUKA</th>
<th>max/min</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>1 GeV proton on Al target</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>0</td>
<td>0.243</td>
<td>0.257</td>
<td>3.67</td>
<td>0.271</td>
<td>0.314</td>
<td>15.1</td>
</tr>
<tr>
<td>15</td>
<td>0.340</td>
<td>0.306</td>
<td>0.357</td>
<td>0.338</td>
<td>0.312</td>
<td>1.17</td>
</tr>
<tr>
<td>30</td>
<td>0.239</td>
<td>0.181</td>
<td>0.236</td>
<td>0.216</td>
<td>0.171</td>
<td>1.32</td>
</tr>
<tr>
<td>45</td>
<td>0.142</td>
<td>0.119</td>
<td>0.154</td>
<td>0.137</td>
<td>0.108</td>
<td>1.42</td>
</tr>
<tr>
<td>60</td>
<td>0.0965</td>
<td>0.0902</td>
<td>0.109</td>
<td>0.0991</td>
<td>0.0734</td>
<td>1.49</td>
</tr>
<tr>
<td>90</td>
<td>0.0442</td>
<td>0.0380</td>
<td>0.0489</td>
<td>0.0390</td>
<td>0.0297</td>
<td>1.49</td>
</tr>
<tr>
<td>120</td>
<td>0.0262</td>
<td>0.0178</td>
<td>0.0234</td>
<td>0.0204</td>
<td>0.0147</td>
<td>1.79</td>
</tr>
<tr>
<td>150</td>
<td>0.0198</td>
<td>0.0103</td>
<td>0.0155</td>
<td>0.0151</td>
<td>0.0103</td>
<td>1.92</td>
</tr>
<tr>
<td><strong>10 GeV proton on Al target</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>0</td>
<td>6.07</td>
<td>9.67</td>
<td>4.89</td>
<td>7.72</td>
<td>5.14</td>
<td>1.98</td>
</tr>
<tr>
<td>15</td>
<td>1.04</td>
<td>1.01</td>
<td>0.853</td>
<td>0.747</td>
<td>0.883</td>
<td>1.40</td>
</tr>
<tr>
<td>30</td>
<td>0.462</td>
<td>0.507</td>
<td>0.461</td>
<td>0.431</td>
<td>0.436</td>
<td>1.18</td>
</tr>
<tr>
<td>45</td>
<td>0.290</td>
<td>0.333</td>
<td>0.342</td>
<td>0.326</td>
<td>0.275</td>
<td>1.24</td>
</tr>
<tr>
<td>60</td>
<td>0.216</td>
<td>0.246</td>
<td>0.270</td>
<td>0.241</td>
<td>0.187</td>
<td>1.44</td>
</tr>
<tr>
<td>90</td>
<td>0.131</td>
<td>0.128</td>
<td>0.168</td>
<td>0.134</td>
<td>0.0999</td>
<td>1.68</td>
</tr>
<tr>
<td>120</td>
<td>0.0889</td>
<td>0.0733</td>
<td>0.142</td>
<td>0.0866</td>
<td>0.0672</td>
<td>2.11</td>
</tr>
<tr>
<td>150</td>
<td>0.0681</td>
<td>0.0532</td>
<td>0.126</td>
<td>0.0683</td>
<td>0.0543</td>
<td>2.38</td>
</tr>
<tr>
<td><strong>100 GeV proton on Al target</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>0</td>
<td>178</td>
<td>205</td>
<td>173</td>
<td>226</td>
<td>228</td>
<td>1.32</td>
</tr>
<tr>
<td>15</td>
<td>2.30</td>
<td>3.98</td>
<td>1.58</td>
<td>1.64</td>
<td>2.03</td>
<td>2.51</td>
</tr>
<tr>
<td>30</td>
<td>1.11</td>
<td>2.16</td>
<td>0.922</td>
<td>1.07</td>
<td>1.09</td>
<td>2.34</td>
</tr>
<tr>
<td>45</td>
<td>0.720</td>
<td>1.47</td>
<td>0.661</td>
<td>0.792</td>
<td>0.674</td>
<td>2.22</td>
</tr>
<tr>
<td>60</td>
<td>0.535</td>
<td>1.09</td>
<td>0.499</td>
<td>0.590</td>
<td>0.465</td>
<td>2.34</td>
</tr>
<tr>
<td>90</td>
<td>0.326</td>
<td>0.595</td>
<td>0.306</td>
<td>0.336</td>
<td>0.258</td>
<td>2.31</td>
</tr>
<tr>
<td>120</td>
<td>0.229</td>
<td>0.366</td>
<td>0.270</td>
<td>0.228</td>
<td>0.178</td>
<td>2.06</td>
</tr>
<tr>
<td>150</td>
<td>0.182</td>
<td>0.276</td>
<td>0.246</td>
<td>0.188</td>
<td>0.145</td>
<td>1.90</td>
</tr>
</tbody>
</table>

**Red**: maximum value, **Blue**: minimum value.  
Table 5. Neutron angular fluence above 20 MeV, Cu target (neutrons/sr per proton)

<table>
<thead>
<tr>
<th>Angle</th>
<th>Geant4</th>
<th>PHITS</th>
<th>MCNPX</th>
<th>MARS</th>
<th>FLUKA</th>
<th>max/min</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>0.239</td>
<td>0.237</td>
<td>1.71</td>
<td>0.229</td>
<td>0.278</td>
<td>7.45</td>
</tr>
<tr>
<td>15</td>
<td>0.351</td>
<td>0.298</td>
<td>0.343</td>
<td>0.302</td>
<td>0.299</td>
<td>1.18</td>
</tr>
<tr>
<td>30</td>
<td>0.268</td>
<td>0.199</td>
<td>0.248</td>
<td>0.215</td>
<td>0.187</td>
<td>1.43</td>
</tr>
<tr>
<td>45</td>
<td>0.181</td>
<td>0.142</td>
<td>0.177</td>
<td>0.149</td>
<td>0.126</td>
<td>1.43</td>
</tr>
<tr>
<td>60</td>
<td>0.131</td>
<td>0.111</td>
<td>0.130</td>
<td>0.110</td>
<td>0.0885</td>
<td>1.48</td>
</tr>
<tr>
<td>90</td>
<td>0.0689</td>
<td>0.0536</td>
<td>0.0647</td>
<td>0.0493</td>
<td>0.0411</td>
<td>1.67</td>
</tr>
<tr>
<td>120</td>
<td>0.0435</td>
<td>0.0270</td>
<td>0.0332</td>
<td>0.0274</td>
<td>0.0229</td>
<td>1.90</td>
</tr>
<tr>
<td>150</td>
<td>0.0327</td>
<td>0.0160</td>
<td>0.0218</td>
<td>0.0202</td>
<td>0.0168</td>
<td>2.04</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Angle</th>
<th>Geant4</th>
<th>PHITS</th>
<th>MCNPX</th>
<th>MARS</th>
<th>FLUKA</th>
<th>max/min</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>5.29</td>
<td>8.38</td>
<td>4.58</td>
<td>6.28</td>
<td>4.49</td>
<td>1.87</td>
</tr>
<tr>
<td>15</td>
<td>1.36</td>
<td>1.18</td>
<td>1.02</td>
<td>0.789</td>
<td>0.974</td>
<td>1.50</td>
</tr>
<tr>
<td>30</td>
<td>0.703</td>
<td>0.642</td>
<td>0.589</td>
<td>0.528</td>
<td>0.544</td>
<td>1.33</td>
</tr>
<tr>
<td>45</td>
<td>0.495</td>
<td>0.439</td>
<td>0.449</td>
<td>0.425</td>
<td>0.370</td>
<td>1.34</td>
</tr>
<tr>
<td>60</td>
<td>0.393</td>
<td>0.333</td>
<td>0.351</td>
<td>0.331</td>
<td>0.268</td>
<td>1.47</td>
</tr>
<tr>
<td>90</td>
<td>0.264</td>
<td>0.186</td>
<td>0.224</td>
<td>0.202</td>
<td>0.160</td>
<td>1.65</td>
</tr>
<tr>
<td>120</td>
<td>0.191</td>
<td>0.112</td>
<td>0.196</td>
<td>0.137</td>
<td>0.115</td>
<td>1.75</td>
</tr>
<tr>
<td>150</td>
<td>0.149</td>
<td>0.0828</td>
<td>0.173</td>
<td>0.109</td>
<td>0.0950</td>
<td>1.81</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Angle</th>
<th>Geant4</th>
<th>PHITS</th>
<th>MCNPX</th>
<th>MARS</th>
<th>FLUKA</th>
<th>max/min</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>155</td>
<td>179</td>
<td>161</td>
<td>161</td>
<td>219</td>
<td>1.41</td>
</tr>
<tr>
<td>15</td>
<td>3.28</td>
<td>4.26</td>
<td>2.14</td>
<td>1.93</td>
<td>2.49</td>
<td>2.21</td>
</tr>
<tr>
<td>30</td>
<td>1.89</td>
<td>2.44</td>
<td>1.33</td>
<td>1.41</td>
<td>1.50</td>
<td>1.83</td>
</tr>
<tr>
<td>45</td>
<td>1.36</td>
<td>1.72</td>
<td>0.968</td>
<td>1.11</td>
<td>1.02</td>
<td>1.78</td>
</tr>
<tr>
<td>60</td>
<td>1.08</td>
<td>1.32</td>
<td>0.750</td>
<td>0.873</td>
<td>0.748</td>
<td>1.76</td>
</tr>
<tr>
<td>90</td>
<td>0.736</td>
<td>0.771</td>
<td>0.478</td>
<td>0.543</td>
<td>0.461</td>
<td>1.67</td>
</tr>
<tr>
<td>120</td>
<td>0.551</td>
<td>0.492</td>
<td>0.431</td>
<td>0.386</td>
<td>0.339</td>
<td>1.63</td>
</tr>
<tr>
<td>150</td>
<td>0.433</td>
<td>0.374</td>
<td>0.388</td>
<td>0.320</td>
<td>0.281</td>
<td>1.54</td>
</tr>
</tbody>
</table>

Red: maximum value, Blue: minimum value

Table 6. Neutron angular fluence above 20 MeV, Au target (neutrons/sr per proton)

<table>
<thead>
<tr>
<th>Angle</th>
<th>Geant4</th>
<th>PHITS</th>
<th>MCNPX</th>
<th>MARS</th>
<th>FLUKA</th>
<th>max/min</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>0.222</td>
<td>0.230</td>
<td>0.717</td>
<td>0.148</td>
<td>0.235</td>
<td>4.83</td>
</tr>
<tr>
<td>15</td>
<td>0.337</td>
<td>0.300</td>
<td>0.323</td>
<td>0.213</td>
<td>0.302</td>
<td>1.58</td>
</tr>
<tr>
<td>30</td>
<td>0.278</td>
<td>0.232</td>
<td>0.262</td>
<td>0.172</td>
<td>0.226</td>
<td>1.61</td>
</tr>
<tr>
<td>45</td>
<td>0.210</td>
<td>0.184</td>
<td>0.208</td>
<td>0.132</td>
<td>0.169</td>
<td>1.60</td>
</tr>
<tr>
<td>60</td>
<td>0.160</td>
<td>0.151</td>
<td>0.164</td>
<td>0.110</td>
<td>0.127</td>
<td>1.49</td>
</tr>
<tr>
<td>90</td>
<td>0.0892</td>
<td>0.0853</td>
<td>0.0946</td>
<td>0.0523</td>
<td>0.0701</td>
<td>1.81</td>
</tr>
<tr>
<td>120</td>
<td>0.0547</td>
<td>0.0479</td>
<td>0.0560</td>
<td>0.0311</td>
<td>0.0452</td>
<td>1.80</td>
</tr>
<tr>
<td>150</td>
<td>0.0394</td>
<td>0.0299</td>
<td>0.0406</td>
<td>0.0230</td>
<td>0.0356</td>
<td>1.76</td>
</tr>
<tr>
<td>0</td>
<td>4.26</td>
<td>6.76</td>
<td>4.51</td>
<td>4.08</td>
<td>3.62</td>
<td>1.86</td>
</tr>
<tr>
<td>15</td>
<td>2.08</td>
<td>1.56</td>
<td>1.19</td>
<td>0.680</td>
<td>1.19</td>
<td>3.06</td>
</tr>
<tr>
<td>30</td>
<td>1.40</td>
<td>1.02</td>
<td>0.749</td>
<td>0.538</td>
<td>0.810</td>
<td>2.60</td>
</tr>
<tr>
<td>45</td>
<td>1.10</td>
<td>0.761</td>
<td>0.590</td>
<td>0.464</td>
<td>0.618</td>
<td>2.37</td>
</tr>
<tr>
<td>60</td>
<td>0.921</td>
<td>0.604</td>
<td>0.476</td>
<td>0.332</td>
<td>0.490</td>
<td>2.78</td>
</tr>
<tr>
<td>90</td>
<td>0.684</td>
<td>0.374</td>
<td>0.322</td>
<td>0.251</td>
<td>0.336</td>
<td>2.72</td>
</tr>
<tr>
<td>120</td>
<td>0.542</td>
<td>0.245</td>
<td>0.293</td>
<td>0.176</td>
<td>0.264</td>
<td>3.09</td>
</tr>
<tr>
<td>150</td>
<td>0.426</td>
<td>0.186</td>
<td>0.267</td>
<td>0.140</td>
<td>0.224</td>
<td>3.04</td>
</tr>
<tr>
<td>0</td>
<td>125</td>
<td>140</td>
<td>143</td>
<td>90.7</td>
<td>201</td>
<td>2.21</td>
</tr>
<tr>
<td>15</td>
<td>5.67</td>
<td>5.35</td>
<td>2.87</td>
<td>1.76</td>
<td>3.60</td>
<td>3.22</td>
</tr>
<tr>
<td>30</td>
<td>3.86</td>
<td>3.49</td>
<td>1.88</td>
<td>1.45</td>
<td>2.56</td>
<td>2.66</td>
</tr>
<tr>
<td>45</td>
<td>3.07</td>
<td>2.65</td>
<td>1.46</td>
<td>1.23</td>
<td>1.96</td>
<td>2.49</td>
</tr>
<tr>
<td>60</td>
<td>2.57</td>
<td>2.12</td>
<td>1.19</td>
<td>1.02</td>
<td>1.57</td>
<td>2.52</td>
</tr>
<tr>
<td>90</td>
<td>1.91</td>
<td>1.36</td>
<td>0.846</td>
<td>0.683</td>
<td>1.11</td>
<td>2.80</td>
</tr>
<tr>
<td>120</td>
<td>1.52</td>
<td>0.924</td>
<td>0.767</td>
<td>0.500</td>
<td>0.879</td>
<td>3.04</td>
</tr>
<tr>
<td>150</td>
<td>1.19</td>
<td>0.713</td>
<td>0.704</td>
<td>0.414</td>
<td>0.749</td>
<td>2.88</td>
</tr>
</tbody>
</table>

Red: maximum value, Blue: minimum value.

Comparison of neutron spectra above 20 MeV at 0 degree

Figures 1-3 show the comparison of neutron angular spectra at 0 degree for Al, Cu and Au, respectively. Vertical scale of all figures is three digits to see the differences clearly. There are large differences between codes especially for 1 GeV proton.

Comparison of neutron spectra above 20 MeV at 150 degrees

Figures 4-6 show the comparison of neutron angular spectra at 150 degrees for Al, Cu and Au, respectively. Vertical scale of all figures is three digits to see the differences clearly. Spectra shape are similar between codes compared with those at 0 degree.
Figure 1. Neutron spectra from the Al target at 0 degree

Al 1 GeV, 0 degree

Al 10 GeV, 0 degree

Au 100 GeV, 0 degree

Figure 2. Neutron spectra from the Cu target at 0 degree

Figure 3. Neutron spectra from the Cu target at 0 degree

Figure 4. Neutron spectra from the Al target at 150 degrees

Figure 5. Neutron spectra from the Cu target at 150 degrees

Figure 6. Neutron spectra from the Au target at 150 degrees

Summary

From the comparisons shown the above, the following results can be summarised: differences between code are less than factor 2 for total neutron and total neutron energy above 20 MeV emitted from target and relatively large differences exist in the comparison of angular spectrum especially at 0 degree.

Future themes

It is desired to check models especially for angular distributions by code developers and reflect to this comparison. Experimental results are necessary for small angle like 0 degree to check models used in each code. It is desired to include results with MCNP6.

References


Measurements and FLUKA simulations of bismuth, aluminium, indium and carbon activation at the upgraded CERN Shielding Benchmark Facility (CSBF)

E. Iliopoulou1,2,*, P. Bamidis1, M. Brugger3, R. Froeschl2, A. Infantino2, T. Kajimoto4, N. Nakao4, S. Roesler2, T. Sanami5, A. Siountas1, H. Yashima6

1Medical Physics Laboratory, School of Medicine, Aristotle University of Thessaloniki, Greece,
2CERN,
3Hiroshima University,
4Shimizu Corporation,
5KEK,
6Kyoto University
*elpida.iliopoulou@cern.ch

The CERN High Energy AcceleRator Mixed Field (CHARM) Facility is situated in the CERN Proton Synchrotron (PS) East Experimental Area. The facility receives a pulsed proton beam from the CERN PS with a beam momentum of 24 GeV/c with 5E11 protons per pulse with a pulse length of 350 ms and with a maximum average beam intensity of 6.7E10 protons per second. The extracted proton beam impacts on a cylindrical copper target.

The shielding of the CHARM facility includes the CERN Shielding Benchmark Facility (CSBF) situated laterally above the target that allows deep shielding penetration benchmark studies of various shielding materials. This facility has been significantly upgraded during the extended technical stop at the beginning of 2016. It consists now of 40 cm of cast iron shielding, a 200 cm long removable sample holder concrete block with 3 inserts for activation samples, a material test location that is used for the measurement of the attenuation length for different shielding materials as well as for sample activation at different thicknesses of the shielding materials.

In total 60 activation samples, of bismuth, aluminium, indium and carbon, were placed in the CSBF in various configurations of the facility in July 2016, September 2016 and August 2017 to characterise the upgraded version of the CSBF and to benchmark the FLUKA Monte Carlo simulation code with respect to the predictions of high energy neutron fluence behind thick shielding. The samples were placed at the removable sample holder concrete block at 10.5 cm, 85.4 cm, 160.35 cm and 200 cm height, respectively. Samples were placed also at the material test location with standard concrete at 0 cm, 80 cm and 160 cm material thickness and with cast iron at 20 cm and 40 cm material thickness. Monte Carlo simulations with the FLUKA code have been performed to estimate the specific production yields of bismuth isotopes (206Bi, 205Bi, 204Bi, 203Bi, 202Bi, 201Bi) from 209Bi, 24Na from 27Al, 115mI from 115I and 11C from C for these samples at the configurations used.

The production yields estimated by FLUKA Monte Carlo simulations are compared to the production yields obtained from γ-spectroscopy measurements of the samples taking the beam intensity profile into account. The agreement between FLUKA predictions and γ-spectroscopy measurements for the production yields is at a level of a factor of 2, confirming the reliability of the FLUKA simulations for the characterisation of the CSBF. This demonstrates that FLUKA is a very suitable tool for deep shielding penetration studies.
Introduction

The CERN High Energy AcceleRator Mixed Field Facility (denoted CHARM) has been constructed in the CERN Proton Synchrotron (PS) East Experimental Area in 2014 (Mekki, 2016). The facility receives a pulsed proton beam from the CERN PS with a beam momentum of 24 GeV/c with 5E11 protons per pulse with a pulse length of 350 ms and with a maximum average beam intensity of 6.7E10 p/s.

The extracted proton beam from the PS impacts on a cylindrical copper or aluminium target and the created secondary radiation field is used to test electronics equipment installed at predefined test positions.

The shielding of the CHARM facility (Froeschl, 2014) also includes the CERN Shielding Benchmark Facility (CSBF) situated laterally above the target (Froeschl, 2014). This facility allows deep-penetration benchmark studies of various shielding materials (Adorisio et al., 2011; Agosteo et al., 2001; Agosteo et al., 2013; Nakao et al., 2008). The CHARM facility at beam line level is illustrated in Figure 1 indicating the direction of the beam coming from the Proton Synchrotron (PS), the CHARM target, the target alcove for storing the target during access and the movable shielding walls.

Figure 1. Horizontal integration drawing of the CHARM facility at beam line level

The CSBF is located laterally above the CHARM target.

Based on the experience gained during the activation campaign in 2015 (Iliopoulou et al., 2015; Iliopoulou et al., 2016; Iliopoulou et al., 2018), the CSBF has been significantly upgraded during the extended year-end technical stop at the beginning of 2016. It consists now of 40 cm of cast iron shielding and up to 400 cm of concrete. The new configuration is composed of a 200 cm long removable sample holder concrete block with 3 slots for activation samples, a material test location that is used for the measurement of the attenuation length for different shielding materials as well as for sample activation at different thicknesses of the shielding materials and a dedicated platform for measuring deep penetration neutron spectra with active and passive detectors, covered by two barite concrete blocks. The CSBF layout under normal operation of the CHARM facility is indicated in the Figure 2.
In July 2016, September 2016 and August 2017 there were three activation campaigns with activation of bismuth, aluminium, indium and carbon samples that were placed in the CSBF to characterise the upgraded version of the CSBF. Monte Carlo simulations with the FLUKA code (Böhlen et al., 2014; Fassò et al., 2005) have been performed to estimate the specific production yields of bismuth isotopes ($^{206}$Bi, $^{205}$Bi, $^{204}$Bi, $^{203}$Bi, $^{202}$Bi, $^{201}$Bi) from $^{209}$Bi, $^{24}$Na from $^{27}$Al, $^{115m}$I from $^{115}$I and $^{13}$C from C for these samples in the upgraded configuration and have then been compared to the production yields obtained from $\gamma$-spectroscopy measurements of the samples taking the beam intensity profile into account.

**Design of the CERN Shielding Benchmark Facility (CSBF)**

The CERN Shielding Benchmark Facility (CSBF) has been significantly upgraded during the extended technical stop at the beginning of 2016. The CSBF upgrade allows for easier manipulation and for having more exploitation possibilities of the facility (Iliopoulos et al., 2018; Froeschl et al., 2016). The design of the upgraded CSBF was based on FLUKA simulations.

During the operational periods of 2016 and 2017, the CSBF consisted of 40 cm cast iron shielding, 400 cm of standard concrete, barite concrete and cast iron shielding that are part of the three main possible configurations of the CSBF. These three main possible configurations allow measurement at the removable sample holder concrete block (which is also the nominal configuration of the facility during the nominal CHARM facility operation), on the CSBF platform and in the shielding material test location.

**Removable sample holder concrete block**

The removable sample holder concrete block was needed for the facilitation of the handling procedure of the activation samples or passive dosimeters in order to place them deep inside the CSBF shielding and irradiate them. For this reason, the removable sample holder concrete block provides 3 slots of 10 cm x 10 cm cross section that are centred along the vertical axis of the block, so that they can be filled with the samples. In Figure 2 the recent layout of CSBF is presented when the removable sample holder concrete block is inserted. The position 1 is located at a height of 10.5 cm, measured from the bottom of the removable sample holder concrete block, the position 2 at 85.4 cm height and the position 3 at 160.35 cm height. There is also a possibility of placing samples on the top of the block, mentioned as position 4 at 200 cm height. The block is easily inserted in and extracted from the CSBF shielding in a specifically designed shaft, with dimensions 40 cm x 40 cm x 240 cm, by means of a remote-controlled hook suspended from the crane. The neutron spectra predicted by the FLUKA Monte Carlo code at the four positions of the removable sample holder concrete block are shown in the Figure 3.
Figure 2. CSBF upgrade layout for measurements with the removable sample holder concrete block inserted in the facility


Figure 3. Neutron fluence spectra predicted by FLUKA at the four different positions on the removable sample holder concrete block for an average beam intensity of 6.7E10 protons per second

**CSBF platform**

The CSBF platform was created at 560 cm above beam line level, indicated in Figure 4, and allows placing active detectors or dosimeters attached to phantoms on the top of the shielding, for measuring their response to a deep penetration neutron spectrum. In order to use the platform, two barite concrete blocks of 120 cm height have to be removed. The CSBF platform measurements can be performed in parallel to activation measurements at the removable sample holder concrete block activation measurements.

**Figure 4. CSBF upgrade layout for the CSBF platform usage**

![CSBF Upgrade Layout](image)

The CSBF platform is indicated with the thin green line in the figure on the left.


**Shielding material test location**

The shielding material test location was designed for measuring the spectrum averaged attenuation length of various shielding materials (e.g. standard concrete, barite concrete, hematite concrete, colemanite concrete, magnetite concrete and cast iron). In addition, the shielding material test location can be used to place activation samples within different materials and thicknesses. The current layout of CSBF when the facility is in the shielding material test location mode, is presented in Figure 5.
Figure 5. CSBF upgrade layout for the shielding material test location


Beam parameters and configurations

This section presents the beam parameters and the facility configurations that were used during the activation experiments in July 2016, September 2016 and in August 2017. The beam intensity was measured with a Secondary Emission Chamber, whose measurement values are logged in the CERN measurement database. An intensity calibration factor was applied to the counts per pulse to obtain the number of protons per pulse. This calibration factor had been previously obtained with aluminium foil activation measurements using sodium isotopes with a statistical uncertainty of 7% of the γ-spectrometry analysis (Curioni et al., 2017).

A beam size of 1.2 cm x 1.2 cm Full Width at Half Maximum (FWHM) was used for the FLUKA simulations as specified in the layout of the beam line and confirmed by online beam profile measurements (Curioni et al., 2017).

The average beam intensity of CHARM, binned in 5 minutes long intervals, from July 6 to July 12 2016, from September 16 to September 22 2016 and from August 23 to August 30 2017 when the experiments were conducted, is shown in Figure 6, Figure 7 and Figure 8 respectively.

The beam passes through the upstream Proton Irradiation facility (IRRAD) before impacting on the CHARM target. During the period of the experiment, silicon samples with a total thickness of 0.2 cm were placed into the beam in IRRAD and these samples were also taken properly into account in the FLUKA simulations.

The chemical composition of the concrete and the cast iron implemented in the FLUKA Monte Carlo simulations for the shielding with their respective densities are listed in Table 1 and in Table 2.

During the activation experiment, the copper target of 8 cm diameter and 50 cm length has been used inside the CHARM facility. Inside the target room, there are four movable
shielding walls, each of 20 cm thickness and made out of concrete or iron. They can be placed between the target and the irradiation positions for electronics components inside the CHARM facility in different combinations, so that the irradiation spectra are adjusted to the desired radiation field (energy and intensity) during the tests. The movable shielding walls are also indicated in Figure 1. For this activation experiment, only one configuration of the four movable shielding walls was used during the different irradiation periods, namely all movable shielding walls retracted from the facility. The configuration has been properly taken into account in the FLUKA Monte Carlo simulations.

Table 1. Chemical composition (Nakao et al., 2008) and density of concrete

<table>
<thead>
<tr>
<th>Element</th>
<th>Weight fraction %</th>
<th>Element</th>
<th>Weight fraction %</th>
</tr>
</thead>
<tbody>
<tr>
<td>Hydrogen</td>
<td>0.561</td>
<td>Silicon</td>
<td>16.175</td>
</tr>
<tr>
<td>Carbon</td>
<td>4.377</td>
<td>Sulfur</td>
<td>0.414</td>
</tr>
<tr>
<td>Oxygen</td>
<td>48.204</td>
<td>Potassium</td>
<td>0.833</td>
</tr>
<tr>
<td>Sodium</td>
<td>0.446</td>
<td>Calcium</td>
<td>23.929</td>
</tr>
<tr>
<td>Magnesium</td>
<td>1.512</td>
<td>Titanium</td>
<td>0.173</td>
</tr>
<tr>
<td>Aluminium</td>
<td>2.113</td>
<td>Iron</td>
<td>1.263</td>
</tr>
</tbody>
</table>


Table 2. Chemical composition (Lazzaroni, 2015) and density of cast iron (Nakao et al., 2008)

<table>
<thead>
<tr>
<th>Cast iron</th>
<th>Density 7.2 g/cm³</th>
</tr>
</thead>
<tbody>
<tr>
<td>Element</td>
<td>Weight fraction %</td>
</tr>
<tr>
<td>Iron</td>
<td>92.3</td>
</tr>
<tr>
<td>Carbon</td>
<td>3.85</td>
</tr>
<tr>
<td>Manganese</td>
<td>0.3</td>
</tr>
<tr>
<td>Silicon</td>
<td>3.4</td>
</tr>
<tr>
<td>Phosphorus</td>
<td>0.08</td>
</tr>
<tr>
<td>Sulfur</td>
<td>0.02</td>
</tr>
<tr>
<td>Cobalt</td>
<td>0.05</td>
</tr>
</tbody>
</table>

Figure 6. Average beam intensity of the CHARM facility during the activation experiments in July 2016 binned in 5 minutes long intervals


Figure 7. Average beam intensity of the CHARM facility during the activation experiments in September 2016 binned in 5 minutes long intervals

Figure 8. Average beam intensity of the CHARM facility during the activation experiments in August 2017 binned in 5 minutes long intervals


Activation samples and their irradiation

Sixty disk samples in total, bismuth, aluminium, indium and carbon samples have been irradiated. The samples were placed in the removable sample holder concrete block of the CSBF at the four positions as indicated in Figure 2. Samples were placed also at the shielding material test location, as indicated in Figure 5, with standard concrete at 0 cm, 80 cm and 160 cm material thickness and with cast iron at 20 cm and 40 cm material thickness.

Comparison of FLUKA simulation results to measured production yields

The simulation results were obtained by first scoring the neutron fluence spectra with FLUKA. Then, the neutron fluence was folded with cross section data for the bismuth isotopes, $^{24}$Na, $^{115m}$I and $^{11}$C (Maekawa et al., 2001), shown in Figure 9, to obtain the predicted production yields per atom per primary proton on the target.

The activities of the bismuth isotopes, $^{24}$Na, $^{115m}$I and $^{11}$C were measured for the bismuth, aluminium, indium and carbon samples respectively using γ-spectrometry, sometimes even at different cool-down times. In case of multiple samples for the same materials at the same position or multiple γ-spectrometry measurements of the same sample, the activities selected were the ones with the lowest uncertainty of the γ-spectrometry measurements. These activities have been converted to the production yields by taking into account the corresponding irradiation profiles with 5 minutes long binning and the corresponding cool-down times.

The production yields predicted by FLUKA and measured by γ-spectrometry are presented in Figures 10, 11, 12, 13 and in Tables 4 and 5. The agreement between FLUKA predictions and γ-spectrometry measurements for the production yields is generally better than a factor of 2.
The contributions that have been taken into account for the uncertainty estimation are shown in Table 6. The uncertainty of the beam size has negligible impact on the results as verified by FLUKA simulations. The materials placed in IRRAD during the period of the experiment were taken into account in the simulations. The uncertainty of the production yields coming from the uncertainty of the materials placed in IRRAD is far below 1%. A hypothetical change in the concrete density would provoke a change on the slope of the dependence of the yields on the depth of the shielding plotted in Figures 10, 11, 12, 13 and the effect of the change would increase with increased shielding thickness.

The cumulative distribution to the production yields as function of the neutron energy is presented in Figure 14 for the sample placed at a concrete shielding thickness of 160.35 cm in the CSBF. From this figure, it can be seen that for sodium-24, the neutron energy range contributing to the production yield is quite large whereas for the bismuth isotopes the energy ranges are narrower and located around the respective cross section peaks. The 10%, 25%, 75% and 90% quantiles of the production yield distribution for the various radionuclides, are presented in Table 3, quantifying this effect.

Figure 9. Production cross-sections of the bismuth isotopes, $^{24}\text{Na}$, $^{115m}\text{I}$ and $^{11}\text{C}$ as a function of the neutron energy (Maekawa et al., 2001)

Figure 10. Comparison of the predicted production yields by FLUKA and the measured production yields by $\gamma$-spectrometry for bismuth, sodium and indium radionuclides as function of shielding thickness, at the removable sample holder concrete block in 2016


Figure 11. Comparison of the predicted production yields by FLUKA and the measured production yields by $\gamma$-spectrometry for bismuth, sodium and indium radionuclides as function of shielding thickness of concrete, at shielding material test location in 2016

Figure 12. Comparison of the predicted production yields by FLUKA and the measured production yields by $\gamma$-spectrometry for sodium and carbon radionuclides as function of shielding thickness of cast iron, at removable sample holder concrete block in 2017


Figure 13. Comparison of the predicted production yields by FLUKA and the measured production yields by $\gamma$-spectrometry for bismuth, sodium and indium radionuclides as function of shielding thickness, at shielding material test location in 2017

Figure 14. Cumulative contribution to the production yields at a concrete shielding thickness of 160.35 cm as a function of the neutron energy

![Cumulative contribution to the production yields at a concrete shielding thickness of 160.35 cm as a function of the neutron energy](image)


Table 3. The 10%, 25%, 75% and 90% quantiles of the production yield distribution for the various radionuclides at a concrete shielding thickness of 160.35 cm

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>Neutron energy (MeV)</th>
<th>Q_{0.1}</th>
<th>Q_{0.25}</th>
<th>Q_{0.75}</th>
<th>Q_{0.9}</th>
</tr>
</thead>
<tbody>
<tr>
<td>Bi-201</td>
<td></td>
<td>79.9</td>
<td>87.8</td>
<td>128</td>
<td>173</td>
</tr>
<tr>
<td>Bi-202</td>
<td></td>
<td>70.6</td>
<td>77.4</td>
<td>114</td>
<td>149</td>
</tr>
<tr>
<td>Bi-203</td>
<td></td>
<td>59.7</td>
<td>66.1</td>
<td>104</td>
<td>141</td>
</tr>
<tr>
<td>Bi-204</td>
<td></td>
<td>48.3</td>
<td>54.4</td>
<td>89.4</td>
<td>126</td>
</tr>
<tr>
<td>Bi-205</td>
<td></td>
<td>38.2</td>
<td>42.5</td>
<td>70.2</td>
<td>110</td>
</tr>
<tr>
<td>Bi-206</td>
<td></td>
<td>29.5</td>
<td>32.7</td>
<td>61.5</td>
<td>100</td>
</tr>
<tr>
<td>Na-24</td>
<td></td>
<td>9.39</td>
<td>12.2</td>
<td>79.7</td>
<td>122</td>
</tr>
<tr>
<td>In-115m</td>
<td></td>
<td>1.63</td>
<td>2.33</td>
<td>7.99</td>
<td>59.7</td>
</tr>
</tbody>
</table>

NEA/NSC/R(2021)2

 263

Table 4. Predicted production yields by FLUKA and measured production yields by γ-spectrometry for the
removable sample holder block and the shielding material test location in 2016
Radionuclide
Bi-206

Bi-205

Bi-204

Bi-203

Bi-202

Bi-201

Na-24

In-115m

Position/
Height
(cm)
1 / 10.5
2 / 85.4
3 /160.35
4 / 200
0
80
160
1 / 10.5
2 / 85.4
3 /160.35
4 / 200
0
80
160
1 / 10.5
2 / 85.4
3 /160.35
4 / 200
0
80
160
1 / 10.5
2 / 85.4
3 /160.35
4 / 200
0
80
160
1 / 10.5
2 / 85.4
3 /160.35
4 / 200
0
80
160
1 / 10.5
2 / 85.4
3 /160.35
4 / 200
0
80
160
1 / 10.5
2 / 85.4
3 /160.35
4 / 200
0
80
160
1 / 10.5
2 / 85.4
3 /160.35
4 / 200
0
80
160

Predicted production
yield by FLUKA
(number/atoms/p)
1.34E-031
1.13E-032
1.60E-033
5.48E-034
7.31E-033
1.11E-033
1.41E-034
1.26E-031
1.03E-032
1.53E-033
5.89E-034
7.22E-033
1.17E-033
1.37E-034
8.99E-032
8.14E-033
1.17E-033
4.62E-034
5.96E-033
9.36E-034
1.07E-034
7.08E-032
7.42E-033
1.02E-033
3.93E-034
6.06E-033
8.55E-034
9.54E-035
4.76E-032
5.44E-033
7.33E-034
2.92E-034
4.47E-033
6.56E-034
7.20E-035
3.41E-032
4.05E-033
5.61E-034
2.26E-034
3.61E-033
5.34E-034
5.94E-035
1.71E-032
1.52E-033
2.28E-034
9.16E-035
1.15E-033
1.56E-034
1.96E-035
2.26E-031
1.40E-032
2.22E-033
8.26E-034
5.72E-034
3.50E-035
4.18E-036


Uncertainties
FLUKA
(%)
±4.01
±5.91
±9.71
±11.55
±10.76
±5.71
±9.51
±4.00
±5.65
±9.40
±11.46
±10.93
±5.84
±9.32
±3.94
±5.53
±9.31
±11.45
±10.01
±5.95
±9.21
±4.11
±5.58
±9.36
±11.07
±10.47
±6.16
±9.21
±4.47
±5.71
±9.68
±11.28
±10.85
±6.74
±9.26
±4.64
±5.78
±9.92
±11.33
±10.99
±7.39
±9.50
±3.30
±5.52
±9.23
±11.19
±9.60
±5.58
±9.23
±2.49
±5.09
±9.04
±10.58
±5.16
±5.23
±8.76

Measured
production yield
(number/atoms/p)
1.76E-031
1.82E-032
2.15E-033
9.91E-034
1.39E-032
2.77E-033
3.13E-034
1.83E-031
1.75E-032
2.36E-033
9.20E-034
1.60E-032
2.55E-033
4.38E-034
9.78E-032
1.35E-032
1.38E-033
7.11E-034
9.23E-033
1.82E-033
2.02E-034
1.00E-031
1.18E-032
1.44E-033
6.79E-034
1.08E-032
1.68E-033
1.79E-034
6.87E-032
1.17E-032
1.09E-033
5.45E-034
9.42E-033
1.62E-033
1.55E-034
5.45E-032
4.85E-033
1.01E-033
2.14E-034
5.42E-033
9.23E-034
1.67E-034
2.98E-032
2.10E-033
2.65E-034
1.13E-034
2.12E-033
3.56E-034
3.63E-035
3.87E-031
2.19E-032
3.13E-033
1.21E-033
1.80E-032
2.16E-033
3.04E-034

Uncertainties
Measurement
(%)
±7.14
±7.25
±7.50
±7.25
±7.31
±7.20
±8.84
±7.58
±7.92
±8.72
±7.96
±7.66
±8.06
±11.56
±7.25
±7.20
±7.50
±7.20
±7.10
±7.10
±8.06
±8.78
±8.54
±9.76
±8.38
±7.54
±7.83
±14.59
±8.01
±9.09
±11.32
±10.41
±7.78
±7.62
±12.70
±13.72
±31.88
±25.77
±44.16
±14.41
±12.54
±22.80
±7.14
±7.47
±7.31
±7.96
±7.25
±7.07
±11.64
±10.04
±7.12
±9.90
±8.32
±9.69
±7.54
±10.78

Ratio
Prediction/
Measurement
0.76
0.62
0.74
0.55
0.53
0.40
0.45
0.69
0.59
0.65
0.64
0.45
0.46
0.31
0.92
0.60
0.85
0.65
0.65
0.51
0.53
0.71
0.63
0.71
0.58
0.56
0.51
0.53
0.69
0.47
0.67
0.54
0.47
0.41
0.46
0.63
0.83
0.56
1.06
0.67
0.58
0.36
0.57
0.72
0.86
0.81
0.54
0.44
0.54
0.58
0.64
0.71
0.68
0.67
0.60
0.51

Uncertainties
ratio
(%)
±8.19
±9.35
±12.27
±13.64
±13.01
±9.19
±12.98
±8.57
±9.73
±12.82
±13.96
±13.34
±9.96
±14.85
±8.26
±9.08
±11.96
±13.53
±12.27
±9.27
±12.24
±9.69
±10.20
±13.52
±13.88
±12.90
±9.96
±17.25
±9.18
±10.74
±14.89
±15.34
±13.36
±10.17
±15.72
±14.48
±32.40
±27.61
±45.59
±18.13
±14.55
±24.70
±7.86
±9.29
±11.78
±13.74
±12.03
±9.01
±14.85
±10.35
±8.75
±13.40
±13.46
±10.98
±9.17
±13.89


Table 5. Predicted production yields by FLUKA and measured production yields by γ-spectrometry for the removable sample holder block and the shielding material test location with cast iron in 2017

<table>
<thead>
<tr>
<th>Radio nuclide</th>
<th>Position /Height (cm)</th>
<th>Predicted production yield by FLUKA (number/atoms/p)</th>
<th>Uncertainties FLUKA (%)</th>
<th>Measured production yield (number/atoms/p)</th>
<th>Uncertainties Measurement (%)</th>
<th>Ratio Prediction/ Measurement</th>
<th>Uncertainties ratio (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Bi-206</td>
<td>20</td>
<td>4.72E-033</td>
<td>±4.77</td>
<td>1.11E-032</td>
<td>±1.74</td>
<td>0.43</td>
<td>±0.98</td>
</tr>
<tr>
<td></td>
<td>40</td>
<td>1.31E-033</td>
<td>±4.30</td>
<td>3.79E-033</td>
<td>±1.58</td>
<td>0.35</td>
<td>±0.87</td>
</tr>
<tr>
<td>Bi-205</td>
<td>20</td>
<td>4.39E-033</td>
<td>±4.27</td>
<td>9.73E-033</td>
<td>±1.21</td>
<td>0.85</td>
<td>±1.44</td>
</tr>
<tr>
<td></td>
<td>40</td>
<td>1.24E-033</td>
<td>±4.33</td>
<td>3.10E-033</td>
<td>±1.09</td>
<td>0.80</td>
<td>±1.17</td>
</tr>
<tr>
<td>Bi-204</td>
<td>20</td>
<td>3.29E-033</td>
<td>±4.44</td>
<td>6.71E-033</td>
<td>±2.06</td>
<td>0.90</td>
<td>±1.32</td>
</tr>
<tr>
<td></td>
<td>40</td>
<td>9.18E-034</td>
<td>±4.26</td>
<td>2.19E-033</td>
<td>±1.78</td>
<td>0.90</td>
<td>±1.17</td>
</tr>
<tr>
<td>Bi-203</td>
<td>20</td>
<td>2.87E-033</td>
<td>±4.92</td>
<td>5.49E-033</td>
<td>±1.50</td>
<td>0.96</td>
<td>±1.25</td>
</tr>
<tr>
<td></td>
<td>40</td>
<td>8.15E-034</td>
<td>±4.50</td>
<td>1.77E-033</td>
<td>±0.89</td>
<td>0.90</td>
<td>±0.96</td>
</tr>
<tr>
<td>Bi-202</td>
<td>20</td>
<td>2.15E-033</td>
<td>±5.74</td>
<td>4.92E-033</td>
<td>±1.48</td>
<td>0.90</td>
<td>±0.96</td>
</tr>
<tr>
<td></td>
<td>40</td>
<td>5.98E-034</td>
<td>±4.77</td>
<td>1.63E-033</td>
<td>±2.01</td>
<td>0.90</td>
<td>±1.07</td>
</tr>
<tr>
<td>Bi-201</td>
<td>20</td>
<td>1.79E-033</td>
<td>±5.47</td>
<td>2.81E-033</td>
<td>±1.97</td>
<td>0.90</td>
<td>±1.18</td>
</tr>
<tr>
<td></td>
<td>40</td>
<td>4.56E-034</td>
<td>±5.28</td>
<td>8.14E-034</td>
<td>±1.19</td>
<td>0.90</td>
<td>±1.07</td>
</tr>
<tr>
<td>Na-24</td>
<td>1 / 10.5</td>
<td>1.71E-032</td>
<td>±3.30</td>
<td>2.98E-032</td>
<td>±1.76</td>
<td>0.90</td>
<td>±1.98</td>
</tr>
<tr>
<td></td>
<td>2 / 85.4</td>
<td>1.52E-033</td>
<td>±5.52</td>
<td>2.43E-033</td>
<td>±3.37</td>
<td>0.63</td>
<td>±2.90</td>
</tr>
<tr>
<td></td>
<td>3 / 160.35</td>
<td>2.28E-034</td>
<td>±9.23</td>
<td>3.48E-034</td>
<td>±7.78</td>
<td>0.86</td>
<td>±12.08</td>
</tr>
<tr>
<td></td>
<td>4 / 200</td>
<td>9.16E-035</td>
<td>±11.19</td>
<td>1.25E-034</td>
<td>±5.22</td>
<td>0.73</td>
<td>±13.89</td>
</tr>
<tr>
<td></td>
<td>20</td>
<td>6.77E-034</td>
<td>±3.64</td>
<td>1.07E-033</td>
<td>±7.14</td>
<td>0.90</td>
<td>±7.97</td>
</tr>
<tr>
<td></td>
<td>40</td>
<td>1.87E-034</td>
<td>±3.85</td>
<td>3.54E-034</td>
<td>±7.87</td>
<td>0.86</td>
<td>±7.60</td>
</tr>
<tr>
<td>In-115m</td>
<td>20</td>
<td>2.08E-033</td>
<td>±2.63</td>
<td>1.75E-032</td>
<td>±1.70</td>
<td>0.90</td>
<td>±7.52</td>
</tr>
<tr>
<td></td>
<td>40</td>
<td>2.41E-033</td>
<td>±3.20</td>
<td>5.72E-033</td>
<td>±1.12</td>
<td>0.90</td>
<td>±7.90</td>
</tr>
<tr>
<td>C-11</td>
<td>1 / 10.5</td>
<td>1.38E-032</td>
<td>±3.17</td>
<td>1.69E-032</td>
<td>±1.70</td>
<td>0.90</td>
<td>±7.80</td>
</tr>
<tr>
<td></td>
<td>2 / 85.4</td>
<td>1.44E-033</td>
<td>±5.35</td>
<td>1.61E-033</td>
<td>±1.70</td>
<td>0.90</td>
<td>±7.97</td>
</tr>
<tr>
<td></td>
<td>3 / 160.35</td>
<td>2.01E-034</td>
<td>±9.03</td>
<td>2.54E-034</td>
<td>±1.54</td>
<td>0.70</td>
<td>±12.43</td>
</tr>
<tr>
<td></td>
<td>4 / 200</td>
<td>8.01E-035</td>
<td>±10.89</td>
<td>1.01E-034</td>
<td>±15.99</td>
<td>0.70</td>
<td>±18.21</td>
</tr>
<tr>
<td></td>
<td>20</td>
<td>6.36E-034</td>
<td>±5.26</td>
<td>7.49E-034</td>
<td>±17.59</td>
<td>0.82</td>
<td>±9.97</td>
</tr>
<tr>
<td></td>
<td>40</td>
<td>1.68E-034</td>
<td>±4.77</td>
<td>2.20E-034</td>
<td>±8.06</td>
<td>0.77</td>
<td>±9.37</td>
</tr>
</tbody>
</table>


Table 6. Uncertainties taken into account for the uncertainty estimation of the production yields

<table>
<thead>
<tr>
<th>Source of uncertainty</th>
<th>Uncertainty on production yield</th>
</tr>
</thead>
<tbody>
<tr>
<td>Simulations</td>
<td>statistical*</td>
</tr>
<tr>
<td></td>
<td>concrete density¹</td>
</tr>
<tr>
<td></td>
<td>cast iron density¹</td>
</tr>
<tr>
<td></td>
<td>0.5 – 10%²</td>
</tr>
<tr>
<td></td>
<td>2%</td>
</tr>
<tr>
<td>Measurements</td>
<td>γ – spectrometry*</td>
</tr>
<tr>
<td></td>
<td>sample mass</td>
</tr>
<tr>
<td></td>
<td>beam intensity (calibration)</td>
</tr>
<tr>
<td></td>
<td>beam intensity (statistical)*</td>
</tr>
<tr>
<td></td>
<td>beam momentum</td>
</tr>
<tr>
<td></td>
<td>beam position and profile</td>
</tr>
<tr>
<td></td>
<td>target density</td>
</tr>
<tr>
<td></td>
<td>target dimensions</td>
</tr>
</tbody>
</table>

*Statistical uncertainty.

¹Uncertainty of the concrete density is 0.05 g/cm³.

²The concrete density uncertainty leads to an uncertainty of the production yield of 0.5% for 10.5 cm of concrete and of up to 10% for 200 cm of concrete.

³Except for 201Bi at 85.4 cm, 160.35 cm, 200 cm, see Table 4.

Summary

The CERN High Energy AcceleRator Mixed Field (CHARM) Facility has been constructed in the CERN PS East Experimental Area. The facility receives a pulsed proton beam from the CERN PS with a beam momentum of 24 GeV/c with 5E11 protons per pulse with a pulse length of 350 ms. The maximum average beam intensity is 6.7E10 p/s.

The shielding of the CHARM facility also includes the CERN Shielding Benchmark Facility (CSBF) situated laterally above the target. This facility allows deep-penetration benchmark studies of various shielding materials.

From the experience gained through a previous activation campaign in July 2015, we decided to upgrade the CSBF in 2016 in order to facilitate the procedure of sample placement and to add more functionalities in the facility. Based on the results mentioned above (Iliopoulou et al., 2015; Iliopoulou et al., 2016), FLUKA was used for the design of the upgrade in CSBF as a reliable Monte Carlo simulation tool.

Several activation foil experiments have been conducted at the upgraded CSBF in 2016 and 2017. In total 60 bismuth, aluminium, indium and carbon cylindrical samples were placed in the removable sample holder concrete block of the CSBF and at the shielding material test location at different heights. The production rates computed from the activities of the irradiated samples measured by $\gamma$-spectrometry have been compared to the estimated production rates from FLUKA Monte Carlo simulations. The agreement is at a level of a factor of 2.

This agreement is good for deep shielding penetration studies and is consistent with previous similar studies at the CERN-EU High Energy Reference Field facility (CERF) (Nakao et al., 2008) and demonstrates that FLUKA is a very suitable tool for deep shielding penetration studies.

Acknowledgements

We thank our colleagues from the IRRAD and CHARM operation teams for their support and for providing beam time and the CERN Experimental Area group for the integration and construction of the CSBF. We would also like to show our gratitude to our colleagues from several Japanese institutes for the excellent collaboration during the experiments. Moreover, we are thankful to the CERN transport group for their assistance. We are also grateful to our colleagues from the CERN $\gamma$-spectrometry laboratory for their support.

References


Monte Carlo simulations of ISIS Target Station 1 using FLUKA code and comparison with the MCNPX reference model

Lina Quintieri*,1, Steven Lilley1
1ISIS Neutron and Muon Source, STFC, Rutherford Appleton Laboratory, Didcot OX11 0QX, United Kingdom
*linaquintieri@stfc.ac.uk

The 800 MeV proton beam from the ISIS synchrotron is serving two target stations: Target Station 1 (TS1) which was up and running by 1987 and Target Station 2 (TS2) following in 2008. The TS1 target is a twelve-plate tantalum-clad tungsten target, irradiated by a 40-pulses-per-second proton beam (average beam current is approximately 160 µA). The neutrons produced in spallation reactions are slowed to energies useful for neutron scattering experiments by moderators (surrounded by a beryllium reflector) which are designed to meet the instrument requirements for neutron flux and resolution. The target, reflector and moderators (TRAM) of ISIS Target Station 1 is currently being redesigned to increase maintainability, extend the life of the target station and (if possible) provide an increase in useful neutron output. The new TRAM design has been accurately reproduced with the FLUKA code based on the latest engineering drawings. The FLUKA model and simulation results (neutron production, energy deposition profile, particle fluence energy spectrum, radionuclide production etc.) will be presented and discussed. Furthermore, several relevant quantities are compared to the corresponding values obtained by the MCNPX reference model for the equivalent TRAM module. The calculations results show generally good agreement for values such as energy deposited although there are, as it will be shown, a few interesting differences.

Introduction

ISIS is one of the world’s most powerful spallation neutron sources for the study of material structures and dynamics. Currently ISIS has two spallation targets, TS1 operating at proton beam powers of up to 200 kW, and TS2 operating to 45 kW.

ISIS TS1 (ISIS, 2019) has been operating for 30 years and serves 19 neutron scattering instruments. The target, reflector and moderators (TRAM) of ISIS TS1 are currently being redesigned to increase maintainability, extend the life of the target station and possibly provide an increase in useful neutron output. The new TRAM module will be installed in 2020-2021 and it is expected to start operating in late 2021 (Gallimore, 2018).

The new target design has mainly aimed at reducing the mass of those materials that can be considered “parasitic” with respect to the neutron production (i.e. required for support but that could affect negatively the neutron economy). The total mass of the target is actually decreased by about 60% (from 153.8 kg for the current target to 61 kg for the new one) but keeping almost unchanged the tungsten mass (from 47.3 kg for the current target to 46.0 kg for the new one), where most useful neutrons are produced. In Figure 1 a 3D rendering of the current target and of the upgraded one are reported. Above the target sit two light water moderator assemblies consisting of aluminium cans, Boral de-coupler (Boral is a mix of boron carbide and aluminium alloy) and Gd poison foils. One of the
water moderators is centrally poisoned, the other is asymmetrically poisoned. Below the target sit two light water pre-moderators, below which the cryogenic moderators are located. One cryogenic moderator is a liquid methane moderator with two poison foils, the other is a liquid hydrogen filled moderator. All of these components are located inside a sintered solid beryllium reflector which is edge cooled. Further information on the upgraded target can be found in “Design and optimisation of the ISIS TS1 Project target” (Wilcox and Jones, 2018) and the upgraded reflector and moderators in “ISIS Target Station One Upgrade Project – An overview of the development work being undertaken to improve the Target, Reflector and Moderator (TRaM) support systems” (Coates, 2018).

Neutronics calculations are essential to design the TRAM components to maximise useful neutron output but also to ensure the TRAM can be operated safely. Several key engineering quantities such as heat deposition and decay heat need to be calculated to be used as inputs to engineering analysis. The baseline model of ISIS TS1 and of the updated TS1 TRAM have be previously simulated in MCNPX (MCNPX, 2010; Škoro et al., 2017), however in this work, FLUKA (Ferrari, 2005) has been used to provide a cross-check and increase our understanding of the uncertainties in the simulation outputs.

**Figure 1. TS1 target: current and upgraded version**

![TS1 target: current and upgraded version](image)


**FLUKA model of the upgraded target, reflector and moderator for ISIS TS1**

**Geometry and material**

Figure 2. shows the FLUKA (Ferrari, 2005) model of the upgraded TS1-TRAM. The model has been built in such a way to be able to have a direct estimation of the average neutron track-length in each TRAM component or sub-assembly whose radionuclide inventory could be actually monitored and measured during either maintenance campaigns or at the end the operation life. This required to implement the beryllium reflector module according the “brick by brick” approach, following as close as reasonable the engineering mechanical drawings of each component.
In order to assess the geometric accuracy of the model, we used as figure of merit the comparison of the volume estimations for all the components made of tungsten, tantalum, stainless steel (SS) and beryllium. For all those components, we found a difference of less than 3% between the CAD volume and the FLUKA volume. The volumes have been calculated in the FLUKA model using a stochastic volume estimation method. The FLUKA 2011.2.x.3-gcc-8.1.0 version has been used for the results here discussed.

**Activated physics options**

In addition to the “Precision” default physics options (Fassò, 2005), the following other options have been activated:

- evaporation (on) → Evaporation from residual nuclei activated with the new heavy fragmentation model;
- coalescence (on) → Coalescence mechanism activated in order to have an accurate description of high energetic light fragments emission. This option has to be put on when it is desired to describe accurately pions, muons and kaons decay;
- pairbrem (on) → Heavy ion e+/e- pair production;
- ionsplit (on) → Ions splitting into nucleons;
- photonuclear (on) → Photonuclear physics over all the energy range;
- heavy ion transport (on) → with the lower energy threshold set at 1 keV;
- biasing for photoneutron production, based on the reduction of the photon inelastic interaction length;
- pion decay activated;
- muon decay activated.
Simulated scenarios

Several simulations have been carried out to calculate secondary particle yield, escaping fluence rate and energy spectrum together with the energy deposition for several selected scenarios. The results presented here refer to the “nominal beam-on” and the “irradiation reference profile” scenario. In the nominal case, only the prompt radiation generated as result of the 800 MeV proton beam hitting the target is simulated, while in the second one, the decay heat, the activity and the radionuclide inventory in the whole TRAM assembly are estimated at several cooling times after the shutdown.

The primary beam characteristics used for all the calculations are summarised in Table 1, while the irradiation profile to whom the results discussed in this paper refer is composed of five “beam-on/beam-off” cycles as reported in Table 2.

Table 1. Primary beam characteristics

<table>
<thead>
<tr>
<th>Proton Beam</th>
<th>Energy profile</th>
<th>Spatial profile</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Energy profile</td>
<td>Spatial profile</td>
</tr>
<tr>
<td>Distribution</td>
<td>Gaussian</td>
<td>Distribution</td>
</tr>
<tr>
<td>Mean energy</td>
<td>800 MeV</td>
<td>Divergence in V</td>
</tr>
<tr>
<td>Energy spread</td>
<td>±0.4%</td>
<td>Divergence in H</td>
</tr>
<tr>
<td>FWHM</td>
<td>7.4 MeV</td>
<td>Angular divergence</td>
</tr>
</tbody>
</table>


Table 2. Irradiation cycles

<table>
<thead>
<tr>
<th>Irradiation History</th>
<th>Status</th>
<th>Time [s]</th>
<th>Current [( \mu \text{A} )]</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Beam-on</td>
<td>1.9e7</td>
<td>200</td>
</tr>
<tr>
<td></td>
<td>Beam-off</td>
<td>4.2e7</td>
<td>0</td>
</tr>
<tr>
<td></td>
<td>Beam-on</td>
<td>1.4e7</td>
<td>200</td>
</tr>
<tr>
<td></td>
<td>Beam-off</td>
<td>4.2e7</td>
<td>0</td>
</tr>
<tr>
<td></td>
<td>Beam-on</td>
<td>1.5e7</td>
<td>200</td>
</tr>
</tbody>
</table>


Results

Neutron and secondary particles yield

Secondary particle generation and their leakage into the vacuum vessel (estimated at 1 m from the geometrical target centre) have been calculated in two configurations:

- the bare target (referred in the following as “bare target” case);
- target embedded in the whole reflector/moderator assembly (referred in the following as “TRAM” case).
The estimated rate of secondary particles produced in direct spallation process or in inelastic reactions with high energy neutrons (E > 20 MeV), in the case of 200 μA primary proton current, is reported in Figure 3. While the neutron production is approximately equal in bare target and TRAM case, in the latter the amount of $^3$He, $^4$He, triton and deuteron increases considerably mainly due to the contribution of inelastic reactions inside the beryllium reflector.

**Figure 3. Secondary particles produced in direct spallation**

![Bar chart showing secondary particle production rates in bare target and TRAM cases.](image)


The values reported in Figure 3 do not take into account the contribution from nuclear reactions initiated by neutrons with kinetic energy lower than 20 MeV. If, in the TRAM case, secondaries generated by low energy neutrons are also taken into account, the average proton generation rate increases from 2.14 to 15.5 proton per primary, since additional 13.4 protons are generated by low energy neutrons. As a consequence, in case of 200 μA current with 800 MeV primary proton, the secondary proton source rate becomes comparable to the neutron one (~ 2E+16 s⁻¹). Similarly, the deuteron production rate in the TRAM case is almost doubled, when the low neutron energy contribution is considered as well.

The net neutron yield, defined as the “production -absorption” balance of neutrons in each region, is detailed in Table 3, where the contribution from the different material is specified for each macro-region of the TRAM module (target, reflector, moderators, poisons, auxiliary supports). The main reduction to the escaping neutron current happens in the Boral layer where an average of seven neutrons per primary proton are absorbed.
### Table 3. Neutron yield* per primary proton

<table>
<thead>
<tr>
<th>Target</th>
<th>800 MeV proton</th>
<th>TRAM [n/p]</th>
</tr>
</thead>
<tbody>
<tr>
<td>Ta</td>
<td>0.81±3%</td>
<td></td>
</tr>
<tr>
<td>W</td>
<td>11.14±1%</td>
<td></td>
</tr>
<tr>
<td>Be</td>
<td>1.43±1%</td>
<td></td>
</tr>
<tr>
<td>SS</td>
<td>-0.28±2%</td>
<td></td>
</tr>
<tr>
<td>Ni (thermocouples)</td>
<td>0.003±19%</td>
<td></td>
</tr>
<tr>
<td>MixWater**</td>
<td>0.023±5%</td>
<td></td>
</tr>
<tr>
<td>Boral</td>
<td>-7.07±1%</td>
<td></td>
</tr>
<tr>
<td>Gd</td>
<td>-0.28±1%</td>
<td></td>
</tr>
<tr>
<td>Liquid CH4</td>
<td>-0.02±3%</td>
<td></td>
</tr>
<tr>
<td>Liquid H2</td>
<td>-0.002±8%</td>
<td></td>
</tr>
<tr>
<td>Water</td>
<td>-0.09±2%</td>
<td></td>
</tr>
<tr>
<td>Al alloys</td>
<td>-0.10±2%</td>
<td></td>
</tr>
<tr>
<td>Ti alloys</td>
<td>-0.17±1%</td>
<td></td>
</tr>
<tr>
<td>SS</td>
<td>-0.10±2%</td>
<td></td>
</tr>
<tr>
<td>Total net balance</td>
<td>~5.3±1%</td>
<td></td>
</tr>
</tbody>
</table>

*The Neutron balance is the net contribution "+production -absorption" of neutrons in each region.

**MixWater is made of 80% D$_2$O + 20% H$_2$O.


The energy spectrum of the neutrons escaping from the TRAM towards the vacuum vessel has been evaluated and compared to the equivalent calculation in the bare target case (as one way exiting fluence, scored on a spherical surface of 1 m radius from the target centre). In Figure 4 the two spectra are compared:

- in the case of *bare* target, the total energy spectrum of the escaping neutrons is a convolution of the spectra of the generated neutrons in each plate, relatively unaffected by the SS vessel and cooling materials around;
- in the TRAM case, the thermal peak (around 25 meV) appears at the expense of the 0.7 MeV evaporation peak that is depopulated as a consequence of the slowing-down of the produced neutrons (mainly in the reflector and moderators) and of the neutron capture reactions, occurring in the TRAM due to multiple parasitic absorptions.
The net counting of the escaping neutrons per proton reduces from 14.2 in case of the bare target to 5.3 in case of the whole TRAM, and among these, only approximately 1 n/primary (that is less than 18% of the total neutron leakage) has energy lower than 1 eV.

The maximum rate of neutrons escaping from the TRAM defines the ISIS TS1 source strength and according the FLUKA simulations, it is equal to $7 \times 10^{15}$ n/s, for a proton beam power equal to 160 kW (corresponding to a power deposited in the target of 109 kW). The values of the integrated (over all solid angles and energy spectrum) current of other relevant particles escaping the TRAM are reported in Table 4.

### Table 4. Rate of secondary escaping particles from TRAM (primary beam: 800 MeV proton – 200 μA)

<table>
<thead>
<tr>
<th>Source strength</th>
<th>Neutron</th>
<th>Proton</th>
<th>Pion +/-</th>
<th>Muon +/-</th>
<th>Photon</th>
<th>Electron</th>
<th>Positron</th>
</tr>
</thead>
<tbody>
<tr>
<td>(s⁻¹)</td>
<td>$7 \times 10^{15}$</td>
<td>$4 \times 10^{13}$</td>
<td>$1.8 \times 10^{12}$</td>
<td>$2.5 \times 10^{11}$</td>
<td>$1 \times 10^{16}$</td>
<td>$7 \times 10^{13}$</td>
<td>$6.5 \times 10^{12}$</td>
</tr>
</tbody>
</table>


**Energy deposition: comparison between FLUKA and MCNPX models**

The energy deposited in the whole TRAM assembly has been estimated, differentiating the contribution of the different particle types that propagate through it (see Figure 5). The estimation of the separated contributions to the total energy deposition by neutrons, gammas and protons respectively constitutes an indirect check of the correctness of the whole TRAM model built with FLUKA.
The separated energy deposition profiles show the expected nuclear processes and spatial distribution which can then be used as an input for further engineering calculations.

The values of the deposited energy in the whole TRAM module by protons, neutrons and gammas respectively are specified in Table 5: most of the energy is deposited inside the target (87% of the total) by protons in the direct spallation process. In comparison, neutrons and prompt gammas are depositing only a negligible amount of energy: neutrons mainly in the moderating and absorbing materials around the target and the reflector and prompt gamma mainly inside the target itself. The remainder of the total energy deposition in the whole TRAM is due to: αs, heavy ion fragments, pions, muons, electrons and positrons.

Table 5. Energy deposited by particles in the whole TRAM assembly and in the embedded target

<table>
<thead>
<tr>
<th>E (MeV/pr)</th>
<th>Total</th>
<th>Proton</th>
<th>Neutron</th>
<th>Gamma</th>
</tr>
</thead>
<tbody>
<tr>
<td>TRAM</td>
<td>625</td>
<td>401</td>
<td>37</td>
<td>1.6</td>
</tr>
<tr>
<td>Target</td>
<td>543</td>
<td>380</td>
<td>4.4</td>
<td>1.5</td>
</tr>
</tbody>
</table>

Table 6 details the energy deposited in the different materials of each region for the TRAM case.

<table>
<thead>
<tr>
<th>Region</th>
<th>800 MeV proton</th>
<th>TRAM [MeV]</th>
</tr>
</thead>
<tbody>
<tr>
<td>Target</td>
<td></td>
<td></td>
</tr>
<tr>
<td>W</td>
<td></td>
<td>427.8±0.2%</td>
</tr>
<tr>
<td>Ta</td>
<td></td>
<td>85.6±0.2%</td>
</tr>
<tr>
<td>SS</td>
<td></td>
<td>19.4±0.7%</td>
</tr>
<tr>
<td>Ni (thermocouples)</td>
<td></td>
<td>0.5±2.3%</td>
</tr>
<tr>
<td>MixWater</td>
<td></td>
<td>9.5±2.3%</td>
</tr>
<tr>
<td>Total in Target</td>
<td></td>
<td>543</td>
</tr>
<tr>
<td>Reflector</td>
<td></td>
<td>54.2±0.4%</td>
</tr>
<tr>
<td>Poisons and liners</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Boral</td>
<td></td>
<td>20.5±0.2%</td>
</tr>
<tr>
<td>Gd</td>
<td></td>
<td>9.5E-3±2.8%</td>
</tr>
<tr>
<td>Cold Moderators</td>
<td></td>
<td></td>
</tr>
<tr>
<td>LCH4</td>
<td></td>
<td>0.5±1.7%</td>
</tr>
<tr>
<td>LH2</td>
<td></td>
<td>8.2E-3±8.3%</td>
</tr>
<tr>
<td>Water moderators</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Water (can +pipes)</td>
<td></td>
<td>1.76±4%</td>
</tr>
<tr>
<td>Structural and auxiliary Materials</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Al alloys</td>
<td></td>
<td>3.4±1.1%</td>
</tr>
<tr>
<td>Ti alloys</td>
<td></td>
<td>0.3±3.3%</td>
</tr>
<tr>
<td>SS</td>
<td></td>
<td>0.4±3.1%</td>
</tr>
<tr>
<td>Pad cooling water</td>
<td></td>
<td>1.12±1.1%</td>
</tr>
<tr>
<td>Total net balance</td>
<td></td>
<td>~625±0.2%</td>
</tr>
</tbody>
</table>


In Table 7, the energy deposited in the target, reflector and moderators have been compared to the values provided by the MCNPX model, as described in reference (Škoro, 2017): the FLUKA model predicts 14% more energy deposition in the target with respect to the MCNPX model.

The energy deposited in all moderators and pre-moderators have been also compared, confirming a quite good agreement between the predictions of FLUKA and MCNPX model.
Table 7. Energy deposition: comparison between FLUKA and MCNPX model

<table>
<thead>
<tr>
<th>Region</th>
<th>Material</th>
<th>FLUKA (MeV/primary)</th>
<th>MCNPX (MeV/primary)</th>
<th>Δ/MCNPX (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Target</td>
<td>Total</td>
<td>543.5 ± 2.3%</td>
<td>474.6</td>
<td>14.5</td>
</tr>
<tr>
<td></td>
<td>W</td>
<td>427.8</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td>Ta</td>
<td>85.6</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td>SS</td>
<td>19.4</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td>MixWater</td>
<td>9.5</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Reflector</td>
<td>Total</td>
<td>82.86 ± 7%</td>
<td>80.77</td>
<td>3</td>
</tr>
<tr>
<td></td>
<td>Be</td>
<td>54.24</td>
<td>58.97</td>
<td>8</td>
</tr>
</tbody>
</table>


In Table 8 and Table 9 the energy deposited in the asymmetric poisoned water moderator and in the liquid methane moderator is detailed for both codes. The FLUKA model of these two moderators is shown in Figure 6, together with the list of the used materials for each moderator.

The energy deposition in water, beryllium and Boral obtained by FLUKA and MCNPX respectively agree within 15% with very few exceptions, among whom the most relevant is the energy deposited inside the poison absorber sub-assemblies (made of Gd foils in Al cases) of both water and cold moderators. Since in these cases, the main contribution in the energy deposited is coming from the aluminium case the discrepancy in predictions could be attributed to the different models or cross sections used for the specific alloys in the two codes (analysis is still in progress).

Table 8. Energy deposition in the symmetric Water moderator

<table>
<thead>
<tr>
<th>Water moderator (symmetric poisoned)</th>
<th>FLUKA (MeV/primary)</th>
<th>MCNPX (MeV/primary)</th>
<th>Δ/MCNPX</th>
</tr>
</thead>
<tbody>
<tr>
<td>Total</td>
<td>3.27</td>
<td>3.36</td>
<td>3%</td>
</tr>
<tr>
<td>Water</td>
<td>0.881 ± 3%</td>
<td>0.84</td>
<td>4.6%</td>
</tr>
<tr>
<td>Boral (decoupler)</td>
<td>2.042 ± 5%</td>
<td>1.84</td>
<td>11%</td>
</tr>
<tr>
<td>Poison (Gd+Al)</td>
<td>0.017 ±1%</td>
<td>0.04</td>
<td>57%</td>
</tr>
</tbody>
</table>

Decay heat, activity and radionuclide inventory

The irradiation history profile used for the calculations is described in Table 2. From the engineering point of view, it is important to have a reliable indication of the decay heat deposited in the system even after the proton beam shutdown in order to be able to predict correctly the temperature distribution in the system during the cooling time (time after the beam shutdown). Recent work at ISIS to attempt to validate the predictions of both decay heat (Findlay et al., 2018) and activity (Findlay et al. 2017) in targets has shown good agreement with MCNPX calculations.

In FLUKA the generation and transport of decay radiation (limited to γ, β-, β +, X-rays, and Conversion Electrons emissions) is possible during the same simulation which produces the radionuclides (one-step method). For that, a dedicated database of decay emissions is used, based mostly on information obtained from the Brookhaven NNDC (BNL, 2019). As a consequence, results for production of residuals, their time evolution and residual doses due to their decays can be obtained in the same run, for an arbitrary number of decay times and for a given irradiation profile.

The activity in the enclosed target in the TRAM has been estimated at several cooling times after the shutdown and the results are reported in Figure 7. The total activity estimated in
tungsten (blue line) represents the higher contribution for the first \(^{25}\) h after the shutdown, while the tantalum activity (green line) is dominating afterwards.

**Figure 7. TRAM case: activity curve for target**

![Activity curve for target](image1.png)


The main parameters that address the design of the target cooling system are respectively: 1) the decay heat density as a function of time (see Figure 8) and b) the spatial distribution of the decay heat density.

In Figure 9, the decay power at cooling time 0 s is shown for each plate of the embedded target, highlighting the specific contribution from the different materials each plate is made of.

**Figure 8. TRAM case: Decay heat density in target as a function of cooling time**

![Decay heat density in target](image2.png)

In case of prompt radiation (i.e. in the nominal “beam-on” scenario), the 2D average spatial profile of the deposited power density in the plane perpendicular to the target axis is Gaussian, with its maximum in the target cross section centre. On the other hand, during the cooling time, the average spatial profile of the decay power density is rather resembling a “sombrero”, evolving in time as shown in Figure 10.

Figure 9. TRAM case: decay heat at t=0 s in the target plates

![Image showing decay heat at t=0 s in the target plates](source: STFC, Rutherford Appleton Laboratory, 2020)

Figure 10. TRAM case: Time evolution of the average spatial profile of the decay heat in the target

![Image showing time evolution of decay heat](source: STFC, Rutherford Appleton Laboratory, 2020)
The higher power density in the external cladding is due to the decay heat of the $^{182}$Ta and is localised in front of the MixWater longitudinal channels. The $^{182}$Ta radioisotope is produced mainly by (n,$\gamma$) capture reactions in $^{181}$Ta, with low-energy neutrons backscattered from the low Z material (e.g. the cooling water and beryllium) around the target.

The longitudinal spatial profile of the decay heat density has been also evaluated and compared to the prompt radiation case in Figure 11. During the beam-on scenario, the longitudinal energy deposition profile is following the proton range shape in the target, so that the energy density profile along the target axis falls down quite abruptly after about 25 cm from the front of the first target plate.

In case of decay heat, the deposited energy density is more uniformly distributed along the target due to the decay of $^{182}$Ta and $^{182m}$Ta, generated by low energy neutrons over all the external target cladding. Inside the tungsten plate core, the decay heat at cooling time 0 s is dominated by the decay of $^{183m}$W (with 5 s half-life), $^{185}$W and $^{187}$W.

**Figure 11. TRAM case: Longitudinal power profile in the target (prompt radiation and decay heat at t=0s)**

* (the decay heat scale is 3 orders of magnitude lower than the prompt radiation scale).


The complete radionuclide inventory has been estimated for all the TRAM components at several cooling times. In Table 10 the highest activity radionuclides at the shutdown are reported, all of which are in the target, together with the corresponding values predicted by the MCNPX model in association with CINDER’90 transmutation code (Gallmeier et al., 2010). The distinction between radionuclides produced mainly by low-energy neutrons interactions (LN) and by direct spallation (SP) has been highlighted. MCNPX/CINDER and FLUKA agree within 12% (4.35E+15 in FLUKA vs 4.94E+15 in MCNPX/CINDER) for the estimation of the total $^{182}$Ta activity (stable + metastable nuclide) at 0s after shutdown. However, the ratio for the specific isomer/ground state contributions is significantly different between the two codes.
### Table 10. TRAM case: Higher activity radionuclides inventory at 0s after shutdown

<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{182}$Ta (LN)</td>
<td>9.89E+06</td>
<td>$\beta$-</td>
<td>$^{182}$W</td>
<td>1.73E+15</td>
<td>3.13E+15</td>
</tr>
<tr>
<td>$^{182^*}$Ta (LN)</td>
<td>2.83E-01</td>
<td>IT*</td>
<td>$^{182}$Ta</td>
<td>2.62E+15</td>
<td>1.81E+15</td>
</tr>
<tr>
<td>$^{183}$W (LN)</td>
<td>5.20E+00</td>
<td>IT</td>
<td>$^{183}$W</td>
<td>1.73E+15</td>
<td>9.54E+14</td>
</tr>
<tr>
<td>$^{185}$W (LN)</td>
<td>8.54E+04</td>
<td>$\beta$-</td>
<td>$^{185}$Re</td>
<td>6.52E+14</td>
<td>5.8E+14</td>
</tr>
<tr>
<td>$^{187}$W (LN)</td>
<td>6.49E+06</td>
<td>$\beta$-</td>
<td>$^{187}$Re</td>
<td>4.55E+14</td>
<td>9.37E+14</td>
</tr>
<tr>
<td>$^{179}$W (SP)</td>
<td>2.25E+03</td>
<td>$\beta^+$</td>
<td>$^{179}$Ta</td>
<td>1.79E+14</td>
<td>2.33E+14</td>
</tr>
<tr>
<td>$^{181}$W (SP)</td>
<td>1.05E+07</td>
<td>EC</td>
<td>$^{181}$Ta</td>
<td>1.73E+14</td>
<td>1.91E+14</td>
</tr>
<tr>
<td>$^{177}$Ta (SP)</td>
<td>5.59E+02</td>
<td>$\beta^+$</td>
<td>$^{177}$Hf</td>
<td>1.12E+14</td>
<td>2.09E+14</td>
</tr>
<tr>
<td>$^{177^*}$Ta (SP)</td>
<td>2.04E+05</td>
<td>$\beta^+$</td>
<td>$^{177^*}$Hf</td>
<td>1.11E+14</td>
<td>1.12E+14</td>
</tr>
</tbody>
</table>

*IT = Isomeric Transition via a gamma or internal conversion.


The analysis to understand the differences between the FLUKA and MCNPX/CINDER activity estimations is still on-going and needs further in-depth investigations.

### Conclusions

A detailed model of the upgraded TS1 TRAM module has been developed in FLUKA to complement the existing MCNPX reference model. This new model is focused on addressing specific engineering issues relating to heat deposition and decay heat.

The preliminary simulation results are in agreement with the MCNPX predictions, considering the differences in nuclear data and physics models, even if some differences have to be investigated further.

The next step for the FLUKA model will be the inclusion of the vacuum vessel and the instrument flight-lines in order to properly validate the model with respect to the experimental neutron flux measured at the instrument.

### Acknowledgements

The authors want to thank: a) Goran Škoro for providing the MCNP results, b) Stephen Gallimore, Philip Harrison, Stephanie Thomson, Dan Coates and Leslie Jones of the ISIS Target Design Division for providing the technical drawings of the upgraded TRAM module and lots of useful explanations and clarifications when needed and c) Jon Roddom and Derek Ross for the computing support provided. Finally, the authors acknowledge the computing resources provided by STFC Scientific Computing Department's SCARF cluster.

### References


Science and Technology Facilities Council (2019), ISIS Neutron and Muon Source, www.isis.stfc.ac.uk/.


Benchmarking tests and models for solid state diamond detectors at LCLS-II

Mario Santana Leitner*, Alan S. Fisher¹, Ted Liang¹, Sayed H. Rokni¹, Christine Clarke¹, Carsten Hast¹, Johannes M. Bauer¹, James C. Liu¹, Anne Harris¹, Erich Griesmayer²

¹SLAC National Accelerator Laboratory
²CIVIDEC Instrumentation
*msantana@slac.stanford.edu

The LCLS-II hard X-ray FEL will soon begin commissioning at SLAC National Accelerator Laboratory. After upgrades now in planning, this facility will accelerate electrons to 8 GeV and operate beams of up to 1 MW at 1 MHz. At such high beam powers, fast detection of beam losses is paramount, not only to limit radiation around the areas of the accelerator that had been designed for a lower-power machine (5 kW at LCLS), but also for protection of beam line components and collimators. Indeed, the highest-power errant beams ought to be shut-off within 200 μs. Such a fast response can be achieved by solid state diamond detectors, which offer nanosecond time resolution due to the high mobility of holes vacated in the valence band when charged particles ionise atoms of the crystal lattice. Moreover, unlike conventional ion chambers, diamonds do not tend to saturate even for high charges at fast repetition rates, and they are also radiation and heat resistant, not requiring cooling or much installation space.

Despite their excellent characteristics, there is little to no experience in using diamonds as radiation detectors for safety purposes. Thus, prior to their use at LCLS-II, extensive tests have been carried out to validate these detectors and to calibrate their signals in different radiation fields. Moreover, a function has been written for FLUKA Monte Carlo code in order to accurately correlate dose rates (or other related quantities) with diamond detector readings. This paper presents benchmarking tests for diamond detectors along with measurements and simulations.
Poster session
Neutron source evaluation for the Neutron Data Production System (NDPS) at RAON

Sangjin Lee1*, Bumjong Kim1, Hyungjin Kim1
1Rare Isotope Science Project, Institute for Basic Science, Daejeon, 305-811, Korea
*sjlee@ibs.re.kr

As NDPS at RAON is neutron data production facility, it can produce various neutron sources from the multiple targets such as C, Li, Co, and W by ~88 MeV, 600 MeV protons and ~53 MeV deuteron beams. The design of a neutron facility and source will depend on the users’ requirements and the users’ demands. A key point to evaluate these sources in relation to a given purpose is to answer the following questions: 1) what kind of spectrum we can provide? 2) what other particles we have? 3) what are the physical properties of this source 4) how accurate is this estimate? We evaluate several neutron sources using MCNP, PHITS and ADVANTG. In this study, we discuss several types of neutron sources available at RAON. This evaluation is based on the actual facility layout of NDPS at RAON.
Shielding and radiological protection for a compact inverse Compton backscattering X-rays source, ThomX

Jean-Michel Horodynski

1iRSD – ingénierie Radioprotection sûreté démantèlement, UPS3364, CNRS/INP, 91898 Orsay, France
*jean-michel.horodynski@u-psud.fr

ThomX (Variola et al., 2014) is a particle accelerator dedicated to the production of high flux and high-energy X-rays (flux up to $10^{13}$ photon.s$^{-1}$ – energy up to 90 keV) using inverse Compton backscattering effect between an electron beam and a highly amplified laser (with the help of a Fabry-Perot cavity [Bonis et al., 2012]). Shielding has been designed to comply with many constraints regarding the surroundings and the technical specificities of the machine. This presentation shows how the multipurpose Monte Carlo code FLUKA (Ferrari et al., 2011) (Bohnen et al., 2014) helps studying miscellaneous topics among those raised by the building of a particle accelerator, above all regarding radiological protection, safety and environmental protection. An overview of the validation campaign will be described too, as the ThomX commissioning will begin at the first semester of 2019.

ThomX will be used in the same room as another particle accelerator, Andromede (Eller et al., 2014), which is a Secondary Ion Mass Spectrometer (SIMS) using a Van der Graaf generator. Thus, hutchs had been designed in order to use both plants independently ensuring an ambient dose rate inferior to 5 µSv.h$^{-1}$ outside the hutchs even if both accelerators are on. As the total weight of the shielding is limited by the maximum load of ground, the width of the walls of the ThomX hutch had been optimised, leading to the reduction of 350 tonnes of concrete compared to the first design (Horodynski et al., 2016). In a same goal, local shielding was designed to reduce radiological impact on workers. Environmental issues were studied, for instance the radiological impact of air release coming from the hutch when the accelerator is used had been assessed, or the radioactive waste production.

Introduction

ThomX is a electron accelerator that will produce high flux and high-energy X-rays (up to 40-90 keV, up to $10^{13}$ photons.s$^{-1}$) using Compton backscattering effect between electron beam and laser (see Figure 1). This facility is commissioned in order to reach three main goals:

- compact design;
- use of mostly standard components;
- development of high performance Fabry-Perot cavity.
A photocathode gun will produce electron bunches (Q=1 nC) which will be accelerated into a LINAC up to 50-70 MeV. Electrons are then injected to the ring during 20 ms. At the interaction point, electrons from the bunch will collide with the photons produced by a laser and amplified in a Fabry-Perot cavity. Then, bunch are extracted from the ring and dumped in a target through the extraction line (see Figure 2). The main parameters of the facility is shown on Table 1.

Various challenges regarding the radiological protection of this facility must be done. The first one is the building where the accelerator will be ran. Actually, two particle accelerators will be used in the same building (see Figure 3): ThomX and Andromede. The latter is a Secondary Ion Mass Spectrometry (SIMS) for nano-domains and nano-objects. A Van der Graaf generator will accelerate nano-particles of Au (Au4004+ up to Au10000n+) and C60n+ fullerenes between 1 to 4 MeV. Ions will be produced by two sources, a Liquid Metal Ion Source (LIMS) as a NanoParticle Ion Source (NIPS) provided by Orsay Physics and an Electron Cyclotron Resonance (ECR) source for multi-charged molecular ions, provided by Pantecknik. This platform is called IGLEX which is a multidisciplinary centre for technologies and applied sciences.

The second challenge was to design efficient shielding with the technical and economical limitations. For instance, the maximum load on the ground limited the weight of the bunker walls. In the same time, the space in the building was limited by the presence of two machines. Optimisation of the shielding must be done in order to reach the radiological protection objectives.

Thus, Monte Carlo code was used to design and to optimise the shielding of the facility. At the same time, it was possible to realise many other studies related to the reduction of the radiological impact of the accelerator on workers, population and environment. In our case, FLUKA code was used, based on the feedbacks for various electron accelerators particle (For instance, ARIEL – TRIUMF [Vancouver, Canada], Spring-8 [Saitama, Japan; Asano et al., 2013], bERLinPRO [Berlin, Germany] [Ott et al., 2011]).
Figure 2. ThomX design layout (Variola et al., 2014)

The footprint is: 10m*5m.

Figure 3. IGLEX with both facilities, Andromede (IPN) and ThomX (LAL)


Table 1. Main parameters of the ThomX facility

<table>
<thead>
<tr>
<th>Parameters</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Electron beam parameters</strong></td>
<td></td>
</tr>
<tr>
<td>Charge</td>
<td>1 nC</td>
</tr>
<tr>
<td>Normalised emittance rms</td>
<td>4.4π mm.mrad</td>
</tr>
<tr>
<td>Energy</td>
<td>50 – 70 MeV</td>
</tr>
<tr>
<td><strong>Interaction point</strong></td>
<td></td>
</tr>
<tr>
<td>Collision angle at the interaction point</td>
<td>2°</td>
</tr>
<tr>
<td>Power in the Fabry-Perot cavity</td>
<td>0.07 – 0.7 MW</td>
</tr>
<tr>
<td><strong>X-rays produced by ICS</strong></td>
<td></td>
</tr>
<tr>
<td>Maximum energy</td>
<td>40-90 keV</td>
</tr>
<tr>
<td>Total flux</td>
<td>$10^{11} – 10^{13}$ photons.s⁻¹</td>
</tr>
</tbody>
</table>

Design and optimisation of the shielding

An iterative method was used to design and to optimise the shielding for the ThomX facility (Figure 4). First step was to define the minimum thickness required to reach the radiological protection objectives. Then, weak points and overschilded parts were identified in order to modify locally the design and the thickness of the accelerator hutch. In the same time, the use of local shielding for specific parts of the accelerator was implemented if necessary. Beam losses localisation was based on the feedbacks of the exploitation of electron accelerators. Five main location were defined (see Figure 5): both beam dumps, one for LINAC, and the other for the ring; the scraper used in the transfer line used to set up the bunches to the right size; the septum used to inject and to extract the beam into the ring; the four straight vacuum chamber of the ring. Slightly overestimated beam losses percentage was used in order to get a reasonable security margin (Table 2).

Figure 4. Iterative process applied to the design and the optimisation of the shielding

Figure 5. Localisation of the beam losses along the ThomX accelerator

![Diagram of the ThomX accelerator showing the localisation of beam losses.](source)


Table 2. Percentages of losses all along the ThomX accelerator

<table>
<thead>
<tr>
<th>Localisation</th>
<th>Percentage of loss (%)</th>
<th>Current load of the primary beam (nC)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Beam - dump LINAC</td>
<td>100</td>
<td>1,5</td>
</tr>
<tr>
<td>Beam – dump ring</td>
<td>100</td>
<td>0,9</td>
</tr>
<tr>
<td>Septum – loading</td>
<td>10</td>
<td>1</td>
</tr>
<tr>
<td>Septum – emptying</td>
<td>10</td>
<td>1</td>
</tr>
<tr>
<td>Straight sections of the ring</td>
<td>4</td>
<td>1</td>
</tr>
<tr>
<td>Scraper</td>
<td>15</td>
<td>1,5</td>
</tr>
</tbody>
</table>


From the first version to the final one (see Figure 6), around 350 tonnes of concrete were removed (see Table 3). Reduction of the thickness of the roof is the main optimisation, requiring prohibited access to this area when the accelerator will be on. In the same time, a block of heavy concrete (around 3 tonnes) and a local shielding was added around the scraper in order to reach the radiological protection objectives outside the accelerator hutch.
Figure 6. Comparison between the first version (left) and the final one (right) of the accelerator hutch after the optimisation process

Red circles located the area where concrete was removed or added.


Table 3. Final results of the optimisation process of the accelerator hutch

<table>
<thead>
<tr>
<th>Zones</th>
<th>Reduction of the thickness of concrete walls</th>
<th>Weight removed (t)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Roof</td>
<td>Thickness reduced by 50 cm</td>
<td>248</td>
</tr>
<tr>
<td>North maze</td>
<td>Thickness of some walls reduced by 50 cm (normal concrete)/80 cm (barite concrete)</td>
<td>33</td>
</tr>
<tr>
<td>South maze</td>
<td>One wall removed and thickness reduced by 50 cm</td>
<td>69</td>
</tr>
<tr>
<td>Total</td>
<td></td>
<td>350</td>
</tr>
</tbody>
</table>


Radiological protection for workers

Monte Carlo code allows radiological protection officers to realise assessment of the radiological exposure for workers and users of the facility studied. In the case of electron accelerators like ThomX, regarding the energy and the current of the primary beam, two scenarios must be studied: when the accelerator is on (prompt radiation) to assess the exposition to the radiation leaking through the shielding; when the accelerator is off (delayed radiation), when workers will be inside the hutch, near the potentially activated parts of the machine.
Five exploitation steps were defined in order to gradually tested the facility. Regarding the beam parameters, some steps will involve radiological exposure due to activated parts of the accelerator, especially when the beam charge will reach 1 nC per bunch and a repetition rate of 50Hz (phase 2bis and 3, Table 4).

It could be demonstrated that people whom will work in the ThomX facility will be exposed to very low level of radiation (Table 5). Shielding of the accelerator hutch was designed in order to reach at least a maximum ambient dose rate equal to 3 µSv.h⁻¹ outside the facility when the accelerator will be on (see Figure 7). Starting from the step 2bis, workers could be exposed to delayed radiations from activated parts of the accelerator (septum and scraper, see next parts). Nevertheless, these assessments will be modified with the feedbacks acquired during the gradual steps of ThomX commissioning with radiological measurements. In particular, radiations coming from activated parts could be reduced again if necessary, with additional local shielding.

### Table 4. Steps defined to test the ThomX facility

<table>
<thead>
<tr>
<th>Step</th>
<th>Parts of the accelerator involved</th>
<th>Maximum energy of the primary beam (MeV)</th>
<th>Charge per bunch (nC)</th>
<th>Repetition rate (Hz)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>LINAC only</td>
<td>50</td>
<td>100⁻³</td>
<td>10</td>
</tr>
<tr>
<td>1bis</td>
<td>LINAC + Transfer Line</td>
<td>50</td>
<td>100⁻³</td>
<td>10</td>
</tr>
<tr>
<td>2</td>
<td>LINAC + Ring</td>
<td>50</td>
<td>100⁻³</td>
<td>10</td>
</tr>
<tr>
<td>2bis</td>
<td>LINAC + Ring</td>
<td>50</td>
<td>1</td>
<td>50</td>
</tr>
<tr>
<td>3</td>
<td>LINAC + Ring</td>
<td>70</td>
<td>1</td>
<td>50</td>
</tr>
</tbody>
</table>


### Table 5. Maximum efficient dose integrated by people working at the ThomX facility in one year

<table>
<thead>
<tr>
<th>Maximum efficient dose integrated in one year by worker (mSv)</th>
<th>Step 1 – 1 bis</th>
<th>Step 2</th>
<th>Step 2bis</th>
<th>Step 3</th>
</tr>
</thead>
<tbody>
<tr>
<td>&lt;0.5</td>
<td>0.8</td>
<td>1</td>
<td>1.5</td>
<td></td>
</tr>
</tbody>
</table>

Protection of the environment

Assessment of the activation of materials, air or water can be done using Monte Carlo code. These studies are useful in order to obtain order of magnitude, uncertainties (composition of materials for instance) or very low statistics do not permit to obtain results with high precision. Nevertheless, they are very valuable in order to identify elements with high activation rate in order to set regular checks on them. Regarding air activation, it is important to assess the level of activity rejected and the potential radiological impact on population.

Assessment of the activation of accelerators parts is necessary in order to set up an efficient radioactive waste management process. For a simple qualitative assessment, standard composition could be used for materials. In the case of ThomX, activation of several parts and area were made: vacuum chamber of the ring, scraper and its shielding, septum, beam-dump targets and concrete walls of the accelerator hutch. Standard composition for stainless steel, copper, tungsten, PET and lead were used. Regarding the concrete, accurate compositions of the standard and heavy concrete of the LURE facility were used in order to get an idea of the level of activation especially for residuals elements like europium, cobalt or zinc for example.

The following pattern of runs was defined: runs on step 2bis beam parameters for three years and runs on step three beam parameters for seven years. A 2-year cooling period was then supposed before the start of the dismantling. The most activated part could be the shielding in tungsten placed around the vacuum chamber just after the scraper (transfer
line, see Figure 8). Total specific activity coming from the production $^{181}$W ($T_{1/2} = 121.2$ days) (see Figure 9). Nevertheless, as this part is very limited in weight and volume, it could be easily managed as a radioactive waste following the French waste disposal.

Regarding concrete walls, only the additional heavy concrete block placed in front of the scraper could be activated to a significative level (see location in Figure 10). The same pattern as for accelerator parts was used. It could be demonstrated that the total specific activity could not be a serious issue in case of dismantling (using cutting process) and final disposal (see Figure 11).

**Figure 8. Top view of the scraper and its shielding**

![Scraper forward shielding (in red - Tungsten)](image)

The forward shielding, made of tungsten (red), is shown in the white circle.


**Figure 9. Specific activity evolution according to one scenario of exploitation of ThomX: 3 years in phase 2bis beam parameters, 7 years in phase 3 beam parameters and 2 years of cooling (facility stopped)**

![Specific activity evolution](image)

Assessment of the air activation inside the accelerator hutch was done using two methods: analytical one and Monte Carlo code. Ratios of radionuclides production between analytical and numerical solutions were very high (up to 26 for $^{13}$N for instance). The very low density of the material (the air) led to poor statistics with the Monte Carlo code. New study could be made using the same methodology developed for the Bremsstrahlung production rate (Ferrari et al., 1993).

The air in the accelerator hutch will be renewed 20 times per hour thanks to the ventilation system needed to use the Fabry-Perot cavity to produce the highly amplified laser beam. Thus, radiological exposure by inhalation of activated air in the accelerator hutch by workers is negligible. Nevertheless, radiological impact on the population and the
environment must be done regarding the air release outside the facility. Thus, total activity released in worst case (8 h.day\(^{-1}\), 5 days.week\(^{-1}\), 52 weeks.year\(^{-1}\)) has been assessed (see Table 6) and the radiological exposure for people working in the building located under the prevailing wind. At least, in step 3, the integrated dose by person will be lower to 2.5 \(\mu\)Sv in one year.

Table 6. Assessment of the total activity released for one year of ThomX runs

<table>
<thead>
<tr>
<th>Radionuclides</th>
<th>Step 1</th>
<th>Step 2</th>
<th>Step 2bis</th>
<th>Step 3</th>
</tr>
</thead>
<tbody>
<tr>
<td>(^{3})H</td>
<td>2.0(^{+1})</td>
<td>1.3(^{+3})</td>
<td>6.3(^{+4})</td>
<td>1.4(^{+5})</td>
</tr>
<tr>
<td>(^{9})Be</td>
<td>4.8(^{-4})</td>
<td>2.8(^{+4})</td>
<td>3.4(^{+6})</td>
<td>3.9(^{+6})</td>
</tr>
<tr>
<td>(^{13})S</td>
<td>1.8(^{-6})</td>
<td>3.1(^{+3})</td>
<td>3.5(^{+5})</td>
<td>2.7(^{+5})</td>
</tr>
<tr>
<td>(^{37})Ar</td>
<td>2.1(^{+3})</td>
<td>6.3(^{+3})</td>
<td>3.2(^{+5})</td>
<td>6.4(^{+5})</td>
</tr>
<tr>
<td>(^{41})Ar</td>
<td>6.1(^{+6})</td>
<td>7.9(^{+6})</td>
<td>4.0(^{+8})</td>
<td>1.2(^{+9})</td>
</tr>
</tbody>
</table>


Status and next steps

A request to obtain the licence to use the ThomX accelerator was sent to the French authority (ASN – Autorité de sûreté nucléaire) in November 2018. First tests of the facility could be done at the end of the first semester of the year 2019. Checking of the shielding, the radiation monitoring system and the personal safety system will be then made by the radiological protection officers. First results will be used to validate or to improve the numerical simulations made.

A complete assessment of radiological exposure for workers, people and environment has been done thanks to Monte Carlo code. It could be possible to cover the whole domain of radiological protection, from dose rate calculations, shielding design, to radioactive waste management and environmental impacts. Feedback provided by this project will be used for other particle accelerator projects where iRSD will be involved.

Acknowledgements

This work is supported by the French Agence Nationale de la Recherche as part of the programme “Investing in the Future” under reference ANR-10-EQPX-51. This work was also supported by grants from Région Île-de-France.

References


Shielding design of a dedicated cabin for the conditioning of a second RF cavity prior to be installed in the BOOSTER ring tunnel of the Synchrotron SOLEIL

Pruvost Jean-Baptiste*
1Synchrotron SOLEIL, France
*jean-baptiste.pruvost@synchrotron-soleil.fr

The injector of SOLEIL storage ring is made of a LINAC (raising electron energy up to 110 MeV @3Hz) and a BOOSTER synchrotron ring (ramping electron energy from 110 MeV to 2.75 GeV @3Hz). This is performed by the BOOSTER with a 5 cells RF cavity from the LEP (CERN) running at 35 kW CW @352 MHz with solid state amplifiers developed in house.

In order to improve the injection efficiency from the BOOSTER to the STORAGE RING of SOLEIL, particularly for the specific Low-alpha filling mode, a second RF cavity had been installed in the BOOSTER ring tunnel of SOLEIL. This second cavity will also have additional benefits in terms of power saving and redundancy in all other modes of operation.

Prior to be installed in the BOOSTER ring tunnel, this second RF cavity was powered with a new 60 kW (VRF=1.8 MV) solid state amplifiers @352 MHz and had previously several conditioning sessions in a dedicated shielded cabin in the storage ring RF emitters room at SOLEIL. This particular room is neither a survey radiation area nor a controlled radiation area as all accessible rooms during operation in the synchrotron building of SOLEIL.

This paper presents the calculation performed in order to design the shielding of this cabin for the radiation issues able to happen during the conditioning sessions. The model of the cabin and of the 5 cells copper RF cavity is described as performed for the use of the FLUKA2011© MC code. The conservative assumptions we assumed for the source term of dark current able to be produced during the conditioning sessions and at the origin of X-rays emission are also presented.

Then, radiation measurements during the RF cavity conditioning sessions are presented and compared to the calculations results.
Shielding and activation calculations for the MYRRHA Accelerator-Driven System design

Anna Ferrari¹, *, Stefan Mueller¹, Joerg Konheiser¹, Diego Castelliti², Massimo Sarotto³, Alexey Stankovskiy²
¹Helmholtz-Zentrum Dresden-Rossendorf, Bautzner Landstraße 400, 01328 Dresden (Germany)
²SCK-CEN, Boretang 200, 2400 Mol, Belgium
³ENEA, Via Martiri di Monte Sole 4, 40129 Bologna, Italy
*a.ferrari@hzdr.de

Accelerator-Driven Systems (ADS) are one of the options studied for the transmutation of nuclear waste in the international community. The design of sub-critical ADS requires high energy and high power proton accelerators, of the order of hundreds MeV and some MW for the proposed demonstration experiments. The use of high-energy Mega-Watt proton beams, in combination with a nuclear reactor core operating in sub-critical or critical mode, presents many challenges for various aspects of the design, being radiation shielding and minimisation of the induced activation key points.

In the frame of the FP7 European project MAXSIMA (2013-2018), an extensive simulation study has been done to assess the main shielding problems in view of the construction of the MYRRHA Accelerator-Driven System at SCK-CEN in Mol (Belgium), which aims to demonstrate efficient transmutation of high level waste and associated ADS technology. The heart of the system is a 100 MW lead-bismuth eutectic (LBE) cooled reactor, working both in critical and sub-critical modes. The neutrons needed to sustain fission in the sub-critical mode are produced via spallation processes by a 600 MeV, 4 mA proton beam, which is provided by a linear accelerator and hits a LBE spallation target located inside the reactor core.

With the goal to assess the shielding of the reactor building and to study the activation of the materials in key points around the reactor and in the vertical part of the proton beam line, an extensive simulation study has been done. Both the Monte Carlo codes MCNPX (version 2.6.0) and FLUKA (version 2011.2) have been used, also with the aim to do a code-to-code comparison and to cross-check the results. Starting from the current MCNPX model of the MYRRHA reactor core in the sub-critical operation mode, which includes the last part of the vertical proton beamline with the spallation target and the LBE coolant material around the core, a proper set of radiation fields has been fully characterised on suitable surfaces on and around the core vessel, and used as input in a second row of Monte Carlo simulations. To characterise the structures around the core, the main primary system components have been represented with a simplified geometry and a more homogeneous material composition. Minor internal elements have been neglected and all components have been described through a radial and axial homogenisation.

These calculations have been done with the FLUKA code, which has the unique possibility to compute, in the same simulation, the transport of both the prompt radiation (due to the ADS in operation) and the residual one (due to the activated materials). The behaviour of the neutron fluence, together with the dose distributions due to the prompt and to the residual radiation, has been then studied. Dose profiles have been evaluated from the core vessel to the external containment and the shielding walls in the horizontal direction, and up to the last magnet of the proton beam line and the final roof in the vertical one.
Moreover, an activation database of all the key structural materials has been built for typical irradiation patterns.

The results of the shielding and activation analysis are illustrated, together with the main implications on the design solutions.
Thermal neutron profiles inside J-PARC main ring tunnel

T. Oyama*1, H. Nakamura1,2, M. Hagiwara1,2, S. Nagaguro1,2, H. Yamazaki1,2, M. J. Shirakata1,2, K. Nishikawa1,2, T. Sanami1

1High Energy Accelerator Research Organization, 1-1 Oho, Tsukuba, Ibaraki, 305-0801 Japan
2J-PARC Center, 2-4 Shirakata, Tokai-mura, Naka-gun, Ibaraki, 319-1195 Japan
*takahiro.oyama@kek.jp

The Japan Proton Accelerator Research Complex (J-PARC) is a proton accelerator facility jointly developed and managed by the High Energy Accelerator Research Organization (KEK) and the Japan Atomic Energy Agency (JAEA). J-PARC consists of three accelerators: a linear accelerator (400 MeV LINAC), a rapid cycling synchrotron (3 GeV RCS), and a slow cycling synchrotron (Main Ring: MR) with designed beam powers of 0.28 MW, 1 MW and 750 kW, respectively. Such a large-scale accelerator complex with high beam intensity (up to 1 MW) and high energy (up to 30 GeV) has many difficult radiation problems in shielding design.

In particular, secondary neutrons produced due to primary beam loss are highly penetrative and act as sources of radioactivation, creating a need for the accurate characterisation of their reactions and transport properties. Most of the secondary neutrons are thermalised in an accelerator tunnel surrounded by concrete walls via multiple scattering processes. These thermal neutrons generate Ar-41 radioactive nuclides in the air through the neutron capture reaction of Ar-40. In comparison with other nuclides observed in the air of high-energy accelerator tunnels, Ar-41 significantly contributes to the radioactivity concentration in the exhaust air after a cooling time of about one hour. The Ar-41 concentration in air has significantly impacts on not only radiation exposure for personnel but also the facility management, because workers cannot access an accelerator tunnel until Ar-41 has sufficiently decayed. For these reasons, the accurate characterisation of the amount of Ar-41 produced during beam operation is a crucial requirement for the radiation shielding design of J-PARC.

In order to investigate the spatial distribution of Ar-41 production in the MR tunnel, the thermal neutron flux was measured using a gold foil activation method. Bare and Cd-covered gold foils were placed at 100 positions on the concrete wall of MR tunnel. The relative activities of the gold foils were simultaneously measured by using an Imaging Plate technique. The radioactivity of one reference foil was also measured with a HP-Ge detector to convert to the absolute activities for all foils measured with the Imaging Plate. In addition, gamma dose rates at the same position of the gold foils were also measured using NaI(Tl) and ionisation chamber survey metres immediately after beam operation to verify its correlation with the thermal neutron flux.

As a result, the thermal neutrons were locally distributed around the beam collimator installed in the injection section, which accounted for 97% of its total value. A strong correlation was observed between the distribution of the thermal neutrons and gamma dose rates.
Optimum design of neutron moderator assembly for accelerator-based boron neutron capture therapy using MCNPX code

Yong-uk Kye1,2*, Sung Gyun Shin3, Won Namkung1, Jung Yun Huang1, Moo Hyun Cho2
1Pohang Accelerator Laboratory, POSTECH, Pohang, Korea
2Division of Advanced Nuclear Engineering, POSTECH, Pohang, Korea
*kyu0610@postech.ac.kr

Accelerator-based Boron Neutron Capture Therapy (ABNCT) is first BNCT facility at Brain Research Center in Korea. ABNCT machine consists of ion source, Radio Frequency Quadrupole (RFQ), Drift Tube Linac (DTL), neutron generation target, moderator assembly and neutron detection system. The maximum proton beam power is of 80 kW with 10 MeV beam energy and 8 mA beam current. The beryllium bare target will be used for generating the fast neutron. The moderator assembly consists of tungsten filter, lead shielding, aluminium reflector, aluminium fluoride as moderator, lead neutron filter, iron shaper and borated polyethylene for generating the epithermal neutron to treat the patients. Material composition and geometry were optimised by MCNP code and epithermal neutron flux from 0.5 eV to 10 keV is $4.2 \times 10^9$ neutron/cm$^2$/sec at the patient position. This presentation will cover how to optimise the moderator assembly for producing the epithermal neutron to satisfy IAEA-TECDOC-1223 criteria and residual radioactivity.

Introduction

Boron Neutron Capture Therapy (BNCT) uses $^4$He and $^7$Li, which are the radiation produced by thermal neutron capture $^{10}$B(n,α)$^7$Li reaction. The range of the generated radiation by neutron capture reaction is less than the size of a cell within 10 um. $^4$He and $^7$Li selectively kill tumour cells, losing all their energy within 10 um (Kreiner et al., 2014; Rasmussen,1956). Dawonsys is building the ABNCT facility at Brain Research Center for the first time in Korea. Proton beam energy and current are 10 MeV and 8 mA respectively, obtained by RFQ and DTL, with accromat optical design for three treatment beamlines.
Moderator optimisation

Beryllium is stable material from irradiation by proton and it has high melting point. Therefore, beryllium was selected as a neutron generation target in this research. There are two main reactions in proton-beryllium interaction, such as neutron generation and charged-particle collision (Rasmussen, 1956). First of all, the fast neutrons are generated in beryllium by $^9\text{Be}(p,n)^9\text{B}$ reaction. Since the generated neutron beam energy is as high as 10MeV proton beam energy, it needed to be moderated down to epithermal range for patient irradiation. Second, the proton beam loses its energy during the penetration in beryllium by soft and hard collision and deposit its whole energy to target (Astrelin et al., 2010; Kumada et al., 2015; Guseva et al., 2000). Finally, all proton beam will stop at target matter. Figure 2 shows energy spectrum of moderated neutron beam at patient position by MCNPX simulation.
Beam shaping assembly (BSA) is to moderate neutron beam energy to epithermal energy range for therapy and include moderator part and shaping part. In case of moderator part, there are lots of candidate materials for moderator assembly such as AlF₃, MgF₂, CaF₂, etc. In order to optimise BSA, not only neutron flux but activation of composing materials should be considered. For example, AlF₃ and MgF₂ generate the 22Na by neutron activation, and 22Na emits the 1.37 and 2.75 MeV photons, with half-life of 15 hours (Fantidis, et al. 2017; Hashimoto et al., 2015; Minsky et al.; 2014). These photons cause irradiation to workers or patients even after treatment. The CaF₂ is suitable material with moderator because 22Na is not generated from neutron activation by 10 MeV proton beam. However, MgF₂ has high efficiency for neutron moderation due to the higher density. For the BSA optimisation, we applied combination of the MgF₂ and CaF₂. Figure 3 shows neutron energy spectrum for each moderator candidate of BSA.

Figure 3. Moderated neutron energy spectrum by material

Source: Pohang Accelerator Laboratory, POSTECH, 2020.

Shaping part is necessary for the flatness of the neutron beam at patient position, and it includes neutron reflectors and absorbers (Fantidis et al., 2017; Hashimoto et al., 2015; Minsky et al., 2014). Shaping part is optimised to irradiate a treatment area with uniform neutron beam density. Figure 4 shows radial distribution of epithermal neutron flux at patient position. The use of neutron reflector material such as lead reduces neutron flux loss and can maintain flatness. The BSA was designed based on IAEA standard (IAEA, 2001).
Palladium inserted target was also designed taking account of the hydrogen blistering (Astrelin et al., 2010; Kumada et al., 2015). Because proton lose all their energy inside the target, and combine by themselves to produce hydrogen gas in the target, trapped hydrogen gas make blistering. Figure 5 shows neutron energy spectrum by target type. Finally, BSA was designed by above-mentioned geometric optimisation process.

Results

Table 1 shows neutron beam properties of present design and compared with the design of other institutes. Finally, AlF₃ was chosen as the material of the main moderator. Lead, aluminium and some other materials were added as shaping materials. All the neutron
properties in this design were satisfied with the IAEA criteria. The epithermal neutron of energy range from 0.5 eV to 10 keV can penetrate about 10 cm in human body. IAEA criteria for BNCT are epithermal neutron flux higher than $1.0 \times 10^9$ n/cm$^2$/sec, and the photon and fast neutron dose less than $2.0 \times 10^{-13}$ Gy/cm$^2$/sec. In this result, ratio of epithermal neutron flux per thermal neutron flux, epithermal neutron flux, photon, fast neutron contamination are 44.74, $4.23 \times 10^9$ n/cm$^2$/sec, $0.91 \times 10^{-13}$ Gy/cm$^2$/sec, $0.41 \times 10^{-13}$ Gy/cm$^2$/sec, respectively.

Table 1. Neutron beam properties by institute

<table>
<thead>
<tr>
<th>Institute</th>
<th>Epithermal flux ($10^9$ n/cm$^2$/sec)</th>
<th>Photon contamination ($10^{13}$ Gy/cm$^2$/epithermal flux)</th>
<th>Fast neutron contamination ($10^{13}$ Gy/cm$^2$/epithermal flux)</th>
<th>Beam diameter (cm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>MIT FCB, US</td>
<td>5.3</td>
<td>3.6</td>
<td>1.4</td>
<td>12</td>
</tr>
<tr>
<td>Studsvik, Sweden</td>
<td>1.4</td>
<td>12.6</td>
<td>8.3</td>
<td>$14 \times 10$</td>
</tr>
<tr>
<td>F-R1, Finland</td>
<td>1.2</td>
<td>0.9</td>
<td>3.3</td>
<td>14</td>
</tr>
<tr>
<td>BMRR, US</td>
<td>1.1</td>
<td>1.5</td>
<td>2.6</td>
<td>12</td>
</tr>
<tr>
<td>Rez, Czech Rep</td>
<td>0.6</td>
<td>10.8</td>
<td>16.9</td>
<td>12</td>
</tr>
<tr>
<td>HFR, EC</td>
<td>0.33</td>
<td>3.8</td>
<td>12.1</td>
<td>12</td>
</tr>
<tr>
<td>THOR, Chinese Taipei</td>
<td>1.7</td>
<td>1.3</td>
<td>2.8</td>
<td>14</td>
</tr>
<tr>
<td>JRR-4, Japan</td>
<td>2.2</td>
<td>2.6</td>
<td>3.1</td>
<td>12</td>
</tr>
<tr>
<td>KURRI, Japan</td>
<td>0.46</td>
<td>2.8</td>
<td>6.2</td>
<td>15</td>
</tr>
<tr>
<td>C-BENS, Japan</td>
<td>1.2</td>
<td>7.8</td>
<td>5.8</td>
<td>-</td>
</tr>
<tr>
<td>i-BNCT, Korea</td>
<td>4.3</td>
<td>0.48</td>
<td>-</td>
<td>15</td>
</tr>
<tr>
<td>ABNCT, Korea</td>
<td>4.23</td>
<td>0.91</td>
<td>0.41</td>
<td>12</td>
</tr>
</tbody>
</table>

Source: Pohang Accelerator Laboratory, POSTECH, 2020.
References


Composition analysis of ordinary concrete to estimate residual isotopes in the decommissioning of particle accelerator

Arim Lee¹, Nam-Suk Jung¹, Hee-Seock Lee¹*
¹Pohang Accelerator Laboratory, POSTECH
*lee@postech.ac.kr

A major issue in the process of the decommissioning of accelerator facilities is the activation of ordinary concrete. The activated concrete is directly associated with decommissioning costs because it accounts for most of the volume of wastes generated from the accelerator facilities. Many of constituent elements contained in the concrete can be activated by neutrons, and the most prominent products are ⁶⁰Co, ¹⁵²Eu and ¹⁵⁴Eu produced by trace elements of the concrete, Co and Eu. Both are long-lived radionuclides with a large thermal neutron capture cross section so the disposal procedure of wastes is mainly decided according to the concentration of these radionuclides. In order to estimate the concrete activation, the information on elemental compositions of trace elements as well as the major constituent elements of concrete are essential. In this study, the elemental composition analyses of ordinary concrete samples, obtained from several construction sites in Korea, were done by Elemental Analysis (EA), X-ray Fluorescence (XRF) and Inductively Coupled Plasma Mass Spectrometry (ICP-MS). The results were compared with well-known composition of ordinary concretes of NIST and ANSI-ANS, which are mainly used in the estimation of concrete activation. The ranges of minimum to maximum compositions for Co and Eu are 8.94 to 32.73 ppm and 0.70 to 1.13 ppm, respectively.

Introduction

In the decommissioning process, the activation of the ordinary concrete is a serious problem in terms of radioactive waste disposal. The activated concrete is directly related with the decommissioning plan and costs because it is expected to account for most of radioactive wastes generated from the accelerator facilities.

The concrete is made of the cement and the aggregates. In particular, trace elements contained in aggregates, such as Co and Eu, produce the long-lived radionuclides, ⁶⁰Co, ¹⁵²Eu and ¹⁵⁴Eu, by thermal neutron capture reaction with a large cross-section. These radionuclides build up over time and are the most prominent during the decommissioning process of accelerator facilities as well as waste disposal procedures will be decided mainly according to the concentration of these radionuclides.

To prepare for decommissioning of accelerator facilities, Monte Carlo codes are widely used for estimating concrete activation to calculate the neutron flux inside the concrete. For this estimation, the information on elemental compositions of the concrete including trace elements are essential. However, it is difficult to find the elemental composition data of concretes, which are casted in Korea.

In this study, the elemental composition analyses of ordinary concrete samples, obtained from several construction sites including large-accelerator facilities in Korea, were done
by Elemental Analysis (EA), X-ray Fluorescence (XRF) and Inductively Coupled Plasma Mass Spectrometry (ICP-MS).

Table 1. Radionuclides observed in activated concrete of cyclotron facility (Phillips 1986)

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Half-life</th>
<th>Reaction</th>
<th>Cross section (barn)</th>
<th>Natural abundance (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Eu-152</td>
<td>13.3 y</td>
<td>Eu-151(n,γ)</td>
<td>5,900</td>
<td>47.7</td>
</tr>
<tr>
<td>Eu-154</td>
<td>8.6 y</td>
<td>Eu-153(n,γ)</td>
<td>390</td>
<td>52.2</td>
</tr>
<tr>
<td>Co-60</td>
<td>5.3 y</td>
<td>Co-59(n,γ)</td>
<td>37</td>
<td>100</td>
</tr>
<tr>
<td>Cs-134</td>
<td>2.1 y</td>
<td>Cs-133(n,γ)</td>
<td>29</td>
<td>100</td>
</tr>
<tr>
<td>Zn-65</td>
<td>244.1 d</td>
<td>Zn-64(n,γ)</td>
<td>0.78</td>
<td>48.9</td>
</tr>
<tr>
<td>Sc-46</td>
<td>83.8 d</td>
<td>Sc-45(n,γ)</td>
<td>26.5</td>
<td>100</td>
</tr>
<tr>
<td>Fe-59</td>
<td>44.6 d</td>
<td>Fe-58(n,γ)</td>
<td>1.15</td>
<td>0.3</td>
</tr>
<tr>
<td>Mn-54</td>
<td>312.5 d</td>
<td>Fe-54(n,p)</td>
<td>0.386(4.9 MeV)</td>
<td>5.8</td>
</tr>
<tr>
<td>Na-22</td>
<td>2.6 y</td>
<td>Na-23(n,2n)</td>
<td>0.017(14.6 MeV)</td>
<td>100</td>
</tr>
</tbody>
</table>

Source: Pohang Accelerator Laboratory, POSTECH, 2020.

Process of composition analysis

Sample information

The concrete samples analysed in this study were procured at six different construction sites of conventional buildings and accelerator facilities in Korea. Three samples were made at the building construction sites in Busan and Changwon city as shown in Figure 1. The compressive strength of these concrete samples were 18, 21 and 27 MPa. Three other concrete samples from large-accelerator facilities in Korea, the Pohang Accelerator Laboratory Free Electron Laser (PAL-XFEL), the Korea Multi-purpose Accelerator Complex (KOMAC) of the Korea Atomic Energy Research Institute in Gyeongju, and RAON of the Institute for Basic Science in Daejeon, were also analysed in this study. The size of the obtained concrete samples is normally 10 cm in diameter and 20 cm in height, all of which are Portland type of the ordinary concrete.

Figure 1. Manufactured samples

Source: Pohang Accelerator Laboratory, POSTECH, 2020.
Table 2. Information on concrete samples

<table>
<thead>
<tr>
<th>Sample name</th>
<th>City</th>
<th>Concrete type</th>
<th>Compressive Strength (MPa)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Conventional building 1</td>
<td>Busan</td>
<td>Ordinary concrete</td>
<td>18</td>
</tr>
<tr>
<td>Conventional building 2</td>
<td>Busan</td>
<td></td>
<td>21</td>
</tr>
<tr>
<td>Conventional building 3</td>
<td>Chanwon</td>
<td></td>
<td>27</td>
</tr>
<tr>
<td>PAL-XFEL (PAL)</td>
<td>Pohang</td>
<td></td>
<td>24</td>
</tr>
<tr>
<td>KOMAC (KAERI)</td>
<td>Gyeongju</td>
<td></td>
<td>-</td>
</tr>
<tr>
<td>RAON (IBS)</td>
<td>Daejeon</td>
<td></td>
<td>27</td>
</tr>
</tbody>
</table>

Source: Pohang Accelerator Laboratory, POSTECH, 2020.

Process of sample preparation and composition analysis

In order to analyse the weight fraction of the constituent elements, a homogeneous and fine powdery sample is required regardless of analysis methods. Thus, six number of concrete samples were crushed using a steel breaker and a steel chisel. Figure 2 shows this process of sample preparation for composition analysis in this work.

The composition analysis was carried out by three spectroscopic techniques that have broad application to mineralogy and petrology, EA, XRF and ICP-MS.

Figure 2. Process of sample preparation

Source: Pohang Accelerator Laboratory, POSTECH, 2020.

EA is the common method of analysing C, H, N and S by combusting sample and does not require acid pre-treatments. In this work, the contents of C, H, N, and S elements were analysed by EA using Flash2000 elemental analyser (Thermo Fisher Scientific, Germany).

The composition of major and trace elements were analysed by XRF using S8 TIGER (Bruker, Germany). XRF can analysed the elements from Be to U. Elements, such as Si, Ca and Al, which are high contents of constituent elements, were mainly analysed by XRF. For one analysis, 0.5 g of sample is required as a representative sample.
ICP-MS was carried out for more precise analysis of trace element contents based on the XRF result as well as the main elements of the concrete activation, such as Co and Eu using NexION 350D ICP-Mass Spectrometer (Perkin-Elmer, US). ICP-MS can also analyse from Be to U. It is possible to analyse a trace amounts in ppb to ppt units so the contents of important trace elements related with the concrete activation, Eu, Cs and Co, were analysed in ppm level. For ICP-MS, pre-treatments are required and 0.1 g of representative sample is required as a minimum.

One of the most sensitive technique of trace analysis is Neutron Activation Analysis (NAA) based on neutron capture reactions and gamma-ray spectroscopy. Since the research reactor in Korea (HANARO) had not been operated until recently, it was impossible to analyse the composition of trace elements by NAA. As the HANARO reactor operations have resumed, NAA will soon be used for the analysis.

**Process of organising the results from EA, XRF and ICP-MS**

The results of three analyses were presented in different units, such as wt%, ppm and ppb. The major element from XRF were given in the form of oxides such as SiO$_2$ and Al$_2$O$_3$, and the given unit is wt%. Using the atomic weight and that of ratio, the result of the oxide form was separated to the weight fraction of constituent elements and the oxygen. The unit of the trace element from XRF and ICP-MS was given in ppm and converted to wt%. The results of ICP-MS were adopted for the elements analysed in common for XRF and ICP-MS, since the ICP-MS was known to be more precise for the analysis of the trace elements. However, for Si, Zr, P and As, it was recommended to use the results from XRF because the process of acid pre-treatments can reduce the reliability of ICP-MS. The results of EA was in the unit of wt% so it was directly used and arranged with other data. Finally, the three results were unified in terms of wt%. The whole process is illustrated as shown in Figure 3.

**Figure 3. Process of organising the results from EA, XRF and ICP-MS**

Source: Pohang Accelerator Laboratory, POSTECH, 2020.
Result – EA, XRF, ICP-MS

**Weight fractions of major elements**

The analysis of major elements of concrete were summarized as Table 3 and compared with the elemental composition of the NIST ordinary concrete (NIST 1996) and the ANSI-ANS concrete (ANS, 2006), which are frequently used for the concrete elements.

In the case of the composition of O and Si, which are contained in the concrete with high contents, the results were 5 to 10% lower than those of references. For other constituents with a weight fraction of around 10% or less, the results are similar or higher than those of references. There was no big difference on the weight fraction of major elements in concrete samples depending on the location of construction site and type of building (conventional buildings and accelerator facilities). The analysis results on the elemental composition in concrete samples was similar to that of the ANSI-ANS concrete.

**Table 3. Weight fraction of major elements in ordinary concrete (unit: wt%)**

<table>
<thead>
<tr>
<th>Element</th>
<th>NIST</th>
<th>ANSI-ANS</th>
<th>This work</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Mean (Min–Max)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>O</td>
<td>57.49</td>
<td>49.83</td>
<td>44.82 (43.13-46.03)</td>
</tr>
<tr>
<td>Si</td>
<td>30.46</td>
<td>31.57</td>
<td>28.53 (25.69-29.83)</td>
</tr>
<tr>
<td>Ca</td>
<td>4.3</td>
<td>8.26</td>
<td>8.84 (6.14-13.50)</td>
</tr>
<tr>
<td>H</td>
<td>2.21</td>
<td>0.55</td>
<td>0.77 (0.56-0.99)</td>
</tr>
<tr>
<td>Al</td>
<td>2.00</td>
<td>1.55</td>
<td>6.59 (5.67-7.16)</td>
</tr>
<tr>
<td>Na</td>
<td>1.52</td>
<td>1.70</td>
<td>2.35 (1.72-4.27)</td>
</tr>
<tr>
<td>K</td>
<td>1.00</td>
<td>1.91</td>
<td>2.10 (1.54-2.77)</td>
</tr>
<tr>
<td>Fe</td>
<td>0.64</td>
<td>1.23</td>
<td>2.41 (1.58-3.18)</td>
</tr>
<tr>
<td>C</td>
<td>0.25</td>
<td>-</td>
<td>1.28 (0.59-2.48)</td>
</tr>
<tr>
<td>Mg</td>
<td>0.13</td>
<td>0.26</td>
<td>1.45 (1.10-1.64)</td>
</tr>
<tr>
<td>S</td>
<td>-</td>
<td>0.13</td>
<td>0.20 (0.13-0.30)</td>
</tr>
</tbody>
</table>

Source: Pohang Accelerator Laboratory, POSTECH, 2020.

**Weight fraction of trace elements**

The important information of concrete activation is the amount of trace elements contained in the concrete. Totally, 23 number of trace elements (< 1 wt%) were detected in concrete samples. The weight fraction of trace elements producing radionuclides observed in activated concretes, such as Eu, Co, Mn, Zn, Sc and Cs, was shown in Table 4. Amongst these trace elements, Eu and Co were prominent trace elements. The mean weight fraction of Co and Eu were 15.25 ppm and 0.95 ppm respectively. The ranges of minimum to maximum weight fraction of Co and Eu are 8.94 to 32.73 ppm and 0.70 to 1.13 ppm. The difference of the weight fraction of Eu between samples was smaller than that of Co. These results were compared with the data of other researchers as shown in Table 5. The results of M. Kinno et.al is similar with the results of this study.
Table 4. Weight fraction of trace elements in ordinary concrete (unit: ppm)

<table>
<thead>
<tr>
<th>Element</th>
<th>This work</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Mean (Min–Max)</td>
</tr>
<tr>
<td>Mn</td>
<td>620 (367-755)</td>
</tr>
<tr>
<td>Zn</td>
<td>302 (181-479)</td>
</tr>
<tr>
<td>Sc</td>
<td>8.02 (5.31-10.31)</td>
</tr>
<tr>
<td>Cs</td>
<td>4.68 (2.24-7.45)</td>
</tr>
<tr>
<td>Co</td>
<td>15.25 (8.94-32.73)</td>
</tr>
<tr>
<td>Eu</td>
<td>0.95 (0.70-1.13)</td>
</tr>
</tbody>
</table>

Source: Pohang Accelerator Laboratory, POSTECH, 2020.

Table 5. Comparison of weight fraction of Co and Eu (unit: ppm)

<table>
<thead>
<tr>
<th>Analyzed by</th>
<th>Cobalt</th>
<th>Europium</th>
</tr>
</thead>
<tbody>
<tr>
<td>Martinez – Serrano et. al. (Martinez-Serrano 2010)</td>
<td>4.8</td>
<td>0.077</td>
</tr>
<tr>
<td>Suzuki et. al. (Suzuki 2001)</td>
<td>21.9</td>
<td>1.08</td>
</tr>
<tr>
<td>Kinno et. al.(Ordinary concrete) (Kinno 2002)</td>
<td>8 – 30</td>
<td>0.7 – 1.0</td>
</tr>
<tr>
<td>Masumoto et. al. (NUSTEC 2014)</td>
<td>5.6 – 21.5</td>
<td>0.4 – 0.9</td>
</tr>
<tr>
<td>NUREG/CR-3474 (Bioshield) (Evans 1984)</td>
<td>9.8 ± 1.03</td>
<td>0.55 ± 0.38</td>
</tr>
<tr>
<td>This work (Average)</td>
<td>8.94 – 32.73 (15.25)</td>
<td>0.70 – 1.13 (0.95)</td>
</tr>
</tbody>
</table>

Source: Pohang Accelerator Laboratory, POSTECH, 2020.

Summary and conclusions

The weight fraction of constituent elements in the Portland type of the ordinary concrete obtained from various construction sites and facilities was measured by three different methods, EA, XRF and ICP-MS. The weight fractions of major elements were similar to those of the ANSI-ANS concrete, and the weight fraction of Co and Eu in this work were similar to those of other studies. The measured fraction ranges of Co and Eu are 8.94 to 32.73 ppm and 0.70 to 1.13 ppm, respectively. The results of this study will be good references in the calculation of the radiative rate of other accelerator facilities. Additionally, NAA for trace element will be performed using research reactor, HANARO in Korea.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety (KOFONS), granted financial resource from the Nuclear Safety and Security Commission (NSSC), and Korea (Grants No. 1603005).

References


A development of calculation module to analyse high-energy reactions in the AR2S code

Do Hyun KIM, Chang Ho SHIN, and Jong Kyung KIM*
Department of Nuclear Engineering, Hanyang Univ. 222 Wangsimni-ro, Seongdong-gu, Seoul, 133-791, Korea
*jkkim1@hanyang.ac.kr

Nowadays, several types of high-energy accelerators for their purposes have been developed. To protect its radiation workers and equipment from residual dose, activation generated by accelerated particle should be properly analysed at designing level using currently available inventory codes. However, there are two limitations. Firstly, a lack of the cross-section (XS): XS for high-energy or specific particles does not exist. It is solved by using reaction probabilities generated by the physics model of transport code. The SP-FISPACT is one of that inventory code. Secondly, the resolution problem: To reduce the flux distortion, the interested area to calculate activation should be well divided. This process leads to huge computational time and efforts for making numerous cells. One of the solutions is the mesh-based activation method. In our previous study, the AR2S system is developed by coupling MCNPX 2.7 code and FISPACT-2010 for mesh-based activation calculation. However, to directly use the SP-FISPACT accompany problems, which are required high computational time and give lack of precision for reaction rate from limited computer memory. In this study, the activation module was developed by using a simple calculation model for each mesh. the developed module use following procedure: i) getting a high-energy spectrum of each mesh; ii) calculating reaction rate using MCNPX code; iii) adding reaction rate in original data of the FISPACT code; iv) run activation code. To verify this module, simple activation problem was estimated. the result with good agreement shows that it can well analyse high-energy activation problem. Also, an application for one part of ISOL facility in RAON accelerator, using high energy particle was presented.
Benchmarking experiments of displacement cross-sections for high-energy proton irradiation with cryogenic-sample

Yosuke Iwamoto

1Japan Atomic Energy Agency
on behalf of the DPA measurement team
*iwamoto.yosuke@jaea.go.jp

Monte Carlo particle transport codes such as PHITS, MARS and FLUKA are used to calculate displacements per atom (DPA) values for prediction of the operating lifetime of target materials in high-energy (> 100 MeV) region at accelerator facilities. For validation of calculated DPA values, one possibility is to measure displacement cross sections in relation to changes in electrical resistivity at cryogenic temperature. This paper introduces our benchmarking experiments of displacement placement cross sections of aluminium and copper using 200 MeV proton beams at the cyclotron facility in Research Center for Nuclear Physics (RCNP), Osaka University. In the benchmarking experiment, a proton irradiation device with a Gifford-McMahon (GM) cryocooler to cryogenically cool two 0.25-mm-diameter wire samples of aluminium and copper was developed. By using this device, the defect-induced electrical resistivity changes related to the displacement cross section of aluminium and copper were measured under irradiation with 200 MeV proton beams at cryogenic temperature at 5 K. For the comparison between the experimental data and calculated results with PHITS, the arc-dpa displacement cross sections providing more physically realistic descriptions of primary defect creation in materials provides better agreements with the experimental data than the conventional NRT-dpa displacement cross sections.

Introduction

To predict the operating lifetime of materials in high-energy (>100 MeV) radiation environments at accelerator facilities, Monte Carlo codes such as PHITS (Sato T., et al., 2018; Iwamoto Y., et al., 2017), FLUKA (Böhlen et al., 2014), MARS (Mokhov et al., 1995; Mokhov and Striganov, 2007), and MCNPX (Water, 2002) are used to calculate the transport of particles, nuclear reactions between particles and materials, distribution of primary knock-on atoms, and the number of displacements per atom related to the number of Frenkel pairs. A Frenkel pair is defined as a vacancy and a self-interstitial atom in the irradiated material. The Norgett Robinson Torrens (NRT) model (Norgett et al., 1975) has been widely used to predict the number of “initial” Frenkel pairs generated in the cascades of atom-atom collisions without the recombination of atoms (NRT-dpa). For more accurate estimation of the actual damage production, athermal recombination-corrected displacement damage (arc-dpa) (Nordlund et al., 2018; Konobeyev et al., 2017), which includes the results of the Molecular Dynamics simulation method (MD), is used as well.

For validation of the calculated displacement cross-sections one can measure changes in electrical resistivity of samples at cryogenic temperatures (around 4 K). The number of surviving defects is then related to defect-induced changes in the electrical resistivity of metals at around 4 K, where the recombination of Frenkel pairs by thermal motion is well suppressed.
In high-energy region, experimental displacement cross-sections in the case of 1.1- and 1.94-GeV proton irradiation of copper and tungsten were obtained at the Brookhaven National Laboratory (BNL) (Greene et al., 2004). In the BNL experiment, the cryostat assembly for sample irradiation consisted of a complicated cryogenics system to deliver a metered flow of liquid cryogen (mixture of liquid nitrogen and liquid helium) for controlling the sample temperature.

To enable us to measure the experimental data at various accelerator facilities without liquid-helium cryogen system, we developed a cryogen-free cooling system by using a Gifford-McMahon (GM) cryocooler and measured experimental displacement cross sections in the case of 125-MeV proton irradiation of copper at the Fixed-Field Alternating Gradient (FFAG) accelerator facility in Kyoto University Research Reactor Institute (KURRI) (Iwamoto et al., 2015) and those in the case of 200-MeV proton irradiation of aluminium and copper at Research Center for Nuclear Physics (RCNP), Osaka university (Iwamoto et al., 2018). In this reports, we introduce our measurements at RCNP and comparison between experimental data and calculated results with PHITS.

**Experiments**

**Cryogenic irradiation chamber**

The cryogenic irradiation chamber developed in this work was installed at the N0 beam line in the RCNP cyclotron facility, Osaka University. Protons with energies of 200 MeV and beam current between 1 and 3 nA were directed to be incident on the irradiation chamber via quadrupole magnets, which control beam size at the sample position. Figure 1 shows a schematic of the cryogenic irradiation chamber with the GM cryocooler (RDK-408D2, Sumitomo Heavy Industries, Ltd.) with a cooling capacity of 1 W at 4 K and the twin sample assembly connected to the 2nd stage of the cold head. The GM cryocooler cooled the sample by means of a conduction coolant via the aluminium plate and the OFHC block. The 1-mm-thick aluminium plates of the thermal radiation shield connected to the 40 K stage of the refrigerator covered the entire sample assembly to intercept any thermal radiation from the ambient irradiation chamber.

For a subsequent annealing study of the accumulated damage products, a 40 Ω electrical resistance heater was attached to the OFHC block. A pre-calibrated electrical resistance thermometer (Cernox thermometer, Lake Shore Cryotronics, Inc.) was attached in the OFHC block and the AlN plate to confirm the cooling performance of the GM cryocooler. To simultaneously measure the changes in electrical resistivity of two samples under proton irradiation, two aluminium plates with the aluminium and copper wire samples were connected to the OFHC block by using bolts, as shown in Figure 1. A detailed drawing of the sample holder with the wire sample is indicated in the paper (Iwamoto et al., 2018).
Table 1 lists the characteristics of the wire sample. Each aluminium and copper wire with a 0.25-mm diameter, purchased from Nilako Corporation, was set on the AlN plate in a serpentine-shaped line. The AlN plate was used because of its excellent electrical insulation and high thermal conductivity (21 W/m K at 4 K). Before irradiation, the aluminium wire and the copper wire were annealed in vacuum for 1 h at 550 °C (823 K) and 1000 °C (1273 K), respectively. The length between the two potential points was 123 mm for aluminium and 134 mm for copper. The wire and the CX1050-SD Cernox resistance thermometer were carefully sandwiched between 1-mm-thick AlN plate and 1.5-mm-thick AlN plate. The sample folder was embedded into a 0.5-mm-deep hollow on the aluminium back plate and compressed by the 1-mm thick aluminium plate, providing good thermal contact between the sample and the AlN sheets.
Table 1. Characteristics of wire sample

<table>
<thead>
<tr>
<th>Material</th>
<th>Aluminium</th>
<th>Copper</th>
</tr>
</thead>
<tbody>
<tr>
<td>Diameter (mm)</td>
<td>0.25</td>
<td>0.25</td>
</tr>
<tr>
<td>Length between two potential points (mm)</td>
<td>123</td>
<td>134</td>
</tr>
<tr>
<td>Purity (%)</td>
<td>99.99</td>
<td>99.999</td>
</tr>
<tr>
<td>Resistivity at room temperature with annealing process (Ω m)</td>
<td>$2.19 \times 10^{-8}$</td>
<td>$1.47 \times 10^{-8}$</td>
</tr>
<tr>
<td>Resistivity at 3 K with annealing process (Ω m)</td>
<td>$4.43 \times 10^{-11}$</td>
<td>$2.02 \times 10^{-11}$</td>
</tr>
<tr>
<td>Residual resistivity ratio (RRR) of resistance between room temperature and 3 K</td>
<td>492</td>
<td>738</td>
</tr>
</tbody>
</table>


The electrical resistance of the wire was measured using a combination of a current source (model 6221, Keithley Instruments, Inc.) and a nano-voltmeter (model 2182A, Keithley Instrument, Inc.) in the same manner as that in the FFAG experiment (Iwamoto et al., 2015). This apparatus is based on the current-reversal method (four-prove technique) in the delta mode, which works by sourcing pulses with opposite polarity and taking one measurement during each pulse. Taking the difference between the positive and negative pulse measurements cancels the effects of the thermal electromotive force. A current of ±100 mA was fed into the copper wire with the polarity changing at a frequency of 10 Hz. The voltage of the copper wire was read at intervals of 5 m. The precision of this resistance measurement at 3 K was ±0.001 μΩ corresponding to an electrical resistivity of ±0.67 fΩ m for aluminium and ±0.60 fΩ m for copper, where the electrical resistivity of the sample is expressed as follows:

$$\rho = RA / L$$  \hspace{1cm} (1)

where R is the measured electrical resistance, L is the length between two potential points, and A is the area of the sample. The temperature of the Cernox resistance thermometer was measured using a temperature controller (model 335, Lake Shore Cryotronics, Inc.). According to the manual of the temperature controller, the accuracy of temperature measurement using the Cernox thermometer is 7.1 mK at 4.2 K. After the cooling test of the cryogenic irradiation device, the minimum temperature of the AlN plate was 3.0 K. The electrical resistivity of aluminium and copper at 3 K reached $4.43 \times 10^{-11}$ Ω m ($1.11 \times 10^{-4}$ Ω) and $2.02 \times 10^{-11}$ Ω m ($5.53 \times 10^{-5}$ Ω), respectively.

**Proton beam fluence**

For imaging with proton beams, a 1-mm-thick ZnS fluorescent screen was attached onto the thermal radiation shield. When protons interact with the fluorescent screen, a light detectable using a camera is emitted. The camera was set near the glass view port of the cryogenic proton irradiation chamber to capture the beam shape with data for the red, green, and blue (RGB) lights. The beam size was adjusted to cover the wire sample by controlling the magnetic fields of the quadrupole and bending magnets upstream of the irradiation chamber. The irradiation lengths of the aluminium and copper wires were 73.5 mm and
82.0 mm, respectively, and the irradiation area was $1.84 \times 10^{-4}$ m$^2$; these numbers were determined using pictures of the beam image.

The number of protons during irradiation was measured at the beam relative monitor in situ by counting the number of events produced by neutron-proton scattering at the 2.2 mg/cm$^2$ thick polyethylene in front of the cryogenic irradiation chamber. For calibrating the beam relative monitor, beam current was measured using a Faraday cup in front of the N0 beam line and associated with the number of events produced by n-p scattering. As a result, we obtained a proportional relationship between the count rate of events produced by n-p scattering and the beam current. The beam current at the beam dump was not used for counting the number of protons because about 60 % protons through the irradiation chamber were not bent to the beam dump correctly owing to energy loss of the beam and scattering in the chamber. As 200 MeV proton beam lost energy in the sample holder to materials such as AlN and aluminium plates on the beam line, the proton energies incident on the samples were estimated to be 185.3 ± 0.9 MeV for the aluminium wire and 195.5 MeV ± 0.5 MeV for the copper wire, as determined using PHITS.

**Experimental results**

*Electrical resistivity changes of aluminium and copper during irradiation*

Figure 2 shows the relationship between the electrical resistance and the temperature of the AlN plate during proton irradiation with an energy of 185 MeV for aluminium and 196 MeV for copper with average currents of 1.35, 2.0, and 3.0 nA. The temperature increase due to beam heating was 1.2 K, 1.5 K, and 2.0 K in the cases of the 1.35 nA, 2.0 nA, and 3.0 nA currents, respectively, and the temperature was maintained below 5 K during beam irradiation. The electrical resistances of the aluminium and copper samples increased during beam irradiation owing to the production of defects in the wires. The jump in resistance at the beam-on and beam-off times resulted mainly from the temperature increase and decrease due to beam heating. In case of the aluminium wire, the electrical resistance decreased during beam-off because the electrical resistance decreased with a decrease in temperature at around 5 K. By contrast, in case of the copper wire, the electrical resistance increased during beam-off because the relationship between the electrical resistance of the copper wire and temperature had a local minimum point at around 5 K (Iwamoto et al., 2018). Any magnetic impurities in the copper wire may affect the increase in its electrical resistance due to the Kondo effect (Kondo, 1964).
Figure 2. Relationship between electrical resistance and temperature of AlN plate during proton irradiation with energy of 185 MeV for aluminium (left) and 196 MeV for copper (right) with average currents of 1.35, 2.0, and 3.0 nA


**Displacement cross-sections**

The displacement cross-section can be related easily to the measured increase in resistivity and the calculated damage energy in the metal. The experimental displacement cross-section $\sigma_{\text{exp}}$ was obtained using the measured damage rate, which is the ratio of the change in resistivity of metal $\Delta \rho_{\text{metal}}$ at around 5 K to the beam fluence $\Phi$:

$$\sigma_{\text{exp}} = \frac{1}{\rho_{FP}} \frac{\Delta \rho_{\text{metal}}}{\Phi}$$

where $\rho_{FP}$ is the change in resistivity per Frenkel-pair density for a particular metal, and the experimental data of $\rho_{FP}$ were summarised and discussed in refs (Konobeyev et al., 2017; Broeders et al., 2004). In this work, $\rho_{FP}$ was set to 3.9 ± 0.6 m for aluminium (Ehrhart and Schilling, 1973) and 2.2 ± 0.5 $\mu\Omega$ m for copper (Haubold and Martinsen, 1978), and the same values were used for estimating the displacement cross section of the 125-MeV-proton-irradiation experiments (Iwamoto et al., 2015), the 1.1- and 1.94-GeV-proton-irradiation experiments (Greene et al., 2004), and the Jung data (Jung, 1983) for the sake of comparison with our experimental data. Figure 3 shows the experimental displacement cross-section data of aluminium and copper obtained in RCNP data (Iwamoto et al., 2018), FFAG data (Iwamoto et al., 2015), BNL data (Greene et al., 2004), and Jung data (Jung, 1983). It is assumed that the increase in resistivity is the sum of the resistivity per Frenkel pair (Norgett et al., 1975). The total uncertainty in experimental displacement cross-section was estimated to be 20% for aluminium and 26% for copper, which originates from the $\rho_{FP}$ error described in refs (Ehrhart and Schilling, 1973; Haubold and Martinsen, 1978) (15% for aluminium and 23% for copper). In terms of the displacement cross section of copper, the displacement cross-section for 125 and 200 MeV irradiation is similar to the values in the experimental data for 1.1 and 1.94 GeV irradiations. Figure 3 also shows the NRT-dpa and arc-dpa displacement cross sections calculated using PHITS. In terms of the cross sections of copper in the experimental data and those determined using arc-dpa, PHITS with the defect efficiency function of Nordlund shows better agreement with the
experimental data than the NRT-dpa. Further works on measuring other metals are in progress, which will allow us to study the basic physics of point defects in the high-energy region.

Figure 3. Displacement cross-sections for proton irradiation of aluminium (left) and copper (right): RCNP data (Iwamoto et al., 2018) (red circles), FFAG data (Iwamoto et al., 2015) (red triangle), BNL data (Greene et al., 2004) (black circles), Jung data (Jung, 1983) (open squares), NRT-dpa cross-section calculated using the NRT model (dashed line), and the arc-dpa cross section (solid line)


Summary

We developed a cryogenic proton irradiation device with a GM cryocooler to cool 0.25-mm-diameter wire samples. For the cooling test before irradiation, temperature of the sample holder was 3 K. Then, we installed the device into the beamline of the cyclotron facility at RCNP, Osaka University, and measured the change in electrical resistivity of aluminium and copper in relation to the displacement cross-sections by using 200 MeV proton irradiation with an intensity of 1-3 nA at around 5 K. After 200 MeV proton irradiation with 3.89 × 10^{18} protons/m², the damage rate of aluminium was 1.31 × 10^{-31} Ωm³/proton at 185 MeV and that of copper was 3.6 × 10^{-31} Ωm³/proton at 196 MeV. A comparison of the experimental displacement cross-sections with the calculated results of NRT-dpa and arc-dpa cross sections indicated that arc-dpa with the defect production efficiencies gave a better quantitative description of the displacement cross-section than NRT-dpa.

Acknowledgements

The authors wish to express their gratitude to Prof Dr M. Fukuda and the staff at RCNP for their generous support for beam operation. They also wish to thank to Prof Dr T. Nakamoto at KEK, Prof Dr T. Ogitsu at KEK and the staff at KEK and JAEA for their generous support for device development. This work was supported by JSPS KAKENHI Grant Number JP16H04638. The members of the DPA measurement team at RCNP were: Y. Iwamoto (JAEA), M. Yoshida (KEK), T. Yoshiie (KURNS), D. Satoh (JAEA), H. Yashima (KURNS), H. Matsuda (J-PARC/JAEA), S. Meigo (J-PARC/JAEA), and T. Shima (RCNP).
References


ELI Beamlines laser facility - Current status

David Horvath¹, Veronika Olsovcova¹, Vojtech Stransky¹, Roman Trunecek, Roberto Versaci¹, Nikhil Shetty¹, Sabrina Bechet¹, Leonel Morejon¹, Alberto Fassò¹
¹ELI Beamlines, Czech Republic
*roberto.versaci@eli-beams.eu

Extreme Light Infrastructure (ELI) is a European Research Infrastructure which benefits from the latest development in new generation laser technology to produce high intensity ultra-short laser pulses. "ELI Beamlines" is the Czech Republic based pillar, which aims at the development of high-brightness sources of X-rays and the acceleration of proton, electron and ion beams, to be used both for pure research and practical applications. Radiation fields generated by laser differ from conventionally generated fields in some characteristics, such as ultra-short pulse length and low repetition rates, which raise some challenges for the reliable implementation of radioprotection systems.

The commissioning of four beamlines is scheduled for early 2019, with gradual increase in source charge, energy and repetition rate. This contribution updates on the current status of the facility, the preparatory radioprotection works, and presents some preliminary results from the first experimental tests.

Introduction

ELI (Extreme Light Infrastructure) is a new Research Infrastructure of pan-European interest and part of the European ESFRI Roadmap. The facility is based on three sites presently being implemented in the Czech Republic, Hungary and Romania, with an investment volume exceeding 850 million Euro, mostly stemming from the European Regional Development Funds. ELI aims at hosting some of the most intense lasers worldwide, develop new interdisciplinary research opportunities with light from these lasers and secondary radiation derived from them.

ELI Beamlines is the Czech Republic based pillar of the ELI project. It mainly focuses on the development of short-pulsed secondary sources of X-rays, protons, and electrons, and on their multidisciplinary applications.

Lasers and particles beamlines

Lasers

ELI Beamlines will host four lasers with different characteristics.

L1 laser is being developed in house. It is designed to generate <20 fs pulses with energy exceeding 100 mJ per pulse at 1 kHz repetition rate. Its commissioning is on-going and is expected to end in 2019 (Batysta, 2014).

L2 laser is also being developed in house. Its goal is to generate <20 fs pulse with 2 J energy per pulse and 10 Hz repetition rate. Its development is scheduled to be completed by 2022.
L3 laser has been developed in co-operation with Lawrence Livermore National Laboratory, United States. Commissioning and beam transport system construction are ongoing. The system is designed to generate >30 J pulse in <30 fs pulses, so reaching 1 PW power, with a 10 Hz repetition rate.

L4 laser has been developed in co-operation with National Energetics, United States. The system is currently being assembled on site and its commissioning is scheduled to start in 2019. It is designed to reach a peak power of 10 PW with a pulse duration of about 130 fs with a maximum repetition rate of 1/60 Hz.

**Beamlines and source terms**

ELI Beamlines will be home to eight separate secondary radiation beamlines.

PXS (Plasma X-ray Source): laser pulses are focused on a solid target to produce plasma which then emits X-rays. This source will be able to emit photons up to 1 MeV. The commissioning of the PXS beamline is on-going.

E2: this source will produce betatron radiation exploiting the oscillation of electrons in the laser-generated plasma. The generated electron beam will have energy up to 2 GeV. The design of this beamline is currently being finalised.

P3 (Plasma Physics Platform): several lasers are very tightly focused on one point to obtain high energy densities plasmas. These plasmas then generate large divergence mixed radiation fields containing protons, electrons, and positrons up to 1-2 GeV and photons with a few tens of MeV (Weber, 2017). The P3 experimental set up is presently being assembled.

ELIMAIA (ELI Multidisciplinary Applications of laser-Ion Acceleration): as per its acronym, this beamline is dedicated to the ion acceleration and aims, among the other goals, at accelerating protons up to 200 MeV (Margarone, 2018). ELIMAIA has been assembled on site and the first commissioning test, using an alignment laser is scheduled for 2019.

LUIS (Laser Undulator Illuminating Source): aims at the production of X-rays pulses generated by laser-accelerated electrons passing through an undulator. This will produce electrons having energies up to 2 GeV. The design of this beamline is currently being finalised.

HELL (High energy ELectron acceleration by Laser): looks forward the development of record-breaking electron acceleration techniques and is expected within few years to be able to accelerate electrons up to 10 GeV (Levato, 2018). The beamline is being assembled on site.

TERESA (TEstbed for high Rereptition rate Source of Accelerated particles): is a small scale proof of concept for the acceleration mechanisms for protons (up to 15 MeV) and electrons (up to 100 MeV). TERESA commissioning is ready to start in first quarter of 2019.

S1: is a small scale version of photon (up to 30 keV) generation from laser-generated plasma and is currently in operation.

**Radiological protection assessment**

The radiological protection assessment for the ELI Beamlines facility has been performed by means of Monte Carlo simulations. The software of choice is FLUKA (Battistoni, 2017;
Ferrari, 2005), which is a recognised standard for radiological protection Monte Carlo simulations. The FLUKA graphical interface, FLAIR (Vlachoudis, 2009) has also been used. Radiation fields generated by individual beamlines have been characterised by means of simulations. Besides ambient dose equivalent rate maps, the possible problems due to the effects of the radiation on the electronics have been investigated, and the activation levels of the main experimental components have been estimated. Last but not least, the shielding design has been based on the FLUKA simulations, using an iterative approach. The obtained characteristics of the field, in terms of spectra and particle fluences, have been used to design Monitoring System, which is to monitor radiation levels and collect relevant radiation related information for the Personal Safety Interlock system.

A detailed FLUKA model of the whole laboratory building (size roughly 120 x 70 m x 30 m) has been created, to a total of more than 30 000 lines. The model includes detailed description of each penetration (very relevant for radiation leakage, especially in a facility designed to transport laser to each separate experimental hall) and of the devices and their materials. Whenever information were not available, a conservative radiological protection assumption has been made. The FLUKA model of the experimental floor is shown in Figure 1.

![FLUKA model of the experimental floor](source: ELI Beamlines, 2020)

Three examples of field characterisation have been presented at the SATIF-14 conference.

**PXS**

The FLUKA simulations of the PXS beamline were used for the design of a walk-in shielding hutch. The hutch has a double purpose: to allow personnel presence in the experimental hall while PXS is in operation and to allow PXS scientist to get close to the experimental set up during the preparatory and post-experiment phases without being impeded in their work by a too small shielding. Repeated simulations were used to identify the minimum amount on material necessary and to investigate the inner coating of the hutch to minimise the backscattering towards the detectors. Simulations have also provided an estimate of the radiation levels outside the hutch. The preliminary results from the first commissioning phase seem to confirm the simulation results. Figure 2 shows the simulated ambient dose equivalent rate map. Figure 3 shows a close up of the FLUKA model of the hutch, while Figure 4 shows a picture of the actual PXS hutch.
**TERESA**

The design shielding for the TERESA beamline was based on the results of many FLUKA simulations taking into account different beams types and energies, as well as the presence of a magnetic field. This shielding is very important since the hall will be shared between TERESA scientist and laser scientists requiring to work in the room during operations. The
implemented solution consists of a thicker concrete shielding in front, plus an additional leg on the side were the beam is bent by the magnetic field, plus a small local shielding for the very low energy particles. An example of the ambient dose equivalent rate map for a specific case is shown in Figure 5.

Figure 5. Ambient dose equivalent rate map in µSv/shot generated by a 40 MeV electron beam


P3

Because of the large divergence of the generated radiation fields and of the constraints due to presence of various devices in the experimental hall, the design of the shielding for the solid target experiment with the 10 PW laser in P3 has been quite challenging. One of the most interesting peculiarity is the size of the P3 experimental chamber (2.3 m radius, 3.3 m height) and the impossibility to locate the shielding close to it. Various options have been considered and finally it has been decided to install a massive 4 m long shielding, to reduce the amount of radiation impinging on technology installations (activation concerns) and also the radiation entering the adjacent hall. Figure 6 shows the ambient dose equivalent rate map obtained with FLUKA.

Figure 6. Ambient dose equivalent rate map in mSv/shot generated by the interaction of L4 with a solid target

Conclusions

A short update of the current status of the facility has been presented, together with few examples of the simulation work performed. The simulations work for the shielding design is fundamental as it allows to compare the effectiveness of different possible designs for various experimental configurations. The physical realisation of the designed shielding is currently ongoing; some are in engineering design phase, while other are already installed and used. Unfortunately, it was not possible to present preliminary experimental results from the commissioning phase, as the experimental data are still under evaluation.

Acknowledgements

The results of the Project were obtained with the financial support of the Ministry of Education, Youth and Sports within targeted support of large infrastructures.

References


