Operational Radiation Safety Program

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National Council on Radiation Protection and Measurements
7941 Woodmont Avenue, Suite 400, Bethesda, Maryland 20814
## Contents

1. Introduction ......................................................................................................................... 10
   1.1 Purpose and Scope of this Report ................................................................................. 10
   1.2 Intended audience ......................................................................................................... 11

2. Updated NCRP Recommendations on Dose Control .......................................................... 12
   2.1 Numeric Protection Criteria (Dose Limits) ................................................................. 12
   2.2 Optimization of Protection (ALARA) ......................................................................... 15
   2.3 Negligible Individual Dose (NID) ............................................................................... 18
   2.4 Exposure to the Public from Facility Operations ......................................................... 18
      2.4.1 Standards and Guidance .................................................................................. 19
      2.4.2 Types of Exposures to Members of the Public ............................................... 19
   2.5 Exposure to Nonhuman Biota ..................................................................................... 20

3. Radiation Safety Program Roles and Responsibilities ......................................................... 22
   3.1 Senior Management ...................................................................................................... 22
      3.1.1 Safety Culture ................................................................................................. 23
      3.1.2 Safety Management System ............................................................................ 24
   3.2 Radiation Safety Officer and Staff ................................................................................ 25
      3.2.1 Radiation Safety Program Procedures ............................................................ 29
      3.2.2 The Regulatory Environment .......................................................................... 30
   3.3 Ancillary Staff ............................................................................................................... 30
      3.3.1 Radiation Safety Committee ........................................................................... 30
      3.3.2 Occupational Medicine ................................................................................... 31
      3.3.3 Legal Support .................................................................................................. 32
      3.3.4 Radiological Training ..................................................................................... 32
   3.4 Supervisors and managers ............................................................................................. 32
   3.5 Radiation workers ......................................................................................................... 32

4. Facility and System Design Considerations ........................................................................ 34
   4.1 Selection of Locations for New or Modified Facilities ................................................. 34
   4.2 Facility Layout .............................................................................................................. 35
   4.3 Equipment and System Design ................................................................................... 36
48  4.4 Shielding ....................................................................................................................... 37
49  4.5 Ventilation .................................................................................................................. 39
50  4.6 Decontamination Considerations .......................................................................... 40
51  4.7 Electrical Design for Instrumentation, Access Control, and Alarm Systems ... 41
52  5. Radiological Work Controls ...................................................................................... 42
53       5.1 Hierarchy of Controls ......................................................................................... 42
54       5.2 Planning Radiological Work ............................................................................. 42
55           5.2.1 Work Scope and Hazard Identification .................................................. 44
56           5.2.2 Dose Management ..................................................................................... 44
57           5.2.3 Safety Procedures and Radiation Work Permits .................................... 46
58           5.2.4 Work Authorization .................................................................................. 50
59           5.2.5 Pre-job briefings ....................................................................................... 50
60  5.3 Radiation Safety Training ......................................................................................... 51
61           5.3.1 Management Responsibilities ................................................................... 51
62           5.3.2 Factors to Consider when Establishing Training Requirements ................ 52
63           5.3.3 Personnel to be Trained ............................................................................. 52
64           5.3.4 Design and Development of a Radiation Safety Training Program ....... 55
65  5.4 Interlocks and Access Control Systems ................................................................. 56
66  5.5 Posting and Labeling ............................................................................................... 59
67           5.5.1 Posting of Areas .......................................................................................... 59
68           5.5.2 Labeling of Items ......................................................................................... 60
69  5.6 Radioactive Material Operations .............................................................................. 61
70           5.6.1 Management of Radioactive Material ...................................................... 62
71           5.6.2 Management of Sealed Radioactive Sources ........................................... 63
72           5.6.3 Dispersible Radioactive Material Controls ................................................ 65
73           5.6.4 Contamination Control ............................................................................... 69
74  5.7 RGD Operations ........................................................................................................ 71
75           5.7.1 Types of RGDs .............................................................................................. 72
76           5.7.2 Management of RGDs ................................................................................... 75
77           5.7.3 RGD Workplaces .......................................................................................... 77
8. Personnel Dosimetry

8.1 External Dosimetry

8.1.1 Criteria for providing dosimeters to workers

8.1.2 Use of electronic dosimeters/real-time monitoring

8.1.3 Accreditation programs (NVLAP, DOELAP)

8.1.4 Area monitoring dosimeter programs

8.1.5 Nuclear accident dosimeters for workers and areas

8.1.6 External Dose Assessments

8.2 Internal Dosimetry

8.2.1 Direct (in vivo) monitoring

8.2.2 Indirect (bioassay) monitoring

8.2.3 Bioassay Program Criteria

8.2.4 Accreditation programs (DOELAP)

8.2.5 Internal Dose Assessment

9. Control of Exposure to the Public

9.1 Determining the Need for Surveillance

9.2 Radiological Effluent Monitoring and Environmental Surveillance

9.2.1 Monitoring Radioactive Airborne Effluents

9.2.2 Monitoring Liquid Effluents

9.2.3 Solid Waste Effluents

9.3 Environmental Surveillance

9.3.1 Preoperational Monitoring

9.3.2 Operational Monitoring

9.4 Measurement Methods

9.5 Dose Assessment

9.6 Reporting

10. Response to Spills and Emergencies

10.1 Spill Response
<table>
<thead>
<tr>
<th>Page</th>
<th>Section</th>
</tr>
</thead>
<tbody>
<tr>
<td>138</td>
<td>10.1.1</td>
</tr>
<tr>
<td>139</td>
<td>10.1.2</td>
</tr>
<tr>
<td>140</td>
<td>10.2</td>
</tr>
<tr>
<td>141</td>
<td>10.2.1</td>
</tr>
<tr>
<td>142</td>
<td>10.2.2</td>
</tr>
<tr>
<td>143</td>
<td>10.2.3</td>
</tr>
<tr>
<td>144</td>
<td>10.2.4</td>
</tr>
<tr>
<td>145</td>
<td>11.</td>
</tr>
<tr>
<td>146</td>
<td>11.1</td>
</tr>
<tr>
<td>147</td>
<td>11.2</td>
</tr>
<tr>
<td>148</td>
<td>11.2.1</td>
</tr>
<tr>
<td>149</td>
<td>11.2.2</td>
</tr>
<tr>
<td>150</td>
<td>11.2.3</td>
</tr>
<tr>
<td>151</td>
<td>11.2.4</td>
</tr>
<tr>
<td>152</td>
<td>11.3</td>
</tr>
<tr>
<td>153</td>
<td>11.4</td>
</tr>
<tr>
<td>154</td>
<td>12.</td>
</tr>
<tr>
<td>155</td>
<td>12.1</td>
</tr>
<tr>
<td>156</td>
<td>12.2</td>
</tr>
<tr>
<td>157</td>
<td>12.3</td>
</tr>
<tr>
<td>158</td>
<td>12.4</td>
</tr>
<tr>
<td>159</td>
<td>12.5</td>
</tr>
<tr>
<td>160</td>
<td>12.6</td>
</tr>
<tr>
<td>161</td>
<td>12.7</td>
</tr>
<tr>
<td>162</td>
<td>Appendix A: Suggested Topics for Radiation Safety Training (NCRP 2000b)</td>
</tr>
<tr>
<td>163</td>
<td>Appendix B: Example of the use of the HotSpot Code for Workplace Selection</td>
</tr>
<tr>
<td>164</td>
<td>Appendix C: Supplemental Information Specific to Different Operational Settings</td>
</tr>
<tr>
<td>165</td>
<td>C.1</td>
</tr>
<tr>
<td>166</td>
<td>C.1.1</td>
</tr>
<tr>
<td>167</td>
<td>C.1.2</td>
</tr>
<tr>
<td>Page</td>
<td>Section</td>
</tr>
<tr>
<td>------</td>
<td>---------</td>
</tr>
<tr>
<td>168</td>
<td>C.1.3 Radioactive Material Licensing and Use</td>
</tr>
<tr>
<td>169</td>
<td>C.1.4 Radiopharmaceutical Therapy</td>
</tr>
<tr>
<td>170</td>
<td>C.1.5 Medical Events</td>
</tr>
<tr>
<td>171</td>
<td>C.1.6 Institutional Review Board</td>
</tr>
<tr>
<td>172</td>
<td>C.2 University/Research Facilities</td>
</tr>
<tr>
<td>173</td>
<td>C.2.1 Organization and Administration of the Radiation Safety Program</td>
</tr>
<tr>
<td>174</td>
<td>C.2.2 Facility and System Design Considerations</td>
</tr>
<tr>
<td>175</td>
<td>C.2.3 Work Planning, Audits, and Training</td>
</tr>
<tr>
<td>176</td>
<td>C.2.4 Emergency Planning</td>
</tr>
<tr>
<td>177</td>
<td>C.2.5 Decommissioning</td>
</tr>
<tr>
<td>178</td>
<td>C.3 Biotechnology Facilities</td>
</tr>
<tr>
<td>179</td>
<td>C.3.1 Principal Radiological Hazards</td>
</tr>
<tr>
<td>180</td>
<td>C.3.2 Facility Design Requirements</td>
</tr>
<tr>
<td>181</td>
<td>C.3.3 Radioactive Waste handling and Disposal</td>
</tr>
<tr>
<td>182</td>
<td>C.4 National Laboratories</td>
</tr>
<tr>
<td>183</td>
<td>C.4.1 Radioactive Material and RGD Use</td>
</tr>
<tr>
<td>184</td>
<td>C.4.2 Administration of the Radiation Safety Program</td>
</tr>
<tr>
<td>185</td>
<td>C.4.3 Principal Radiological Hazards and Facility Design Considerations</td>
</tr>
<tr>
<td>186</td>
<td>C.4.4 Radiological Monitoring Equipment</td>
</tr>
<tr>
<td>187</td>
<td>C.4.5 Internal and External Dosimetry</td>
</tr>
<tr>
<td>188</td>
<td>C.4.6 Radiological training</td>
</tr>
<tr>
<td>189</td>
<td>C.4.6 Effluent Monitoring and Radioactive Waste Handling</td>
</tr>
<tr>
<td>190</td>
<td>C.5 Commercial Nuclear Power Reactors</td>
</tr>
<tr>
<td>191</td>
<td>C.5.1 Radiation Safety Program</td>
</tr>
<tr>
<td>192</td>
<td>C.5.2 Facility Design</td>
</tr>
<tr>
<td>193</td>
<td>C.5.3 Radioactive Material and Controls</td>
</tr>
<tr>
<td>194</td>
<td>C.5.4 Work Planning</td>
</tr>
<tr>
<td>195</td>
<td>C.5.5 Training</td>
</tr>
<tr>
<td>196</td>
<td>C.5.6 Personnel Monitoring</td>
</tr>
<tr>
<td>197</td>
<td>C.5.7 Posting and Labeling</td>
</tr>
</tbody>
</table>
C.5.8 ALARA ........................................................................................................... 212

C.6 Nuclear Gauging Devices .................................................................................. 212
C.6.1 Radiological Hazards ....................................................................................... 214
C.6.2 Radiation Safety Program ............................................................................... 215
C.6.3 Facility Design ................................................................................................ 216
C.6.4 Radiation Monitoring Equipment .................................................................. 216
C.6.5 Physical security monitoring and surveillance .............................................. 216
C.6.6 Dosimetry ....................................................................................................... 217
C.6.8 Training ........................................................................................................... 217
C.6.9 Radioactive Waste Handling and Disposal ..................................................... 217

C.7 Industrial Radiography ...................................................................................... 218
C.7.1 Radiographic Sources ...................................................................................... 220
C.7.2 Hazards ............................................................................................................ 220
C.7.3 Radiation Safety Controls ............................................................................... 222
C.7.4 Equipment ....................................................................................................... 222
C.7.5 Training ........................................................................................................... 222
C.7.6 RSO Duties ...................................................................................................... 223
C.7.7 Operating and Emergency Procedures ............................................................ 223
C.7.8 Personnel Monitoring ..................................................................................... 224
C.7.9 Surveys ............................................................................................................ 224

C.8 Irradiator Facilities ............................................................................................. 224
C.8.1 Facility Design ................................................................................................ 225
C.8.2 Monitoring Requirements ............................................................................... 226
C.8.3 Access Controls ............................................................................................... 226
C.8.4 Security monitoring and surveillance .............................................................. 226
C.8.5 Dosimetry ....................................................................................................... 227
C.8.6 Training ........................................................................................................... 227
C.8.7 Effluent Monitoring ......................................................................................... 227
C.8.8 Radioactive Waste Handling and Disposal ..................................................... 227

C.9 Large Accelerator Facilities ................................................................................ 227
<table>
<thead>
<tr>
<th>Page</th>
<th>Section</th>
</tr>
</thead>
<tbody>
<tr>
<td>228</td>
<td>C.9.1 Types of Accelerators and Types of Radiation Produced</td>
</tr>
<tr>
<td>229</td>
<td>C.9.2 Specialized Categories of Large Research Facilities</td>
</tr>
<tr>
<td>230</td>
<td>C.9.3 Types of Radioactive Materials Present</td>
</tr>
<tr>
<td>231</td>
<td>C.9.4 Principal Radiological Hazards</td>
</tr>
<tr>
<td>232</td>
<td>C.9.5 Facility Design Considerations</td>
</tr>
<tr>
<td>233</td>
<td>C.9.6 Radiation Monitoring</td>
</tr>
<tr>
<td>234</td>
<td>C.9.7 Training</td>
</tr>
<tr>
<td>234</td>
<td>C.9.8 Effluent monitoring and radioactive waste management</td>
</tr>
<tr>
<td>235</td>
<td>C.10 Oil and Gas Industry</td>
</tr>
<tr>
<td>236</td>
<td>C.11 Decommissioning/Cleanup Sites</td>
</tr>
<tr>
<td>237</td>
<td>C.11.1 Regulatory Guidance and Cleanup Criteria</td>
</tr>
<tr>
<td>238</td>
<td>C.11.2 Types of Cleanup Sites</td>
</tr>
<tr>
<td>239</td>
<td>C.11.3 Phases of Decommissioning</td>
</tr>
<tr>
<td>240</td>
<td>C.11.4 Facility Historical Site Assessment and Characterization Surveys</td>
</tr>
<tr>
<td>241</td>
<td>C.11.5 Work control</td>
</tr>
<tr>
<td>242</td>
<td>C.11.6 Background Reference Areas and Instrumentation</td>
</tr>
<tr>
<td>243</td>
<td>C.11.7 Remedial Action Survey Support</td>
</tr>
<tr>
<td>244</td>
<td>C.11.8 Waste Management</td>
</tr>
<tr>
<td>246</td>
<td>Abbreviations, Acronyms, and Symbols</td>
</tr>
<tr>
<td>247</td>
<td>Glossary</td>
</tr>
<tr>
<td>248</td>
<td>References</td>
</tr>
<tr>
<td>249</td>
<td></td>
</tr>
</tbody>
</table>
1. Introduction

The objective of an operational radiation safety\(^1\) program is to enable radiological work to proceed safely and compliantly. Institutions and organizations that use nonexempt quantities of radioactive materials and ionizing radiation generating devices (hereafter referred to as ‘RGDs’) are required by their regulatory authority to specify the policies and practices necessary to control radiation exposure to their employees and the public within the prescribed dose limits and to levels that are as low as reasonably achievable (ALARA). These requirements are generally implemented through the operational radiation safety program.

A radiation safety program consists of preoperational and operational components. The preoperational components generally include site-selection, facility design, the design of equipment and shielding, and licensing requirements. The operational component generally addresses all aspects of ongoing operations (e.g., planning, execution, oversight, facility modifications). Although this Report focuses primarily on operational radiation safety, it does provide some general guidance for preoperational activities.

1.1. Purpose and Scope of this Report

The primary purposes of this Report are to describe the essential elements, basic principles, and practices that comprise an operational radiation safety program, and to update the guidance provided in NCRP Report No. 127, *Operational Radiation Safety Program*. Note that nonionizing radiation (e.g., laser safety) and radiation safety associated with space travel are outside the scope of this Report.

Since Report No. 127 was published in 1998, the NCRP has published 19 additional reports covering radiation safety in specific settings such as veterinary medicine (NCRP 2004c) and educational institutions (NCRP 2007) or impacting specific aspects of operational radiation safety. Where appropriate, these reports have been summarized or referenced in this Report. Of

\(^1\) Except when referencing text from other NCRP reports, this Report uses the term ‘radiation safety’ (as opposed to ‘radiation protection’ or ‘radiological control’), as ‘radiation safety’ is generally considered a broader term inclusive of elements such as training, safety culture, and work control processes.
particular note is NCRP Report No. 180, *Management of Exposure to Ionizing Radiation: Radiation Protection Guidance for the United States* (NCRP 2018), wherein the NCRP updated its recommendations for limiting and managing exposure to ionizing radiation. In Report No. 180, NCRP recommends numeric protection criteria that are suitable for use as annual and cumulative regulatory limits for occupational exposure of individuals. These numeric protection criteria are discussed in Section 2.

The essential elements of a radiation safety program are described in Sections 3 through 10. These essential elements are appropriate for all radiation safety programs and should be applied in a graded approach, commensurate with the hazards presented by radiological operations. Appendix C provides an expansion of Report No. 127 in that it presents unique considerations for establishing and implementing operational radiation safety programs in specific occupational settings. This Report also includes a comprehensive reference list, acronym list, and a glossary.

This Report discusses specific aspects of operational radiation safety and references more detailed information in other NCRP reports, publications of the International Commission on Radiological Protection (ICRP), and other consensus bodies such as the American National Standards Institute (ANSI). This Report is not intended to be a design manual (e.g. for radiation shielding or ventilation systems), however, it does provide references for those topics. This Report also does not specifically address the myriad of regulatory and licensing requirements that may be imposed on a radiation safety program by state, local or federal authorities, but it does address regulatory requirements in several areas.

1.2. **Intended Audience**

This Report is intended for individuals with responsibility for establishing and implementing an operational ionizing radiation safety program. These individuals are assumed to have some formal education, training, and experience in radiation safety. The Report provides guidance for the development of new radiation safety programs and serves as a useful tool for assessing mature radiation safety programs. This Report may also be useful to the management of an organization or institution regarding the basic elements and requirements of an operational radiation safety program (hereafter referred to as a “radiation safety program”).
2. Updated NCRP Recommendations on Dose Control

The guidelines in NCRP Report No. 180, Management of Exposure to Ionizing Radiation: Radiation Protection Guidance for the United States (2018), are “designed to prevent the occurrence of acute and chronic radiation-induced tissue reactions (deterministic effects) in humans, and to reduce the probability of stochastic effects (primarily cancer) in radiation-exposed persons while maintaining the benefits to the individual and to society from the activities that generate such exposures” (NCRP 2018).

The recommendations contained in Report No. 180 implement the basic radiation safety principles of justification, optimization of protection and numeric protection criteria (for the management of dose to an individual). The actions to add, increase, reduce or remove a source of exposure to humans require justification (i.e., the action does more good than harm). Optimization of protection is applied in all exposure situations to manage doses to be well below the numeric protection criteria. Numeric protection criteria are derived for specific exposure situations characterized by the nature of the ionizing radiation source, the individuals exposed, the circumstances of exposure and the ability of those with authority to control both the source and the actions of persons at risk of exposure.

2.1. Numeric Protection Criteria (Dose Limits)

NCRP Report No. 180 (NCRP 2018) recognizes that nonoccupational radiation exposure should not be governed by “dose limits” when the source of exposure is outside the control of the regulating authority (e.g., following an accident such as occurred at Chernobyl and Fukushima) and when control of the exposed individuals is not practicable (e.g., the general public or workers outside areas subjected to radiation safety controls). Hence, it provides “numeric protection criteria” for several classes of exposure. However, when sources of radiation exposure are subject to preplanning and control, as is true with occupational exposure, Report No. 180 states these criteria are suitable for use as regulatory limits. Table 2.1 summarizes the numeric protection criteria for human exposure as recommended in Report No. 180. This Report refers to the numeric protection criteria for occupational exposures as dose limits.
### Table 2.1—Numeric protection criteria (for management of dose to an individual).

<table>
<thead>
<tr>
<th>Exposure Situation</th>
<th>Numeric Protection Criterion (mSv)$^b$ (effective dose, except where noted)</th>
<th>Suitable for Application as a Regulatory Dose Limit?</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Occupational Exposure—Stochastic Effects</strong></td>
<td></td>
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</tr>
<tr>
<td>Annual (&gt;18 y of age)</td>
<td>Should not exceed 50 mSv</td>
<td>Yes</td>
</tr>
<tr>
<td>Cumulative (&gt;18 y of age)</td>
<td>Should not exceed 10 mSv times current age in years</td>
<td>— $^c$</td>
</tr>
<tr>
<td>Individual under 18 y of age</td>
<td>Should not exceed 1 mSv y$^{-1}$</td>
<td>Yes</td>
</tr>
<tr>
<td>Embryo or fetus of pregnant worker following declaration of pregnancy</td>
<td>Should not exceed 0.5 mSv per month (equivalent dose in the embryo or fetus)$^d$</td>
<td>Yes</td>
</tr>
<tr>
<td>NORM (including TENORM)$^e$</td>
<td>If action is necessary, include with the annual value of 50 mSv</td>
<td>No</td>
</tr>
<tr>
<td>Radon in the workplace</td>
<td>Include if activity concentrations in air $&gt;$300 Bq m$^{-3}$ after application of radon mitigation measures</td>
<td>No</td>
</tr>
<tr>
<td><strong>Tissue Reactions</strong></td>
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<tr>
<td>Skin and extremities</td>
<td>Should not exceed 0.5 Gy (absorbed dose$^f$ in skin or extremities per year, averaged over the most highly exposed 10 cm$^2$ of skin</td>
<td>Yes</td>
</tr>
<tr>
<td>Lens of the eye</td>
<td>Should not exceed 50 mGy (absorbed dose$^f$ in lens of the eye) per year</td>
<td>Yes</td>
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<tr>
<td><strong>Public Exposure—Stochastic Effects</strong></td>
<td></td>
<td></td>
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<tr>
<td>Source is stable, characterized, and subject to an advance control program</td>
<td>Should not exceed 1 mSv y$^{-1}$ (per source)</td>
<td>Yes</td>
</tr>
<tr>
<td>Radioactive material not previously subject to control$^g$</td>
<td></td>
<td></td>
</tr>
<tr>
<td>First year (after identification of situation)</td>
<td>Should not exceed 20 mSv</td>
<td>No</td>
</tr>
<tr>
<td>Exposure Situation</td>
<td>Numeric Protection Criterion (mSv)</td>
<td>Suitable for Application as a Regulatory Dose Limit?</td>
</tr>
<tr>
<td>--------------------------------------------</td>
<td>-----------------------------------</td>
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<tr>
<td>Later years</td>
<td>Further reduce dose using optimization of protection</td>
<td>No</td>
</tr>
<tr>
<td>Radon in dwellings</td>
<td>For activity concentration in air &gt;300 Bq m⁻³, take mitigative measures</td>
<td>No</td>
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</tbody>
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**Tissue Reactions**

<table>
<thead>
<tr>
<th>Tissue Reactions</th>
<th>Numeric Protection Criterion (mGy)</th>
<th>Suitable for Application as a Regulatory Dose Limit?</th>
</tr>
</thead>
<tbody>
<tr>
<td>Skin and extremities</td>
<td>Same as for occupational exposure; should not exceed 0.5 Gy (absorbed dose in skin or extremities per year, averaged over the most highly exposed 10 cm² of skin)</td>
<td>Yes</td>
</tr>
<tr>
<td>Lens of the eye</td>
<td>Should not exceed 15 mGy (absorbed dose in lens of the eye) per year, applied for each exposure situation under the control of a responsible organization</td>
<td>Yes</td>
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</tbody>
</table>

**Medical Exposure**

<table>
<thead>
<tr>
<th>Medical Exposure</th>
<th>Numeric Protection Criterion (mSv)</th>
<th>Suitable for Application as a Regulatory Dose Limit?</th>
</tr>
</thead>
<tbody>
<tr>
<td>Comforters and caregivers</td>
<td>Should not exceed 5 mSv per episode</td>
<td>No</td>
</tr>
<tr>
<td>Other patients, visitors, and facility staff considered members of the public</td>
<td>Should not exceed 1 mSv y⁻¹</td>
<td>No</td>
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<tr>
<td>Biomedical research subjects</td>
<td>See Table 5.1 in Report No. 180</td>
<td>No</td>
</tr>
</tbody>
</table>

**Exposure of Emergency Workers**

<table>
<thead>
<tr>
<th>Exposure of Emergency Workers</th>
<th>Numeric Protection Criterion (mSv)</th>
<th>Suitable for Application as a Regulatory Dose Limit?</th>
</tr>
</thead>
<tbody>
<tr>
<td>During lifesaving activities or actions to prevent a catastrophic situation. Includes other urgent rescue activities</td>
<td>Managed by a decision dose of 0.5 Gy (cumulative whole-body absorbed dose), implemented at the command level</td>
<td>No</td>
</tr>
<tr>
<td>For other emergency activities, including extended activities following initial lifesaving, rescue, and damage control response</td>
<td>Should not exceed 100 mSv for the duration of the emergency operation; reduce using optimization of protection</td>
<td>No</td>
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</tbody>
</table>

\[a\]For exposure of nonhuman biota, the guideline ([Report 180] Section 5.6) is that assessment of the radiological impact on an ecosystem based upon a daily absorbed-dose rate (in tissue of nonhuman biota) is not necessary at absorbed-dose rates <0.1 mGy d⁻¹. Where existing conditions or proposed actions result in exposures at or above this absorbed-dose rate, additional evaluation may be considered necessary. The 0.1 mGy d⁻¹ absorbed-dose rate is not to be viewed as a target for remediation, nor do absorbed-dose
Exposure Situation | Numeric Protection Criterion (mSv)$^b$ (effective dose, except where noted) | Suitable for Application as a Regulatory Dose Limit?
--- | --- | ---

rates at or above this value indicate that harm is occurring.

$^b$In all cases, the phrase “should not exceed” conveys that the first objective for management of dose to an individual is to meet the applicable numeric protection criterion, and then to apply optimization of protection. The phrase “should not exceed” is not intended to mean that the value is suitable as a regulatory dose limit. NCRP recognizes: (1) that there may be exposure situations in which initial doses to individuals are greater than the applicable numeric protection criterion, and (2) that the values are not a boundary between safe and unsafe. Suitability of a numeric protection criterion as a regulatory dose limit is indicated in the last column by “Yes” or “No.”

$^c$NCRP acknowledges that, in practice, the costs and logistics of tracking doses may make cumulative lifetime recording difficult. In concept, when the costs and logistics of tracking doses are feasible, this cumulative lifetime recommendation would be suitable for application as a regulatory dose limit.

$^d$Situations in which a worker who has declared her pregnancy may be exposed to radioiodine should be minimized or avoided if possible because of the risk of congenital hypothyroidism.

$^e$NORM = naturally occurring radioactive material TENORM = technologically enhanced NORM

$^f$If it is necessary to apply this recommendation to high-LET radiation, NCRP recommends that the absorbed dose in the skin or extremities, or the lens of the eye, should be multiplied by the biological effectiveness of the high-LET radiation as described in [Report No.180] Section 5.2.2.1 and Appendix A.2.

$^g$For example: elevated levels of NORM; environmental contamination.

$^h$Numeric protection criteria do not apply to medical exposure of patients.

$^i$See definition of “episode” in [Report No. 180] Section 5.4.2.

$^j$A suggested classification scheme is provided in Section 5.4.3 for use of effective dose as a qualitative indicator of radiation detriment for balancing against expected individual or societal benefit for biomedical research studies.

$^k$See [Report No. 180] Section 5.5.2.

2.2. Optimization of Protection (ALARA)

The basic radiation safety assumptions and objectives recommended by the NCRP are given in Report No. 180 (NCRP 2018). In Report No. 180, the concept of optimization largely encompasses and replaces what has historically been referred to as the ALARA philosophy$^2$. The concept of “Optimization of Protection” (also referred to as “Optimization”) is described as follows:

“The likelihood of incurring exposures, the number of individuals exposed, and the magnitude of the dose to an individual should be kept as low as reasonably achievable, taking into account societal, economic and environmental factors. More generally, optimization of

$^2$ While acknowledging that the term “optimization” is more encompassing than “ALARA,” this Report largely uses the term ALARA because ALARA is integral to the nomenclature of current radiation safety programs.

“OPTIMIZATION OF PROTECTION IS FUNDAMENTALLY EXAMINING ALL ASPECTS OF AN EXPOSURE SITUATION TO DETERMINE IF PROTECTION CAN REASONABLY BE IMPROVED AND IS DIRECTLY RELATED TO REDUCING HARM WHILE MAINTAINING THE BENEFITS THAT ARE PROVIDED. OPTIMIZATION OF PROTECTION SHOULD BE APPLIED TO EACH PARTICULAR SOURCE BY THE INDIVIDUAL OR ORGANIZATION RESPONSIBLE FOR MANAGING EXPOSURE TO THAT SOURCE. THE PROCESS INCLUDES ASPECTS THAT ARE BOTH RELATED TO THE SOURCE, AND RELATED TO THE INDIVIDUALS THAT ARE EXPOSED, WITH PREFERENCE GIVEN TO EXERCISING CONTROLS ON THE SOURCE BEFORE CONTROLS THAT ARE APPLIED TO INDIVIDUALS. IF THERE ARE EXPOSURES THAT ARE GREATER THAN THE RECOMMENDED NUMERIC PROTECTION CRITERIA (NCRP 2018 SECTION 3.3.3), IT IS CRITICAL THAT PROTECTION BE IMPROVED IN AN EFFORT TO SATISFY THE RECOMMENDED CRITERIA. THE NUMERIC PROTECTION CRITERIA SERVE TO AVOID PROTECTION OUTCOMES THAT ARE INEQUITABLE, AND THUS SERVE AS A BOUNDARY FOR JUDGING THE ACCEPTABILITY OF THE PROTECTION OPTIONS AVAILABLE. THE DEGREE OF BALANCE BETWEEN IMPROVING PROTECTION AND THE SOCIETAL, ECONOMIC AND ENVIRONMENTAL FACTORS IS PRESUMPTIVELY NOT ACHIEVED WHEN DOSES ARE ABOVE THE RECOMMENDED NUMERIC PROTECTION CRITERIA.

NOTE THAT NUMERIC PROTECTION CRITERIA DO NOT APPLY TO MEDICAL EXPOSURE OF PATIENTS OR EXPOSURE TO UBQUITOUS BACKGROUND RADIATION (WITH THE EXCEPTION OF ELEVATED LEVELS OF RADON IN DWELLINGS AND THE WORKPLACE, AND SOLAR AND COSMIC RADIATION IN CERTAIN OCCUPATIONAL CIRCUMSTANCES). THE ENTIRE PROCESS OF IMPROVING PROTECTION IS REFERRED TO AS OPTIMIZATION OF PROTECTION.

“More often, particularly in continuing work with a source, optimization of protection will occur through the pragmatic continued application of good radiation safety practice, and the continuation of a good radiation safety culture that asks if something can be improved, and if it is reasonable to make that improvement (NCRP 2018, Section 3.4). Stakeholders, such as occupationally exposed workers, are often in the best position to know how to do things better and need to actively participate in their own safety. However, optimization of protection is not minimization of exposure. There is always the consideration of what is reasonable in the context of all the hazards presented in an exposure situation. Hence, what constitutes “reasonably achievable” is related to costs associated with radiation safety (including those resulting from negative health or environmental outcomes) as well as societal good that is achievable.

“It is important to note that selection of the optimal protection solution will generally be related to the specific exposure situation and is most often a judgment rather than a quantitative result. Such judgments are informed by the science and engineering of an all-hazards approach, but are driven by the societal, institutional, economic and human values of those involved in decision-making processes. The outcome of the optimization of protection will be unique to the exposure situation, and the particular circumstances therein. The outcome of optimization of protection is not to be considered as a regulatory dose limit.

“All exposures that may occur should be considered, including both from sources internal and external to the body. Furthermore, other hazards that may be present are also considered, so that protection is afforded for all hazards. In addition, implementation of the optimization of protection principle should consider not only exposure that is expected or anticipated to occur but should also include consideration of less likely, but plausible, exposure that might occur due to unintended or unplanned circumstances or accidents. In any situation, there is the possibility that the activities that are planned will not proceed as expected, and it is important to include reasonable contingencies to avoid unintended or excessive exposures. For example, it is reasonable to expect that a source might not return to its shielded position, and plans need to be in place to respond to such an incident. During the design, construction, installation and operation of a new source of radiation, a formal assessment of the risks may be necessary to reasonably anticipate failures of safety systems and processes and provide additional measures.”
Although most of the optimization decisions regarding facility design will have been made during the planning and design phase, constant awareness and attention is essential during operations to avoiding unnecessary exposures. Perhaps the most important approach to achieving optimization of protection is to encourage a “safety culture” where managers, supervisors and workers are encouraged to routinely question whether a particular level of exposure is necessary. Supervisors and managers should remain open and actively solicit worker input because they are often in an ideal position to offer practical suggestions relative to dose optimization.

2.3. Negligible Individual Dose (NID)

As specified in NCRP Report No. 180 (2018), “NCRP continues to recommend a negligible individual dose of 0.01 mSv annual effective dose per source or practice as a guide for evaluating when efforts to reduce further the dose to an individual may not be warranted. This recommendation is based on a judgment by NCRP that under virtually any exposure situation the radiological risks engendered by an annual effective dose of 0.01 mSv can reasonably be considered as not warranting further reduction efforts to achieve optimization of protection. NCRP also recognizes that the organization responsible for implementing the optimization of protection principle for a specific exposure situation, with stakeholder interaction, may decide to apply a different approach or reach a different conclusion.” For example, 0.1 mSv annual effective dose may be the de facto NID in facilities where, due to the type of dosimeter in use, the minimum recordable dose is 0.1 mSv.

2.4. Exposure to the Public from Facility Operations

NCRP Report No. 180 recommends a numeric protection criterion for a source of exposure to control the dose to members of the public, who may be exposed to sources of ionizing radiation produced or enhanced by human activities. Other limitations on public exposure may be specified in the license or registration under which a facility operates or by other legal authorities. Important goals for the radiation safety staff are to ensure that public exposures due to operation of the facility meet these exposure limits and, further, that public exposures are ALARA.
These goals are achieved by controlling radiation fields emanating within and from the facility and by controlling releases from the facility to the environment. Control begins with proper equipment and facility design and includes appropriate procedures for operation within the design envelope. Assurance that appropriate levels of control have been achieved is frequently provided by environmental monitoring programs (see Section 9.3 and NCRP 2010c). The size and complexity of the monitoring program depend upon the types and quantities of radionuclides present in the facility and the radiation fields that its operation generates.

2.4.1. Standards and Guidance

The NCRP recommends that ongoing exposure of members of the public be limited to an annual effective dose of 1 mSv per source (NCRP 2018). For individuals exposed infrequently, NCRP recommends a maximum annual effective dose limit of 5 mSv (NCRP 2018). These recommendations exclude exposures from natural background radiation and radiation exposure associated with medical diagnosis and treatment of individual patients. The total estimated effective dose equivalent from both internal and external exposure must be less than the stated limit and should be kept ALARA. Note that the limit applies to the sum of all the sources controlled by a given facility, not to each source individually located at a single facility.

Further limitations on radiation exposure of members of the public may be dictated by the federal, state or local agencies that regulate the facility’s operations. Specific environmental protection regulations may also apply.

2.4.2. Types of Exposures to Members of the Public

On-site radiation exposures to members of the public may result from exposure to radioactive materials and/or ionizing radiation-generating devices (RGDs). Exposure from natural sources in the soils or building materials is excluded when determining dose to members of the public, but exposure to technologically enhanced naturally occurring radioactive material (TENORM) is included if the enhancement is an integral or intentional part of the facility operations.

Radiation exposure received by medical patients is not included when determining dose to members of the public. However, healthcare providers, caregivers, family members, or others
who may be in contact with the patient are considered members of the public. These others
(nonpatients) are subject to the standards applicable to members of the public, unless they are
specifically designated and trained as radiation workers, as may be the case for radiation
oncology staff.

When the operational setting is a medical facility, dose control to comforters and
caregivers may need to be considered. The group is limited to individuals who are unpaid, not
occupationally involved in providing medical care, and are typically parents, other family
members, or close friends of the patient. Other individuals who may be inadvertently exposed as
a result of proximity to a treated individual are to be protected as members of the public.

NCRP (2018) recommends that the effective dose for a comforter or caregiver should not
exceed 5 mSv per episode, where an episode is defined as a single medical facility visit or stay
and includes the care received after release from the medical facility. The total exposure received
by comforters and caregivers includes exposure from diagnostic and therapeutic procedures
during the duration of care in the medical facility and, after release from the facility, during care
at home or elsewhere.

Off-site exposures are those occurring beyond the facility boundary. These exposures
include, but may not be limited to: radiation emanating from the facility at the site boundary;
contact with radioactive material deposited in the environment outside the site boundary;
radiation fields produced at a temporary job-site or during transport of radioactive material by a
private carrier; contact with material carried beyond the site boundary by an externally or
internally contaminated person; or contact with a person who has received a dosage of a
radiopharmaceutical as part of a medical procedure.

2.5. Exposure to Nonhuman Biota

Exposure of nonhuman biota may or may not concurrently result in exposure of humans.
As specified in Report No. 180 (2018), “the objective of radiation protection for nonhuman biota
is associated with population maintenance of the affected species rather than protection of
individual members of the species, and is considered within the framework of the National
Environmental Policy Act (NEPA 1969). Such assessments are in addition to the existing
practice of comparing radionuclide concentrations in various media (air, water) for the purpose of informing protection of such resources for future use.”

NCRP concludes that an assessment of the radiological impact on an ecosystem based upon a daily absorbed-dose rate (in tissue of nonhuman biota) is not necessary at absorbed-dose rates <0.1 mGy d$^{-1}$. At or above this absorbed dose rate, additional evaluation may be considered necessary. Note that the 0.1 mGy d$^{-1}$ absorbed-dose rate is not to be viewed as a target for remediation, or as a regulatory limit, but as a prudent value which will, in practice, result in most situations being deemed as not requiring further evaluation. For more information, see NCRP Report No. 180 (2018).
3. Radiation Safety Program Roles and Responsibilities

In addition to the radiation safety staff, other groups within an organization contribute to the overall success of the radiation safety program. These groups include senior management, radiation workers and their supervisors and managers, the radiation safety committee, the training organization, the medical staff, the legal staff, and others, depending on how a program is organized. This section discusses the roles and responsibilities of the key groups that support the radiation safety program.

3.1. Senior Management

For a radiation safety program to be successful, it must have the support of senior management. Senior management is typically responsible for:

- Establishing a documented radiation safety policy. The policy should include commitments to the protection of workers and the public, the application of the ALARA principle, compliance with regulations, and engagement with the workers. It should outline responsibilities, expected behaviors, and accountability for all individuals involved in the radiation safety program. The radiation safety policy is typically drafted by the radiation safety staff, reviewed by the radiation safety committee (if applicable), and approved and issued by senior management.

- Working with the Radiation Safety Officer (sometimes referred to as the Radiological Control Manager or Radiation Protection Manager) to define the goals of the radiation safety program. Program goals include establishing ALARA goals, minimizing the amount of radiological waste produced, supporting the involvement of the radiation safety staff in the planning phase of radiological work.

- Establishing the roles and responsibilities of radiation safety program staff and delegating the necessary authority to them to implement the radiation safety program.

- Committing the necessary resources (financial, human, and organizational) to implement the radiation safety program.
518  • Establishing a radiation safety committee (RSC) (sometimes referred to as an
519     ALARA Review Committee, or an ALARA Committee) as needed to review
520     proposed radiological operations, policies, and procedures.
521  • Assuring that radiological operations are performed with safety and cost
522     effectiveness.
523  • Establishing the safety culture and a safety management system that drives
524     continuous improvement of the radiation safety program.

3.1.1. Safety Culture

Management involvement is essential in establishing the organization’s radiation safety
526     culture (referred to as the “radiation protection culture” in NCRP Report No. 180). As outlined in
528     “Radiation protection [safety] culture includes the core values and behaviors resulting
529     from a collective commitment by leaders and individuals within an organization3 to emphasize
530     radiation protection over competing goals (nonsafety related) to ensure protection of people and
531     the environment. The overriding principle is to “put safety first.” Positive radiation protection
532     culture fosters traits to effectively establish and implement a radiation protection program that
533     applies to an organization’s environment.
534     “Radiation protection [safety] culture is consistent with and may be included within an
535     overall safety culture (NRC 2016). The specific focus in this Report [No. 180] on fostering a
536     radiation protection culture is to recognize its intrinsic value as a force multiplier in supporting
537     and sustaining effective implementation of other recommendations contained in this Report.
538     “Guiding principles for establishing a positive radiation protection [safety] culture have
539     been developed by the International Radiation Protection Association (IRPA 2014) and have
540     been endorsed by radiation protection professional organizations throughout the world.

3 As used here, the term organization refers broadly to any enterprise in which there is work with radiation
or decisions impacting radiation exposures are made.
“Some key attributes of a positive radiation protection [safety] culture include the following (adapted from IRPA 2014):

- leaders demonstrate commitment to radiation protection in their decisions and actions;
- individuals throughout the organization take responsibility for radiation protection of themselves and others;
- radiation protection issues are promptly identified, evaluated, and addressed commensurate with their significance;
- radiation protection is integral to the planning and control of work activities;
- opportunities to learn from radiation protection experience are sought out and identified improvements are implemented;
- processes for raising radiation protection concerns are open, nonjudgmental and responsive;
- communication on radiation protection matters is timely and effective;
- shared trust and respect permeate the radiation work environment;
- individuals continually demonstrate a questioning attitude about radiation protection conditions and activities;
- individuals take responsibility for their own exposure; and
- all parties share radiation protection information through a genuine dialog.”

3.1.2. Safety Management System

 Organizations should implement a safety management system to drive continuous improvement of the radiation safety program. Stakeholders in developing, implementing and reviewing this system are management, workers (especially those who will be working with and near radiation sources), and in some instances, the public. Some of the major elements of a safety management system are presented below and shown in Figure 3.1; several references
Planning. The planning stage includes identification of the work to be completed, the steps needed to accomplish the work, and the associated barriers and risks (e.g., physical hazards, regulatory compliance). The organization should identify and prioritize longer-lead time items such as amending a license or modifying a facility, and assign staff to manage the tasks.

Implementation and Operation. In the implementation stage, the developed policies, procedures, or improvement plans are clearly communicated to ensure the impacted staff understand what is changing and why. The operational stage involves conducting the prescribed work.

Checking & Corrective Action. The checking and corrective action phase extends throughout the conduct of work and involves monitoring the work and the workplace; program audits/assessments; and improvement activities. Management should use worker feedback and workplace indicators to adjust the process, increase safety, and foster personnel accountability.

Management Review. This element deals with the annual review of the program by top management. This review, in conjunction with radiation safety stakeholders, should be thorough enough to ensure that the program is meeting objectives and provide recommendations for areas of improvement where needed.

Figure 3.1 illustrates an example of a Safety Management System. Feedback from the Management Review leads to continuous improvement of the program with the concomitant increase in safety and decrease in risk.

3.2. Radiation Safety Officer and Staff

The radiation safety program is typically directed by a Radiation Safety Officer (RSO). The RSO should report at a sufficiently high level to carry out their responsibilities, with ready access to senior management and to all levels of the organization for any radiation safety program concern. The RSO should have the authority to terminate any operation when justified by radiation safety concerns.
Fig. 3.1. Example of a safety management system.
The minimum qualification of the RSO will depend on the magnitude of the potential hazards and complexity of the operations. At a minimum, the RSO should possess an appropriate academic background (e.g., a degree, or substantial education, in one of the physical or biological sciences or engineering) together with practical radiation safety experience germane to the operation. As the complexity of the operations increases, a degree or specialized education in health physics, combined with practical experience, is warranted. For programs that use a variety of radiation sources in different modalities, a health physicist who is certified by the American Board of Health Physics should fill the RSO role (in a health care setting, this may be a qualified medical physicist [see definition in glossary]).

The RSO has responsibilities to the radiation safety staff as well as to the organization. The RSO’s responsibilities to the radiation safety staff include:

- developing and supervising a qualified staff of radiation safety professionals and technicians of a size and level of expertise appropriate for the activities of the organization.
- providing continuing education opportunities to maintain high skill and knowledge levels. Continuing education should utilize the expertise within the organization as well as opportunities provided by professional societies, universities and other organizations.
- supporting and encouraging staff members to become certified by the American Board of Health Physics, the American Board of Medical Physics, the American Board of Radiology, or the National Registry of Radiation Protection Technologists, as appropriate.

Utilizing the radiation safety staff, the RSO’s principal responsibilities include:

- Representing the radiation safety program to key stakeholders.
  - advising senior management, managers and supervisors, and radiation workers concerning radiation safety practices and regulations
  - interacting with regulators (e.g., discussing interpretation and implementation of regulations, discussing proposed changes to regulations, participating in and responding to audits by the regulatory authority). See Section 3.2.2.
interacting with others, as needed (e.g., public affairs staff, concerned workers)

- Establishing the requirements for use of radioactive materials and RGDs.
  - knowing the regulatory requirements and integrating them into site procedures.
  - developing and maintaining the radiation safety manual (see Section 3.2.1).
  - specifying required operational controls (e.g., personal protective equipment (PPE), dosimetry, shielding, instrumentation).
  - specifying/approving facility designs that impact radiological operations (e.g., shielding, fume hoods, ventilation).

- Specifying requirements for radiological support services.
  - establishing quality assurance requirements for survey and sampling data.
  - ensuring the integrity of dosimetry and other radiation safety program records
  - establishing the initial inventory and budget for portable radiation detectors (see Section 7.1)

- Providing oversight
  - authorizing the use of RGDs and radioactive materials within the facility
  - establishing and tracking ALARA goals
  - maintaining a list of the radiation sources in the facility
  - conducting periodic and as-needed surveillance of radiation safety program activities (e.g., acquisition and disposal of radioactive materials; radiation safety training for facility employees and users).

The radiation safety staff might consist of a few people, or hundreds of people, depending on a variety of factors, including:

- the magnitude of potential radiological hazards, which is generally determined by
  - the amounts, types, and forms of radioactive material being handled, and
  - the operational complexities involved
the number and types of RGDs in use, and
the size of potentially exposed populations (e.g., workers, visiting scientists, members of the public)

- how radiological work is distributed
  - geographically (i.e., in one or more facilities, and the proximity of those facilities to each other),
  - temporally (i.e., over the course of one, two, or three shifts), and
  - between routine or frequently changing operations (as with research and development)

- whether radiological support services (e.g., dosimetry, instrument calibration and maintenance, analytical laboratories) are provided in-house or outsourced.

Regardless of the size of the radiation safety staff, it should remain organizationally independent from those receiving its services.

### 3.2.1. Radiation Safety Program Procedures

The complexity of the radiation safety program’s implementing policies and procedures should be commensurate with the size and complexity of the radiation use in the organization.

A radiation safety manual is a collection of procedures and processes to help workers use radioactive material and RGDs safely and in compliance with the organization’s policy and regulatory requirements. The radiation safety manual should include:

- top management’s commitment to proper radiation safety practice.
- a description of the RSC, the radiation safety staff, and the radiation safety program.
- specific policy and regulatory requirements.
- specific procedures that ensure compliance with these requirements.

This information may be provided in formats other than a manual; regardless of the format, the RSO should ensure the material is periodically reviewed and revised as necessary to
ensure its relevance to current operations and requirements. The assigned author should solicit input from the users of the procedure, the radiation safety staff, and the RSC.

3.2.2. The Regulatory Environment

There are many federal entities that govern radiological activities, including the Nuclear Regulatory Commission (NRC), the Department of Energy (DOE), the Department of Defense (DOD), the Food and Drug Administration (FDA), the Occupational Safety and Health Administration (OSHA), the Mine Safety and Health Administration (MSHA) and the Department of Transportation (DOT). In some cases, States may develop their own radiation safety regulations, or adopt the NRC regulations if they are an Agreement State. In addition, states have the authority for regulating the use of accelerators and x-ray machines not covered by federal regulations. It is essential that the RSO and staff know which regulatory authorities and statutes govern the radiological operations at their site(s).

3.3. Ancillary Staff

3.3.1. Radiation Safety Committee

A radiation safety committee can effectively expand the impact of the radiation safety staff. Certain regulations may specify the establishment, membership, and responsibilities of the RSC, but it is typically comprised of individuals who are knowledgeable about the use of radioactive materials or RGDs in the facility and may include individuals who are knowledgeable about the overall organization and its legal, financial, procurement, and other business functions. The RSO should be an ex-officio member of the RSC.

The RSC is accountable to management for developing and implementing a radiation safety program that meets the radiation safety policy and regulatory requirements, and that supports the implementation of the ALARA principle. If an organization doesn’t have an RSC, this responsibility is shared by the RSO and management (of radiological operations).

The responsibilities of the RSC should be documented in a committee charter or charge statement, typically issued by senior management. Although they vary from one organization to another, typical RSC responsibilities include:
periodically reviewing the radiation safety policy and recommending updates as necessary.

- reviewing (and sometimes approving) the organization’s radiation safety manual.

- reviewing and auditing the effectiveness of the radiation safety program.

- reviewing prospective radiological work, including the purpose, safety, and compliance with the radiation safety program and regulatory requirements. In academic or healthcare organizations, this may include reviewing research protocols for the Institutional Review Board or the Institutional Animal Care and Use Committee.

- approving new uses or users of radiation sources.

- assisting the RSO in the escalation of corrective actions for noncompliant behavior.

- reviewing incidents involving radiation and approving corrective actions.

- providing guidance to the RSO on the operational uses of radioactive materials and RGDs.

Members of the RSC should be qualified in their normal field of endeavor. Management (of radiological operations) is responsible for assuring that members, and especially the chairperson, have the necessary experience and qualifications to meet the responsibilities of the RSC.

3.3.2. **Occupational Medicine**

The medical staff in an occupational medicine department provides important services such as medically approving individuals to wear certain types of respiratory protection (e.g., Self-Contained Breathing Apparatus) and administering the Declared Pregnant Worker program. They also provide care to workers accidentally exposed to very high external radiation doses, significant intakes of radioactive material, and high levels of skin contamination. Large facilities may have an Occupational Medicine organization that can address these issues; smaller facilities should identify other resources (e.g., an independent contractor, an affiliated organization with the needed skills) that can address and respond to these issues as they arise.
3.3.3. **Legal Support**

At times, the RSO or RSC may need legal support to properly implement a regulation, review correspondence to the regulatory agency or represent the organization during escalated regulatory enforcement actions. If the organization does not have a legal staff, the RSO should identify legal support that can be accessed on an as-needed basis.

3.3.4. **Radiological Training**

Radiological training is an essential part of the radiation safety program. Depending on the size of the program, radiation safety training may be performed by staff members or a separate training group. See Section 5.3.

3.4. **Supervisors and managers**

Managers and supervisors of radiation workers are responsible for:

- providing a safe work environment.
- ensuring that an optimization process has been completed to assure individual and collective radiation doses will be ALARA.
- establishing performance expectations and good work practices in the areas that they supervise.
- ensuring workers understand their responsibilities relative to the radiation safety policy and procedures.
- Ensuring workers are appropriately trained (see Section 5.3.1 for detailed training responsibilities).

3.5. **Radiation workers**

Radiation workers are responsible for:

- ensuring their own safety. That is, ensuring they understand work procedures before enacting them, maintaining a questioning attitude, and stopping any time they perceive or anticipate a problem arising.
• complying with the radiation safety policy and procedures

• engaging with their supervisor and the radiation safety staff, sharing ideas for improvement and voicing any concerns they may have.
4. Facility and System Design Considerations

Facility and system design is not normally considered a part of the operational radiation safety program, as the facility has already been designed, constructed and commissioned before radiological operations commence. However, operational radiation safety management or staff may be called upon to provide input into the design of additional buildings, facility modifications, or the evaluation of new or modified systems, components, and equipment. This section contains the basic principles and considerations for radiological design review of new and modified buildings.

It is essential that a radiological engineer or health physicist participate in the planning and design of new buildings and modified facilities, to ensure that appropriate radiation safety features are incorporated into the design for the protection of workers and any potentially-exposed members of the public. Competent input and review in the design stages will facilitate operations and ensure radiation doses are below the limits and ALARA. Early involvement also provides the lead time necessary for updating the site license and decommissioning plan, if needed.

In most cases, passive controls (e.g., concrete shielding) are preferable over active/engineered controls (e.g., warning lights and interlocks). Using a comprehensive radiological design review checklist can aid in ensuring the ALARA principle is applied to all facets of the proposed design.

4.1. Selection of Locations for New or Modified Facilities

When choosing the location of a new facility, or a new radiological operation within an existing facility, the general goals are to:

- minimize the potential dose to co-located workers who are not involved with radiological work.
- minimize the potential dose to members of the public (e.g., by locating operations away from public access areas and as far as practicable from the site boundary).
consolidate radiological operations in as small a footprint as practicable, with ready access to necessary materials and services (e.g., radioactive material, radiation instruments, change rooms, PPE, waste accumulation areas).

4.2. Facility Layout

The facility layout is integral to efficient operations and an effective ALARA program. As it is important to plan both for operations and radiation safety support services, space should be allocated for the following based on the types of operations involved:

- change rooms.
- work areas sufficiently large to support radiological operations (e.g., including workers and radiation safety technicians, waste receptacles, monitoring equipment)
- break rooms.
- radioactive materials and radioactive waste storage areas.
- storage for monitoring instruments and other radiation safety supplies (e.g., protective clothing, respirators, plastic sheeting, tape).
- analysis of surveillance samples.
- packaging/storage of radioactive wastes.
- equipment decontamination.
- personnel decontamination.
- personal contamination monitoring at egress points.
- Calibration of radiation instrumentation

These spaces should be optimally located for efficient operations, ease of movement of people and materials into and out of radiologically controlled areas, dose control, and routine and as-needed maintenance. In some cases, choices between cost and convenience may have to be made, such as locating radioactive source storage areas remotely rather than investing in costly shielding to allow the material to be stored in a centrally located, occupied area.
When practical, the facility should be designed with zones based on the risks associated with potential external radiation exposure, activation, airborne radioactivity, and surface contamination. The number and the size of these zones should be minimized to that necessary for efficient operations. Clean areas such as lunchrooms, offices and conference rooms should be accessible without traversing either radiation or contaminated areas, and workers wearing PPE should not be co-mingled with workers who are not wearing PPE. Areas with a low potential for personnel radiation exposure should be segregated from those containing high dose rate areas. Storage areas for nonradioactive supplies and parts should be located outside areas in which these materials might become activated or contaminated. Where activation is a potential concern, construction materials should be chosen to minimize the potential for activation. Persons going from one clean area to another, or between low-risk areas, should not be required to pass through zones of greater risk of exposure or contamination.

Calibration facilities should be designed for the specific requirements of the instruments that are used and the sources that are measured at a particular institution, unless the facility is established to provide calibration services to clients outside the institution. Care should be given to controlling the interfering background and scattered radiation.

4.3. Equipment and System Design

It is important to consider radiation safety and operational efficiency in the selection and design of fixtures, equipment, systems and components. Equipment and components that may become highly radioactive or contaminated should be designed for accessibility, ease of maintenance, ease of installation and removal, and ease of decontamination (ASTM 2016). Fixtures designed to support movable shielding and containment enclosures can reduce the time spent in the vicinity of activated or contaminated equipment. At accelerator and reactor facilities that produce particle beams or neutron radiation, it is prudent to choose materials with a low activation cross-section to minimize induced radioactivity.

The planned operations will guide the selection of installed radiation monitoring and surveillance equipment (see Section 7.1). These monitoring systems are typically installed if operations could cause unexpected dose rates in excess of 1 mSv h\(^{-1}\) in potentially occupied areas (e.g., around some types of RGDs), and for remote monitoring of highly radioactive filters, ion
exchange resins, tanks and other equipment. The instruments must be selected based on the
radiation fields and dose rates they are expected to monitor.

Equipment and systems used in areas where dispersible radioactive materials are handled
should be designed with contamination control and decontamination in mind. Wherever possible,
it is prudent to locate equipment control panels outside of potentially contaminated areas, and to
cover items such as cable trays that must run through these areas. Installed contamination
monitors (e.g., personnel contamination monitors (PCMs), hand and shoe monitors) may be
needed at the exits from dispersible radioactive material handling areas.

For operations that produce extremely high doses of radiation (e.g., a high-energy
accelerator), an access control and warning system is required to preclude inadvertent access
(NCRP 1986). Care must be used when selecting electronic instruments for use in these areas, as
very high radiation levels may interfere with their proper functioning.

Regardless of the type of radiological operation, fire detection/suppression capabilities
are normally required; security devices (e.g., for theft detection) may also be required for certain
radioactive sources or materials. These systems should be designed for ease of maintenance and
use, considering the specific radiological environment in which they will be used.

4.4. Shielding

Radiation shielding is usually the primary means of limiting radiation doses to acceptable
levels and should be an integral part of the planning for new or modified facilities. The selection
of shielding material is dependent upon both its radiation attenuation properties for the type of
radiation present, its engineering characteristics such as weight, cost, structural stability and
materials compatibility; operational parameters such as the use factor of the source, the
occupancy time of individuals in the area outside the shield, the potential changes in these
parameters; and in some cases, its potential for radioactivation. Specific guidance for selecting
materials for neutron shielding is contained in NCRP Report No. 38 (NCRP 1971) and Report
No. 72 (NCRP 1983b). The ultimate removal and disposal of the shielding material should be
considered during the selection and design phase.

Both the cost and spatial extent of shielding should be considered, but radiation safety
should not be compromised. Placing shielding close to the source will generally reduce the
amount of shielding needed. Consideration should also be given to the amount of time the source
is exposed or the RGD is energized. When deciding upon the amount of shielding to install, it is
prudent to consider the possibility of increased use of the source, increase in the radiation field
intensity, changes in beam direction, and future design changes that would increase occupancy
near the source shielding.

The field of radiation shielding design encompasses radiation physics, materials
properties, and structural engineering. Shielding is usually designed by radiological or nuclear
engineers with specific expertise in this area. Important radiation shielding references include:
Foderaro (1976), Glasstone and Sesonske (1994), Johnson (2017), Lamarsh and Baratta (2017),
McGinley (1993), Rockwell (1956), Shultis and Faw (2000), and Shultis and Faw (2016). Data
necessary for shielding design and recommendations concerning shielding techniques are also
provided in NCRP reports (NCRP 2004b, 2005) and ANSI standards (ANSI/ANS 2006a, 2006b,
1991). Detailed information on shielding design and considerations for medical facilities is
provided in NCRP Report No. 147 (NCRP 2004a) and NCRP Report No. 151 (NCRP 2005);
although written to address medical use of RGDs, these reports may also be useful for other
applications of RGDs.

Computer codes are frequently used to model the transport of radiation fields through
materials to define the types and amounts of shielding required. Codes such as MicroShield®
(Grove Software) are intended for use with relatively simple geometries (e.g., cylindrical
sources, slab shields, point sources). These codes have the advantage of saving considerable time
and usually contain data for many materials and radionuclide sources. More complex codes, such
as the Monte Carlo N-Particle transport code (MCNP) (Werner 2018; NCRP 2003b) can be used
to evaluate detailed source-shielding configurations. These codes typically require extensive user
training to apply. Proper training on the use of such codes is imperative to prevent errors that
may be missed by those unfamiliar with the calculational limitations of the codes.

Radiation shielding should be designed such that doses to exposed individuals are not
expected to exceed the annual dose limit, taking into account design, construction, occupancy
factors, and measurement uncertainties. Shielding design should also incorporate the application
of the ALARA principle. The design criterion for dose depends on whether the facility is
accessible to the general public.
4.5. **Ventilation**

Proper ventilation is essential to control the movement of airborne radioactive materials, to prevent or minimize the spread of contamination within the facility, and to prevent overheating of RGDs. Ventilation is second only to encapsulation in importance for preventing worker exposure to airborne radioactivity. The ventilation system in new and/or modified facilities should be designed to control airborne radioactivity; personal respiratory protection equipment should typically be relied upon only when normally-present encapsulation is to be breached, or prophylactically for higher-risk operations (e.g., when bagging an item out of a glovebox).

Ventilation design is typically performed by HVAC engineers with specific expertise in this area. Extensive work has been done on the design and engineering of ventilation for hazard control. Some important references dealing with ventilation systems are: American Conference of Governmental Industrial Hygienists (ACGIH 2007), ANSI/American Society of Safety Professionals (ANSI/ASSP 2018), ANSI/American Industrial Hygiene Association/ASSP (ANSI/AIHA/ASSP 2012a), ANSI/American Society of Heating, Refrigerating and Air-Conditioning Engineers (ANSI/ASHRAE 2019), Cooper and Alley (2010), International Organization for Standardization [ISO 2004 (R2019)], Kathren et al. (1980), McDermott (2001), Plog (1988), and Underwriters Laboratories (UL 2009).

Ventilation systems must provide acceptable pressure differentials between work areas and the outside environment. In the system design, a pressure gradient should be established with the air flow directed from clean areas to contaminated areas and with the lowest pressure and collection points in areas with the highest potential for release of dispersible material. Proper air flow needs to be consistently maintained, regardless of whether the doors are opened or closed.

Ventilation systems typically include some type of filtration or air cleaning device to capture gaseous and dispersible particulate radioactive materials. These include single and multi-stage High-efficiency Particulate Air (HEPA) filters for particulates and charcoal adsorbers for volatile gases and organic compounds. Filtration systems should be designed and located to facilitate ease of independent testing and filter replacement.
Exhaust vents and stacks should be located to avoid entrainment of exhaust air through ventilation system intakes. The design should also include provision for modifying the ventilation during an accident (e.g., containment, use of a redundant system, use of a by-pass system, or change in flow rates). Controls for the ventilation system should be located in areas that are readily accessible in the event of an accident. Effluent monitoring requirements and criteria should also be considered during the design of the ventilation system (NRC 2009, 2007a).

Ventilation systems associated with fume hoods or gloveboxes containing radioactive materials are often operated continuously to minimize the potential for release of these materials. The design should incorporate a standby fan and backup electrical power to ensure continued operation of the system despite component or power supply failures. Airlocks are often incorporated in the design of glovebox pass-in/pass-out ports; the airlock should remain under negative pressure relative to the room when the airlock door to the room is opened. Design and operational considerations for chemical fume hoods are discussed in Section 5.3.2 and in the American Conference of Governmental Industrial Hygienists (ACGIH) Industrial Ventilation design manual (ACGIH 2019). Glovebox design considerations are described in standards published by the American Glovebox Society (AGS) in AGS-G001 (AGS 2007).

4.6. **Decontamination Considerations**

Materials for surfaces of floors, walls, fixtures, equipment and work surfaces in areas where dispersible radioactive materials may be used should be selected with decontamination in mind, taking into account the chemical and physical environment. If porous materials (e.g., concrete) are used in construction, exposed surfaces should be faced with nonporous materials or painted with several layers of nonporous coating (ASTM 2016). Outer layers of strippable paint can also be useful in areas that need frequent decontamination. Cracks and holes in surfaces and sharp corners should be minimized or eliminated. In some cases, the use of disposable materials and equipment may be preferable to relying repeated decontamination.

Containment for radioactive liquid waste should be provided in radioactive material work areas. Depending on the amount of liquid waste produced, either stand-alone systems (e.g., a glass or plastic container in a 5-gal metal can) or plumbed facility systems can be used. In the
latter case, holding and sampling tanks, as well as processing or radioactive material removal systems, may be required for contaminated waste drains and sinks to ensure that radioactive effluents do not exceed permissible levels. Piping systems should be designed to minimize connections between clean and potentially contaminated systems.

4.7. Electrical Design for Instrumentation, Access Control, and Alarm Systems

The electrical design for any new buildings or modification to existing facilities should include wiring, outlets and other components adequate to support initial and projected loads for radiation monitoring instrumentation, access control and alarm systems. The design should include adequate power for fixed radiation monitoring instruments and outlets for portable equipment and instrumentation (e.g., counting equipment for radioactive samples). Uninterruptible or backup power supplies should be provided for critical equipment and instrumentation. Interlocks and the associated circuits that are part of access control and warning systems should be designed by skilled electrical engineering professionals with redundancy and to be “fail-safe.” The controls, relays and wiring should be installed in secure and protected conduits to prevent tampering and unauthorized modification. Computer-controlled programmed logic systems must be protected against accidental and unauthorized changes. Additional design information and recommendations are provided in NCRP Report No. 88 (NCRP 1986).

Wired and wireless systems that process data may require evaluation for cybersecurity issues. Cybersecurity features protect the systems from unauthorized access that could compromise the integrity, data, or function of the system. This protection is particularly important for systems that use telemetry for wireless transmission of data, such as dosimeter readings, personnel information, and radiation levels.
5. Radiological Work Controls

As discussed in Section 2, the radiation safety principles of justification, dose limitation, and the optimization of dose are the basis of the radiation safety program that implements the engineering, administrative controls and program elements provided in this Report.

5.1. Hierarchy of Controls

There is a well-established hierarchy of controls which should be used when determining which radiological controls are appropriate for a process or operation. This hierarchy, which is shown in Table 5.1 and discussed at length in NCRP Report No. 176 (NCRP 2017), has five steps, the first of which is to evaluate whether the process can be effectively implemented without the use of radioactive materials or RGDs. If not, the second step is to determine whether a less hazardous radioactive material or device can be substituted. The third step is to apply engineered controls, which includes any physical safety control (e.g., shielding, gloveboxes, hoods, interlocks, signs). Passive engineered controls (e.g., shielding) are preferable to active engineered controls that can be defeated or fail (e.g., interlocks, alarm systems). If a radiological risk remains after engineered controls are applied, the fourth step is to apply administrative controls (e.g., work planning, training, policies). The final control method is to apply physical protection to the worker through the use of PPE. This section focusses on the last three levels in the hierarchy of controls.

5.2. Planning Radiological Work

Careful planning of work activities coupled with a safety procedure, training, pre-job briefing, and the selection of appropriate equipment and PPE is essential to control the exposure of workers to radiation and other hazards. Post-job reviews can provide valuable information that can be used to reduce future exposures from similar tasks.
Table 5.1 – *Hierarchy of exposure controls [adapted from NIOSH (2009) and NCRP (2017)]*.  

<table>
<thead>
<tr>
<th>Control Method</th>
<th>Process, Equipment or Job Task</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Elimination</td>
<td>Change the design to eliminate hazard</td>
</tr>
<tr>
<td>2. Substitution</td>
<td>Replace a high hazard with a low hazard</td>
</tr>
<tr>
<td>3. Engineered</td>
<td>Shielding, isolation/enclosure (gloveboxes, hoods), ventilation (local, general), effluent filtration, physical barriers, interlocks, warning lights, signs</td>
</tr>
<tr>
<td>4. Administrative</td>
<td>Safety and operating procedures, training, job scheduling</td>
</tr>
<tr>
<td>5. Personnel protective equipment</td>
<td>Respirators, clothing, gloves, goggles</td>
</tr>
</tbody>
</table>

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5.2.1. **Work Scope and Hazard Identification**

The first step in work planning is to define the scope of the work to be performed. The scope should specify what work will be done, as well as where, and if relevant, who will do the work and when the work will be performed (e.g., during nightshift).

Based on the work scope, potential radiological and nonradiological hazards should be identified, paying special attention to work that is conducted either infrequently or for the first time, and to potential hazards that either have not been previously encountered or entail a greater potential hazard than previously encountered. For example, work with radioactive nanomaterials may be both new and entail greater radiological hazards than work with a commensurate amount of non-nanomaterial radioactivity. NCRP Report No. 176 (NCRP 2017) provides useful information on this topic.) Nonradiological potential hazards include such things as toxic chemicals, inert gases, high pressure, high voltage, fire, working at elevation, working in very hot or cold environments, and moving heavy equipment.

The appropriate occupational health and safety disciplines (e.g., health physics, industrial hygiene, industrial safety, fire safety) should analyze the potential hazards and work cohesively to identify controls that reduce the hazards to acceptable levels of risk. The safety disciplines should deconflict any conflicting controls (e.g., the type of gloves to be worn). Involving other health and safety disciplines is essential when nonradiological hazards potentially predominate.

5.2.2. **Dose Management**

During the work planning phase, potential internal and external radiation doses to workers should be evaluated and controls that will maintain individual doses ALARA (see Sections 2.2) should be identified. The extent and formality of exposure planning and dose reduction activities should be commensurate with the hazard of the task and the potential dose savings.

5.2.2.1. **Administrative Dose Guidelines**

Administrative dose guidelines (ADGs) are a tool for focusing on and managing accumulated annual dose so that individual doses that approach or exceed the ADG receive prompt management attention. ADGs should be established to reduce the likelihood of workers
exceeding the regulatory dose limits and to ensure that individual doses are optimized for the
work to be performed. Although ADGs are not “limits”, they should not knowingly be exceeded
without proper review and approval except during emergencies. ADGs should be established
with line management involvement and may be established for a particular task, a portion of the
year, or an entire year. A procedure for responding to doses that exceed the ADGs should be
established.

To maintain doses ALARA and optimize safety, ADGs should be established if the work
is likely to result in individual doses above some pre-established value. On a practical level,
ADGs are typically warranted if workers are likely to receive an annual whole-body dose in
excess of 1 mSv; establishing lower guidelines can have an overall negative impact by diverting
management attention from issues of greater importance. Given the lag between a worker
receiving a dose and the dosimeter being processed, a worker could go from being well below
the ADG to exceeding it without preapproval. Such situations should be followed up promptly
by the radiation safety staff, working in conjunction with the individual and their supervisor;
however, they do not necessarily warrant removing the worker from radiological work during the
follow-up phase, especially where low and challenging ADGs are established.

5.2.2.2. Investigation Levels

In addition to ADGs, radiation safety program management should consider establishing
investigation levels, which are used to identify outlier dosimeter and bioassay results so they can
be promptly evaluated. Investigation levels are independent of both an individual’s year-to-date
dose and their ADG and should not be confused with dose limits. Investigation levels can be
established to flag anomalous data (e.g., a closed window thermoluminescent detector (TLD)
reading that exceeds the corresponding open window dose) as well as dosimeter and bioassay
results that appear valid and are out-of-the ordinary or otherwise of concern.

Radiation safety program management typically sets the investigation levels and
monitoring results are compared to the investigation levels as they are produced. Investigation
levels typically have an associated reporting timeframe (e.g., a whole-body dosimeter result
exceeding X mSv must be reported to the RSO (or designated staff) within 24 h; a dose
exceeding a regulatory dose limit must be reported to the RSO within 2 h). A radiation safety
professional should promptly investigate dose-related measurements that exceed the
investigation level and report findings to the RSO, the individual, and the individual’s supervisor. Anomalous measurements such as the closed window TLD dose exceeding the open window TLD dose should be evaluated as routine, not urgent, work, unless the dose otherwise surpasses a dose-related investigation level.

Because monitoring results are compared to the investigation level in real time, it is important to have uncomplicated dose-related investigation levels. A general set of investigation levels with increasing investigation urgency (e.g., a whole-body dosimeter reading in excess of 3 mSv, 10 mSv, or 50 mSv; an extremity dosimeter reading in excess of 10 mGy or 100 mGy; a bioassay result greater than predetermined, isotope-specific level) may be more effective than frequently changing or job-specific investigation levels, which tend to be impractical to manage. The selected dose-related investigation levels should be sufficiently high that they warrant and receive prompt investigatory action.

5.2.3. Safety Procedures and Radiation Work Permits

Specific safety requirements should be clearly documented to ensure the information transmitted to the workers is unambiguous. Documents such as safety procedures and radiation work permits (RWPs) are typically used for this purpose, although different organizations may call them by other names (e.g., Use Permits, Work Permits, Integration Work Sheets, Radiation Use Authorizations, Long-term RWPs) or integrate the controls into other technical work documents. Safety procedures are typically used to describe and authorize tasks that are repetitive and will be performed over an extended period of time. RWPs are typically used to describe and authorize specific tasks and jobs with relatively short (e.g., days, weeks) duration, where specific radiological controls are needed. For complex tasks, a step-by-step safety procedure or RWP may be needed to ensure that all the necessary steps are performed in the correct order and with the expected results.

The thresholds for which safety procedures and RWPs are required should be documented, and both operations management and the RSO should have the authority to require a safety procedure or RWP where questions arise. Thresholds for requiring a safety procedure or RWP may vary amongst different organizations.

Typical thresholds for requiring a safety procedure include:
• routine radiological work
• routine work with accountable levels of dispersible radioactive material
• routine work with IAEA Category 1, 2, or 3 sealed radioactive sources (IAEA 2005, 2006)
• use of RGDs that require licensure

Typical thresholds for requiring an RWP include:
• nonroutine work in radiological areas
• routine work with changing radiological conditions (e.g., maintenance on a contaminated system)
• work with a potential for higher radiological risks (safety procedure or RWP)

As an acceptable alternative to a specific RWP or safety procedure, workers may be trained and qualified on operating and maintenance procedures which incorporate the required radiological controls. Workers should periodically requalify on these procedures (typically annually).

Safety procedures and RWPs should be written by individuals knowledgeable of the work to be performed, conditions in the work area, and radiation safety; include both radiological and nonradiological hazards; and at a minimum, be reviewed and approved by both operations supervision and radiation safety, as well as other safety disciplines if nonradiological hazards are present. Safety procedures typically have an annual or 2 y review cycle, whereas RWPs are typically terminated when the work is completed.

Workers should be familiar with the contents of the safety procedure and RWP, either by reading the document or receiving a briefing, and should acknowledge their understanding by signing or initialing the document (or electronic equivalent). RWP and safety procedure briefings/retraining for routine work should be required at least annually. Briefings on the safety procedure or RWP may need to be repeated more frequently (e.g., daily, weekly, quarterly) depending on the risk, complexity and duration of the work, changing radiological conditions as the work progresses, and the turnover of workers.
Safety procedures and RWPs should contain information for controlling radiological and nonradiological hazards, as well as the following types of information, as applicable to the work to be conducted.

Administrative information:

- a unique procedure/RWP identifier (e.g., title, number), a revision number, effective date and any expiration date, as applicable
- identification of who to contact with questions about the procedure, as well as contact information
- the individuals responsible for making sure that the work activities are conducted in accordance with the safety procedure or RWP
- worker qualifications including any specialized training that is required

Scope of work and hazards analysis, including:

- the work location (e.g., building, room, or area)
- a description of the work authorized by the safety procedure or RWP
- identification of the work procedures to be used, if applicable
- a description of the potential hazards that will be encountered in performing the work, including potential radiation dose rates, identification of the sources of radioactive material, the potential and likely radioactive contamination levels, and the potential for intake of radioactive material

Work controls to mitigate the identified hazards, including requirements for

- use of safety equipment (e.g., hoods, gloveboxes, tongs, portable shielding)
- PPE (e.g., protective clothing, respiratory protection, hard hats)
- worker stay time and dose limitations
- external dosimetry (e.g., whole body dosimeters, extremity dosimeters, personal alarming dosimeters with specified alarm levels)
- job-specific bioassay
1145 • radiation monitoring
1146 • the radiation safety coverage
1147 • nonradiological hazard controls
1148 • actions to be followed in the event of an emergency

1149 Action Levels and Stop Work thresholds. For low-hazard operations, action levels may
be based on a multiple of the area background reading, such as two- or three-times background.
For higher-hazard operations, action levels should be set such that radiation workers recognize
that something unexpected has occurred. Action levels should be low enough that they do not
pose a risk of workers exceeding administrative dose guidelines or the provided levels of
contamination control (e.g., PPE), but not so low that they routinely interrupt the work being
performed. Action levels may be generic (e.g., for routine operations, Continuous Air Monitor
(CAM) alarms, radiation area alarms, minor injuries) or specific (e.g., for nonroutine work).
Examples of Action Levels and Stop Work thresholds include:

1158 • Dose rates exceeding the expected (specified) levels
1159 • Contamination exceeding the expected (specified) levels
1160 • Exposure to airborne radioactivity exceeding a specified number of derived air
concentration-hours (DAC-h)
1162 • Unanticipated area radiation monitor alarms or personal dosimeter alarms
1163 • Unanticipated CAM alarms
1164 • Unanticipated elevated air sample results
1165 Environmental protection, including
1166 • consideration of the potential for effluents and the need for effluent or waste
generation controls
1168 • the generation of radioactive and hazardous waste
1169 • waste minimization as required by 10 CFR 20.1406 (NRC 2020a)
In addition to work procedures for the occupational setting, medical facilities may have need for special types of safety procedures that provide instructions and precautions to caregivers and patients subject to nuclear medicine intakes as well as patients receiving high skin doses from fluoroscopic procedures. See Appendix C.1 for more information regarding radiation safety programs in medical settings.

5.2.4. Work Authorization

Prior to authorizing work, operations management should ensure that:

- prescribed safety controls for radiological and nonradiological hazards are in place;
- workers have completed all required training;
- workers are familiar with the safety procedure or RWP and know what to do if a modification is needed during performance of the task;
- any work hold points or suspension points are understood by the workers; and
- all of the required approvals have been obtained and documented.

For high consequence, potentially high hazard, or complex work, it is often useful to perform a dry run (where practicable) to identify any potential deficiencies in the safety procedure, RWP or other work control documents. A pre-start readiness review should be conducted for high consequence and potentially high hazard work to ensure that all the necessary procedural requirements are in place.

5.2.5. Pre-job briefings

Pre-job briefings should be conducted for nonroutine work in radiological areas, routine work with changing radiological conditions, or work with a high potential for radiological issues. These briefings should include, at a minimum, the participants, planners, supervision, radiation safety and other safety personnel involved in the work. The rigor of the briefings should be commensurate with the risk of the work.

The briefings should include

- the scope of work,
radiological and nonradiological hazards,
• safety procedure or RWP requirements,
• radiological conditions upon job start,
• any changes expected to radiological conditions during the work,
• any Action Levels and Stop Work thresholds, and
• responses to abnormal conditions.

Workers should be encouraged to identify and bring to the attention of management any previously unidentified hazards and to suggest actions to mitigate the hazards. For work covering extended periods, the pre-job briefings should be provided periodically, with a higher frequency for higher risk and complex work.

5.3. Radiation Safety Training


5.3.1. Management Responsibilities

Management is responsible for providing radiation safety training for individuals with unescorted access to facilities or other areas where there is potential for exposure to elevated ionizing radiation levels. This training can range from a pamphlet, to a computer-based course, to instructor-led training, and should be commensurate with the facility’s potential radiological risk. Instructor-led training should be provided by qualified individuals who have the requisite combination of academic training and practical experience. The training must be compliant with federal and state regulations and any facility licenses or permits.

Management is also responsible for evaluating the effectiveness of the training program and identifying areas requiring improvement. Guidance on the self assessment of radiation safety programs (including radiation safety training programs) is provided in NCRP Report No. 162 (NCRP 2009b).
Training program records should be kept current and easily retrievable. These records should include individual training records and training status and examination results. They should also include course descriptions, lesson plans for classroom, computer-based training, and on the job training. NCRP Report No. 114, *Maintaining Radiation Protection Records* (NCRP 1992) provides additional guidance on maintaining radiation safety training records.

### 5.3.2. Factors to Consider when Establishing Training Requirements

The depth and breadth of radiation safety training requirements for different groups of workers will depend on several factors, including the potential for and magnitude of radiation exposure, the complexity of the task, and regulatory requirements. Additional factors include the degree of supervision received in performing the tasks, the amount of previous training received, and the degree to which the trainees will instruct or supervise others. Also, some workers may have extensive training on the production, interaction, and the biological effects of radiation so some of this more basic material may be omitted from their training.

### 5.3.3. Personnel to be Trained

Since the potential for exposure to radiation generally varies widely for different categories of workers, the appropriate scope and depth of training for each worker category varies widely as well. The typical worker categories are listed in this section and the recommended training topics for each worker are presented in Appendix A, Table A.1 of NCRP Report No. 134 (NCRP 2000b).

#### 5.3.3.1. General Employees

General employees usually do not work with radioactive material or RGDs and have minimal potential for exposure to radiation in the course of their work. Individuals who do not work with radioactive material or RGDs but are likely to receive $> 1$ mSv y$^{-1}$ due to their work location are considered radiation workers. Workers who are not otherwise radiation workers but work in close proximity to radioactive material or RGDs (*e.g.*, custodial staff, or facility maintenance crews) should be provided with radiation training that is commensurate with their job responsibilities.
5.3.3.2. **Radiation Workers**

Radiation workers generally have the most potential for exposure to radiation during their work assignments. The potential exposure could be from radioactive material (e.g., a radiochemist) or from an RGD (e.g., a radiographer). These workers will typically be required to have the most extensive training.

For complex work with a high potential for radiological exposure, the workers should receive specialized radiological training above the normal training for radiation workers. This training may include classroom instruction, or participatory training using mockups of the work area or in the actual work area if radiological conditions are benign. This training should cover the scope of work, conduct of the work, tools required, RWP requirements, potential abnormal conditions, and expected response to abnormal conditions. If mockups are used, simulation should be held to a minimum.

5.3.3.3. **Declared Pregnant Workers**

Declaration of Pregnancy is a formal process requiring the worker to sign a “Declaration of Pregnancy” form. Participation in the program offers a lower dose limit for the embryo-fetus and is always voluntary (NRC 1999 Rev 3, Reviewed 2017; DOE 2011a). General information regarding the Declared Pregnant Worker program should be included in general employee training and radiation worker training. Detailed information about the risks of prenatal exposure and controlling exposures during pregnancy should be provided when a worker is considering whether or not to declare her pregnancy.

5.3.3.4. **Packaging and shipping personnel**

DOT (2015) requires individuals whose normal duties may include packaging or shipping radioactive materials to be trained to do so.

5.3.3.5. **Contractor Personnel**

Contract workers are generally employed by a contractor at a facility for either limited or extended periods of time. This category could include maintenance workers, radiation safety technicians, security personnel, and consultants. The level of training they receive should be commensurate with their likely exposure to radiation.
5.3.3.6. **Minors**

Individuals under the age of 18 y may be permitted access to facilities where there is a potential for exposure to ionizing radiation. Minors may also come into contact with radiation sources as part of school science programs, outreach programs, and internships (e.g., programs that offer training to become a medical radiographer (x-ray technologist) or other similar profession). However, specific recommendations and regulations apply regarding their exposure to radiation. (NCRP 2018; DOE 2011a; NRC 2020a).

5.3.3.7. **Management and Supervisory Personnel**

Managers and supervisors should be classified as radiation workers or general employees based on their own work assignments. If these individuals directly supervise radiation workers, they should receive radiation safety training commensurate with the amount of direct supervision they provide to their employees. That is, if they are “hands on’ during the radiological work, they should be trained or qualified as a radiation worker. If they are “hands-off”, they should be competent to provide the supervision (e.g., by experience or prior training), but they should not necessarily have to complete the classes required for those being supervised.

5.3.3.8. **Emergency Response Personnel**

Most facilities have plans and procedures for responding to the various types of emergencies (e.g., fire, medical, chemical, radiological) that could occur at their facilities. Personnel who have been assigned responsibilities for responding to radiological emergencies (e.g., firefighters, medical staff, health physicists, radiation safety technicians, and radiation workers) should receive training on the types of radiological hazards that could be encountered and use of PPE, measurements and instrumentation, and protective strategies such as time, distance and shielding. The training objective is to provide sufficient training to allow informed judgments by responders in emergency situations.

5.3.3.9. **Visitors**

Visitors include individuals who may enter a facility as part of a tour group, to service equipment, to visit patients, or to confer with employees. Visitors who require access to areas controlled for purposes of radiation protection are typically provided access after receiving
information as to the area they are entering and appropriate instructions (e.g., wearing a
dosimeter, the need to follow the instructions of escort personnel, and what to do in an
emergency). Visitors such as visiting scientists, students, and others working in the facility
should receive training commensurate with their likely exposure to radiation, and comparable to
that received by workers performing the same tasks.

5.3.4. Design and Development of a Radiation Safety Training Program

The basic steps in the development of a radiation safety training program are discussed in
detail in NCRP Report No. 134 (NCRP 2000b). A brief summary of these steps follows:

1. A job task analysis is performed to determine the level of training and skill
   required to safely complete each task at the desired level of competence.

2. Based on the job task analysis, the training objectives, course structure,
instructional methods, and testing criteria are developed.

3. Based on the information developed in the previous step, a lesson plan and
   training materials are prepared. Examples of training plans for different categories
   of workers at different types of facilities are presented in Appendix B of NCRP

4. An evaluation plan should be developed concurrently with the lesson plan. The
types of evaluation can include feedback from the trainees on the usefulness and
effectiveness of the training, a written or oral test, and an evaluation of the
worker’s performance on the specific task.

5. Instructors should have experience in teaching and an adequate knowledge of the
   subject. Training may include lectures, demonstrations, and “hands-on” training as
   appropriate. In many cases computer-based training programs can be an efficient
   and effective training method.

6. A process should be developed to periodically evaluate the task performance and
   provide feedback to improve the training process.

Periodic retraining should be required for most worker categories. The retraining
frequency should be based on the potential for exposure to radiation, any changes in the task
being performed, and the level of responsibility of the worker. Periodic worker retraining may also be required by regulations (e.g., DOE 2011a and NRC 2020a).

5.4. Interlocks and Access Control Systems

Access control systems are designed to prevent inadvertent or unauthorized access to areas where elevated dose rates can exist. Alarm systems are designed to alert people when the radiation dose rate increases above a predetermined level and may also provide instructions and initiate mitigating actions. A detailed discussion of access control and alarm systems, and the criteria for their selection and use is presented in NCRP Report No. 88 (NCRP 1986). A brief summary relevant to external dose control is provided here.

The degree of sophistication needed for access control and alarm systems is determined by the potential for inadvertent external exposure and theft prevention. For example, the access control system for a calibration facility in which a small source is exposed and the radiation hazard is relatively minor may be a simple warning sign and a rope barrier. Whereas, the access control system for facilities, such as an accelerator, where life-threatening levels of radiation may be present, would be more elaborate and include interlocked barriers, audible and visual warnings, and explanatory signs.

An access control system can include any combination of signs, visual and audible signals, physical barriers, interlocks, run-safe switches, emergency shutdown switches, pre-startup notification and search procedures, administrative procedures, and special instructions.

Visual signals should be flashing or rotating beacons that are magenta, safety purple, red, or for dimly lit areas, black on white. A sign at each signal should indicate the purpose of the light and what actions are required.

Audible signals should be loud enough and distinctive enough to be heard over the ambient noise level, but not so loud or irritating that workers are tempted to disable them. A pulsed chime is often found to be satisfactory.

Barriers can range from something as simple as a rope to something as complex as an interlocked concrete door. However, the barrier should not prevent rapid egress in an emergency. The intrusion protection provided by a barrier should be commensurate with the potential
radiation exposure hazard. For external sources of radiation, shielding around a radiation source may be an effective access barrier.

An interlock is a device on an access control barrier that automatically prevents radiation exposure or reduces the dose rate when the barrier is opened and access to the area is permitted. Interlocks should be used as part of an access control system for areas in which there is the potential for an individual to receive an effective dose exceeding 50 mSv (Table 5.2). Interlocks should be “fail-safe”; that is, if the interlock fails, it will fail in the “safe” position and prevent high radiation levels in potentially occupied areas.

Run-safe switches and search procedures should be used in large or complex radiation exclusion areas to prevent the presence of radiation until the areas are known to be unoccupied and secured. A run-safe switch is switch that, when toggled to “safe” ensures that a specific piece of equipment (e.g., an accelerator) cannot be operated and it is safe to be in the area. When toggled to “run”, the specific piece of equipment is capable of running and potentially causing harm to those in the area. The run-safe switch is typically set to “safe” upon first entry into the area and reset to “run” as part of the prestart search of the area. Emergency shutdown switches should also be installed in radiation exclusion areas to prevent or terminate prompt radiation if the area is inadvertently occupied.

Pre-startup notification is a visual and audible indication that radiation is imminent in an exclusion area. The notification should consist of a dimming of the lights or activation of a revolving beacon to alert any persons who might remain in the area following the search. This should be followed by an oral announcement or a unique sound, or both, to indicate that radiation is imminent in the area.

Alarm systems are normally activated directly by radiation sensors that trigger a warning device or disable the radiation source when a preset radiation level has been reached. Such systems include area monitors or criticality accident alarm systems.

Area monitors are used to monitor ambient radiation levels in potentially occupied areas and will activate an alarm when a predetermined level of radiation has been reached or exceeded. They are generally used for penetrating radiations such as x and gamma rays or neutrons; however, with suitable detector design they can be used for detecting beta-particle radiation.
### Table 5.2—Alarm and access control systems as a function of effective dose that an individual might receive from an inadvertent exposure.\(^a\)

<table>
<thead>
<tr>
<th>Effective Dose Category (mSv)</th>
<th>Effective Dose Category (rem)</th>
<th>Alarm System</th>
<th>Access Control System</th>
</tr>
</thead>
<tbody>
<tr>
<td>&lt; 1</td>
<td>&lt;0.1 rem</td>
<td>None</td>
<td>Signs and ropes</td>
</tr>
<tr>
<td>&gt;1 to 50</td>
<td>&gt; 0.1 to 5</td>
<td>Visual and audible</td>
<td>Signs and barriers (^b)</td>
</tr>
<tr>
<td>&gt;50 to 250</td>
<td>&gt;5 to 25</td>
<td>Visual and audible</td>
<td>Interlocks, barriers (^b) and signs</td>
</tr>
<tr>
<td>&gt;250 to 1,000</td>
<td>&gt;25 to 100</td>
<td>Visual and audible</td>
<td>Interlocks, barriers (^b), signs, lights and audible alarms</td>
</tr>
<tr>
<td>&gt;1,000</td>
<td>&gt;100</td>
<td>Visual and audible</td>
<td>Interlocks, barriers (^b), signs, lights, audible alarms, run-safe and emergency off switches</td>
</tr>
</tbody>
</table>

\(^a\)From NCRP Report No. 88 (NCRP 1986)

\(^b\)Barriers in these dose categories normally include shielding materials such as concrete, steel, earth and lead to reduce the radiation levels in occupied areas to acceptable levels.
Criticality accident alarm systems are a special class of area monitors that detect the ionizing radiation from a nuclear criticality. Such devices must respond to bursts of high intensity neutron or gamma radiation associated with a criticality accident.

The selection of access control and alarm systems is based on the potential effective dose that an individual might receive in the event of an inadvertent exposure. In selecting these systems, the potential equivalent dose rate, the portion of the body exposed, and the potential exposure time should be used to determine a potential effective dose. General guidance for this selection process is shown in Table 5.2, which is extracted from NCRP Report No. 88 (NCRP 1986).

5.5. Posting and Labeling

Postings, which are used to identify areas, and labels, which are used to identify items, inform workers and visitors of potential radiological hazards. Requirements for radiological postings and labels are specified in federal and state regulations and typically require a standard format that includes:

- a heading that indicates the degree of hazard (e.g., “Caution”, “Danger”)
- a statement of the type of hazard (e.g., Radioactive Material Area, Contamination Area, Radiation Area)
- the standard radiation symbol as specified in ANSI N2.1 (ANSI/HPS 2011c).

In addition, signs may contain other information such as entry requirements (e.g., PPE, training, dosimetry, RWP) and contact information.

A color conflict may arise when posting Danger and Grave Danger hazards, as the ANSI format (ANSI/NEMA 2011) for these signs is a white background with a red and black header. One method of resolving this conflict is to mount the radiation trefoil symbol on a yellow circle, thus meeting both the ANSI standards and federal, and state requirements.

5.5.1. Posting of Areas

The access point to radiological areas should be conspicuously posted to inform workers of the radiological hazards in the area. For large areas delineated with boundary tape, chains, or
rope, the boundary should also be posted so that a person approaching the area is made aware of the hazard.

The posting regulations offer discretion regarding use of a CAUTION versus a DANGER heading when posting High Radiation Areas, High Contamination Areas, and Airborne Activity Areas (NRC 2020a; DOE 2011a). In general, a DANGER sign should be used if the condition being posted is sufficient to cause deterministic effects (acute injury, death); otherwise, a CAUTION sign should be used.

“No Access’ signs, such as “Caution, High Contamination Area – Do Not Enter’, or “Caution, High Contamination Area – Authorized Entry Only’ may be used following spills or accidents. Areas posted with this type of sign should be inaccessible to all except those directly authorized by the radiation safety program. “No Access’ signs should be removed as soon as the situation has been rectified. “No Access’ signs may also be used on a “permanent’ basis at entries to accelerator housings, for example, as long as the potential exists for a serious radiation exposure.

As appropriate to the hazards present, areas should be posted with a sign bearing the radiation symbol (ANSI/HPS 2011c) and one or more of the following warnings, as appropriate:

- Caution Radioactive Material
- Caution – Radiation Area
- Caution/Danger High Radiation Area
- Grave Danger – Very High Radiation
- Caution/Danger Airborne Radioactivity Area

NRC and DOE regulations vary somewhat at the thresholds for posting, but are very specific regarding the situations that require posting.

5.5.2. Labeling of Items

Radiation labels or tags should be provided on radioactive material containers, laboratory equipment used with radioactive materials, radioactive waste containers, equipment components that are radioactive or contaminated with radioactive material, and RGDs. The labels should be
conspicuous and readily identify the radiation hazard. Labels should have a yellow background and either a black or magenta trefoil, and black or magenta lettering. The label on packaged radioactive material should be visible through the package or affixed to the outside. If the outer packaging is likely to be removed, or is easily removed, the inner package should also bear the radiation label.

Radioactive material tags and labels should only be removed by a member of the radiation safety organization and the label should be defaced or treated as radioactive waste.

Specific labeling requirements are found in the applicable federal, state, and local regulations (e.g., NRC, DOE, DOT, OSHA) and the radiation safety program may impose additional labeling requirements appropriate for the facility. There are regulatory exceptions to labeling individual containers when certain conditions are met (NRC 2020a; DOE 2011a):

- The item contains less than specified quantities or concentrations of radionuclides
- The containers are attended by a qualified individual
- The containers are in transport and packaged and labeled in accordance with the regulations of the DOT
- Containers are accessible only to individuals authorized to handle or use them
- Containers are installed in manufacturing or process equipment

RGDs (e.g., x-ray diffraction units) should be posted with a sign bearing the radiation symbol and the words “Caution (or Danger) – X rays: this equipment produces X rays (or ionizing radiation) when energized”.

5.6. **Radioactive Material Operations**

Radiation safety organizations are responsible for developing and implementing programs that control radioactive contamination and maintain the security and accountability of the radioactive material at their facility. This Section presents guidance for the control of dispersible (unencapsulated) and nondispersible radioactive material, sealed radioactive sources, and contaminated areas, items, and equipment as part of an operational radiation safety program.
5.6.1. Management of Radioactive Material

Control of radioactive material is essential to ensure the safety of workers, the public, and the environment. Effective radioactive material control relies on a combination of acquisition controls, labeling (discussed in Section 5.5), limitations on where and how much radioactive material is handled and stored, material inventory and use logs, and security measures to protect against unauthorized use or removal.

The radiation safety staff should review and approve the acquisition of radioactive material and a log for tracking the use and disposal of the material should be maintained. The log should provide details of the receipt/production, usage, and disposal information. Radioactive material removed from the original container should be recorded on the log noting the date, the volume or activity removed, approximate percent of activity disposed as radioactive waste, identification number of the waste containers, and the radiation worker’s initials. When all of the radioactive material has been removed from a container, or no additional material will be removed, the radiation worker should enter the remaining activity on the log and dispose of the original container in a designated waste disposal container. Workers should record the disposal container identification number on the usage log (see Section 6, Radioactive Waste Management).

A routine physical inventory of the radioactive material possessed or stored in each location should be performed to confirm that the material has been entered in the log, authorized for use and is properly secured. The surveillance should also verify that the material is being used as approved. Unauthorized transfer or removal of any radioactive materials from the authorized location should be promptly reported to the radiation safety organization for evaluation. Inventories should verify the radionuclide, quantity, and physical form and the adequacy of the labeling and packaging and storage location. The individual location inventories should be used to establish a facility-wide inventory to support radioactive material license(s) or other applicable reporting requirements.

Radioactive material should be secured in a manner that prevents unauthorized use or removal of the source. Unoccupied storage and use areas should be locked to prevent removal of licensed materials. Control should be maintained over radioactive material not in storage. Facilities that contain high-activity radioactive material require a more secure environment and
should employ more advanced physical protection systems, such as electronic access keys, alarms and multiple layers of security, as discussed in Sections 5.4 and 5.12.

Radioactive material should only be stored in approved locations; outside storage should be avoided to prevent container degradation. The dose rate on the outside of containers of radioactive material and storage cabinets should be periodically surveyed and be commensurate with the levels allowed in the area where the material is located. If the material in storage is or contains liquid, secondary containment large enough to contain the liquid in the primary container should be used. If the radioactive material container or the contaminated item has sharp edges or projections, it should be taped or additionally protected to ensure package integrity.

Storage areas adjoining the exterior wall of a building should be evaluated to ensure the exposure rates outside the building are consistent with the radiological posting level for the area. To the extent practicable, radioactive material should not be stored with flammable or combustible material, such as cardboard containers. Fire protection measures (e.g., smoke detectors, fire extinguishers, area sprinklers) should be installed and functioning in the designated radioactive material storage area.

Transfer of radioactive material or contaminated equipment between organizations with different regulators (e.g., a DOE facility and a university) can prevent challenges if the surface contamination thresholds differ, as an item may be released as “clean” from the sending organization yet found to be “radioactive” by the receiving organization. Resolving these situations can take a significant amount of staff time and result in negative publicity and unwanted attention from the regulator. Therefore, it is prudent to know the release thresholds of each of the organizations involved prior to releasing items that have potential radiological contamination. Agreements reached between the sending and receiving facilities prior to shipment are very helpful in laying the groundwork for problem-free transfers.

5.6.2. *Management of Sealed Radioactive Sources*

A sealed radioactive source is a manufactured item containing a discrete quantity of radioactive material to be used as an ionizing radiation source that is encapsulated, plated or bonded in a matrix to prevent the dispersal of the radioactive material under the conditions of use and normal wear for which it was designed. Sealed radioactive sources are used in a variety of
occupational settings and can contain low (e.g., kBq) to very high (e.g., TBq) levels of
radioactivity. Control of sealed radioactive sources is accomplished through source
accountability and security and implementation of a program that ensures the source is used,
handled, stored, and disposed in a manner commensurate with the hazards.

NCRP Report No. 182, Radiation Safety of Sealed Radioactive Sources, (NCRP 2019)
provides comprehensive guidance on the operational radiation safety aspects of sealed
radioactive sources. Report No. 182 includes guidance for the safe handling, tracking and control
of sealed radioactive sources and provides example procedures for confirming inventories, leak
testing, labeling, safety, training, periodic inspection, and emergency response. Following is a
summary of the guidance in NCRP Report No. 182:

- Source containers (and sealed sources) should be labeled, with labels durable enough
to remain legible for the expected usage life of the source;

- Locations where sources are used or stored should be posted with the type of
information and warning based upon the degree of potential radiation hazard that the
source presents. Guidance on posting radiation warning signs is provided in NCRP
Report No. 88 (NCRP 1986). Postings should also conform to applicable regulatory
requirements of DOE (2011a) and NRC (2020a). DOE also provides guidance on the
storing and posting of sealed source storage locations (DOE 2011a).

- Sealed sources should be protected from hazards such as damage, fire or flooding.
For low-activity calibration sources, storage in a fire-resistant cabinet or safe is
normally sufficient. The level of protection should be commensurate with the hazard
potential.

- The use of flammable liquids and other materials in or near source handling or
storage areas should be minimized.

- Information should be made available to potential emergency responders as to the
location, type of radioactive material, the type of encapsulation (if any) and the
degree of potential hazard that the sources could present.
To ensure an accurate inventory of sealed radioactive sources, each source should be assigned a unique (numbered or alphanumeric) identifier and be properly labeled indicating the radionuclide, activity and date of activity determination.

Routine leak tests, inspections and inventories should be performed and documented as required by the applicable regulation, license or permit. Verification of source integrity through routine leak tests ensures that the radioactive sealed source is not leaking, which could result in the spread of contamination.

If a sealed radioactive source is lost, damaged, or missing, or otherwise unaccounted for, the radiation safety program manager, or designee, should be notified. An investigation should be conducted that documents the circumstances of the loss and efforts taken to locate the source. Proper notification of the appropriate regulating agency within specified time frames is required for higher-activity sources, as specified in the NRC license or other regulating documents.

5.6.3. **Dispersible Radioactive Material Controls**

This Section provides guidance for handling dispersible radioactive material while managing the spread of radioactive contamination and minimizing or preventing internal radiation dose.

Internal radiation exposure can occur via inhalation, ingestion, absorption, and injection. Most potential human intakes occur via inhalation, although significant internal doses have occurred via injection (e.g., after inadvertently stabbing oneself with a screwdriver or cutting oneself with a contaminated tool). Absorption through the skin can occur with tritium dioxide (HTO), radioiodine, and radioactive materials coupled with Hydrofluoric Acid (HF), but rarely occurs with other forms of radioactive material. Ingestion also occurs rarely in the occupational setting, largely as a result of administrative controls preventing eating, drinking (except from water fountains), and smoking where dispersible radioactive materials are handled.

Aside from being an internal exposure concern, uncontrolled contamination or extensive contamination creates significantly greater operating costs due to increases in training, use of PPE, operational inefficiency due to donning, wearing, and doffing anti-contamination clothing and gear, decontamination and decommissioning costs, radioactive waste generation, and
regulatory oversight. Clearly, it is in every organization’s best interest to limit radioactive contamination to the extent possible.

5.6.3.1. **Engineered Controls**

Engineered controls should be used before administrative controls and PPE to limit potential worker intakes and limit the spread of contamination. Generally, there are two types of engineered controls for dispersible radioactive material: containment and ventilation.

**Containment**: An effective method to limit the spread of contamination and the potential for airborne radioactivity is to provide devices or systems that will contain the radioactive material. A containment device could be something as simple as absorptive lay-down paper, a plastic bag, a vial, a metal can, or a glovebag; a containment system could also be many different things, including a hard-walled glovebox, a piping system, or a hot cell. Selection of the appropriate containment device or system should be based on an assessment of the type and amount of material at risk, the potential for release of material, the relative hazard of the material, the length of time the material must be contained, the type of operation, and other hazards such as the radioactive material’s associated dose rate.

**Ventilation**: Ventilated containments (e.g., glovebags, gloveboxes) can be maintained at a negative pressure with respect to the area around it to prevent release of the material through any leaks in the containment. Ventilation can also be used to reduce the concentration of airborne radioactivity by exhausting the air through a filter and diluting contaminated air with clean make-up air. The contaminated air is typically exhausted from the area via the facility’s permanent ventilation, cleaned by filtration or other means, and the clean air is resupplied to the area or discharged, often via a stack. Another use of ventilation is to draw contaminated air away from the worker’s breathing zone using facility ventilation (e.g., as with a chemical fume hood) or by using local exhaust ventilation.

Workplaces used for handling dispersible radioactive material in a laboratory environment utilize containment and ventilation, as well as surfaces and construction methods that enhance decontamination. The three basic workplace types are a bench top, a chemical fume hood, and a glovebox. There are many variations of these workplace types and modifications may need to be made based on the particular work to be performed. As an example, a hot cell is a
heavily shielded version of a glovebox that has its glove ports replaced with manipulators. As discussed in the following subsections, each basic workplace type provides a different level of protection.

5.6.3.2. *Bench top operations*

The laboratory bench top provides the least protection for the worker and the work area. The bench top and the surrounding floor area should be constructed of smooth and impermeable materials to facilitate cleanup in the event of a spill and the work area should be visually identified (*e.g.*, with tape, lay down paper, a tray). Local radiation shielding is provided when necessary. Typically, bench top operations are set up within a laboratory-type workplace with at least 6 air changes per hour and restricted access.

5.6.3.3. *Chemical fume hoods*

A chemical fume hood provides a single barrier between the worker and the radioactive material and is suitable for higher levels of radioactivity than a benchtop. In this case, the “barrier” is the flow of air into an enclosure which carries airborne radioactive material away from the worker. The velocity of the air flow into the fume hood should be adjusted to minimize the airborne material both inside and outside the fume hood. The recommended face velocity is 0.4 to 0.6 m s\(^{-1}\) (80 to 120 feet min\(^{-1}\)) (NIOSH 2012, NCRP 2017). One common fume hood design maintains a constant volume of air flow into the enclosure so the face velocity varies inversely with sash height. This design can result in significantly different air velocities entering the enclosure when the sash is raised or lowered, which may result in air velocities which are too high, thus causing turbulence around the enclosure opening and unnecessary resuspension of material in the fume hood. Conversely, air velocities that are too low may not maintain adequate confinement of suspended materials inside the fume hood. A preferable design is the constant-velocity fume hood where the air volume into the enclosure is varied to maintain the same air velocity across the fume hood opening regardless of the sash height.

The surfaces in the fume hood should be smooth and impermeable and have rounded corners to facilitate decontamination. The floor of the fume hood should have raised edges to contain any spilled liquids. The surrounding floor area should be smooth and impermeable to facilitate decontamination. The exhaust from the fume hood is typically equipped with one or
more filters depending on the nature of the material being handled. As is the case for bench top operations, fume hoods are typically set up in a laboratory-type workspace with restricted access and radiation shielding is provided as necessary.

5.6.3.4. **Gloveboxes**

Gloveboxes are suitable for handling very high levels of radioactivity (particularly alpha particle emitters), as they provide two barriers between the worker and the radioactive material. Most gloveboxes provide minimal shielding, however, so they are not necessarily suitable for large quantities of gamma ray emitters (e.g., $^{60}$Co, $^{137}$Cs). The primary barrier provided by the glovebox is the physical enclosure, which is typically equipped with glove ports. The second barrier is the negative pressure maintained inside the glovebox by a HEPA-filtered exhaust system. The negative pressure is maintained by controlled air flow into the glovebox through a filtration system. Other gas mixtures may be required to ventilate the glovebox depending on the material being handled (e.g., pyrophoric materials). The ventilation flow rate into the glovebox should allow the gloves to remain flexible enough to perform the necessary work and maintain an adequate air flow into the enclosure in the event of the loss of a glove. Current guidance is that velocities should be $0.64 \pm 0.13$ m s$^{-1}$ through an open glove port to adequately limit the release of material from the glovebox in the event of a glove failure (DOE 2003; NCRP 2017).

The surfaces inside the glovebox should be smooth and impermeable and have rounded corners. Gloveboxes should be housed in a laboratory-type workspace with restricted access. Typically, the room with the glovebox is maintained at negative pressure relative to the hall or adjacent areas so that if the room becomes contaminated, the ventilation assists in keeping it contained within the room. Depending on the nature of radioactive material being handled in the glovebox, the exhaust from the workspace may be equipped with one or more HEPA filters to minimize the release of radioactive material to the environment in the event of a spill.

General guidelines on glovebox design and operation are available from a number of sources including the American Glovebox Society (AGS 2007) and the NCRP (2017), and specific advice on gloveboxes in nuclear applications is also available (AGS 2017; NCRP 2017).
5.6.3.5. **Workplace Selection Criteria**

Dispersible radioactive materials are used in a wide variety of applications and it is typically the health physicist’s responsibility to recommend a suitable workplace for a proposed operation using these materials. This recommendation should be based on an estimate of the potential radiological hazard posed, considering the following basic factors:

- the quantity of the radioisotope(s) being handled
- the specific activity of the radioisotope(s)
- the radiotoxicity of the radioisotope(s) based on the annual limit on intake (ALI)
- the dispersibility of the material (e.g., particle size, solid or liquid form)
- the processes to be performed (e.g., grinding, mixing, transferring, dissolving, heating), and
- the mass of material to be handled.

In addition to the potential radiological hazard, other factors should be considered, such as the material’s ability to penetrate the skin, volatility, pyrophoric properties, chemical toxicity, and external dose rate from beta and gamma radiation. These factors may influence the type of workplace selected, the types of PPE used by workers, the type of gloves installed on a glovebox, the atmosphere in a glovebox, and the level of shielding required.

Several techniques have been proposed to aid in selecting a suitable workplace for working with dispersible radioactive materials (Homann and Aluzzi 2014; Delacroix et al. 2002; ILO 1989). Each of these techniques considers the toxicity of the radionuclides and the likelihood that the radioactive material will become airborne. The HotSpot Health Physics Codes (Homann and Aluzzi 2014) are available online in a computerized version and incorporate all of the factors in the bullet list above. Appendix B provides an example of the use of the HotSpot software tool.

5.6.4. **Contamination Control**

Radioactive material that has escaped its containment usually results in surface and/or airborne contamination. In addition, some routine activities (e.g., work with uncontained
dispersible radioactive material, maintenance of contaminated equipment or systems,
decontamination) have the potential to contaminate facilities, equipment, and people. Regulators (e.g., NRC, DOE, DOT, individual states) establish contamination thresholds above which an item or area must be controlled as “radioactive”, or “radioactively contaminated”. These thresholds vary by types of emission, radionuclide, and by the regulating agency.

Airborne radioactivity may consist of radioactive gases and their daughters (e.g., radioiodine, radon and radon progeny) or suspended particulates inadvertently released from their containment (e.g., via a hole in a glovebox glove, a break in a contaminated duct) or resuspended from area and equipment contamination. The potential for surface and airborne radioactivity can be reduced by keeping the amount of material at risk (i.e., the amount of material that is out of the storage container) to a minimum, removing surface contamination through decontamination of areas and equipment, and minimizing or eliminating resuspension in air by covering or fixing surface contamination.

Decontamination should be conducted starting at the edges and working inwards to the highest levels of contamination. Methods used to decontaminate tools, equipment, and surfaces include dry wiping, rinsing with water, washing with water and detergent, treatment with special decontamination solutions, ultrasound treatment, and abrasive decontamination. In some cases, special detergents or equipment are required, such as sand-blasting, hydro-lasing, and other techniques used to remove contaminated materials from some surfaces. Stripping materials are also quite effective in removing surface contamination and have the added benefit of tying down the material so that it cannot become airborne. Selecting the appropriate method requires consideration of the monetary value of the contaminated item, degree and type of contamination, the materials, size, and construction of the item, and the criteria for acceptable levels of residual contamination. In selecting a decontamination method, consideration should also be given to the potential radiation exposures of all workers involved, including those performing the decontamination, and the potential for generating additional radioactive or mixed wastes. Following effective decontamination and prior to release of an item for unrestricted use, radioactive material labels/tape should be removed or defaced. Documentation of the survey or evaluation should be maintained as required by the applicable regulation or license requirement.
If decontamination is not practical, contamination should be prevented from spreading by covering the area or bagging the item, fixing the contaminant with paint, or removing forces that are likely to spread the contaminant (e.g., by closing a vent that is blowing on the contamination). When used, fixative agents should be applied with a method that does not create airborne radioactivity. If paint is used, the area should be labeled with “under paint contamination” and, if the spot is on the floor, it is preferable to use two layers of different color paint so it is visually obvious when the first layer is wearing away and needs to be repainted.

Workplace monitoring is an essential component of contamination control and is discussed in Section 5.9; contamination monitoring is discussed in Section 5.9.3.

Internal radiation dose monitoring may be recommended or required for workers handling dispersible radioactive materials. See Section 8.2.

Use of PPE is essential when workers are handling dispersible radioactive material, working with contaminated items, and working in contaminated areas. See Section 5.8.

The response to spills of radioactive material is provided in Section 9.1. Management of contaminated persons during an accident or an emergency is outside the scope of this discussion, but NCRP Report No. 161 (NCRP 2009a) provides a detailed discussion of the management of persons contaminated with radionuclides. Other documents (IAEA 1988; ICRP 1978; NCRP 1980) also address the subject.

5.7. RGD Operations

An RGD is an electric or battery-powered device (including accelerators and certain high-power lasers) that generates ionizing radiation either incidentally or intentionally. Some programs choose to include radiography and backscatter devices that contain a radiation source (e.g., ¹⁹²Ir, ¹³⁷Cs, ⁶⁰Co) in their RGD program, as these devices can effectively be turned on and off like an x-ray machine. However, in this Report, such devices will be considered “radioactive material” and are therefore not included in the RGD program.
5.7.1. **Types of RGDs**

RGDs comprise a wide array of devices, including those that produce radiation intentionally (e.g., x-ray diffraction and fluorescence analysis systems, flash x-ray machines, cabinet RGDs, industrial radiography equipment, medical diagnostic and therapy equipment, and particle accelerators), as well as those that produce radiation incidentally (e.g., electron microscopes, high-voltage electron guns, electron arc-welding machines, evacuated high-voltage electronic devices, electron beam devices with energies greater than 5 kV).

Some high-powered lasers can also produce ionizing radiation (e.g., those where the intensity times the wavelength-squared exceeds $10^{18}$ W). Ultra-intense laser interactions with solid materials must be evaluated on a case-by-case basis as it can lead to the production of energetic electrons (up to 100 MeV), high-intensity bremsstrahlung, charged particles, particularly energetic protons, and photo- and proton-induced neutrons.

Unmodified commercially available components or devices that retain the manufacturer labels and have the following characteristics are typically not included in RGD programs if:

- radiation is produced incidentally; and
- the potential across the terminals is less than 15 kV; and
- radiation fields are no more than twice background when measured at five centimeters from the device surface (or at the closest accessible surface) when operated at the maximum operating parameters.

Examples include high-voltage switches, planar triodes, power supplies containing various types of thermionic valves installed in shielded cabinets or racks, mass spectrometers, vacuum switches, spark-gap devices, and electronic components such as cathode ray tubes.

5.7.1.1. **Cabinet-type RGDs**

A cabinet RGD is an x-ray system with the x-ray tube installed in an enclosure (i.e., the cabinet), which, independently of existing architectural structures except the floor on which it may be placed, is intended to contain at least that portion of the material or item being irradiated, provide radiation attenuation, and exclude personnel from its interior during generation of x rays (FDA 2008).
Cabinet RGDs come in many different configurations and the following factors should be addressed when establishing a cabinet RGD program.

The RGD enclosure should be designed so that the radiation level does not exceed 0.5 mR in 1 h at any point 5 cm outside the external surface (FDA 2008).

Enclosure access points should be interlocked so that the radiation source is terminated if the enclosure is opened. Each access point should have two independent (i.e. separate circuits to activate termination) interlocks to prevent a single point failure.

A labeled key-activated switch should be used so that radiation generation is only possible when the key is in place. The keys used to turn on the RGDs should be strictly controlled to prevent any unauthorized use. In the case of software-controlled RDGs, procedural safeguards should be implemented to prevent operation of the RGD by unauthorized personnel (e.g., the access should be password protected), and to ensure that all required safety systems are functioning and properly set.

A warning light (e.g., “x ray on”) indicating when radiation is being generated should be visible from any potential access point to the enclosure and should be illuminated only when the RGD is energized.

Depending on the potential radiation hazard if the key switch, interlock system, or other component should fail, consideration should be given to installing an area radiation detector to alert workers of the failure.

The RGD enclosure should be labeled to indicate that radiation is produced when the machine is energized.

5.7.1.2. **Open-Beam/Diffraction-Type RGDs**

There are some operations that make the use of a fully enclosed RGD impractical. Examples include x-ray diffraction and analysis machines and other RGD operations where there is the need to adjust the component configuration with the beam on (e.g., target and detector locations, beam slits, and collimators). In these situations, beam shutters are typically used to intercept the primary beam to prevent it from reaching the locations inside the enclosure when component adjustments are being made. Shutters should be interlocked with the enclosure’s access ports to prevent access if the shutter does not close. Since these adjustments will be made
inside the shielded enclosure, the radiation dose rates could be much higher inside than outside
the enclosure during closed beam operations. In these situations, the radiation dose to extremities
should not exceed 0.25 mSv in 1 h and should not exceed 0.025 mSv in 1 h to the whole body
(ANSI/HPS 2001b).

Also, provisions (e.g., a strategically placed, installed radiation detector) should be made
to detect the failure of a primary beam shutter. Normally, the operation of these RGDs should be
covered by a safety procedure (see Section 5.2.3).

Open-beam RGDs should also have labeled key-activated switches (or, if software
controlled, a warning meant to alert operators that RGD is being activated), warning lights, area
radiation detectors, and appropriate labeling.

When the RGD is de-energized, a hand-held survey meter should be used (and may be
required by regulations) to verify that radiation levels have returned to normal or to alert
operators of unexpected radiation levels.

5.7.1.3. Accelerators

There are many different types of accelerators, ranging from single stage electron
accelerators to large, heavy-particle accelerators. A useful summary of the different types of
accelerators and their applications is provided in NCRP Report No.144 (NCRP 2003b).
Accelerators are typically capable of producing very high intensity beams of radiation which
require elaborate safety systems for personnel protection. In most cases, accelerators are operated
in specially designed facilities with many built-in safety features (e.g., shielding, interlock
systems, warning lights, access control systems, ventilation systems, and area radiation
monitoring systems). Except in medical diagnostic or therapeutic applications, accelerator rooms
are normally evacuated during operation. See Appendix C.2 for more information regarding
radiation safety programs in accelerator facilities.

5.7.1.4. Portable RGDs

Portable, hand-held RGDs are often battery-powered, and have low primary beam
intensities that use back-scattered x rays to determine which elements may be present in the
sample, or the concentration of a particular element or compound (e.g., determining the presence
or concentration of lead in paint). Due to their nature, the sides and back of these portable RGDs are shielded to the extent practicable, and the beam is collimated. Some units have a plunger-type or other proximity switch to ensure close proximity to the sample before x rays are generated, but nevertheless, workers should be instructed in their proper use of the portable RGD.

5.7.1.5. Medical RGDs

RGDs in a medical setting may comprise all of the types of RGDs included in the previous subsections, including cabinet-type RGDs for tissue sample analysis, and open-beam RGDs and accelerators for the diagnosis and treatment of illness, all of which may be stationary or portable. Because these RGDs may be used in highly-trafficked areas, and are used to intentionally expose humans to radiation, both the shielding plan and machine output need evaluations specific to the medical use of the equipment. Additionally, administrative controls should be established regarding the use of mobile RGDs in a medical setting, where members of the public (e.g., other patients) may be in close proximity. Guidance regarding shielding for medical facilities to protect both workers and the public is referenced in Section 4.4, and guidance regarding additional worker protection in these settings is referenced in Section 5.8.2.

5.7.2. Management of RGDs

Detailed guidance addressing radiation safety, facility design and installation, testing and radiological surveys of RGDs is provided in several ANSI/HPS standards (ANSI/HPS 2001b, 2005, 2008, 2015).

5.7.2.1. RGD acquisition

The RSO or RGD program manager should ensure workers are aware of which types of devices are within the scope of the RGD program, and review and approve the acquisition of RGDs. The operating (line) organization should promptly notify the RSO of receipt of an RGD.

5.7.2.2. RGD inventory

The RSO should assign a unique identification number to each RGD and enroll it in the RGD inventory. At a minimum, the inventory should include administrative information (e.g.,
the name and contact information of the responsible individual, the location of the RGD; technical information (e.g., the make and model of the RGD; the energy and current operating parameters); and inventory information (date last inventoried, due date for next inventory; date of last survey/tests, due date for next survey/tests).

It is appropriate to have a longer inventory frequency (e.g., 1 y) for lower hazard RGDs and those that are nonoperational or in storage, and a higher inventory frequency (e.g., 6 months) for higher-hazard RGDs. It may be efficient to couple the inventory with the periodic survey, but if the RGD is temporarily out of service, the periodic (official) survey will become separated from the inventory.

5.7.2.3. Periodic surveys and tests

Prior to routine use and periodically thereafter, each operational RGD should be surveyed and the protective devices (e.g., interlocks, warning lights) tested. Additionally, these surveys and tests should be repeated whenever changes or repairs are made that could adversely impact the radiation safety controls or dose rates in accessible areas.

Some RGDs are subject to federal, state or local regulations including registration, licensing and compliance with regulatory requirements. Regulatory requirements address operator training, area posting and labeling requirements, surveys, and reporting and must be implemented. For example, state regulations and the Joint Commission (formerly, Joint Commission on Accreditation of Healthcare Organizations (JCAHO)) directives require certain medical RGDs (e.g., accelerators, mammographic x-ray units) to be calibrated before they may be used on humans. These calibrations are to be performed at the time of installation and at least annually, or upon repair of certain components. Additionally, medical RGDs may have daily, weekly, monthly, or quarterly testing associated with them, based on manufacturer recommendations or imposed by state regulation to ensure beam output, collimators, or other elements important to safe use are functioning as expected.

5.7.2.4. Operating procedures

In addition to a safety procedure, RGDs should have written operating procedures that addresses:
1891  • Normal operating instructions
1892  • Methods and occasions for locking and securing the RGD
1893  • Calibration and alignment
1894  • Interlock test procedure
1895  • Documentation of operational checks, inspections, and testing records
1896  • Control of keys or computer log-on/passwords used for RGD operation
1897  • Emergency response, including alarms, set-points, and required notification
1898  • Documentation and verification of adequate shielding.

1899  **5.7.2.5. Use logs**
1900  RGDs should have a use log for recording the date and time of use, the operator’s name, and the power settings. For portable RGDs, the use location should also be recorded.

1902  **5.7.2.6. Relocating, storing, and disposing of RGDs**
1903  The responsible RGD “owner” should notify the radiation safety organization prior to relocating or disposing of an RGD. Radiation-generating devices that will be placed in storage or relocated should be securely disconnected from a power source.
1906  RGDs that are to be disposed in municipal landfills should be rendered permanently inoperable to prevent the device from inadvertent use. RGDs may contain regulated hazardous components including beryllium and printed circuit boards that must be disposed of via specific waste streams. Operator hazardous waste management personnel should be consulted to determine the appropriate methods for disposal of RGDs.

1911  **5.7.3. RGD Workplaces**
1912  There are many factors that influence the selection of a suitable workplace for operating RGDs. These factors include:
1914  • the type of radiation produced (e.g., x rays, gamma rays, and neutrons);
1915  • the energy and intensity of the radiation produced,
• the expected on-time \((e.g., \text{hours energized per day})\);

• the complexity of the work to be performed;

• the size of the beam to which an individual could be exposed in an accident situation
\((e.g., \text{a 1 cm diameter collimated beam of x rays versus a wide-angle beam of gamma rays from an accelerator target})\).

• The inherent shielding provided by the room walls, floor, and ceiling.

There are some situations \((e.g., \text{the radiography of large pieces or assemblies})\) where an overhead crane is necessary to emplace the items to be radiographed. In these situations, a dedicated, shielded enclosure is often designed and constructed, with walls suitably high (three meters or more), but with no roof. The RGD is located on a mobile stand or assembly inside the enclosure, and the controls are on the outside of the enclosure. The crane operator delivers the piece and assists in the proper positioning for the radiography to be performed, and then moves away. The beam orientation is factored into how far the crane (with its operator) is to be removed from the enclosure. This “safe-distance” is usually pre-determined such that the crane operator is not exposed to radiation levels above 5 µGy per hour. Were the crane to move toward the enclosure, the RGD operator should be immediately and automatically notified so RGD operations can be terminated, or the crane should otherwise be prevented from moving into the higher radiation field.

In situations where there is the potential to inadvertently expose other workers to high doses of radiation, warning lights that illuminate when the RGD has been energized, key-activated switches, and area radiation detectors should be positioned around the enclosure to alert personnel of pending x ray generation.

Some situations require the use of RGDs in the field—\(i.e., \text{outside of a specifically designed enclosure})\). These may include radiography, mobile medical radiography, and the use of hand-held RGDs \((e.g., \text{for sample analysis})\). In these situations, the operator must be concerned not only with the proper operation of the RGD but maintain a high level of situational awareness \((\text{NCRP 2000a})\). Depending on the hazard associated with the RGD, a hand-held survey meter may be necessary.
5.8. Personal Protective Equipment

Personal protective equipment can provide protection from both radioactive contamination and from external sources of radiation and should be provided by management to supplement engineered and administrative controls. In this Report, PPE includes any item worn by the worker for the purpose of protection (e.g., lab coat, coveralls, shoe covers, gloves, leaded vests, leaded gloves, respirator, hard hat, safety glasses, face shield). Protective clothing (PC) is a subset of PPE and generally includes any combination of lab coats, hoods, coveralls, shoe covers, and gloves.

5.8.1. General Considerations

PPE should be specified by the radiation safety staff and the requirements should be recorded in an RWP or procedure (see Section 5.3). Because PPE can become damaged with use or in storage, and lead shielding in vests and aprons can shift with time, it should be inspected before use and periodically thereafter to verify continued integrity and usefulness. Damaged PPE should be replaced.

Care should be taken when specifying PPE because of its potential to cause the wearer physical strain (e.g., heat stress when wearing double or waterproof PPE in hot or humid environments, limited ability to see peripherally when wearing a full-face respirator).

5.8.2. Protective Clothing for External Radiation Control

PC for external radiation is effective primarily against beta particles and x and gamma rays with energies less than 200 keV. The most common types of PC for external radiation are lead-impregnated gloves, aprons, vests, and thyroid collars. The weight of lead limits its practical thickness in aprons and vests to about 1 mm, which provides attenuation in excess of a factor of three for photons less than 200 keV. The lead thickness in gloves is limited to about 0.03 mm to limit the loss of flexibility. Specific recommendations for medical workers are given in NCRP Report No. 102 (NCRP 1989a).

Plastic glasses or face shields are useful for reducing the dose from beta particles. The thicknesses of the devices should be selected such that they are effective in absorbing the beta particles emitted from the source being handled.
A lab coat, or a single or double layer of coveralls worn to prevent direct contamination of the skin, can also provide some protection against beta particles. The actual protection afforded depends on the thickness of the material and the energy of the beta particles (Farrell and Hudson 1985; Hudson 1983; Shonka et al. 1990).

When selecting PC for external radiation, an evaluation should be made of the potential increase in working time and radiation dose caused by the loss of dexterity and the added weight of the clothing. It is possible that the anticipated dose reduction provided by the clothing will be more than offset by the concomitant increase in the time spent in the radiation field.

5.8.3. Protective Clothing for Contamination Control

PC worn to prevent skin and personal clothing contamination should be durable and effective. Tightly-woven cotton or synthetic fabrics and disposable paper clothing are usually acceptable for work in dry areas. Waterproof garments are necessary when liquids are likely to splash onto the body. Shoe covers prevent tracking contamination into uncontaminated areas. Gloves protect the hands from becoming contaminated during work. Hoods are used when the potential exists for contamination of the head.

The effectiveness of PC depends on:

- selecting the appropriate PC for the job
- training in the proper techniques for donning and doffing PC
- proper laundering (for reusable PC).

Lab coats provide a minimal level of protection and should be used in the areas with only low levels of dispersible contamination, and where the work does not involve kneeling, sitting, or other actions that would be likely to transfer contamination to unprotected areas of the body.

In areas where the contamination is higher, and for work that requires kneeling or sitting on potentially contaminated surfaces, a single coverall is typically worn over street clothes or modesty garments. Most coveralls are donned via a front zipper, but some are donned by stepping into the coverall, pulling the coverall up, inserting arms in sleeves, and then pulling the neck opening closed with a draw string. For work in highly contaminated areas, the zipper opening should be secured with tape or Velcro.
In areas where contamination levels are sufficiently high to require respiratory protection to prevent the inhalation of resuspended contamination, a second layer of PC should be used. When doffing, the outer layer should be considered potentially contaminated. Hence, the outer PC is normally disposable because handling and laundering the used clothing may generate airborne radioactivity or exceed allowed levels in the laundry facility. For highly contaminated PC, a fixing agent may be applied to the outer layer before doffing to mitigate resuspension of the contamination. The outer PC should be removed before doffing respiratory protection equipment.

Launderable PC should be clearly marked for radiological use and should remain under radiological control even after laundering.

### 5.8.4. Respiratory Protection

The respiratory protection program should be administered by a qualified individual to ensure necessary components of the program are established and implemented. The program should be audited annually by the program administrator and periodically audited by a knowledgeable, objective person not directly associated with the program. The frequency of this outside audit should be determined by the size and complexity of the respiratory protection program and previous audit findings (ANSI/ASSP 2015).

Respirators should be physically controlled so they are available only to qualified users. Qualifications should be checked before each issuance of a respirator. The user should perform a seal check prior to each use of a tight-fitting respirator.

Caution and judgment must be exercised in determining the need for respiratory equipment because it may increase physical stress, impede the worker’s vision and extend the time required to complete the work. Thus, the use of respiratory equipment may lead to increased risk of injury and increased external radiation exposure (Dooley and Barresi 1994; Lee 1994). The goal of the respiratory protection program is to control the total radiation dose and to limit the total risk to the individual.

When respiratory protection is required, the respiratory protection program should have written procedures for the key aspects of the program discussed below. Additional details are provided by ANSI/ASSP (2015); ASTM (2019, 2020); Colton et al. (1991); and the NRC (2001).
5.8.4.1. **Medical Evaluation**

Wearing a respirator imposes physiological demands on the wearer. The medical evaluation identifies those conditions which may endanger the user when wearing a respirator and should be performed by a licensed physician experienced in occupational medicine. The physician should be aware of the potential environmental conditions in which the respirator would be worn (e.g., full-face negative pressure respirator with double coveralls in high heat and humidity). The medical evaluation should be conducted before respiratory fit testing and initial respirator use in the field.

5.8.4.2. **Respirator Selection**

In making the decision to use respirators, all airborne hazards including (nonradiological hazards) should be assessed and protected against. The type of respirator selected should be based on the required protection factor and the potential level of airborne radioactivity, while considering other factors such as the impact to set up and use the device; the impact on field of view, communication, and mobility; and stress to the wearer. For example, a tight-fitting respirator limits the visual field of view of the wearer potentially creating trip and fall hazards.

5.8.4.3. **User Training**

Respirator users should be trained before fit testing and annually thereafter. At a minimum, the training should include inspection, donning and doffing, seal checking of tight-fitting respirators, capability and limitations of the respirator, physical or medical reasons that preclude use, changes in facial features affecting the seal of tight-fitting respirators, and retreat from the respirator use area after respirator malfunction or user distress.

5.8.4.4. **Fit Testing**

Respirator users should be fit tested on each type and size of respirator which they are qualified to wear; fit testing should be conducted annually thereafter, and after changes affecting face shape (e.g., significant changes in weight, jaw surgery). Before the fit test is conducted, the wearer should have a current medical evaluation and training on the respirator. Skin should be clean shaven in the sealing areas for tight-fitting respirators. The test can be qualitative or quantitative, but quantitative is preferred because it gives a numeric indication of the degree of
protection to the wearer and the qualitative fit test should only be used when protection factor of 10 or less is needed. The test should be designed to challenge the fit of the respirator. Title 29 CFR 1910.134 (DOL 2018) lists eight exercises to perform during a fit test to challenge the fit of the respirator.

5.8.4.5. Maintenance and Cleaning

Respirators should be maintained and repaired by individuals specifically trained to perform this work. Respirators that are reused by different individuals should be cleaned and sanitized before use by another individual and sealed in an appropriate container to maintain that state between uses. Cleaning and sanitization should be performed in accordance with the manufacturer’s instructions to preclude damaging the respirator.

5.8.4.6. Air Sampling

When wearing respiratory protection equipment, air samples should be taken to characterize the airborne radioactivity hazard to which the wearer is exposed. These samples should be representative of the breathing zone air.

5.9. Radiological Monitoring and Surveillance

Radiation and contamination surveys should be conducted in areas where radioactive material is handled or stored and where RGDs are operated. Air sampling or monitoring should be conducted if work with dispersible radioactive material could result in airborne radioactivity and internal doses at or above the monitoring threshold. The frequency and extent of the surveys should be reviewed periodically and adjusted as needed for evolving radiological conditions.

The instrumentation used to perform these surveys must be capable of accurately measuring the types of radiation, at the levels expected, and under the expected environmental conditions (e.g., in radiofrequency (RF) fields, in high humidity, at very high or low temperatures) Radiation safety instrumentation is discussed in Section 7.1

5.9.1. Purpose of Radiological Monitoring/Surveillance

The purposes of radiation and contamination surveys and air monitoring are to:
characterize the radiation and contamination levels and boundaries and airborne radioactivity so the area can be properly posted and controlled;

provide the information needed for work planning so that radiation doses and contamination levels can be maintained ALARA;

record long-term trends in the work environment;

verify the integrity and effectiveness of engineered controls (e.g., shielding, ventilation) and operating procedures;

ensure changes in radiological conditions are discovered in a timely manner.

provide supplemental data for dose assessments; and to

establish the extent to which radioactive material is present on surfaces of equipment and facilities and the extent to which that material may be transferable.

5.9.2. Frequency of Radiological Monitoring/Surveillance

Surveys should be conducted at a frequency that is commensurate with the potential for changes in the radiological environment and the magnitude of those changes. For example, weekly contamination surveys might be appropriate in routinely accessed contamination areas, whereas quarterly contamination surveys might be appropriate in radioactive material storage areas. Quarterly or semi-annual contamination surveys might be appropriate to verify the boundary of inactive, generally inaccessible contamination areas.

As-needed (nonroutine) surveys should be performed to evaluate conditions that have not been previously surveyed or when there is the potential for a change in radiological conditions. For example:

- during the initial operation of a newly installed RGD or radiation source;
- following the modification of an RGD or radiation source;
- during and following radiological operations that potentially alter the status quo (e.g., when opening a system or container with known or suspected internal contamination);
• following the introduction of radioactive material into an area;
• following modification of the shielding around a radiation source; and
• following an incident in which an elevated external radiation exposure or a loss of containment of radioactive material is suspected or has occurred.

5.9.3. Contamination Monitoring of People and Equipment

Contamination monitoring of people and equipment (as well as other items) is essential to prevent the spread of contamination into clean areas and to control potential internal exposures. Equipment monitoring should include smears (a.k.a. “swipes”) to determine the extent of removable contamination, and direct monitoring to determine the level of total contamination. In this context, “direct monitoring” refers to the use of hand-held radiation detector used to survey a person and equipment. Direct monitoring generally does not require formal survey documentation unless unexpected readings are detected or the equipment is being released from radiological control.

People should be monitored for contamination:
• upon exiting contamination and airborne radioactivity areas;
• after working in radiological gloveboxes, containments, and fume hoods;
• when breaching contaminated or potentially-contaminated systems;
• following decontamination or decommissioning work,
• after monitoring for release or clearance of materials,
• after disturbing contaminated soil.

Equipment should be monitored for contamination:
• prior to release from a contamination-controlled area
• as needed during work evolutions to determine the extent or level of contamination.

The number of survey measurements should be sufficient to characterize the contamination on or in the item, considering the potential for nonuniform contamination.
5.9.4. Area Monitoring for External Radiation Exposure

Area monitoring (as opposed to monitoring with hand-held radiation detectors) can provide an important adjunct to the monitoring and surveillance program. There are two types of area monitors: active and passive. Active area monitors are powered instruments that provide real-time measurements and usually have an alarm capability with local and/or remote readout. Some active monitors record the radiation levels over time and the data can be used to document radiological conditions. Other active area monitors simply provide an alarm, thus notifying workers of rapidly changing radiological conditions. Passive monitors are devices such as TLDs and optically stimulated luminescence dosimeters (OSLDs) that provide a retrospective measurement that integrates the radiological conditions over the time period they are deployed. See Section 8 for detailed information on radiation dosimeters.

Active, fixed area monitors are typically used to monitor ambient radiation levels in potentially occupied areas where the potential exists for a significant increase in the ambient radiation levels (e.g., around some types of RGDs). These monitors provide an ongoing measurement at their predetermined and fixed locations and normally activate an alarm or disable the source when a predetermined radiation dose or dose rate is exceeded.

Passive area monitors are typically used in areas where the radiation environment is expected to be relatively stable and low (e.g., in certain Radioactive Material Areas (RMAs) and where RGDs are used). When placed in such areas, they can be used to supplement or replace personal dosimeters, especially in areas where the annual external effective dose is expected to be less than 0.1 mSv. Passive area monitors can also be used to verify the effectiveness of engineered and administrative controls in areas adjacent to Radiation Areas (RAs), High Radiation Areas (HRAs), and Very High Radiation Areas (VHRAs). The design and implementation of an area monitoring program should include a technical basis document that describes, as appropriate:

- the purpose of the area monitoring program
- the type radiation to be detected (e.g., gamma rays, neutron radiation)
- the range of dose rates that may be encountered
- the target location of the detectors relative to the workers and the radiation source
2163  • the operating and response characteristics of the dosimeters
2164  • effects of the environment on the dosimeters (e.g., if used in outdoor or very hot
2165  locations)
2166  
2167  Area and environmental dosimeters should have associated action levels that trigger a
2168  review by the radiation safety staff. These action levels should be set at some fraction of the
2169  applicable dose limit, while accounting for the occupancy of the area.

2169  5.9.5. **Assessing Airborne Radioactivity**
2170  Measurements of airborne radioactivity can be either active or passive. In this Report,
2171  “air monitoring” refers to measurements with CAMs that have real-time alarm capability, and
2172  “air sampling” refers to passive samples (e.g., air samples collected on filter paper or silica gel
2173  (for detecting HTO in air)) that are collected for retrospective laboratory analysis. CAMs may
2174  also be used as part of the passive monitoring program insofar as their filter papers can be
2175  submitted for laboratory analysis.
2176  “Routine” air monitoring/sampling means ongoing sampling/monitoring not associated
2177  with a specific task or job, whereas “task-specific” or “job-specific” sampling/monitoring refers
2178  to short-term sampling/monitoring that lasts for the duration of the task or job.
2179  Areas where there is a potential for airborne contamination should have an air monitoring
2180  or sampling program. There is no specific threshold for when to use air monitoring, which
2181  typically necessitates the use of relatively expensive equipment: DOE (2011) requires air
2182  monitoring “as necessary to warn workers of airborne concentrations that warrant immediate
2183  action to terminate inhalation of airborne radioactive material,” and the NRC (1993b) indicates
2184  air monitoring is warranted as needed to “alert staff when concentrations rise far above and
2185  remain above the DAC” (i.e., the Derived Air Concentration). Due to the cost of air monitors, it
2186  is important to optimize the number of CAMs used in an area. Whicker, et al. (2003) provides a
2187  quantitative method for optimized placement of CAMs and shows that in large rooms, five or
2188  more CAMs provide no better protection than four CAMs. On a practical level, CAMs are
2189  generally warranted in areas that require a glovebox for worker protection; a CAM alarm is
2190  frequently the first indication that an intake may have occurred.
The air sampling/monitoring location is important for the evaluation of potential exposures to airborne radionuclides. Fixed sampling locations, especially if they are located close to the breathing zones of the workers, may be acceptable for approximating worker exposures. However, studies show that fixed air samplers typically underestimate exposures on average by a factor of 10, and air samplers placed at ventilation exhaust between a factor of 25 and 330 (NRC 1993a). Therefore, personal air samplers that are worn by the worker are the method of choice for monitoring workers in areas of airborne radioactive material.

Whereas air monitors are used to warn workers of airborne radioactivity and personal air samplers are preferred for workers in areas with airborne radioactive material, air sampling is often used to document that engineered controls are working effectively and that no airborne radioactivity is present. In any case, the placement of the air sampler/monitor is essential for obtaining the needed information. The NRC (1993b) and Whicker (2010) provide detailed information for conducting airflow studies and determining the placement of air samplers/monitors. Table 5.3. [taken from NUREG-1400, Table 2.1 (NRC 1993b)] shows the general locations for airflow testing and placement of samplers/monitors, depending upon the purpose of the sampling/monitoring.

The frequency of filter changes should be commensurate with the likelihood of changes in airborne radioactivity levels and the ambient dust loading. When air sampling for alpha emitters, the higher the ambient dust loading, the shorter the filter exchange frequency, as accumulated dust will attenuate the radiations emitted from the filter.

As-needed air sampling should be performed to evaluate conditions that have not been previously assessed; when there is an expectation that airborne radioactivity will be present; or an expectation that airborne radioactivity levels may have changed. Low volume air sampling should be used to determine radioactivity levels for the protection of workers. High volume air sampling should be used in situations in which airborne radioactive material concentrations need to be determined rapidly or the specific activity of the material is low. For example, surveys for airborne radioactivity should be performed:
Table 5.3—*General locations for airflow testing and placement of air sampling/monitoring equipment (from NUREG-1400).*

<table>
<thead>
<tr>
<th>Purpose of Sampling/Monitoring</th>
<th>General Locations for Airflow Testing and Placement of Samplers/Monitors</th>
</tr>
</thead>
<tbody>
<tr>
<td>Identify area needing confinement control</td>
<td>Located in airflow pathway near actual or potential release point</td>
</tr>
<tr>
<td>Provide early warning of elevated airborne release</td>
<td>Continuous air monitors placed between workers and release points(s)</td>
</tr>
<tr>
<td>Verify the integrity of engineered controls</td>
<td>Located downstream of confinement-control area</td>
</tr>
<tr>
<td>Determine total concentration from many potential release points</td>
<td>Located downstream at exhaust point</td>
</tr>
<tr>
<td>Determine if an airborne radioactivity area exists</td>
<td>At locations of workers</td>
</tr>
</tbody>
</table>
• during tasks that involves opening a system that is known to contain (or has contained) dispersible radioactive material
• while conducting cutting, grinding, or welding operations on radioactive or radiologically contaminated materials
• during initial entry and periodically thereafter into any area that is known or suspected to contain airborne radionuclides or significant loose surface contamination
• immediately following the discovery of widespread radioactive contamination
• whenever respiratory protection equipment is used to control the intake of radioactive materials

The air monitoring program should be described in a technical basis document and address the:

• radiations emitted by the radioactive materials to be sampled
• range of concentrations that are likely to be encountered
• sampler location relative to workers and the source of contamination
• portability, operating and response characteristics of the sampling system
• chemical reactivity of the contaminant and the temperature sensitivity of the sampling system.

For certain applications (e.g., assessing an internal dose) it may be important to determine the chemical and physical state of the contaminant and the aerodynamic characteristics of the particulate material.

5.9.6. Survey directions and documentation

Direction for routine survey should specify the type of survey (e.g., radiation survey, contamination survey), the frequency, and survey locations. The surveyor should be permitted to include additional locations based on actual workplace conditions. The surveyor should be able to recognize and respond to changed radiological conditions and keep their dose ALARA during the conduct of the surveys.
Survey results should be documented according to established procedures and should include:

- the purpose of the survey (if a special/nonroutine survey)
- a description or picture/map showing each measurement location
- the measured dose rates or contamination levels at each measurement location
- the manufacturer, model number, serial number, and calibration date of the instruments used
- the name and signature or the electronic signature of the surveyor
- the name and signature or the electronic signature of the person who reviewed/approved the survey
- the date the survey was performed, and the time, if needed to distinguish surveys conducted in the same area and on the same day
- comments by the surveyor and the person evaluating the survey that are relevant to the interpretation of the survey data.

5.10. **Responding to Elevated Survey and Monitoring Results**

The establishment of Action Levels or Stop Work thresholds in safety procedures and RWPs is discussed in Section 5.2.3. Whereas it is important to provide workers with clear guidance as to when action is required, it is equally important to provide them with the expected actions to take. Actions that workers should take when an action level is exceeded typically include the following:

- Stop (or don’t resume) the work until the source of the elevated monitoring level has been identified and corrected, and appropriate actions have been taken (e.g., decontaminating the affected area).
- Notify the radiation safety staff and obtain help, if necessary.
- Notify the work supervisor.
Actions that should be coordinated with the radiation safety staff when an action level is exceeded may include the following:

- Post and control access to the area.
- Decontaminate the area or item until the contamination is below action level.
- Investigate the cause of the event or conduct a formal critique, if required.
- Identify steps to take to reduce levels below the action level and to prevent recurrence.
- Notify outside authorities, as needed.
- Document the event, including survey and monitoring results, and actions taken to correct and prevent recurrence.

5.11. Management of Contaminated and Activated Items and Equipment

Items and equipment used in areas where they may have become radioactive (activated) or contaminated should be monitored after use and properly controlled if found to be activated or contaminated.

Radioactive contamination of items, equipment, and surfaces can be either fixed or removable. The primary hazard from fixed contamination is external radiation as fixed contamination doesn’t readily transfer to people or other surfaces. The primary hazard from dispersible (unfixed) surface contamination arises from the potential for it to be resuspended and inhaled, or to transferred to the mouth or skin. Because of these hazards, there are regulatory limits for fixed and removable contamination under DOE and license limits under the NRC and Agreement States.

Radiological monitoring should be performed before handling potentially activated items (so the worker is aware of potential dose rates), and promptly after handling potentially contaminated items (so the worker is aware if their hands are contaminated). Surveys should be performed and documented in accordance with ANSI N13.49. (ANSI/HPS 2001c) prior to releasing potentially contaminated or activated items from radiological control. Section 7.1 provides details regarding radiation and contamination surveys.
ANSI N13.12 (ANSI/HPS 2013) provides a standard for clearance of items with potential surface and volumetric contamination from a restricted or controlled area. The standard was developed in a manner consistent with the recommendation of the IAEA and provides derived screening levels (DSLs) for surface and volume radioactivity for groups of radionuclides. The DSL is the residual volume and surface radioactivity level below which items or equipment can be managed without regard for their residual radioactivity. The standard assigns 50 radionuclides to one of four groups, 0.1, 1, 10, or 100 Bq g\(^{-1}\) (volume) and Bq cm\(^{-2}\) (surface) DSLs. The grouping is based on conservative scenarios and considers the detectability of the radionuclide and the ALARA principle. See Table 5.5 for the clearance levels.

DOE (2002) also provides guidance for the control and release of property with residual radioactive material.

Items identified as contaminated must be controlled. Typical control methods include:

- bagging and tagging the item;
- segregating the item for use in a designated, posted area;
- holding the item for decay;
- decontaminating the item; and
- disposing of the item as radioactive waste.

If decontamination to less than the DSL is not possible, an assortment of tools and equipment may be reserved solely for use in areas where contamination or activation is possible. Such items such should be clearly labeled, and procedures should be established for their control and use. See Section 5.6.4 for a discussion of decontamination methods.
Table 5.5—Screening levels for clearance, from N13.12 (ANSI/HPS 2013).

<table>
<thead>
<tr>
<th>Radionuclide groups(^b)</th>
<th>Surface (Bq/cm(^2))</th>
<th>Volume (dpm/100 cm(^2))</th>
<th>Volume (pCi/g)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Group 1 High-energy gamma, radium, thorium, transuranics, and mobile beta-gamma emitters: (^{144})Na, (^{57})Co, (^{60})Co, (^{109})Cd, (^{137})Cs, (^{137})I, (^{222})Rn, (^{222})Th, (^{224})Ra, (^{226})Ra, (^{228})Th, (^{232})Th, (^{238})U, (^{239})Pu, (^{240})Pu, (^{241})Am, (^{242})Am, (^{243})Am, (^{245})Cm, (^{246})Cm, (^{247})Cm, (^{248})Cm, (^{249})Cf, (^{251})Cf, (^{254})Es, and associated decay chains(^d,) and others(^b)</td>
<td>0.1</td>
<td>600</td>
<td>3</td>
</tr>
<tr>
<td>Group 2 Uranium and selected beta-gamma emitters: (^{144})Ce, (^{155})Ba, (^{159})Eu, (^{162})Tb, (^{181})Hf, (^{185})Os, (^{189})Ir, (^{192})Ir, (^{204})Tl, (^{206})Bi, (^{233})U, (^{235})U, (^{238})U, natural uranium(^e,) (^{227})Np, (^{232})Th, (^{236})Pu, (^{237})Pu, (^{240})Cm, (^{240})Cm, (^{248})Cf, (^{250})Cf, and associated decay chains(^d,) and others(^b)</td>
<td>1</td>
<td>6,000</td>
<td>30</td>
</tr>
<tr>
<td>Group 3 General beta-gamma emitters: (^{14})C, (^{40})K, (^{58})Co, (^{75})Se, (^{89})Sr, (^{95})Zr, (^{99})Tc, (^{103})Ru, (^{114})In, (^{129})Sn, (^{131})I, (^{133})Ba, (^{144})Ce, (^{153})Gd, (^{181})W, (^{203})Hg, (^{207})Tl, (^{210})Po, (^{220})Ra, (^{226})Ra, (^{228})Ra, (^{231})Pa, (^{236})Pa, (^{238})U, (^{241})Pu, (^{242})Cm, and others(^b)</td>
<td>10</td>
<td>60,000</td>
<td>300</td>
</tr>
<tr>
<td>Group 4 Low-energy beta-gamma emitters: (^{140})Ce, (^{144})Ce, (^{148})Ce, (^{153})Nd, (^{171})Tm, (^{191})Os, (^{233})Pa, (^{249})Cf, and others(^b)</td>
<td>100</td>
<td>600,000</td>
<td>3,000</td>
</tr>
<tr>
<td>Group 5 Low-energy beta-gamma emitters: (^{56})Fe, (^{136})As, (^{136})As, (^{226})Ra, (^{226})Ra, (^{228})Ra, (^{228})Ra, (^{232})Pm, (^{235})Sm, (^{236})Sm, (^{237})Sm, (^{237})Sm, (^{238})W, and others(^b)</td>
<td>100 (surface)(^c)</td>
<td>600,000 (volume)</td>
<td>30,000</td>
</tr>
</tbody>
</table>

\(^a\)The screening levels for clearance have been rounded to one significant figure and are assigned to both surface and volume radioactivity (assuming an average surface to mass ratio of 1:1, as discussed in Annex A), unless otherwise noted. Note: regulatory authorities may increase all volume and surface screening levels by one order of magnitude when clearing bulk quantities of less than 1 metric ton or 1 m\(^3\).

\(^b\)To determine the specific group for radionuclides not shown, a comparison of the screening factors, by exposure scenario, listed in Tables B. 1, C.1, and D.1 of NCRP Report No. 123I (NCRP 1996) for the radionuclides in question and the radionuclides in the general groups above should be performed and a determination of the proper group made, as described in Annex A.

\(^c\)Because of potential ground-water concerns, the volume or surface radioactivity values for \(^{129}\)I should be lowered by one order of magnitude when disposal to landfills or direct disposal to soil is anticipated.

\(^d\)For decay chains, the screening levels represent the total activity (i.e., the activity of the parent plus the activity of all progeny) present.

\(^e\)The natural uranium screening levels for clearance shall be lowered from Group 2 to Group 1 if decay-chain progeny are present (i.e., uranium ore versus process or separated uranium, for example, in the form of yellowcake). The natural uranium activity equals the activity from uranium isotopes (48.9% from \(^{235}\)U, plus 48.9% from \(^{238}\)U, plus 2.2% from \(^{239}\)U). This approach is consistent with summing radionuclide fractions discussed in Section 4.4.

\(^f\)For radioactivity control considerations, surface radioactivity screening levels for Group 5 radionuclides are controlled to the Group 4 surface radioactivity screening levels.
5.12. Security Program for Radioactive Material and RGDs

The primary objective of the security program is to deter the loss, theft, or unauthorized use of radioactive material and RGDs. The extent and scope of the security program should be commensurate with the potential risks presented by the radioactive material/RGDs. For example, the security program should be detailed and comprehensive at a large research laboratory that uses many types and quantities of radioactive material and a variety of RGDs; it can be much more focused and limited at a facility with a single density gauge, RGD, or small quantities of radioactive materials.

Provisions should be in place to protect radioactive material/RGDs from unauthorized access, loss, damage or theft. Access should be limited to those individuals who are authorized to use, inventory, or transport the item, with the level of security commensurate with the potential radiation hazard, should it fall into unauthorized hands.

Guidance on security measures for sealed radioactive sources is contained in NCRP Report No. 182 (NCRP 2019). This guidance is generally applicable to radioactive material (not just sealed sources), as well as RGDs. In addition to NCRP recommendations, the International Atomic Energy Agency (IAEA) has developed sealed radioactive source classifications that address the level of security recommended for different types and quantities of radioactive material. These recommendations are detailed in NCRP Report No. 182.

The IAEA classification scheme for radioactive sources is incorporated into the Code of Conduct on the Safety and Security of Radioactive Sources (IAEA 2004a). The Code specifies the recommended levels of security for each of the source categories. Since the United States (U.S.) is a signatory to the Code, the U.S. provides similar recommendations and implements requirements for source security for Category 1 (e.g., irradiators, teletherapy sources, multi-beam teletherapy, radioisotopic thermoelectric generators) and Category 2 (e.g., industrial radiography sources) sources and aggregates of smaller activity sources reaching the threshold values.

In the United States, the NRC and the Agreement States implement the IAEA recommendations via 10 CFR Part 37 and the corresponding state regulations (NRC 2013). DOE implements the sealed source tracking requirements through DOE Order 231.1B, Environment, Safety, and Health Reporting (DOE 2011b).
6. Radioactive Waste Management

Radioactive wastes can be generated in a facility during normal operations, radiological incidents, and decommissioning activities. Examples of these wastes include activated components from accelerator operations, contaminated equipment and tools, contaminated glassware and gloves, and animal carcasses used in research activities. Mixed wastes are those that contain both radionuclides and other hazardous substances such as organic solvents, asbestos, heavy metals, or reactive chemicals. The treatment and regulation of mixed waste is complex; EPA regulates the hazardous component of the waste under the Resource Conservation and Recovery Act (RCRA) (EPA 1976) and the radioactive component is regulated by the NRC, DOE and the states.

In general, the management, transportation, and disposition of hazardous wastes, including radioactive wastes, can be matters of public concern and are strictly regulated by federal and state agencies. When designing a program for control of radioactive wastes, it is important for the facility management to become thoroughly familiar with currently applicable regulatory requirements and community concerns.

The guidance in this section is focused primarily on the management and disposal of low-level waste that might be generated in educational, research, medical, and industrial facilities. In some types of facilities, there will be additional waste streams with more complex treatment/disposal considerations such as nuclear power plants and DOE facilities.

6.1. Minimizing the Production of Radioactive Waste

Facilities that use, manufacture, or generate radioactive materials are required to establish a program for minimizing the generation of radioactive waste, as specified in 10 CFR 20.1406 (NRC 2020a) and DOE Order 435.1 (DOE 1999). The most effective way of addressing the radioactive waste management issue is to implement institutional practices that eliminate or significantly reduce the generation of these wastes. A hierarchy of steps recommended to minimize the production of radioactive wastes is shown in Table 6.1, and is discussed in detail in NCRP Report No. 143 (NCRP 2003a). These steps include Level 1 where the main processes or
Table 6.1—*Hierarchical waste minimization steps.*

<table>
<thead>
<tr>
<th>Level 1</th>
<th>Level 2</th>
<th>Level 3</th>
<th>Level 4</th>
</tr>
</thead>
<tbody>
<tr>
<td>Source Reduction</td>
<td>Recycling</td>
<td>Treatment</td>
<td>Disposal</td>
</tr>
<tr>
<td>• Material substitution</td>
<td>• Reclamation reuse</td>
<td>• Compaction</td>
<td>• Land disposal in a licensed and/or permitted facility or release as permitted under applicable regulations</td>
</tr>
<tr>
<td>• Waste reduction</td>
<td></td>
<td>• Incineration</td>
<td></td>
</tr>
<tr>
<td>• Process modification</td>
<td></td>
<td>• Chemical treatment</td>
<td></td>
</tr>
<tr>
<td>• Management of material going into process</td>
<td></td>
<td>• Encapsulation</td>
<td></td>
</tr>
<tr>
<td>• Segregation</td>
<td></td>
<td>• Solidification</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>• Storage for decay</td>
<td></td>
</tr>
</tbody>
</table>
activities are evaluated for the opportunity for partial or total elimination of the waste stream. In Level 2, the processes and materials are managed, recycled, or reclaimed reduce the hazard and or the volume of the anticipated waste. In Level 3, the wastes intended for disposal are treated to reduce the hazards and/or volume, enhance the long-term stability of the waste form, and comply with the waste acceptance criteria of the disposal facility. In Level 4, a disposal method is selected that is protective of public health and the environment and in accordance with federal and state laws and regulations.

In addition to providing guidance for the options in the hierarchical waste minimization levels, NCRP Report 143 provides guidance on applicable laws and regulation, implementing an effective waste minimization program, specific waste minimization methods (with examples), and designing facilities for waste minimization.

The waste minimization program should include procedures that separate materials that may become radioactive from other potentially hazardous materials to avoid the generation of mixed wastes.

An assortment of tools and equipment should be reserved solely for use in areas where activation or contamination is possible. These reserved tools and equipment should be identified, conspicuously marked, issued and used under strict controls. When it is cost-effective to do so, contaminated equipment should be decontaminated and reused. Routinely disposed materials such as protective clothing should be compatible with available waste-processing systems, volume reduction, and waste disposal requirements.

Potentially activated or contaminated material should be segregated from nonradioactive or uncontaminated material by providing separate containers for nonradioactive and radioactive trash. Separate containers can also be used to segregate reusable radioactive items such as tools, instruments, protective clothing and respirators from disposable radioactive wastes. Background information on waste minimization is presented in documents of the NRC (1994a) and EPA (1993).

Radioactively contaminated and activated metal scrap can be generated during normal operations at several types of industrial facilities (e.g., research and development facilities, nuclear reactors, accelerators, and fuel cycle facilities). Affected materials include machine components and internals composed of Inconel Alloys, zircaloy, or stainless steels. The
decommissioning of existing nuclear power plants, research test facilities, and the defense
nuclear weapons production facilities of the (DOE generates large quantities of scrap metal that
requires effective waste management.

NCRP Report No. 141 (NCRP 2002) provides specific guidance concerning the
management of potentially radioactive scrap metals, some of which may be highly contaminated.
Other useful references are provided by ANSI/HPS (2013), the NRC (2003) and DOE (2001, 2016).

6.2. Collecting, Packaging and Volume Reduction

Conspicuously marked or colored bags, or metal drums or cans lined with a plastic bag
should be used to collect radioactive solid wastes. Depending on the materials present and the
quantities expected, separate and easily distinguishable containers should be provided for
different classes of waste, (e.g., nonradioactive, potentially radioactive or contaminated, known
radioactive or contaminated, reusables and recyclables). All but the nonradioactive trash
containers should be clearly marked with “radioactive materials’ labels. Sharp objects should be
specially packaged before collection to prevent punctures of the container.

Automated devices that detect surface or volume radioactivity may be useful for
confirming the status of material that has been segregated as nonradioactive. Tests of these
devices should be conducted prior to their being used to ensure that radioactivity would be
detected at the desired level of detectability. Frequent checks should be made of the operation
and detection sensitivity of the system.

Low-level wastes that are wet, flammable or corrosive should be dried and neutralized
before they are placed in containers for disposal. Spray cans that cannot be decontaminated and
released should be vented or punctured to ensure that they will not explode when compacted.

Wastes that are nonradioactive but that contain materials generally associated with
radiation control, such as yellow plastics, protective clothing, and items that are marked with the
radiation symbol or other radiation control indicators, should be shredded or defaced before they
are disposed as nonradioactive waste.
Solid waste volume reduction techniques should be adopted to minimize the space needed for storage and disposal of radioactive waste. These techniques include compaction and incineration. Compaction can substantially reduce the volume of solid waste that is destined for shipment for disposal off-site. Where large numbers of scintillation vials are used, wastes can be processed by crushing the vials. The radioactive scintillation fluids can then be separated from the fragments of the vials which can be cleaned and disposed as nonradioactive waste.

Incineration is an effective volume reduction method for materials such as animal carcasses. It is also useful for disposing of oils and fluids that are used in liquid scintillation counting. Incineration can also be used for volume reduction of combustible trash. If incineration results in radioactive ash, the ash must be disposed of as radioactive solid waste, or if it also contains hazardous materials, as mixed waste. Similarly, if incinerator off-gases are filtered, contaminated filters must be disposed as solid radioactive waste. ANSI Standard N13.45 (ANSI/HPS 2012) provides detailed guidelines for the incineration of combustible low-level radioactive and mixed wastes including the licensing and permitting requirements specified by the EPA, and other Federal, State and Local agencies.

6.3. Storage of Waste

Radioactive animal carcasses should be stored frozen while awaiting disposal. If space permits, they should be stored long enough to allow any short half-lived radionuclides to decay away.

Radioactive or mixed wastes that are not stored for radioactive decay should be prepared and packaged for disposal as soon as collection and sorting are completed. They may then be stored in those packages until they are shipped for disposal. Mixed waste are generally packaged for off-site treatment and disposal by specialized contractors.

If only short half-lived radionuclides are present in the waste material, it may be feasible to store the waste in containers for a long enough time to allow the activity to decay to a level at which the material may be disposed as nonradioactive waste. Radioactive materials that are stored for decay or packaged for disposal should be protected sufficiently to prevent deterioration of the containers and to minimize the likelihood of the spread of contamination. Storage locations should be clearly identified as RMAs and periodically inspected. When
practical, waste should be stored indoors. The selected storage area should be designed to provide security and reduce radiation exposure. Note that for mixed waste subject to RCRA (EPA 1976), the 90 d waste storage limitations apply.

6.4. Shipment and Disposal

Many factors must be considered in the selection of appropriate waste disposal methods, including technical feasibility, cost, regulatory requirements, liability, and public and environmental impacts. Disposal methods include on-site treatment (e.g., incineration) and off-site disposal. Incineration primarily burns the nonradioactive constituents of the waste and often is used as a volume reduction technique because the resulting ash requires disposal as radioactive waste.

Radioactive wastes that will be shipped for disposal off-site must be: packaged, labeled, and shipped in a manner that meets DOT requirements (DOT 2011); shipped in a manner that ensures the integrity and the identity of the package; and within the acceptance criteria of the receiving organization. One option which can be convenient and cost effective, particularly for smaller generators, is to use a waste broker which is a company that is licensed to treat, package, and transfer radioactive waste to licensed radioactive waste disposal or treatment facilities.

Prior to shipment of radioactive waste, it is necessary to document the quantities of radionuclides present in the waste and to assure that external exposure rates from shipping containers meet DOT requirements (DOT 2011). Records of the weight and volume of waste containers, as well as quantification of the radionuclide content to be shipped are an integral part of the documentation that is needed, but many additional requirements may be specified by federal or state regulatory agencies. There are special requirements for estimating the amounts of long-lived radionuclides and fissile materials present in the wastes.

If the radionuclides in waste material are energetic gamma ray emitters (i.e., energetic enough to be detected outside of the container), it is possible to use external measurement techniques to assess the activities of those radionuclides in the waste container. In addition, techniques for assaying the fissile material content of waste containers have been developed. However, if the waste is contaminated primarily with low levels of pure alpha or beta particle emitters, or low-energy photon emitters (e.g., $^{57}$Co), sampling of the waste will likely be
necessary. The amount of effort devoted to characterization of such wastes should be
commensurate with the level of radioactivity present and applicable regulations. The most
important factor in estimating the content of radionuclides in the waste is the collection of a
representative sample, although in some cases (e.g., where small quantities of limited isotopes
are used in a small number of labs), it may be possible to use “process knowledge” to estimate
the radionuclide content.

The sampling technique of choice will depend on the form of the waste material. Coring
has been used to obtain samples of wastes in semisolid form prior to solidification. For
heterogeneous wastes, the component waste streams may be analyzed prior to consolidation or
composites of grab samples may be used. It may also be possible to collect representative core
samples of heterogeneous wastes following compaction. Initially, an assessment of the variability
from sample to sample should be conducted to provide guidance for routine sampling of such
waste materials.
7. Radiological Instrumentation

This Section describes the specification, calibration, maintenance and use of the instruments that are used to implement a radiation safety program. Useful references include Lee and Burgess [1999 (R 2014)] and NCRP Commentary No. xxx, Recommendation for Instrument Response Verification and Calibration for Use in Radiation Emergencies (NCRP 2021kls). Record keeping requirements for instrumentation are discussed in Section 12.5.

Depending on the type of operations conducted, radiation safety programs may need:

- hand-held instruments for surveys of radiation fields and contamination on surfaces;
- stationary instruments (e.g., radiation area monitors) for monitoring ambient radiation levels and airborne radioactivity (e.g., CAMs);
- personnel monitors (e.g., electronic personal dosimeters (EPDs) (discussed in Section 8.1.2);
- hand and shoe monitors;
- portal monitors;
- laboratory instruments for assay of samples (e.g., oils, soil, vegetation);
- environmental monitors for measuring effluent streams (e.g., in air or liquid discharge systems).

7.1. Field Instrumentation

When measuring external radiation fields, instruments must be capable of measuring both the primary and scattered radiation fields, including skyshine, that may be present in and around the facility. When measuring radionuclides in the air or in other effluents, instruments must be capable of detecting the major radionuclides that are used within the facility at levels that are at, or below, administrative dose guidelines. The instruments must be appropriately calibrated and maintained, and proper records must be kept of the instrumentation types, use, calibrations and maintenance.
The RSO should establish the initial instrumentation inventory and budget, including extra instruments to cover operations when instruments are sent for calibration, and emergency response considerations. The RSO should estimate the expected lifetime of instruments based on repairs and usage, and maintain an annual replacement budget.

7.1.1. Selection of Instruments for Various Applications

There are two categories of instruments:

- those that measure ambient radiation fields, and
- those that measure the radioactivity in a sample or on a surface.

NCRP Report No. 57 (NCRP 1978a) discusses instruments that are appropriate for use in radiation field measurement and NCRP Report No. 50 (NCRP 1976b) discusses instruments that are useful for environmental radiation measurements and sample analysis. Extensive information about various detector and instrumentation systems can be found in NRC (2020c) Attix (1972, 2008); Attix and Roesch (1966); Attix and Tochilin (1969); Kase et al. (1985, 1987, 1990); Knoll (2010); and L’Annunziata (2012).

7.1.1.1. Photon and Beta Radiation Surveys

Ionization chamber instruments are commonly used to measure photon radiation fields as their response is nearly energy-independent over a wide range of photon energies. These instruments can also be designed to cover a very wide range of dose rates, and they operate accurately in pulsed radiation fields. Care must be taken, however, to be certain that at the higher energies (e.g., >1 MeV) charged particle equilibrium is achieved (by providing sufficient wall thickness), and that excessive attenuation of photons does not occur at low energies (e.g., < 0.1 MeV) (by providing a thin entrance window) (Schauer et al. 2004). These two conditions may not be satisfied with the same instrument. For measurements at normal environmental radiation levels (i.e., as typically found outside the facility), the ionization chamber must be large or pressurized to achieve adequate sensitivity.

Organic (e.g., anthracene, stilbene, plastic) and inorganic (e.g., NaI, LaCl, LaBr, CeBr, SrI) scintillation detectors are very sensitive and can therefore be used to measure very low (e.g., μR h⁻¹) levels of photon radiation. The response of organic scintillators is relatively independent
of photon energy. When used in a spectroscopy mode, algorithms can convert the energy spectrum into integrated dose or dose rate, and at the same time determine the nuclide(s) causing the dose. Because of their large mass, scintillation detectors are useful in survey meters to detect the presence or to confirm the absence of small sources, and sometimes for very low dose rate measurements. Special thin window and large area scintillation detectors (e.g., Field Instrument for the Detection of Low Energy Radiation (FIDLER) probes) are also very useful to detect very low-energy radiation (e.g., $^{241}$Am) because of their high sensitivity to low-energy photons.

Pulse counting instruments, such as GM counters, proportional counters, and scintillators used in counting mode are very sensitive and useful for surveys to indicate the presence of alpha/beta particle emitting surface contamination and increased photon radiation fields. In general, these instruments should not be used for dose measurements unless it is specifically calibrated for the exact radiation field that is to be measured as these instruments are quite dependent upon the energy of the radiation. But with the use of special processing algorithms, or special energy compensation external filtration, the energy response can be suitably flattened such that GM survey meters and area monitors are suitable for dose rate measurements above approximately 20 keV, depending on the specific detector and energy compensation filtration. Care must be taken when using pulse-counting instruments to measure pulsed radiation fields commonly found around accelerators; due to the finite resolving time of each radiation interaction, pulse-counting instruments can grossly under-respond to the recorded count rate and dose rate of pulsed radiation fields. It is possible for conventional GM tubes to saturate and for the instrument to indicate zero when in an intense radiation field, however the electronics in the instrument circuitry may be designed to detect a saturated GM tube and provide a full-scale response to indicate the presence of an intense radiation field.

Measurements of beta radiation fields can usually be made using the same instruments described for measuring photon radiation fields.

7.1.1.2. *Contamination Surveys for Alpha and Beta Radiation*

Surveys for radioactive contamination require instruments that are very sensitive to beta and/or alpha particles. Usually these will be instruments that have pulse counter detectors such as proportional counters, GM counters, or scintillators, and indicate a count rate rather than a dose rate.
Instruments used to detect alpha particle radiation from surface contamination require a very thin window so that the alpha particle can penetrate to the sensitive volume. The readout electronics should incorporate a pulse-height discrimination circuit so that the instrument can be made insensitive to beta and photon radiation. Silver-activated ZnS (ZnS(Ag)) scintillators are very insensitive to photons and are good alpha particle detectors. Some instruments have both a ZnS(Ag) alpha particle scintillator followed by a thin plastic beta particle scintillator. Pulse-height discrimination separates the alpha particle pulses from the lower energy beta particle pulses, allowing simultaneous alpha and beta particle counting.

Instruments used to detect beta particle radiation from surface contamination are usually GM tubes or thin plastic scintillators. The GM counters require a thin window (as with GM “pancake” probes), but the window can be made thick enough to exclude alpha particles. However, some very low-energy beta-particle emitters (e.g., $^3$H) may not be detected at all, or at low enough levels (e.g., $^{14}$C), with these instruments, so other techniques (e.g., liquid scintillation counters, windowless proportional counters) may be required to detect contamination from such low-energy beta particle emitters.

While finding beta particle-emitting surface contamination is rather easy, accurately measuring the activity is more complicated, and determining the dose or dose rate from such sources can be quite challenging. It is necessary that the nuclide be known and that the instrument be calibrated using a known source in the same physical geometry and with a nearly identical beta-radiation spectrum (NCRP 1991b). Dose rate measurements are best done by specialists using an extrapolation ionization chamber.

7.1.1.3. Neutron Radiation Surveys

Area surveys for neutrons in neutron or in mixed neutron - photon radiation fields require instruments specifically designed to detect neutrons while being relatively insensitive to photons. These instruments usually have either a proportional counter or a scintillator as the detector using (n, alpha) or (n, p) reactions. They respond to neutrons that have been thermalized by passing through a specially designed moderator that surrounds the detector. The moderator design will determine the energy dependence of the detector response. Thus, it is important to calibrate these instruments using a neutron spectrum that is representative of the spectrum that is to be surveyed. Since these instruments are pulse counters, and since most neutron detecting instruments also
have some minimal response to gamma rays, they can be saturated by intense photon fields, or
by intense photons in a pulsed radiation field. Care must be taken to be certain that the reading
obtained truly represents the neutron dose rate in the radiation field and not simply the photon
dose rate or the pulse rate of the source.

Usually, the described type of neutron survey instruments indicates the equivalent dose
rate, as they have been designed to respond across the neutron energy spectrum in accordance
with an accepted energy-dependent fluence-to-equivalent dose conversion (ICRP 2015, 2016a,
2016b, 2017, 2019). Since these fluence-to-dose relationships are defined by regulatory or
advisory bodies and have changed over the years as the relationship between dose and equivalent
dose is refined, it is, in principle, better to measure the neutron fluence and energy spectrum. The
fluence measurement is relatively simple to make with the described instruments, but the
spectrum measurements are very difficult. Regardless of the difficulty, if neutron exposures are
likely to contribute significantly to worker dose, measurements of the neutron spectra should be
made. Techniques for making such measurements have been described by Cross and Ing (1987).

7.1.1.4. Surveys Using Stationary Radiation Detectors

Instrumentation for stationary monitoring of work areas, personnel contamination
monitors, airborne radionuclides, effluent streams, and the general environmental dose rate use
the same detectors used in survey instrumentation. Stationary area monitors that are intended to
accurately measure high dose rate or pulsed fields should be designed and type tested for such
operation. Ionization detectors, scintillators or other solid state detectors operated in the current
mode, or GM instrument using time-to-count, are all capable of measuring high dose rate photon
radiation fields. Current-mode instruments can minimize the problems of saturation in pulsed or
high-dose rate fields. The readouts should be continuously recorded so that both short- and long-
term changes in the radiation field can be documented.

7.1.1.5. Surveys Using Passive Radiation Detectors

Passive detectors can also be used for monitoring the general radiation environment in
the work area as well as at the boundary of, or in the vicinity of the facility (see Section 5.9.4).
Typically, the detectors that are appropriate for this type of monitoring are small, resistant to
changing environmental conditions, and relatively independent of the energy of the radiation.
TLDs, such as LiF or CaSO₄, are very useful for monitoring ambient photon radiation. Combinations of \(^{6}\)LiF and \(^{7}\)LiF or etchable track detectors can be used for neutron radiation measurements (Griffith and Tommasino 1990). Other detectors may also be useful for this purpose, e.g., alanine dosimeters that are interpreted by measuring their retained electron spin resonance signal (Hansen et al. 1987; Regulla and Deffner 1982), and superheated liquid drop (bubble) dosimeters that are sensitive to neutron radiation (Apfel 1979, 1992; Ing and Birnboim 1984; Ipe et al. 1990; Schulze et al. 1992). OSLDs are a type of dosimeter similar to a TLD, except that a laser (instead of a lamp) is used to liberate electrons from their traps. Unlike a TLD, OSLD can be read multiple times, without significant loss of signal (e.g., there is a loss of less than 1 % upon the second reading). OSLD do not need to be annealed before use and have insignificant fading except in extreme temperatures. OSLD can read doses of 10 µSv or so, but uncertainties in background bring into question the validity of reporting doses on the order of tens of µSv.

**7.1.1.6. Surveys for Airborne Radioactivity**

Airborne radioactive material is usually monitored by passing air through a filter and measuring the activity on the filter. The design of the collection and measurement system depends on the radiation that is to be detected. If alpha particle-emitting radionuclides are to be detected, the particles should be collected in a thin layer on the surface of a membrane filter or impactor plate. A semiconductor surface barrier detector is an efficient detector that can discriminate against alpha particle-emitting background, such as radon decay products, using a pulse height analyzer. Material that is collected on a fiber filter paper can be monitored for airborne beta-particle emitters by using a GM or proportional counter. Certain radionuclides, such as \(^{125}\)I and \(^{3}\)H, require specialized monitoring techniques that are less amenable to continuous monitoring. Iodine may be trapped on activated carbon or other appropriate absorbent filters and generally, the photon emission is detected. Tritium can be monitored by passing air through an ionization chamber, but the sensitivity is low and the reading must be corrected for background interference. Stationary monitoring methods can be used to obtain an indication of a release of airborne radionuclides into an environment, but they are usually not quantitative. Accurate quantification of airborne radionuclides requires that samples be taken that can be
related to a precise volume of air, and radioactivity of the samples should be measured using a
calibrated counting system.

7.1.1.7. *Surveys for Radioactivity in Liquid Effluents*

Continuous monitoring of liquid effluent streams can be done by immersing an
appropriately designed GM or scintillation detector into the waste stream. Alpha particle emitters
and low-energy beta-particle emitters will not be detected, and the overall sensitivity is low.
Buildup of contamination on the detector will cause significant uncertainty in the result. Offline
liquid monitors as well as offline liquid process monitors are available as well. The offline liquid
monitors draw a sample and perform an analysis in a shielded configuration while offline process
monitors are configured with their own pumping systems to provide constant flow through a
sample volume with an associated detector. Representative sampling of the waste stream and
measuring the sample in the laboratory is a more common method for detecting radionuclides in
liquid effluent. Laboratory spectroscopic assays can be more accurate and more sensitive;
however, they take labor and time to collect and analyze the sample, and the results are not
usually available until weeks later.

7.1.2. *Instrument Specifications*

A number of factors determine which instrument is most appropriate for radiation
measurements; carefully written specifications can ensure the proper instrument will be
purchased. The following performance specifications and characteristics should be considered
(ANSI 2003, 1989):

- **Response to radiation:**
  - types and energies of radiation to be measured
  - whether the instrument measures rate, or integrates, or both
  - the precision, accuracy, and sensitivity of measurements
  - energy and angular dependence
  - limits on dead time and energy resolution

- **The use environment:**
• temperature, pressure, and humidity

• wet environments

• presence of electric and magnetic fields, as well as RF and microwave radiation

• routine use vs. emergency use only

• Readout parameters:

• range and type of readout (analog/digital)

• computer interface requirements

• audible alarms and status indicators

• Other physical parameters:

• power supply stability, lifetime, type

• weight, portability, durability

• ease of decontamination

Acceptance of any radiation measuring instrument should include a program for evaluating the performance of the instrument against its specifications. If accuracy and precision over the dose or dose-rate range that will be used are the only important specifications, one or more calibrated sources will be needed. If other specifications are also important, more complex testing will be necessary. Depending on the specifications, a test facility or protocols may need to include a chamber to test the effects of pressure, temperature and humidity; a method for testing the detector over a large dynamic range and a large energy range; testing instruments for the useful measurement time powered only by batteries. Some of this acceptance testing can be accomplished by reviewing the manufacturer’s type (compliance) testing data or by requiring other special tests by the vendor in the purchase contract.

7.1.3. Calibration

Calibration of instruments requires consideration of a number of factors depending on the use of the instrument. These factors include (NCRP 1991b; Wagner 1983):

• level of calibration
The calibration of field survey instruments is thoroughly discussed in NCRP Report No. 112 (NCRP 1991b); a brief summary follows. The calibration requirements of other types of radiation safety program instrumentation are mentioned briefly, but are not covered in detail.

Calibration refers to the determination and adjustment of instrument response in a particular radiation field of known intensity. Three levels of calibration are full characterization, specific acceptance characterization, and routine instrument calibration. Full and specific acceptance characterizations are generally for a class of instruments, whereas routine calibrations are for each individual instrument.

Full characterization is a process that would normally be done by the manufacturer of an instrument to determine the response of the instrument as completely as is possible. This is commonly referred to as type (or compliance) testing. It typically includes evaluations of the useful dynamic range of the instrument, the energy dependence of the response, the linearity of the response, the effects of radiation types other than those for which the instrument is designed, environmental influences, nonionizing radiations, instrument orientation, sensitivity to mechanical shock, dose-rate dependence, and angular dependence.

Specific acceptance characterization refers to the need to determine the instrument response, or to calibrate the instrument, under environmental or other conditions that are different from those included in the type testing process, and which may be encountered in facility operations. Specific acceptance characterization may be necessary if the instrument is to be used under extremes of temperature, pressure, humidity, electric or magnetic fields, and energies outside the designed range.

Routine calibration is sufficient when the instrument will be used for measuring radiation under the conditions for which it was designed and characterized; thus, routine calibration is
appropriate for most individuals and institutions. Routine calibration commonly involves the
determination and adjustment of instrument response in known radiation fields from sources that
cover the energy and dose-rate range in which the instrument will be used.

The response of an instrument refers to the indication it gives when exposed to a
radiation source or field. Calibrations should be made in radiation fields that are specified in the
same units as the instrument readout. Typically, the instrument response should be determined
for various conditions of intensity, radiation energy, and orientation in the field and environment.

Uncertainties in the calibration process will be both random and systematic and will
involve uncertainties in the calibration of the radiation sources, or fields, as well as uncertainties
in the instrument positioning in the field and the instrument response. The random uncertainties
can be estimated and treated by standard statistical techniques. The systematic uncertainties
should be kept as small as possible by careful procedures and elimination of biases. NCRP
Report No. 112 (NCRP 1991b) recommends that survey meters and surface contamination
meters be calibrated to indicate the true dose-related quantity or radioactive contamination with
overall uncertainties in the range from 10 to 35 percent, depending on the intended use of the
instrument. Counting instruments that are used for analyzing environmental samples or
instruments that are used for area, environmental or personal monitoring may have calibration
uncertainties that fall in the same range (Budnitz et al. 1983; NCRP 1985; Wagner 1983).

The frequency of calibration will depend on the use, purpose and required accuracy of the
instrument. All instruments should be under a Quality Assurance Program and have a periodic
Quality Control testing process sufficient to ascertain that the instrument is currently suitable for
use. Historical records of these Quality Control instrument performance tests may also be useful
in deciding on calibration frequency. Historically, the common default has been to calibrate an
instrument at least once per year; however, modern instruments could have much longer
calibration intervals if justified by measurement data. Alternatively, for extra precaution, the
calibration interval could be arbitrarily made shorter for those instruments with harsh use or used
in areas with higher potential exposure. Each time that an instrument is repaired or fails a
performance check, the cause should be investigated and the problem fixed. A simple fix (e.g.
change of batteries, adjustment of amplifier gain) may only require a satisfactory performance
check before returning the instrument to service. More extensive repairs or component
replacements should require that a calibration be made before the instrument is returned to service.

Calibration sources should be carefully chosen to provide the appropriate ranges of dose rate and energy. Proper equipment should be available to assure calibration of the radiation fields and to provide adequate safety for the staff. The staff should be highly qualified and well trained in the operating characteristics of the instrumentation as well as in the techniques and procedures of instrument calibration.

7.1.4. Instrument Maintenance

Instrument maintenance requirements include establishing criteria for:

- frequency of maintenance operations
- performance checks
- power supply checks

The frequency at which maintenance is performed depends on the use of the instrument and the accuracy required. Usually it is sufficient to repair the instrument when it is damaged or when a component fails. However, in some cases, especially if the instrument is the only one available, a preventive maintenance inspection schedule should be established to limit equipment failures.

A procedure should be established for performance checking the instrumentation a regular basis. Ideally this check should be made prior to each use of the instrument, or daily if the instrument is used frequently. The power supply for almost all survey instruments is a battery, and generally there is a built-in battery test function which should be used to check the battery prior to using the instrument. The performance check can be made using a small sealed source or a counting standard. If the instrument fails the performance check, it should immediately be sent for repair and, if other than a simple fix, then a recalibration. A background test should be a part of the performance test as the background reading may give an indication of a possible problem with an instrument. Users should be familiar with the approximate background for their instrument and should note the background reading each time the instrument is used. Fixed area monitors and counting instruments may have high voltage power
supplies that operate off the main power line. If a stable voltage supply is needed, the power supply line should include a surge protector and a voltage regulator.

7.1.5. Use of Instruments and Acceptable Uncertainty

The choice of instrument and the acceptable uncertainty in its calibration depends on the purpose for which the instrument will be used.

For personnel monitoring using dosimeters or integrating radiation survey measurements, NCRP Report No. 57 (NCRP 1978a) recommends the measurement precision to be within ±10 percent, and accuracy criteria based on expected exposures:

- at levels near the annual limit, a measurement accuracy of ±30 percent should be achieved
- at levels below one-fourth of the annual limit, a lower level of accuracy (e.g., a factor of two) is acceptable
- for levels of exposure higher than the annual limit, an accuracy of at least ±20 percent is acceptable (NCRP 1978a).

For instrumentation used for conducting surveys, NCRP Report No. 112 (NCRP 1991b) specifies the accuracy acceptance criteria in photon, beta radiation, and neutron fields and for surface contamination measurements, and provides specific guidance on methods to quantify uncertainties associated with survey measurements. These recommendations are summarized in NCRP Report No. 112, Table 10.1, shown below as Table 7.1. ANSI also provides performance criteria for radiation safety survey instrumentation (ANSI 2003, 1989). In addition, the NCRP provides Recommendations for Instrument Response Verification and Calibration for Use in Radiation Emergencies (NCRP 2021kls).

For instrumentation used as area monitors, the magnitude of acceptable levels of calibration accuracy and precision specified in NCRP Report No. 57 (NCRP 1978a) remain appropriate.
<table>
<thead>
<tr>
<th>Instrument Application</th>
<th>Short-Term Stability(^a)</th>
<th>Accuracy Acceptance Criteria</th>
</tr>
</thead>
<tbody>
<tr>
<td>Dose-related measurements in photon radiation fields</td>
<td>+5 % at ( H &gt; 1 ) mSv h(^{-1})</td>
<td>+10 %(^b)</td>
</tr>
<tr>
<td></td>
<td>+10 % at 0.05 &lt; ( H &gt; 1 ) mSv h(^{-1})</td>
<td>+20 %(^b) when corrections are provided with the instrument</td>
</tr>
<tr>
<td></td>
<td>+20 % at ( H &lt; 0.05 ) mSv h(^{-1})</td>
<td>+30 %(^b) for a rate &lt;10 ( \mu)Gy h(^{-1})</td>
</tr>
<tr>
<td>Dose-related measurements in neutron radiation fields</td>
<td>Same as above</td>
<td>+20 %(^b)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>+30 %(^b) for a rate &lt;0.02 mSv h(^{-1})</td>
</tr>
<tr>
<td>Dose-related measurements in beta radiation fields</td>
<td>Same as above</td>
<td>Product of calibration factor and instrument reading must yield true dose rate with overall uncertainty within ±20 %(^a).</td>
</tr>
<tr>
<td></td>
<td></td>
<td>For beta radiation(^c) &lt;0.03 MeV, uncertainty may be ±30 %</td>
</tr>
<tr>
<td>Assessment of surface contamination</td>
<td>±20 %</td>
<td>±30 %(^a)</td>
</tr>
</tbody>
</table>

\(^a\) “Short-term” refers to measurements made in a short window of time under fixed and specified conditions. Stability values represent the percent relative experimental standard deviation obtained from \( n \) readings; where \( \sigma_n \) is the standard deviation and \( \bar{I} \) the mean response, the given values are 100 \( \sigma_n / \bar{I} \).

\(^b\) At 95 percent confidence level.

\(^c\) Beta-particle energies determined at the position of the instrument rather than at the source.
7.2. Counting and Analytical Laboratories

An operational radiation safety program should have the ability to analyze discrete samples. This generally involves extracting a representative sample of something and taking it to a laboratory to be analyzed by instruments such as proportional counters, scintillators, and semiconductor detectors that are connected to counting electronics or pulse height analyzers.

Since in many cases traditional laboratory instruments are now rather light and low powered, equivalent measurements are sometimes performed in the field. References relevant to this subject include NCRP Report No. 57 (NCRP 1978a) and NCRP Report No. 58 (NCRP 1985).

The scope of laboratory equipment is highly dependent upon the amount, quantity, and nature of the radioactive materials in the facility. However, most facilities will have the following types of instruments either in the radiation safety laboratory, or shared access to such equipment in other parts of the facility. The most common types of laboratory equipment are:

- Liquid scintillation counters (LSC); most commonly used to assay low energy beta particle emitters like $^3$H, $^{14}$C, $^{35}$S.
- Alpha/beta particle counters; typically, proportional counters, which are most commonly used for total beta and total alpha (a.k.a. gross beta and gross alpha) particle counting.
- Gamma ray and alpha particle spectrometers; to determine or confirm the identity of the gamma ray or alpha particle emitting radionuclides in the item, and to quantify the activity.

The number and quality of each type of instrument will vary widely between facilities, depending upon the number of samples to be analyzed and the complexity of the radioactive materials process within the facility. Table 7.2 provides common examples of how each instrument could be used in a radiation safety program.
### Table 7.2—Characteristics of common methods of sample analysis.

<table>
<thead>
<tr>
<th>Sample or Item to be Assayed</th>
<th>Liquid Scintillation Counter</th>
<th>Alpha/Beta Particle Counter</th>
<th>Gamma Ray and Alpha Particle Spectrometers</th>
</tr>
</thead>
<tbody>
<tr>
<td>• Air filters, smears or wipes</td>
<td>• Excellent for $^3$H, $^{14}$C, $^{35}$S;</td>
<td>• Not for $^3$H, but OK for $^{14}$C and higher energies;</td>
<td>• For gamma ray or alpha particle emitters;</td>
</tr>
<tr>
<td></td>
<td>• Some systems can separate 2 or 3 different nuclides;</td>
<td>• Total activity only, no energy information;</td>
<td>• Can identify radionuclides;</td>
</tr>
<tr>
<td></td>
<td>• Some systems can also do alpha particle assay</td>
<td>• Can be very low background;</td>
<td>• Can determine quantity of nuclides</td>
</tr>
<tr>
<td>• Air activity measurements; particulates on filters, gas samples, radioidines</td>
<td>• $^3$H via bubblers in water</td>
<td>• Can differentiate alpha from beta radiation</td>
<td></td>
</tr>
<tr>
<td>• Liquid sample measurements; Filterable, nonfilterable, bioassay of urine</td>
<td>• For beta particle emitters only;</td>
<td>• Can identify and quantify gamma ray or alpha particle activity on particulate filters;</td>
<td>• Can identify and quantify gamma ray activity with minimal sample preparation;</td>
</tr>
<tr>
<td></td>
<td>• Minimal sample preparation.</td>
<td>• Liquids can be concentrated and evaporated in the planchet.</td>
<td>• Liquids may be concentrated for better detection of gamma ray and alpha particle emitters.</td>
</tr>
</tbody>
</table>
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<table>
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<th>Gamma Ray and Alpha Particle Spectrometers</th>
</tr>
</thead>
<tbody>
<tr>
<td>• Bulk items like soils, vegetation, waste items, other contaminated items</td>
<td>• Not normally</td>
<td>• Small quantities of prepared soils and vegetation</td>
<td>• Can assay representative samples of these items in the laboratory (gamma only); • Can assay full items with portable gamma ray spectrometers.</td>
</tr>
</tbody>
</table>

2874
7.2.1. **Liquid Scintillation Counters**

LSCs are a valuable tool for facilities using lower energy (e.g., <200 keV, max) beta particle emitters and are essential for facilities using $^3$H, $^{14}$C, or $^{35}$S, which are impossible or very difficult to analyze on a proportional counter. More sophisticated LSCs can also be calibrated to detect both alpha particle emitters and beta particle emitters in the same LSC vial.

For LSC counting, samples (e.g., wipes, aliquots of liquids) are placed directly into a vial containing liquid scintillation cocktail. A wide variety of LSC cocktails are available to either dissolve or suspend various types of samples. The radiation emitted from the sample interacts with the cocktail and produces visible light, which is detected by one or more photomultiplier tube (PMT) light detectors, typically housed in a shielded location. Since the sample is intimately mixed with the cocktail, the detection efficiency can be very high (e.g., approaching 100 % for beta particle emitters). However, many things (e.g., the color of the sample, the chemicals in the sample, the amount of sample added to the cocktail) can greatly affect the efficiency by quenching the amount of light reaching the photodetectors. There are various methods for calibrating an LSC and accounting for quenching, with newer LSCs having more sophisticated, less laborious methods.

The lowest cost LSC will be one with a single PMT and minimal shielding. One advantage of these units is that they are light weight, and therefore can be moved and used where the samples are collected.

Most LSC instruments are more sophisticated, with 2 or 3 PMTs operated in coincident counting mode. They are heavily shielded with either passive or active shielding or both and have software for sophisticated quench corrections; they may also be refrigerated and have automatic sample changers. These LSCs have much lower backgrounds, higher efficiencies, and thus shorter count times, but they also have higher cost. Researchers and laboratories normally use these type of units so they may be nearby in the facility and perhaps available to the radiation safety department on a shared use basis.

Most of the more sophisticated units are capable of alpha particle counting and separating them from beta particles. Some units with very high light collection are capable of alpha particle spectral analysis.
7.2.2. *Alpha/Beta Particle Counters*

One of the simplest and most widely used instrument is the alpha/beta particle counter, hereafter referred to as “alpha/beta counters”. They come in a wide variety of types and costs and are available in hand-held field portable units for one sample at a time and sophisticated, heavily shielded automatic sample changer units for the laboratory. ANSI (1997) provides information regarding the calibration and use of alpha/beta counters.

At the low end of the instrument spectrum is a simple, low cost, light weight, modestly shielded, portable, 2-inch diameter pancake GM tube connected to a sample drawer and a counter-timer unit. The smear (a.k.a., a swipe) or filter is placed in the drawer and pushed under the GM tube and counted for the specified duration. While they are normally beta particle-only detection devices, some pancake GM tubes are very thin and can also detect alpha particles. Since GM tubes have no energy discrimination, separating alpha particles from beta particles requires two counts, one with a thin alpha particle absorber on top of the sample.

Somewhat more expensive would be similar manual units, but with a PMT and thin plastic scintillator for beta particles, or a ZnS(Ag) phosphor scintillator for alpha particles, or with both together on the same PMT with pulse height discrimination to separate alpha from beta particles.

The highest performance alpha/beta counters use proportional counters as the detector, which can have both high efficiency and low background. The proportional counter has a very thin window that provides good efficiency for detecting the $^{14}$C beta particle. Since proportional counters have a linear gain, they allow alpha particles to be reported separately from beta particles. The background is very low due to using very low background material in the detector construction, the low mass of the detector, and the combination of passive shielding (low background lead) and active shielding (a second guard detector operated in anti-coincident mode). Proportional counter detectors are available for two inch, and five inch diameter samples. Most proportional counters use continuously flowing gas, typically very pure methane or P-10 gas (10 % methane 90 % argon). The newer alpha/beta gas flow proportional counters have incorporated features to only use about 10 % as much gas as earlier units; with these units a single bottle of gas can last a year or more.
Alpha-beta proportional counters are available in arrays of multiple (4, 8, 16) manually loaded sample chambers. These operate in parallel and are best suited for assays with very long counting times. They are also available with a single counting chamber and with automatic sample changers capable of holding up to 100 samples.

Low background alpha-beta counters are also available using large area (2-inch diameter) silicon spectroscopy detectors incorporated with multi-channel pulse height analyzers for alpha particle spectroscopy. Since the samples are “thick” and not in a vacuum, a high-quality alpha particle spectrum is not produced, but it is adequate when used with special algorithms to determine and remove the radon-daughter fraction on the sample. This system allows early detection of other long-lived alpha emitters, without having to wait several days for the radon daughters to decay. Silicon detectors are available in single-sample modestly-shielded portable manual units or traditional automatic-sample-changer low-background units.

Windowless flow proportional counters have the highest efficiency for detection of very low energy beta particles, but this high efficiency comes with the increased possibility of the sample contaminating the detector. Windowless flow proportional counters are available in single-sample manual units and automatic-sample-changer units and are calibrated by dispensing a known quantity of a traceable source in the planchet. The efficiency is a function of beta particle energy. If the nuclide being measured is known, it is best to use that nuclide for the calibration source. More typically, several different sources covering a wide energy range are used: commonly $^{14}$C, $^{99}$Tc, $^{36}$Cl, and $^{90}$Sr/Y. The calibration sources need to be the same diameter as the sample, typically the full diameter of the planchet, and must have the same material behind it as used for the sample, since the backscatter from steel is greater than that from aluminum, and both are typical planchet materials.

The calibration is typically done with a weightless source. But since samples may not be weightless, and since both beta particle sources and especially alpha particle sources have significant self-attenuation, a set of attenuation correction factors must be developed and used. A series of calibration sources are prepared where increasing amounts of a nonradioactive carrier are added to the calibration source (ANSI 1997).
7.2.3. Gamma Ray Spectrometers

Most radiation facilities need some capability to identify and quantify unknown gamma ray emitters. The exceptions might be very small facilities that only work with beta particle emitters or only a single gamma ray emitter; however even those facilities might be faced with a sample that clearly is radioactive as shown by an elevated survey meter reading and needing to confirm that the activity is from naturally radioactive radium, or thorium, or potassium.

The most common and lowest cost gamma ray spectrometer will have a NaI(Tl) scintillation detector, coupled to a PMT, followed by a multi-channel analyzer (MCA). Normally the detector and PMT and sample are placed inside a lead shield to greatly reduce the gamma ray background from radium, thorium, and potassium in the environment. NaI(Tl) detectors typically have an energy resolution or FWHM (Full Width at Half of the Maximum height of the peak) of about 50 keV at the 662 keV gamma ray energy of $^{137}$Cs. That resolution is quite satisfactory for confirming the presence/absence of a single nuclide, or even analyzing several different nuclides mixed together, if their energies are separated by several hundred keV. Because of the rather poor energy resolution, it is difficult for the human user or the gamma ray spectroscopy software to reliably analyze NaI spectra except for well-separated peaks. Improved multi-nuclide spectroscopy comes with higher resolution detectors.

Several scintillators (e.g., LaBr, CeBr, SrI, phoswich) are available with better energy resolution than NaI, and with higher density and therefore better stopping power. However, the cost of scintillators is considerably higher than NaI. Scintillators are capable of FWM energy resolution in the 20-25 keV range at $^{137}$Cs energies. Phoswich ("phosphor sandwich") detectors use a combination of scintillators with dissimilar pulse shape characteristics to measure low-intensity, low-energy gamma rays and x rays simultaneously and are often used to measure actinide elements in the lungs.

Cadmium-Zinc-Telluride (CZT), a semi-conductor that operates at room temperature, is another improved-resolution detector. CZT detectors are routinely capable of 13 keV FWHM at 662 keV. CZT arrays several cm in size are available, but are costly. Smaller sizes (i.e., 1 to 1.5 cc) are less costly and are adequate for modestly radioactive samples. Since the detector is normally packaged with a small MCA, both can be placed in a small and light-weight shield. The
full system (detector, MCA, shield) can be light enough to be called portable, and about the same
price as a NaI detector system.

For the highest quality spectral analysis, the germanium detector is still the best choice. The Ge detector [also called HPGe for High-Purity Ge detector, to distinguish it from the earlier vintage lithium drifted detector or Ge(Li)] is also a semi-conductor detector, but with a very low band-gap. It must be operated cooled to nearly liquid nitrogen (LN) temperature, and should not be located in areas where there is a significant neutron background as neutron interactions with Ge can cause a time varying background. Ge detectors typically have 1.5 keV FWHM at 662 keV. Unlike scintillators, the energy resolution is fairly constant, ranging from 2 keV at 1332 keV and <1 keV at 60 keV. This exquisite energy resolution makes it quite easy for the human spectral analyst or the computer analysis software to analyze quite complex mixtures of nuclides.

Whereas scintillation and CZT detectors are specified in terms of actual detector size, most Ge detectors are characterized according to their “relative efficiency’ (RE). The RE (sometimes referred to as the “percent relative efficiency”) is the comparison of the peak efficiency of the Ge detector versus a 3 × 3 NaI(Tl) detector when measuring a point source of 1,332 keV photons located 25 cm from the detector. Detectors today are commonly available with RE values ranging from 10 to 200 %. A “typical” 40 % RE detector can be made in a wide variety of cylinder diameters and heights. The more common detectors have a diameter that is somewhat smaller than the height. A new class of High Aspect Ratio detectors is available with the diameter 2 - 3 times larger than the height. With these detectors, the 1,332 keV relative efficiency might be unchanged, but the efficiency at 100 keV increases by about a factor of two.

All spectroscopy detectors need a multi-channel analyzer (MCA). Most MCAs today are fully integrated units comprising the amplification, pulse-height determination, and multi-channel storage and display. As the energy resolution gets better, more channels are needed. For good quality spectral deconvolution, there should be at least three but preferably five channels within the FWHM. NaI detectors perform well with 512 channels to cover 50 to 3,000 keV. However, a good quality Ge detector covering 15 to 3,000 keV should have 16,000 channels.

All laboratory spectroscopy systems should have a good shield. They are typically made from lead, and are typically 4 inches thick, but sometimes are made of six inch thick steel. For samples that are moderately radioactive and where short counting times are adequate, then a two
inch thick lead shield is satisfactory. Lead shields commonly have a single- or double-layer inner liner to attenuate and mostly remove the ~95 keV lead characteristic x-ray peak from the background. Historically the shield has a graded liner with a higher-Z material inner liner (like Cd or Sn) to absorb the Pb x rays, and a Cu or Fe liner closest to the sample to absorb the Cd or Sn x rays. A thicker liner of Cu or Fe works just about as well. Most users would be better served with a large sample cavity so that they can assay a wide variety of samples and container sizes and shapes.

While simple gamma ray spectra can be analyzed manually, most users should obtain a good computer software package to do the analysis.

Gamma ray spectrometry systems require several types of calibrations, including Energy vs. Channel, and FWHM vs. Energy; Efficiency vs. Energy; and Cascade Summing.

Due to the complexity and expense of source-based efficiency and cascade summing calibrations, most laboratories only calibrate a few sample containers, perhaps a small one, a middle-sized one, and a large one. Then they will use laboratory procedures to take each sample type and process it to make it look like the calibrated geometry. For example, dilute or concentrate the sample to the calibrated volume; compress or add low-density materials to get to the calibrated density, which is typically water.

7.2.4. Alpha Particle Spectrometers

Alpha particle spectroscopy is less commonly performed than gamma spectroscopy but it extensively used at facilities that handle transuranics for analysis of bioassay samples and suspect air samples. Although alpha spectroscopy does not require a shielded enclosure, extensive chemical processing is required to extract and transfer the alpha radioactivity in the bioassay sample onto a planchette for alpha spectroscopy counting.

Like gamma spectroscopy, alpha spectroscopy couples surface barrier or newer ion-implanted silicon detectors with an MCA to identify the energy of alpha peaks, and a software package to identify and quantify the radionuclide. Alpha spectroscopy is particularly useful in that it can identify and subtract the alpha particle radiation emitted by the naturally occurring radionuclides, including the omnipresent radon progeny.
8. Personnel Dosimetry

There are three purposes for assessing an individual’s occupational dose received from external and internal sources of radiation. First, it provides the information necessary to compare a worker’s occupational dose with the dose limits and administrative dose guidelines. Second, it provides information on the effectiveness of the radiation safety program. Third, it provides the feedback required to implement a dose optimization program. A detailed discussion of personal monitoring techniques is presented in NCRP Report No. 57 (NCRP 1978a); an updated summary is presented below.

8.1. External Dosimetry

Effective dose, which is typically used to specify the occupational dose from external radiation fields, is not measurable directly. The operational quantity for assessing the effective dose (and organ and tissue doses) is the personal dose equivalent \( H_p(d) \) at an appropriate depth \( d \) below a specified point on the human body. For routine monitoring, the following quantities are used as a sufficiently precise assessment of effective dose for radiation protection purposes (NCRP 2018):

- For assessment of effective dose, \( H_p(10) \) (\( d = 10 \) mm) is used. \( H_p(10) \) is commonly referred to as the “deep dose”.
- For assessment of dose to the skin and to the extremities, \( H_p(0.07) \) (\( d = 0.07 \) mm) is used. \( H_p(0.07) \) is commonly referred to as the “shallow dose”.
- For assessment of dose to the lens of the eye, \( H_p(3) \) (\( d = 3 \) mm) has been proposed, but in practice \( H_p(0.07) \) typically is used.

An integrated measurement of the personal dose equivalent \( [H_p(d)] \) can be obtained using TLDs, OSLDs, Direct Ion Storage (DIS) dosimeters, film dosimeters, CR-39 polycarbonate foils, and materials such as alanine and superheated liquid drop (bubble) detectors. To obtain the dose, these passive dosimeters require processing (e.g., heating for TLD, laser reading for OSLD, developing for film, etching and counting tracks for track detectors, acquiring a spin resonance
spectrum for alanine, and counting bubbles in the bubble detector). The newer DIS dosimeters can be read using a USB connection or Bluetooth and a computer connected to the internet. Normally, passive dosimeters are used to provide the primary and permanent record of a worker’s radiation dose because they provide an economical and acceptably accurate estimate of the integrated dose from various components of complex radiation fields.

Active detectors such as pocket ionization chambers, GM detectors and bubble detectors can be built into instruments that provide a direct and real-time indication of the radiation dose rate or the radiation dose as it accumulates. Active detectors are typically equipped with audible alarms and are used when an immediate indication is needed that a dose or dose rate control level is being approached.

The radiation dosimeter used for the worker’s permanent record must be capable of measuring the types and energies of radiation likely to be encountered with sufficient accuracy, sensitivity and precision. ANSI has recommended dose-dependent performance criteria for personal dosimeters that specify acceptable levels of accuracy (bias) and precision (ANSI/HPS 2009).

The dosimeter exchange cycle (i.e., the length of time that a passive personal dosimeter is worn before it is evaluated) should be established based on the potential dose that could be received in the monitoring interval, the ability of the dosimeter to accurately integrate the radiation dose over the monitoring interval, and the need to obtain timely information about radiation exposures. Monthly or quarterly personal dosimeter exchange cycles are common for occupational radiation workers.

If the radiation fields are reasonably uniform, a single personal dosimeter worn on the trunk of the body is usually sufficient to properly characterize the worker’s radiation dose. However, if the radiation fields are not uniform or if the head or extremities are likely to be exposed to higher radiation fields than the trunk, additional dosimeters (e.g., extremity dosimeters, or a second dosimeter placed on the head (eye glasses, typically) or on the outside of a lead apron) may be required (NCRP 1995, ANSI/HPS 2011b). Workers should be instructed on how to correctly wear, store, and exchange their dosimeter(s).
8.1.1. Criteria for Providing Dosimeters to Workers

The NRC, the Agreement States, and DOE recognize several categories of people (e.g., occupational workers, Declared Pregnant Workers, minors, members of the public) for whom dosimeters may be required (NRC 2020a; DOE 2011a). The monitoring thresholds for the NRC and the DOE are the same for all categories except the occupational worker, where the NRC threshold is 5 mSv, and the DOE threshold is 1 mSv. Some workers are required (by statute or company policy) to wear dosimeters based on their job tasks (e.g., entering high radiation areas, working with some types of RGDs, performing certain nonroutine maintenance, medical or industrial radiography), while others are required to wear a dosimeter based on the likelihood of their receiving an occupational dose above the monitoring threshold. Facility management may also choose to monitor more people than required and issue dosimeters to demonstrate compliance with their license, program, or the regulations even though the monitoring threshold is unlikely to be approached. Each radiation safety program should describe the threshold at which workers are to be assigned dosimeters and the frequency of dosimeter exchange. Use of dosimeters at a facility should be periodically revisited, particularly if there are changes to radioactive material use, work locations, applications, or the frequencies at which workers perform tasks.

8.1.2. Use of Electronic Dosimeters/Real-Time Monitoring

If a worker has the potential to be exposed to higher levels of radiation, it may be necessary to provide a dosimeter that can provide doses in real time. These active dosimeters are required under certain situations (e.g., entering a high or very high radiation area (DOE 2011a; NRC 2020a), when using some irradiators, when conducting industrial radiography (NRC 2020b) and may be a good practice in others (e.g., emergencies or accident response). There are many different types of electronic/self-reading dosimeters. Some have audio and/or alarm functions so that the worker does not have to continue looking at the display to determine the situation and can concentrate on the task at hand. Some electronic dosimeters have an integration feature which allows the user to see the total dose received while the instrument was on prior to shutting it off.
It is critical that workers log the use of electronic dosimeters to document exposures when they use them. For dosimeters (such as ones that are read via a computer program) that track dose for significant periods of time, it is recommended that workers read them and document the dose frequently enough to be able to recall the origin of any dose and the circumstances under which the exposure occurred. No worker should go beyond a quarterly frequency for documentation/read out of any direct-reading monitoring device. In industries with a higher dose potential, a monthly frequency may be necessary to ensure ALARA/dose optimization. The radiation safety program should set standards for workers use and read out of electronic dosimeters. These dosimeters should be periodically recalibrated (typically annually).

Older versions of direct-reading dosimeters exhibited responses to external forces that were not dose related (e.g., bumping a pocket ion chamber, nonionizing radiation affecting certain electronic ones). Radiation safety staff should be aware of and educate workers on the limitations on their devices.

8.1.3. Accreditation Programs (NVLAP, DOELAP)

Both the NRC and DOE require dosimetry programs that are used to demonstrate compliance with regulatory dose limits to be accredited (DOE 2011a; NRC 2020a); the NRC accreditation program is called the National Voluntary Laboratory Accreditation Program (NVLAP), and the DOE accreditation program is called Department of Energy Laboratory Accreditation Program (DOELAP). An alternative is to be excepted from accreditation by the regulating agency, or use an equivalent process approved by the regulating agency. Accreditation requirements become more sophisticated as the radiation fields to be measured become more complex.

NRC licensees are required to use a NVLAP accredited vendor (NRC 2020a). The situation is different for DOE, where each site must maintain its own accreditation; DOE does not accredit vendors.

8.1.4. Area Monitoring Dosimeter Programs

An area radiation monitoring program can be used for several purposes (as discussed in Section 5.9.4), including to:
Verify (by monitoring inside RMAs and areas where RGDs are used, and Radiological Buffer Areas (RBAs)) that occupational and public dose in these areas remain below the threshold which requires individual monitoring, taking into account the occupancy factor for unmonitored workers.

Verify (by monitoring in areas adjacent to RAs, HRAs, and VHRAs) the effectiveness of engineered and administrative controls.

Detect changes in radiological conditions in Radiologically Controlled Areas (RCAs) and areas adjacent to RAs/HRAs/VHRAs.

In addition, area monitoring programs can be used to support dosimetry investigations, supplement monitoring programs in RCAs, verify nonmonitored personnel are receiving doses less than the monitoring threshold and provide additional information for dose reconstruction in the case of radiological incidents.

### 8.1.5. Nuclear Accident Dosimeters for Workers and Areas

Nuclear criticality accident dosimeters should be issued to workers when fissile material ($^{233}$U, $^{235}$U, $^{239}$Pu) is handled in sufficient quantities that an uncontrolled chain reaction could occur. Because the neutron and gamma radiation doses from criticality accidents can range from grays to hundreds of grays, specialized dosimeters are required to measure these radiation doses accurately.

### 8.1.6. External Dose Assessments

Passive personal dosimeters typically contain a variety of filters that allow the readings on the dosimeters to be interpreted to estimate the effective dose (and organ and tissue doses) from the various components of the radiation field. Care must be taken to calibrate the dosimeter to respond with acceptable accuracy to the radiation fields that will be encountered in the workplace.

If a worker is exposed to nonuniform radiation fields, the determination of the dose received can be a very complex process. In general, for doses well below the recommended limits, a single measurement taken at the appropriate location on the surface of the trunk provides enough information to estimate the dose with sufficient accuracy. However, as the dose
approaches, or exceeds, the annual limit, it may be necessary to determine the effective dose
more precisely. This can be done using the recommendations and information provided in NCRP
Report No. 122 (NCRP 1995). In addition, the accuracy of dose measurements in some situations
with nonuniform radiation fields may benefit from the use of multiple dosimeters. Guidance for
this process may be found in NRC Regulatory Guide 8.40 (NRC 2010) and ANSI/HPS N13.41 –

8.2. Internal Dosimetry

An internal dosimetry program should be considered when there is the potential for the
internal deposition of radionuclides through inhalation, ingestion, skin absorption, or wounds.
The committed effective dose, which is typically used to specify the dose from radionuclides
inside the body, requires estimating the type and amount of radioactivity in the body as a
function of time. Direct and indirect bioassay are the two measurement techniques used to make
this estimate.

Direct bioassay (also called \textit{in vivo} bioassay) involves the “direct” measurement of the
radioactivity in organs or tissues, or the entire body. This measurement is accomplished by
positioning radiation detectors near the body and detecting the radiation that escapes the body
(see Section 8.2.1). Indirect bioassay (also called \textit{in vitro} bioassay) includes a number of
techniques that are designed to measure the concentration of radioactive material in biological
samples, including urine, feces, exhaled breath, perspiration, saliva, blood, hair, fingernails, and
biopsy samples (Section 8.2.2). The method that is selected depends on the type and energy of
the radiation, route of entry into the body, the solubility (or transportability) of the material, the
biokinetics of the material, knowledge of the route of excretion, the sensitivity of the
measurement technique, and many other factors. Typically, only one bioassay measurement
technique is selected for routine monitoring. However, it is common to use as many different
methods as are appropriate to evaluate significant exposures.

8.2.1. \textit{Direct (in vivo) Monitoring}

Direct monitoring is used primarily to detect photon-emitting radionuclides. Information
on direct bioassay techniques is available (NCRP 1987; NRC 2014; Toohey et al. 1991), as well
as information that provides a means for estimating intakes of radionuclides based upon an
evaluation of the appropriate bioassay compartment through direct measurements (Potter 2002;
NRC 1988; IAEA 2004b).

The advantages of *in vivo* bioassay include:

- direct measurement of activity deposited in the body
- a relatively high accuracy for high-energy photon emitters
- ability to measure radioactivity in specific organs (*e.g.*, lungs, thyroid)
- prompt data acquisition and analysis (prompt feedback to the affected individual is
  very helpful in allaying unnecessary fears)
- simultaneous detection of many radionuclides
- preclusion of the need to give and handle biological materials.

The disadvantages of *in vivo* bioassay include:

- for nuclides with low gamma ray abundance, a lower detection sensitivity than
  excretion analyses
- difficulty in discriminating between internal and external contamination
- a need for elaborate and generally expensive equipment; however, some organ-
  specific counters (*e.g.* thyroid counters) can be less expensive
- special equipment required for radionuclides that emit photons with energy of less
  than about 100 keV
- inability to directly detect alpha and beta-particle radiations, unless there is also a
  gamma ray emission fraction

8.2.2. *Indirect (bioassay) Monitoring*

A fundamental knowledge of the biokinetics of the radionuclide in the body and the
relationship of the concentration in the bioassay sample to the intake quantity is required to
select the appropriate bioassay technique.
The advantages of indirect bioassay include:

- detection of radiations that are not easily measured by \textit{in vivo} measurements (e.g., alpha and beta particles)
- large numbers of people can be monitored, usually with minimal interference with the work schedule
- external contamination can be excluded, if careful sample collection techniques are used
- in some cases, the cost of the analysis is relatively low
- detection of the presence of almost all radionuclides is possible with standard analytical techniques and procedures

The disadvantages of \textit{in vitro} bioassay are:

- results are not typically known for days to weeks
- sample collection is often not looked upon favorably by the subjects
- samples must be transported carefully, and often shipped to off-site laboratories
- sample chain of custody records must be maintained
- some analytical techniques can be time-consuming
- presence of biohazards and the need to handle biological waste.

One approach that can be used to expedite results is to establish a simple, inexpensive procedure to screen the bioassay samples. A more extensive, complex and often expensive procedure can be used to analyze samples that are identified by the screening process as significant or those samples that are obtained as the result of a known or suspected uptake. More detailed information on indirect bioassay techniques may be found in other sources (Boecker et al. 1991; NCRP 1987). The reports by Potter (2002), NRC (1988), and IAEA (2004b) are very useful for estimating intakes based on measurements of activity in selected bioassay compartments at known times after the intake.
8.2.3. **Bioassay Program Criteria**

A formal internal radiation exposure control program should be established when there is the potential for workers to experience an annual intake resulting in a committed effective dose in excess of 1 mSv (NCRP 1998). If a worker is likely to receive an intake resulting in 10% or more of the regulatory dose limit ($\geq 5$ mSv), bioassay monitoring is required (DOE 2011a; NRC 2020a).

The general types of bioassay that should be considered in establishing an appropriate sampling process are:

- baseline
- termination
- diagnostic (e.g., following a suspected uptake)
- routine or periodic.

The purpose of the baseline bioassay is to establish the pre-existing levels of incorporated radionuclides for each worker. The termination bioassay is used to provide a final record of any incorporated radionuclides. Diagnostic bioassays are taken to evaluate the intake and retention of radioactive material from occupational exposures. Information from the diagnostic bioassay samples is used in dose assessments and evaluating the efficacy of dose reduction therapies. Diagnostic bioassay results are also used to determine additional actions that may be appropriate such as additional engineered controls, protection equipment, or procedural changes.

Routine or periodic bioassays are obtained to ensure compliance with dose limits and administrative dose guidelines and to evaluate the effectiveness of the protection program. This sampling must be done on a regular basis and it is important that an appropriate sampling frequency be established.

A qualified individual should assess the potential for intakes of radionuclides and implement a bioassay program as necessary. Some of the factors determining if it is necessary to implement a bioassay program are the types, quantities, and chemical and physical forms of the radioactive material in use and the potential doses that could be received during routine operations and accidental conditions. Additional factors to consider would be the potential for
the radioactive materials to be dispersed into the workspace, the use of engineered controls and personal protective equipment (e.g., fume hoods, respirators), and frequency of use.

The frequency of routine bioassay should be based on the activities being performed, air sampling results, the potential magnitude of an exposure, the retention characteristics of the radionuclides in the body, and the ability of the bioassay technique to detect an uptake of the radionuclide(s). Detection should be at a level that is acceptable for the protection of the health and safety of the worker. This last point requires careful consideration of the radiation dose limits, the use of administrative dose guidelines, the acceptable uncertainty in the estimated intake and the biokinetic model for the radionuclide. The frequency selected should ensure that an intake that exceeds the established administrative dose guidelines will not go undetected by the bioassay technique used. Additional recommendations for bioassay monitoring programs are found in ICRP Publication 78 (ICRP 1997) and ANSI/HPS N13.39 (2001a).

Selection of the appropriate bioassay monitoring technique will be dependent on the radiation type emitted by the nuclides of interest, cost, convenience to workers, and required detection levels.

8.2.4. Accreditation Programs (DOELAP)

When used to demonstrate compliance with DOE regulatory dose limits, an internal dose monitoring program must be accredited by the DOELAP for Radiobioassay (DOE 2011a). DOELAP accreditation involves performance testing and documentation of program elements important to the long-term quality assurance of a radiobioassay program, and assessment of the program’s ability to accurately perform, record, and report the measurement of radioactive material within the human body and in human biological samples. An alternative is to be excepted from accreditation by the DOE or use an equivalent process approved by the DOE.

Useful implementing information is provided by DOE (2019) and ANSI/HPS (2011c).

8.2.5. Internal Dose Assessment

When radioactive materials enter the body, a direct measurement of the radiation dose is not possible. The assessment of effective dose requires an analysis of the type and amount of material taken into the body, as well as its metabolism and excretion from the body. To perform
this assessment reliably requires a program that specifies sampling frequencies and measurement procedures in accordance with accepted methodologies. The program must also be adequate to acquire the necessary information for the assessment.

The information required to assess the internal dose following an intake of radioactive materials is:

- radionuclide identity
- the route of entry of the radionuclide or radioactive compound into the body
- the time of intake
- the chemical form of the radioactive compound
- biokinetic models for interpreting bioassay results
- particle size distribution
- the physical properties of the radiations emitted by the radionuclide

These parameters are used to determine the intake of the radionuclide and this value is subsequently used to calculate the dose. Published calculations of the internal committed equivalent dose to tissues of the body are currently normalized to a unit intake of activity, \( \text{Sv Bq}^{-1} \) (ICRP 2015, 2016a, 2016b, 2017, 2019). These calculations use averaged metabolic data and are adequate for assessing routine exposures that are below the committed effective dose limit. For routine exposures the qualified expert must provide a reasonable estimate of the total intake of activity of a particular radionuclide. Estimates of the committed equivalent doses to various tissues and the committed effective dose can be derived from the published calculations.

The assessment of nonroutine exposures to internally deposited radionuclides that approach or exceed the effective dose limit should be based on the actual metabolism of the material in the exposed individual as provided by the bioassay measurements. These data may be supplemented with workplace information such as air sampling, particle size and solubility information, and monitoring results, as well as operational information provided by the worker. The evaluation for these non routine internal dose estimates are typically performed by an internal dosimetrist using sophisticated biokinetic models and computer codes. A useful
summary of these biokinetic models used for intakes via inhalation, ingestion, and wounds is provided in NCRP Report No. 176 (NCRP 2017).

When bioassay information is unavailable, internal dose can be estimated using air sampling and monitoring results, in combination with worker stay times. This methodology is inherently imprecise and should not be used as the primary means of internal dose assessment.
9. Control of Exposure to the Public

Control of radiation exposure to the public from an operating facility begins with an understanding of the types of radiation and radioactive materials used at the facility and extends through the facility design, operation, and decommissioning. Facility design elements should be tailored to provide shielding of penetrating radiation, including skyshine, and management of airborne, liquid, and solid effluents, during routine operations and anticipated accidents, so that public radiation doses can be demonstrated to be compliant with regulations and ALARA.

Verification that public radiation doses meet regulatory limits is provided through environmental surveillance. This section describes important considerations for conducting environmental surveillance.

9.1. Determining the Need for Surveillance

The need for, and extent of, an on-site and offsite environmental surveillance program to monitor exposures to members of the public should be determined by an initial assessment of the types of radiation and the quantities and forms of radioactive materials expected to be present in the facility. It may be possible to demonstrate that the maximum potential exposure to a member of the public (either on-site or offsite) would be too small to justify a substantive surveillance effort if:

- the radiation fields are expected to be relatively small, well shielded, or of very short duration,
- the quantities of radioactive material are relatively small;
- the containment of radioactive materials is robust;
- the processes and activities are well known and adequately documented;
- public access to the source(s) is restricted.

Routine audits of the facility should verify the initial assumptions and a new or revised assessment should be made if deviations from the initial assumptions are significant. A new assessment as to whether or not environmental surveillance is necessary is warranted if new
radiation sources or significantly larger quantities of radioactive material are added, and if
procedures, workloads, or operating parameters change significantly. If adjustments to the initial
assessment assumptions are needed, a license amendment may also be needed.

Mobile or portable sources of radiation and radioactivity must be included in the
assessment of potential exposures to the public. The assessment of mobile sources can be
streamlined by identifying and analyzing the limiting exposure scenarios. In addition, sources
transported by a private carrier (as opposed to contract or common carriers) must be packaged
and shipped in accordance with DOT regulations to assure compliance with public dose limits
during transport.

At facilities with multiple RGDs, the contribution of each RGD to the exposure in a
public access area must be assessed. Expected radiation fields may be calculated, or one may
rely on the manufacturer’s representation of the output, and reasonable assumptions may be
made regarding occupancy in public areas. The assumed parameters should be verified before, or
shortly after, operations begin, to determine if ongoing surveillance is necessary in areas
accessible to the public.

At facilities where discharges will, or could, be made to the environment, screening
models to determine the need for surveillance are available in NCRP Report No. 123 (NCRP
1996). The input to these models consists of data on the:

- types, quantities, and physical/chemical forms of the individual radionuclides present
  in the facility,
- locations and types of potential releases, and
- generic assumptions about the characteristics of the surrounding environment and
  population.

When applying screening models, the generic assumptions should be reviewed to
determine if they are consistent with the conditions in the facility to assure that the resulting dose
estimate will provide reasonable assurance that the public dose limits will be met. If the
conditions in the facility are more complex than the screening models can accommodate, the
models should be modified and more data collection may be required to produce suitable site-
specific models.
In some cases, such as for RGDs or the use of sealed sources, process knowledge may be relied upon in lieu of routine surveillance. However, the surveillance that might be needed in the event of accidental exposures or release of radioactive material to the environment should be considered. Systems installed for routine surveillance are typically designed to also provide emergency monitoring capabilities, but if no routine surveillance system is planned, emergency surveillance should be considered as an independent requirement.

9.2. Radiological Effluent Monitoring and Environmental Surveillance

Radiological effluent monitoring refers to radionuclide and radiation measurements performed at points of radionuclide origin or release (e.g., in a discharge stack, at the site boundary), while environmental surveillance refers to such measurements performed throughout the environment. Radiological effluent monitoring and environmental surveillance programs are described in NCRP Report No. 169 (NCRP 2010c), which provides a detailed examination of, and recommendations for, monitoring airborne and liquid radioactive effluents, and for establishing a comprehensive environmental surveillance program. As radiological effluent monitoring and environmental surveillance are complementary activities, they are ideally designed and operated as a single program at large facilities. A synopsis of information from Report No. 169 is provided in the following sections and may be used to determine the extent and scope of radiological effluent monitoring and surveillance required at a given facility.

9.2.1. Monitoring Radioactive Airborne Effluents

When establishing a radiological monitoring program for airborne effluents, it is essential to:

- understand the processes within the facility;
- understand the design and flow rates of air through the facility’s ventilation and effluent treatment systems;
- understand the form of the potential discharges (e.g., noncondensable gases, particles suspended in gas, vapors, liquid droplets);
- define the point(s) where discharges to the environment can occur; and
• determine the potential for routine and accidental releases from each discharge point
  (including from unexpected sources such as a breach caused by over-pressurization of
  a system), and the release of radioactive materials those conditions would generate.

  Knowledge of the air flow rates at the points of release is essential to assessing the
  quantities of radionuclides that are released over time. Because air flow rates may vary with
  operating conditions and with time (e.g., due to maintenance, changes in blower capacity and
  filter loading), routine records of fan operations and periodic measurements of discharge flow
  rates are important components of the airborne radionuclide effluent monitoring program.

  Radioactive airborne effluents are normally monitored at a point that is downstream of all
  effluent treatment systems and of the exhaust fans, just prior to discharge to the environment.
  The primary consideration in selecting a monitoring location is the requirement that the air
  withdrawn from the effluent stream contains a representative sample of that stream. Consensus
  guidance for selection of monitoring locations may be found in ANSI/HPS N13.1-2011
  (ANSI/HPS 2011a) and ISO-2889-2010 (ISO 2010); however, specific governmental regulations
  that differ may apply. Monitoring location essentials in these guidance documents include a
  demonstration of contaminant mixing; velocity uniformity; and an average flow angle parallel
  with the axis of the stack and sampling probe. For large ducts or stacks, the guidance documents
  describe the number of withdrawal points needed to assure that the composite sample will
  represent the average concentration being discharged. In exhaust systems that have several ducts
  that are combined in a plenum, uniformity of flow across the exhaust duct is not a sufficient
  condition for a uniform distribution of contaminants in the discharge.

  The presence of the exhaust blower upstream of the sampling location will help to assure,
  but not guarantee, that the radionuclides are well mixed in the exhaust. Nonreactive tracer gases
  (e.g., He and SF₆) can be used to reliably evaluate the degree of mixing in the exhaust gases of
  contaminants released from various parts of the facility. The effectiveness of air mixing caused
  by duct features is documented by Han et al. (2007), McFarland et al. (1999a, 1999b), and Seo et
  al. (2006). These studies provide guidance on the distance to suitable sampling locations
  downstream of mixing features.

  Measurements of concentration profiles in the exhaust duct may be necessary to ensure a
  representative sample of the effluent is collected when there are constant releases from several
process areas. These measurements can also be used to determine if multiple sampling probes are required. Evaluations of the requirements for representative sampling for normal operating conditions and those expected following an accident (when ventilation system flows may be quite different) are both important.

Transport of the sample from the monitoring location to the point of collection is another factor that may affect the choice of monitoring location. Short sampling lines are generally recommended to reduce the potential for material loss due to deposition or impaction on the wall of the line. It is important that any bias due to loss of material in transport be understood and accounted for when monitoring results are reported. An experimental determination using the installed sampling line presents challenges but can be performed (Glissmeyer 2006). Modeling approaches to this problem suffer difficulties; however, there have been demonstrations that under certain conditions modeling compares well with experimental data (Wong, et al. 1996), and methods for modeling may be found in the literature (Brockman 1993; Glissmeyer 2010).

Guidance on radioiodine transport lines is given by Glissmeyer and Sehmel (1991). If the effluent contains radionuclides in particulate form, it is important to collect information about the particle size distribution and the rate of deposition. Models for estimating particle loss during sample transport, as a function of particle size and other parameters, are available and can be used to select sampling line parameters that will minimize deposition and impaction losses (Anand, et al. 1993). Rodgers (1995) suggests that for most applications, a 10 µm aerodynamic diameter particle size is appropriate for the design and evaluation of sampling systems for both normal and off-normal conditions.

If the sample collected from the effluent is not or cannot be analyzed in real time, a sampling frequency must be established. The selected frequency will depend on the variability of the radionuclide release rate, the chance that unplanned releases will occur, the likely magnitude of any unplanned releases, and other factors such as the half-lives of the radionuclides in the effluent. In some situations (e.g., a revised radionuclide mixture from a breached source or a reactor accident involving the fuel), a separate monitoring system may be necessary for post-accident conditions because the demands for information about routine effluents and about accidental releases are not achievable in a single system. Information on sample analysis is provided in Section 7.2.
9.2.2. Monitoring Liquid Effluents

When establishing a radiological monitoring program for liquid effluents, it is essential to understand:

- the sources of radioactivity within the facility and the processes that affect it;
- the design of the effluent piping and treatment systems;
- determination of the liquid flow rates at continuous release points; and
- volumes of liquid wastes that are discharged in batches.

There may be multiple liquid waste streams with different characteristics, waste generation rates, and treatment requirements prior to discharge. Releases of some liquid wastes may be continuous, but collection of liquid wastes in a tank prior to batch discharge is also common. In some cases, liquids are collected and solidified or evaporated by a waste broker before disposal as low-level radioactive waste. Both the NRC and DOE specify effluent monitoring requirements (NRC 2009, NRC 2007a; DOE 2011c) and reporting requirements (NRC 2007b; DOE 2011b).

Wastewater not subject to RCRA that has been stored prior to batch discharge can be sampled and analyzed for a broad range of radionuclides, which will frequently provide greater sensitivity than on-line monitoring and account for radioactive decay of short-lived radionuclides. Collection of representative samples of discharged liquids can be challenging because some radionuclides may be in solution while others may be present in suspended solids or as sediments in the tank, with the proportions dependent upon variables such as pH and temperature. Prior to grab sampling, the liquid in the tank should be thoroughly mixed. If the tank cannot be mixed, the inlet stream should be sampled proportionally or monitored throughout the time that the tank is being filled. It may also be desirable to obtain multiple samples from the waste transfer or discharge line to confirm release estimates based on sampling of a tank or the inlet flow.

If liquid waste discharges occur continuously but flows are variable, proportional sampling of the effluent is recommended. Monitoring may be accomplished using instrumentation that provides data in real time. While this technique offers the possibility of
prompt notification and automatic shut-off capability if elevated releases are detected, the
method is not uniformly applicable. Information on sample analysis is provided in Section 7.2.

9.2.3. Solid Waste Effluents

An often-overlooked source of environmental releases is the discharge of solid radioactive waste, which is discussed in Section 5.11. Most radioactive materials licenses require reporting of the quantities of radioactive waste generated and the radionuclide inventory of those wastes. The radionuclide concentrations in drums or shipping containers provides manifest information used to determine compliance with DOT shipping regulations, and compliance with low-level waste disposal site waste acceptance criteria. The radionuclide content of waste containers can be measured and recorded as a container is filled, or from sampling the radionuclide content if the waste is homogeneous, or by direct measurement of the external dose rates from the containers. Depending on the facility and the potential for generating solid radioactive waste, the most common option is for onsite processing, typically compaction, incineration, or solidification of liquids, prior to shipment under DOT regulations to a licensed low-level waste disposal facility. For small waste generators, onsite processing, packaging, shipment, and disposal can be contracted to a licensed waste broker service.

9.3. Environmental Surveillance

Environmental surveillance is the measurement of external radiation and the collection and analysis of samples of air, water, soil, foodstuffs, biota, and other media in the environs of nuclear facilities or contaminated locations (NCRP 2010c). The need for environmental surveillance at any given facility is driven by the expected radiation field at the facility's site boundary or public access points, planned discharges from the facility, the potential for accidental discharges, and the magnitude of those planned or unplanned discharges.

Facilities with a low potential to contaminate the surrounding environs may demonstrate compliance with applicable public dose limits through calculations using limits contained in 10 CFR 20, Appendix B (NRC 2020a), or by using available computer codes (e.g., COMPLY (EPA 2016), CAP88-PC (EPA 2014). Large facilities with the potential to measurably impact the environment are expected to develop a robust sampling and monitoring plan for off-site
deposition, as described in NCRP Report No. 169 (NCRP 2010c), as well as for radiation dose at
the site boundary or in publicly accessible areas.

9.3.1. Preoperational Monitoring

Obtaining a baseline understanding of radiation levels or pre-existing radioactive materials at the site may be important and required, even for relatively small and low-use facilities. Contribution to background levels may come from the local setting (geology and altitude), building materials (e.g., bricks with significant naturally occurring radioactive material content), high radon levels, or adjacent facilities that are licensed or registered to use sources of radiation. Each of these sources may impact area and environmental dosimeters, as well as influence routine surveys made during operations to demonstrate compliance with public dose limits.

Preoperational monitoring is imperative for larger, more complex facilities that have the potential for routine discharges to the environment or for a significant accidental release. Conducting an environmental monitoring program prior to facility start-up allows for documentation of preoperational environmental radionuclide levels and their variability, and for testing of the necessary monitoring techniques and analytical procedures. In addition to naturally occurring radionuclides, remnants of the long-lived radionuclides released during the above ground nuclear weapons testing era are still found in soils, sediments, and to some extent in other media; occasionally, these radionuclides may be significant.

Some investigation and knowledge of other facilities near the site is also warranted as hospitals, industries, or other institutions may release radionuclides into a waterway shared by the new facility. The importance of such releases and of residual radionuclides from those facilities to the environmental monitoring program will depend on the type of facility and the radionuclides it produces or handles.

Depending on the nature of a new facility, the preoperational period is also the time for investigating and documenting other factors that could influence public radiation doses. These factors may include land and water uses in the vicinity of the facility, the regional meteorology, the regional population distribution, drinking and irrigation water sources, local agricultural production, private gardening activities, local recreation, sport and commercial fishing, and other
special land use or population factors that could impact the expected public dose after a release. Knowledge of the mechanisms of potential radionuclide releases from the facility and investigation of the environmental behavior of those radionuclides will identify the most important radionuclides and environmental exposure pathways that lead to exposure of the population. This information, together with results of meteorological and hydrological investigations, will aid in the design of the environmental surveillance program and in the selection of environmental monitoring locations.

9.3.2. Operational Monitoring

Operational monitoring includes both on-site and off-site monitoring of radiation levels, and estimation of the concentrations of radioactive materials in the facility and in the environment in general. Monitoring data can provide verification of the assumptions made during the preoperational phase regarding potential radiation exposures; ongoing confirmation that radiation dose limits are being met for both radiation workers and members of the public; and serve as an indication of an off-normal condition such as when radiation levels or effluent concentrations are above those expected.

The scope and extent of an ongoing monitoring program depends on a number of factors, including the direct radiation levels produced within and around the facility, the time-varying nature of these levels, the quantities of radionuclides released within or from the facility, the isotopic composition, the chemical and physical forms of effluents, the points of release to the environment, land use and population demographics, and the occupancy of the areas where radiation or radioactive material may be encountered.

Facilities where the use of radiation and radioactive material is relatively low, and where there is no potential for high exposures or large accidental releases of radionuclides, may be able to demonstrate compliance with dose limits for members of the public solely by calculation, or by use of the screening models identified in Section 9.1. Nevertheless, it is prudent to maintain the capability for periodic confirmatory measurements and for accidental release assessment.

Coordination of the surveillance program with the dose assessment procedure will assure that the monitoring data collected include those needed for dose assessment. In general, it is desirable to measure exposure along a critical pathway at the point closest to the potentially
exposed person. For example, in the air-grass-cow-milk-child pathway, measurements of radionuclides in impacted milk will be more reliable indicators of the dose to the child than measurements of air or pasture grass contamination. When media or organisms that concentrate radionuclides are not themselves consumed by humans, but are used as indicators for potential concentrations in a human food item, the relationship between the indicator and the item must be defined in preoperational or early operational surveys. Measurements of such correlations in the local environment are preferred over those obtained from the literature unless it is known that the two environmental settings are very similar. Even then, confirmatory measurements are desirable.

The operational environmental surveillance program may also provide a check on the effluent monitoring program. Periodic review of the two data sets and relevant dispersion estimates is necessary to determine whether measured environmental levels are consistent with those expected from effluent measurements and food chain concentrations from modeling. In most cases this may not provide a measurable correlation since most effluents are low concentration and since dispersion downwind will likely result in environmental levels that are below detection limits.

Monitoring locations that are selected based on site-specific average meteorological conditions may not provide the useful information for a short-term accidental release because the downwind concentrations and locations for an accidental release will be dependent on the meteorological conditions during the accident. However, direct exposure rates or measures of stack concentrations designed for operational monitoring may be used with the meteorological data during the accident to determine those locations that may receive the highest off-site concentrations or doses through modeling.

9.4. Measurement Methods

The physics underlying radiation measurement techniques and several radioactivity measurement procedures are described in NCRP Report No. 57 (NCRP 1978a) and NCRP Report No. 58 (NCRP 1985). The techniques presented in those reports can be applied for analyses of measurements and samples from both on-site and off-site monitoring of radiation levels and radioactive material releases.
Measurements of environmental radiation fields and environmental sample collection, preparation and analysis procedures are also discussed in NCRP Report No. 50 (NCRP 1976b). References to many radiochemical procedures are included as are discussions of common laboratory instrumentation. The manual of procedures of the DOE Environmental Measurements Laboratory (EML) is an excellent source of information on techniques of radiochemical analysis, sample collection, field measurements of radiation fields, and radioactivity measurements (EML 1997).

The literature on radiation and radioactivity measurement techniques, procedures for sampling and analysis, and solution of special problems is voluminous. Selection of appropriate sampling, analysis, and measurement techniques will depend on facility-specific factors, including the anticipated radiation fields, the radionuclides that may be released, their concentrations, the nature of the site and its surroundings, and other factors. The references given are intended to serve as general guides and starting points into the investigations into methods suitable to the needs of a facility.

9.5. Dose Assessment

Among the goals of radiation and radioactivity monitoring programs is the assessment of radiation doses received by members of the public at, in the vicinity of, or otherwise impacted by, the facility. The complexity of the dose assessment procedure depends on the type of facility, the number of release points, the level and nature of the associated radiation fields, and the types and amounts of radionuclides released to the environment during facility operations.

For facilities using only RGDs, or low levels of radioactivity without the potential for a significant release (e.g., sealed sources, low-activity sources, or very short half-life radionuclides), exposure may be assessed by the use of continuous monitoring, periodic confirmation surveys, or by calculation based on radiation fields produced for each work-evolution performed, or potential worst-case contamination events in combination with a validated estimate of public occupancy in unrestricted spaces that could be impacted.

For facilities with the potential to release relatively small amounts of radioactive materials to environmental soils, screening level assessments may suffice to show reasonable assurance of compliance with regulatory limits. The NCRP (1996) has provided generally
conservative screening models for environmental releases (also see (NCRP 1989b, 1993)). These off-the-shelf models, with conservative parameters, may be satisfactory to determine upper bounds for exposure of members of the public and provide reasonable assurance that the public dose limits will be met.

Other calculational methods that may be required include the use of computer codes produced by the U.S. Environmental Protection Agency (EPA) for dose estimates due to releases to the atmosphere, such as CAP88-PC (EPA 2014) or COMPLY code (EPA 2016). Similarly, the use of limits on air and water effluents found in 10 CFR 20, Appendix B (NRC 2020a) and DOE Order 458.1 (DOE 2011c) may be used to estimate doses to a maximally exposed member of a critical group of the public, based on the upper bounds of potential releases from the facility.

Large, complex facilities making routine releases, or having the potential for making a large, accidental release to the environment will require more complex modeling than screening models using parameters that are representative of actual site (and surrounding) conditions. Data to support the choices of parameters can be obtained from site-specific environmental studies in the pre-operational or operational monitoring phase. Facilities that require more complex modeling may have on-site meteorological and hydrological monitoring programs that provide data for estimating dispersion of effluents.

Translating environmental data to public dose in the vicinity of large, complex facilities may be accomplished using various computer codes, such as the RESRAD family of codes (ANL 2021). RESRAD-ONSITE (Kamboj et al. 2018) was originally developed for DOE for use in determining site-specific cleanup levels for environmental restoration, and the NRC has adopted an updated version, RESRAD-OFFSITE (Yu, et al. 2020), for estimating off-site doses from the effluents from operating nuclear facilities. Whether the dose assessment is performed using a simple screening model or a complex, site-specific model, it is essential that the assumptions and results be performed (or verified) by a qualified expert, and thoroughly documented.

Doses from patients who have undergone systemic therapy are a special case and specific guidance on assessing dose to members of the public, including caregivers, family members, breastfeeding infants and others may be found in NCRP Report No. 155 (NCRP 2006). For further discussion see Appendix C.1.
9.6. Reporting

Regulatory agencies may require reporting using a prescribed reporting format if analysis indicates radiation dose from a facility exceeds applicable regulatory limits or certain regulatory constraints in publicly accessible areas. The information required for the report will be identified in regulation, or in the terms and conditions of a facilities license or registration documents.
10. Response to Spills and Emergencies

Radiation incidents and emergencies can occur wherever radioactive material or RGDs are used, may be of a wide range of magnitudes, and may require a wide variety of responses. Emergencies can be either unintentional (accidental), or intentional (such as an incident caused by a terrorist). Intentional emergencies are thoroughly outlined and discussed in previous NCRP reports (NCRP 2010a, 2014) and will not be covered here. Also, accidents that could affect large numbers of people are outside the scope of this report. Emergency planning is discussed in detail in NCRP Report No. 111 (NCRP 1991a) and the general methodology is summarized in this subsection. Although some of the information in NCRP Report No. 111 is dated (e.g., it references ICRP 30), the general methodologies remain relevant.

Examples of incidents that could result in a radiation emergency include:

- Spills of liquid and/or solid radioactive material
- Release of radioactive dusts, mists, fumes, and gases
- Personal injury with or without personal contamination
- Overexposures of workers
- Loss of a radioactive source
- Fire in or near a radiological area
- Major natural disaster resulting in release of radioactive material
- Nuclear power plant accident resulting in release of radioactive material

The overriding purpose of radiological response to abnormal conditions and emergencies is to protect human health and critical infrastructure. Each facility should establish action levels for determining when an incident or accident requires a localized, facility-wide, or off-site support response. For a detailed description on investigating these events see Section 11.3, Investigation of Radiological Incidents.
10.1. **Spill Response**

Spills can occur in any facility where radioactive material in a liquid or dispersible solid form is present. Many facilities classify spills as “minor” or “major”. The descriptions below suggest one basis for this classification, but others may be used as appropriate when considering a particular facility.

### 10.1.1. Minor Radioactive Material Spills

A minor spill involves a relatively low level of radioactivity and a relatively small physical area (*e.g.*, discrete areas within a room). Generally, the cleanup can be performed by radiation workers at the facility, in coordination with the radiation safety staff. A minor spill would generally result in minimal personnel contamination and no contamination spread outside the work area. Actions to take for a minor spill include:

**Prompt actions**

- Stop or confine the spill.
- Warn people in immediate area of spill. Nonessential people should leave the area and not reenter until decontamination has been completed.
- Isolate the contaminated area.
- Minimize the spread of contamination, if practicable (*e.g.*, place absorbent paper towels over a liquid spill).
- Secure or control access to the area; properly post the area if it must be left unattended prior to decontamination.

Some organizations use the acronym SWIMS as a convenient way of remembering these prompt actions.

**Decontamination**

- Collect supplies needed for the decontamination, including protective clothing.
- Clean up the spill, consistent with established procedures and waste minimization requirements. Simple cleaning techniques and procedures using water and soap are
adequate for most decontamination tasks; commercially available cleaning supplies could be used if necessary. Clean from the outer region inward to reduce the possibility of further spread of the contamination.

- Survey the area with an appropriate survey meter or take and analyze swipes. Repeat cleanup until contamination is no longer detected or is below the level requiring control. If the area cannot be sufficiently decontaminated, contact the radiation safety staff.

- Before leaving the area, monitor the area, hands, and shoes for contamination.

Notifications

- Workers should report the incident to the radiation safety staff and the area supervisor.

10.1.2. Major Radioactive Material Spills

A major spill involves more activity than a minor spill and could result in significant personnel contamination, possible airborne releases, and potential for widespread contamination outside the immediate work area. For these spills, the radiation safety staff should be involved in the response. Actions to be taken for a major spill include:

Prompt actions (usually done by the worker or the radiation safety staff, if initially present)

- Attend to seriously injured or contaminated persons and remove them from areas with potential for significant radiation exposure. Call 911 as appropriate.

- Notify others in or near the immediate contamination area to exit the area, segregating people who are likely to be contaminated.

- Request all workers remain in the area until their contamination status has been assessed by radiation safety staff.

- For liquid spills, prevent the spread of contamination by covering with absorbent pads or diatomaceous earth, but do not yet attempt to clean it up.
• If necessary, stop/shield the spill, but only if doing so does not risk further contamination or significantly increase one’s radiation exposure.

• Close the room and lock or otherwise secure the area to prevent entry. Properly post the area (e.g., with a High Contamination Area or Radiation Area sign) and restrict access to authorized personnel only.

Notifications

• Workers should immediately notify the radiation safety staff and follow their guidance for subsequent actions.

• Workers or the RSO should notify area supervisors and management.

• If needed, the RSO should contact the Radiation Emergency Assistance Center/Training Site (REAC/TS) (https://orise.orau.gov/reacts/index.html) for emergency medical response and subject matter expertise on the medical management of radiation incidents.

• Management or the RSO should notify the regulatory authority, as appropriate.

Decontamination (usually performed under the direction of the RSO)

• Develop and implement a reentry and decontamination plan.

• If the contamination cannot be reduced to levels acceptable under the license and/or regulations, or if it is likely that activity remains but cannot be accessed (e.g., in a floor drain system), the area may have to be maintained as a controlled or restricted area.

• Records of this event, relevant actions, surveys, and final status should be (and in some cases are required to be) maintained.

10.2. Radiation Incidents and Emergencies: Planning and Preparedness

Every facility should have a site-wide emergency plan. For small, nonradiological facilities, the plan may be as simple as an evacuation plan in case of fire. If the facility uses radioactive material or RGDs, the plan should address radiological emergencies as well. The
response plan for radiological emergencies should be integrated into the facility’s overall emergency plan as an incident may include a number of disparate hazards (e.g., fire, chemicals, biohazards). This subsection will focus on planning for radiological emergencies, understanding that other hazards may take priority in the emergency response.

Although the resultant emergency plans may be very simple or very complex depending on the facility, the process of emergency planning is the same for all facilities and includes the following basic steps:

1. Prospective classification of potential emergencies
2. Development of an Emergency Plan (based on the prospective classifications), including:
   • Then organizational structure
   • Assignment of responsibilities for those who respond to accidents
3. Development of Emergency Plan Implementing Procedures (EPIPs), including:
   • Training of the responders and other personnel who may be impacted by the incident
   • Coordination with outside agencies
4. Evaluation of the Emergency Plan, including:
   • Drills and exercises involving many or all of the responding units.

10.2.1. Prospective Classification of Potential Emergencies

Before developing an emergency plan, one must know which emergencies to plan for, as well as their potential severity. While many potential emergency situations can be postulated, plans should be limited to credible events. Steps for classifying potential emergencies are as follows:

1. **Identify the locations** of radiation sources (i.e., RGDs and radioactive material) within the facility and their associated hazards (Table 10.1).
Table 10.1—*Sources and types of radiation hazards.*

<table>
<thead>
<tr>
<th>Type of Radiation Hazard</th>
<th>Sources of radiation hazards&lt;sup&gt;a&lt;/sup&gt;</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>RGDs</td>
</tr>
<tr>
<td>External whole or partial body radiation exposure</td>
<td>X</td>
</tr>
<tr>
<td>Skin or personal clothing contamination</td>
<td>(X)</td>
</tr>
<tr>
<td>Internal radiation exposure resulting from:</td>
<td>(X)</td>
</tr>
<tr>
<td>• ingestion or inhalation</td>
<td></td>
</tr>
<tr>
<td>• puncture wound or laceration</td>
<td></td>
</tr>
<tr>
<td>• submersion in a radioactive cloud</td>
<td></td>
</tr>
<tr>
<td>Area contamination</td>
<td>(X)</td>
</tr>
<tr>
<td>Environmental release</td>
<td>(X)</td>
</tr>
</tbody>
</table>

<sup>a</sup>"X" denotes “likely”; “(X)” denotes “possible, but less likely.”
1. **Identify potential emergencies** (*i.e.*, accident scenarios), consistent with the type of work conducted and the amount of radioactivity handled.

2. **Identify the potential severity of emergencies.** NCRP Report No. 111 (NCRP 1991a) primarily uses committed effective dose or committed equivalent dose to determine the severity of the accident. As NCRP Report No. 111 references ICRP 30, current biokinetic models should be used to calculate potential doses to workers.

3. **Classify events as** Incident, Level 1 Emergency, or Level 2 Emergency. NCRP Report No. 111 defines these levels according to their dose implication (see row 1 of Table 10.2) and provides examples of conditions that would fall into each level, as shown in the remainder of Table 10.2. However, professional discretion is necessary when classifying radiological events, as a given scenario may not fit neatly into a single column. While other classification schemes may be used, the general concept of binning potential accident scenarios is helpful when the emergency plan needs to cover a variety of potential accidents.

Once the scope of potential accidents is defined and the emergency planning reference level is selected, an appropriate sequence of responses may be integrated into an emergency plan.

### 10.2.2. Development of the Emergency Plan

An emergency plan should describe the emergency organization structure, the lines of authority, and the functions and duties of individuals who will respond to an emergency. Key individuals should be assigned to address the following topics, depending on the complexity of the plan, and in the event of an emergency, should report to the emergency director:

- industrial safety and hazardous substance control
- security and traffic control
- public information
- physical plant services
- medical services
- legal counsel
Table 10.2—Classification of radiological emergencies.

<table>
<thead>
<tr>
<th>Attribute</th>
<th>Incident</th>
<th>Emergency Level 1</th>
<th>Emergency Level 2</th>
</tr>
</thead>
<tbody>
<tr>
<td>Committed effective dose or committed equivalent dose to workers and emergency responders</td>
<td>● Less than the applicable dose limit</td>
<td>● Exceeds the applicable dose limit</td>
<td>● Deterministic effects (i.e., tissue effects) are possible</td>
</tr>
<tr>
<td>Personnel injury and/or contamination</td>
<td>● Radioactive contamination of personnel</td>
<td>● Admission of an injured and contaminated patient to a medical treatment facility</td>
<td>● An accident involving serious injury and extensive personnel contamination and exposure</td>
</tr>
<tr>
<td>Area contamination</td>
<td>● Minor injuries</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>● Ingestion or inhalation of radioactive material</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Area contamination</td>
<td>● Contamination in laboratories</td>
<td>● Facility multi-laboratory contamination</td>
<td>● Contamination possible outside the facility and off-site.</td>
</tr>
<tr>
<td></td>
<td>● Unplanned release of radioactive material</td>
<td>● Unplanned release of radioactive material that could exceed the public dose limit of 5 mSv</td>
<td>● Potential for release of a large quantity of radioactivity</td>
</tr>
<tr>
<td>Radioactive material control</td>
<td>● Loss of radioactive material</td>
<td>● Loss or theft of radioactive material</td>
<td>● Loss or theft of a Category 1 or 2 quantity (IAEA 2004a)</td>
</tr>
<tr>
<td>Emergency response personnel (e.g., fire, police)</td>
<td>● Not needed</td>
<td>● Involvement of nonfacility personnel such as fire fighters and police</td>
<td>● Emergency personnel entering areas which are significantly contaminated</td>
</tr>
<tr>
<td>Example events</td>
<td>● Ruptured glove box glove resulting in room contamination and contamination of workers</td>
<td>● Floods or fires in low-level radioactive waste areas</td>
<td>● Nuclear Power Plant loss of coolant accident</td>
</tr>
</tbody>
</table>
The emergency plan should be inclusive of all the hazard types as an emergency might include several hazards (e.g., fire, hazardous material releases, radiation, and physical injuries), and depending on the details, one or more hazards might warrant far more attention than the others.

The radiological section of the emergency plan should describe several types of emergency conditions and provide an appropriate classification and response for each. However, it should be remembered that the classification scheme described in Section 10.3.1 is based solely on the potential radiological impact. A fire or explosion would be classified as a serious emergency irrespective of the radiological consequences.

To assure that there is an appropriate level of preparedness, it is imperative that the plan be supported by the level of management that has the authority to commit resources and implement policy. For larger facilities, senior management may need to appoint an emergency coordinator to supervise the development and maintenance of the emergency plan. The emergency coordinator should also be the contact person for outside assistance organizations such as fire and police and medical.

10.2.3. Development of Emergency Plan Implementing Procedures

Procedures should be developed to address the range and type of emergencies determined through the classification process. If the emergency plan is relatively simple, a single implementing procedure that covers the following topics may be sufficient.

- Identify who is responsible for formally classifying the emergency and implementing response procedures
- Describe training and retraining required for responsible individuals
- Have a list of contact names and contact information for relevant internal and external individuals, organizations, and responders
- Outline precautions that may be necessary for each type of response
• Describe which radiation detection instruments will be needed and how to access them
• Describe other PPE that may be needed for each type of emergency
• Develop checklists for each identified type of emergency.

If the emergency plan is complex, it should be implemented through EPIPs which are described fully in NCRP Report No. 111 (NCRP 1991a). These are documented instructions that describe the actions necessary to achieve the objectives of the plan. EPIPs should be written to cover the range of credible emergencies. They should cover the actions needed both during the emergency and during restoration following the emergency. Each EPIP should contain:

• a statement of the purpose of the EPIP
• identification of the person or group who is responsible for the EPIP
• identification of the communication techniques to be used
• a description of the action sequence required to achieve the purpose
• a description of the prerequisites to the specified actions
• the precautions necessary and the limitations of the prescribed tasks
• guidelines to be followed in the exercise of judgment
• training requirements
• copies or examples of the forms to be used
• sign-off sheets, checklists and data sheets

In general, EPIPs should be prepared for the emergency director, security and police personnel, RSO, public information officer, plant services director, institutional fire marshal, medical officer, and legal counsel, to the extent these positions exist within the facility. It is important that each EPIP describe the initial training and periodic retraining programs that are required for primary and backup persons who will be responsible for the EPIP.

It is the responsibility of the site management to make available to the emergency response teams the facilities and equipment that are required by the EPIPs. Equipment that may
be needed for radiological responses includes radiation detection instrumentation, sampling
devices, personal dosimeters, personal protective equipment, decontamination supplies and
communications systems. Equipment should be ready and operable and measurement and
sampling equipment should be calibrated, consistent with the requirements of the plan.

10.2.4. Evaluation of the Emergency Plan (drills and exercises)

Once the emergency plan has been drafted, it should be reviewed, evaluated and tested.
To be effective, the completed plan should be explained to leaders of all the functional groups
(e.g., emergency director, security and police chief, the RSO, public information officer, plant
services director, institutional fire marshal, medical officer, and legal counsel) and their advice
and input should be solicited and incorporated (as appropriate) on the section of the plan in
which they will participate. The draft plan should be approved by top-level management and, if
appropriate, by the external agencies that will participate in it.

When it is complete, the plan should be thoroughly tested. Drills, where parts of the
emergency plan are practiced, should be conducted before an integrated, full-scale exercise so
the participants have knowledge and experience in carrying out their part of the plan. Drills and
exercises should have a stated purpose of what is being exercised and the expectations of what is
to happen. Gaps in individual drill/exercise performances, facility issues, and the emergency plan
should be identified and the gaps should be evaluated and entered into the corrective action
program. Individual performance gaps should be remediated by training or other appropriate
means.

Following the drills, a full-scale, integrated exercise should be conducted, and selected
evaluators should observe what actually happens. Upon completion of the exercise, a formal
critique should be conducted and the evaluators and participants should be asked to comment on
the strengths and weaknesses of the plan, the observations of what actually happened, and their
impacts on the stated purposes. Adjustments to the plan should be made, as needed. The plan
should be reviewed annually and modified as necessary to adjust for changes in operations.
Tabletop exercises should be conducted annually and whenever the plan is modified, and a full-
scale exercise should be conducted at least every 5 y.
11. Quality Assurance

Quality assurance is a systematic evaluation of activity to measure outcomes against expectations. Audits, inspections, surveillance and statistical evaluations (quality control checks) are all basic quality assurance tools used to make systematic evaluations.

Management should ensure that there is a quality assurance program in place to provide oversight of the Radiation Safety Program. A quality assurance program encourages and supports self-assessments by work groups and Radiation Safety program staff. Independent audits are developed when necessary to gain an outside perspective of aspects of the Radiation Safety program. The overall goal of the quality assurance program is to improve performance and it should not be an adversarial or fault-finding activity.

Management has a responsibility not only to assure that corporate goals are met, but also to assure that operations are performed with safety, reliability and cost effectiveness. The assurance of safe, reliable and cost-effective performance depends on management, including the RSO. Reviews of operating records and personal interviews with the staff are necessary components in the discharge of this responsibility.

11.1. Measurement Quality

The results of measurements made as part of a radiation safety program must be reliable. The facility operator should assure that the measurement of radiation fields is consistent with standard radiation safety practices; that environmental and area monitoring dosimeters are analyzed by a qualified laboratory; that collection, analysis and interpretation of facility, effluent and environmental samples are performed in an appropriate way; that measurement instrumentation is properly calibrated, and that the use of the results in a dose assessment procedure is consistent with the nature of the information.

All aspects of the program should be carried out by qualified individuals using appropriate procedures that have been reviewed and approved to meet the goals of the program. The instrumentation used for measurements of radiation and radioactivity must be of adequate quality and calibrated using reference standards prepared by the National Institute of Standards
and Technology (NIST) or other qualified sources. Routine checks of the instrumentation, analysis of duplicate or replicate samples, and comparisons with independent laboratories of the results of analyses of split samples are examples of quality control activities that contribute to data reliability. Quality assurance activities should be recorded and periodically audited and written reports of the work should be prepared routinely.

The quality control activities that are performed by the facility operator will depend upon whether the sampling and analysis steps are performed in-house or by a contractor. However, overall responsibility for quality assurance of the monitoring data rests with the facility operator. Some aspects of the quality assurance program may be dictated by the license for the facility or by guidance issued by the licensing or other governmental authority.

11.2. Assessment of Radiation Safety Program Performance

NCRP recommendations for performing self assessments of radiation safety programs are discussed in detail in NCRP Report No. 162 (NCRP 2009b). A brief summary of these recommendations is presented in this section.

Self assessments are planned and controlled internally by an institution. They differ from external reviews, which are generally referred to as inspections, audits or appraisals and are controlled from outside the institution, often by regulatory agencies.

Self assessments can identify and correct deficiencies and can improve the performance of the radiation safety program. Another benefit of the self assessment program is to encourage worker involvement in improving the radiation safety program. The elements of the self assessment process include:

- establishing an overall self assessment program plan;
- assigning responsibilities for implementing the self assessment program;
- identifying the purpose of the self assessment;
- identifying the type of self assessment to be performed (i.e., performance, risk, or compliance);
developing an individual self assessment plan (e.g., areas to be assessed, criteria used for assessment, level of self assessment);

conducting the self assessment;

documenting and reporting the results of the self assessment;

determining the causes of any deficiencies;

developing corrective actions; and

implementing and verifying the implementation and effectiveness of corrective actions.

Self assessments also vary based on the level at which the assessment occurs (task, process or program) and on the degree of desired formality and rigor. While each primary type has a different objective, elements of each may be used, especially when conducting a comprehensive self assessment of a large radiation safety program. Each type may be used to assess specific aspects of a radiation safety program element in one or across multiple line organizations in a horizontal slice. The assessment types may also be combined to review radiation safety in a particular line organization in a vertical slice such as waste management or operations in a particular laboratory. The types of self assessment are discussed in the following paragraphs.

11.2.1. Performance-Based Self Assessment

A performance-based self assessment evaluates the overall effectiveness and efficiency of a radiation safety program. That is, in addition to evaluating compliance with regulations, it takes a broad look at the radiation safety program (or an element of a radiation safety program) to assess how well it is performing. This could include assessing how well the ALARA program is implemented, the adequacy of and adherence to internal procedures, the ability to respond to incidents and accidents, the quality and implementation of radiation safety training, ongoing efforts to improve the radiation safety program, the quality of the radiation safety staff and the guidance they provide, and management commitment to and involvement in the radiation safety
program. In summary, a performance-based self assessment evaluates the effectiveness of the radiation safety program beyond just compliance with regulations.

11.2.2. Risk-Based Self Assessment

A risk-based self assessment is the evaluation of the radiation safety program to determine if the radiological risks have been reduced to those deemed acceptable by the institution. These risks could be health risks to workers, the public, or the environment. In addition to the potential health risks of a radiological incident or condition, the potential financial costs of suspending operations, mitigation efforts, adverse public perception, regulatory fines and impacts, and the negative impact on business should be considered.

A risk assessment considers both the consequences of a hazardous incident and the probability that the incident could occur. The radiation safety program should be designed and implemented to reduce the radiological risk from primary program activities (e.g., nuclear power plant, nuclear-medicine laboratory, particle accelerator) to an acceptable level. If the radiation safety program is not properly designed or executed, it can result in unacceptable risks to workers, the public, the environment, and to the facility itself (financial, reputation). For example, the improper analysis of the contents of a waste-retention tank could result in a significant release to the environment or the improper use of a survey instrument could lead to worker overexposure.

In a risk-based self assessment, the questions that should be asked are, “What could go wrong?” and “Have mitigating measures been put in place to keep the risk at acceptable levels if something does go wrong?” It is important that senior management establish expectations and standards for acceptable risk, based on regulatory and other stakeholder considerations.

11.2.3. Compliance-Based Self Assessment

All radiation safety programs must satisfy applicable regulatory requirements and license or permit conditions. The regulations that apply to a radiation safety program are determined by the type of operation; the size, complexity and potential radiological hazards associated with a program; and the regulatory agency that has jurisdiction. Regulations are issued by NRC and its Agreement States, the states (for accelerators and x-ray machines), DOE, EPA, OSHA, FDA,
DOT, MSHA, and other government agencies. These requirements should be incorporated in top-level policy documents and reflected in implementing procedures. A compliance-based self assessment evaluates whether a radiation safety program complies with regulatory requirements and license or permit conditions. This process first requires the identification of the applicable radiation safety regulations including license requirements. Next, it requires a review of the radiation safety program to evaluate whether all regulations and other external requirements (e.g., contractual, permits, registrations) are being met. Since a compliance-based self assessment focuses solely on compliance with regulations, it is typically less comprehensive than a performance-based assessment and may not be a valid assessment of the overall safety program.

Finally, there should be a process in place to bring the program into compliance with new regulatory requirements. This can be a particularly challenging process when an institution has facilities at more than one site and involves different state regulatory agencies.

NCRP Report No. 162 (NCRP 2009b) provides examples of lines of inquiry for each of the three types of self assessment for many of the elements of a radiation safety program (e.g., training, engineered safety controls, emergency preparedness, environmental protection, and waste management).

11.2.4. External Assessments

In addition to self assessments, organizations should anticipate and be prepared for external assessments, where the assessors will normally be less familiar with the details of the radiation safety program. External assessments can be compliance-, performance-, or risk based, and are often a combination thereof. Whereas outside reviewers sometimes identify important issues which may have been overlooked or underprioritized by the organization, they can also serve to validate the findings of the self assessment program and highlight issues identified internally. External validation of issues can raise the profile of and facilitate solving the more difficult or costly issues. External assessments can also impact public perception, contract fees, and the license to operate.
Investigation of Radiological Incidents

A radiological incident is an abnormal occurrence that may adversely affect the health and safety of workers, members of the public, or the environment. It can cause property or environmental damage, interrupt program activities and result in noncompliance with regulations. Radiological incidents can occur wherever radioactive material is handled, stored, or transported or where radiation generating equipment is operated. Many radiological incidents have minor consequences and may not require an investigation. Radiological incidents that have a significant adverse impact on the health and safety of workers or members of the public, or adversely impact the environment should be investigated. In addition, regulatory agencies require that radiological incidents be investigated if regulatory limits are exceeded.

The primary purposes of an incident investigation are to determine what happened, establish the cause(s), contributing factors, and consequences of the incident, as well as to recommend corrective measures. Another purpose is to improve work processes to minimize the likelihood of future incidents, as opposed to finding fault with or fixing blame on individuals. The extent and rigor of an investigation should be tailored to the severity and complexity of the incident.

NCRP Report No. 173 (NCRP 2012) provides specific guidance on the following aspects of the investigation process:

- definition of a radiological incident, description of the investigation process and proposes of an incident investigation;
- determination of whether the incident warrants an investigation;
- responsibilities for conducting or participating in incident investigations, including upper management, line management, the radiation safety committee, radiation safety program personnel including the radiation safety program manager or radiation safety officer, and workers;
- initial response to the incident, including the procedures for controlling the incident scene to prevent loss of information, recovering any physical items that may have been removed, and gathering information related to the incident;
• coordination of facility recovery activities and the incident investigation;

• appointment of an individual or an incident investigation team to perform the incident investigation, including recommendations for the training and qualifications of investigators and the use of consultants and specialists in conducting the investigation;

• conduct of the incident investigation, including the initial team meeting, interviewing facility representatives and personnel involved in the incident, and collecting physical evidence;

• performance of the cause analysis, including which type of cause analysis to perform (i.e. direct cause, apparent cause or root cause);

• use of various cause analysis techniques including barrier analysis, task analysis, work change analysis, events and causal factors charting, process analysis, and human performance analysis;

• identification of the cause(s) of the incident;

• determination of the consequences of the incident including radiation doses to workers, patients, and member of the public, releases of radioactivity to the environment, property damage, and interruption of program activities:

• development of a corrective-action plan;

• preparation of the investigation report including legal consideration, and

• follow-up actions, including scheduling corrective actions, reviewing, tracking, and trending the effectiveness of the corrective actions, and lessons-learned distribution.

Finally, Appendix A in NCRP Report No. 173 contains an introductory discussion of Root Cause Analysis Tools and Techniques.
Corrective Action Identification and Tracking

Radiation safety program modifications and improvements are warranted as a result of deficiencies discovered during operations, surveys, self assessments, regulatory inspections, incident analyses, or facility reviews, as well as changes in:

- organizational mission
- regulatory or contractual requirements for which the institution is responsible
- work objectives, priorities or efficiency
- technical knowledge or support
- personnel
- the radiological source term.

Corrections of deficiencies and improvements in operations often arise from reviews of the program by the RSO and from self assessments. Some may arise from the independent audits or the recommendations of government regulators. In any case, changes in the program and personnel responsible for effecting the changes must be recorded. This is especially important to ensure that there is a clear understanding of responsibility, a reasonably uniform approach to safety, and a well-documented response to the identified deficiency.

Changes or corrective actions must be communicated to the RSC, management, and to appropriate workers. This is especially important when common services such as personal monitoring, waste management, training, and emergency response are affected. Records help to limit the confusion that may arise during independent audits and ease the quality assurance efforts of the RSO. Good records are important for effective tracking of changes by the RSO and management.
12. Records

The amount and detail of radiological records that should be maintained has become substantial and their maintenance can consume an appreciable portion of the effort of the radiation safety staff.

Records that should be maintained in a readily retrievable form include the following:

- “master” copies of the radiation safety policy manual and procedures,
- operating procedures,
- personnel dosimetry records,
- area survey records,
- internal self assessments and external audit reports (with deficiencies and corrective actions),
- instrument calibration records,
- waste management records,
- worker training records,
- licenses/state registrations,
- license applications,
- regulatory inspection reports,
- inventories of radioactive materials and RGD’s, and
- any regulatory or legal actions related to the radiation safety program.

The capability for long-term safe retention of these records must be developed and maintained. The institution needs to develop a written retention and disposition schedule for records related to the radiation safety program that should include record material, relevant data and supporting information. Records must be protected from exposure to fire, water damage and other physical damage. A program must also be developed to ensure back-up and retrievability.
of record information and duplication/storage in a separate location. If records are kept on
electronic storage media, the following should be considered (DOE 2017):

- Quality control during data entry and analysis
- Identification of software applications used with the data
- Software validation and verification
- Quality audits of software
- Prevention of unauthorized manipulation of data
- Assurance that stored information is retrievable and useable after system modifications
- Provisions for converting the data to new storage media and software before the current storage media and software become obsolete.

The NCRP has provided guidance for maintaining radiation safety records in Report No. 114 (NCRP 1992). Additional guidance may be found in ANSI/HPS N13.6, Practice for Occupational Exposure Records Systems (ANSI/HPS 2010). The NRC and DOE both specify the length of radiological records retention (NRC 2020a; DOE 2011a), but due to the latent period between exposure to radiation and possible effects, NCRP (NCRP 1992) recommends a longer (75 y) retention period for records that may be useful for epidemiological studies than for other occupational records (50 y).


As discussed in Section 3.2.1, there should be one comprehensive set of Radiation Safety policies and procedures that cover not only the routine, day-to-day activities conducted by the staff who use either radioactive materials or RGDs, but also anticipated accident or emergency conditions. The procedures and policies can be combined into a single Radiation Safety Manual, if desired. These policies and procedures should be reviewed and updated periodically, with distribution of any modifications, amendments, and new items. The Radiation Safety Manual is frequently delivered electronically, and a record copy of each revision should be maintained either in hardcopy or electronically in a retrievable format.
12.2. **Dosimetry Records**

The personnel dosimetry program records should include the types of dosimeters used, the dosimeters exchange frequency, the lower limit of detection, and the individuals or classes of individuals who are monitored.

Radiation dose records should be maintained to demonstrate compliance with dose limits and administrative dose guidelines, and to assist in the evaluation of the effectiveness of the external dose control program. Recommendations for the content of occupational dose records are given in NCRP Report No. 114 (NCRP 1992). Additional guidance is contained in ANSI/HPS N13.6, *Practice for Occupational Exposure Records Systems* (ANSI/HPS 2010). Guidance in these documents includes the following:

- the annual occupational radiation dose for each year that the worker was monitored, including the doses received during previous employment and concurrent employment, if the worker was simultaneously employed at more than one facility.
- the cumulative occupational radiation dose received by the worker
- the respective doses for those parts of the body that were monitored separately
- any significant accidental exposures including the circumstances of the accident, the cause of the accident and the evaluation of personal doses received
- the circumstances of any planned special exposures, and in particular, those that exceeded the dose limits for normal operations
- any special evaluations of personal dose that resulted in an adjustment to the measured dose to obtain an assigned dose
- a description of the dosimetry system used and a discussion of the calibration methodology;
- records related to accreditation by NVLAP or DOELAP;
- evaluations for the application of the ALARA principle.

Internal radiation dose records should be maintained to demonstrate compliance with dose limits and administrative dose guidelines and to assist in the evaluation of the effectiveness
of the internal dose control program. Recommendations for the content of occupational dose
records are given in NCRP Report No. 114 (NCRP 1992) and ANSI/HPS N13.6 (ANSI/HPS
2010) and include the following:

- annual intakes of radioactive material
- radionuclides deposited in the body
- exposure routes
- physical and chemical forms of the radionuclides
- purpose of bioassay measurement (e.g., baseline, termination, diagnostic or routine)
- date and location of suspected or confirmed intake
- type of bioassay measurement
- a listing of the bioassay data used in the determination of the equivalent dose
- information to enable linkage to procedures, calibration factors, geometry,
  background and energy resolution checks, and confidence levels
- metabolic and dosimetry models used
- assumptions used to estimate intake of radioactive material and the committed
effective dose
- magnitude and location of the deposition of specific radionuclides
- identification of the individuals who were involved in making the estimates of intake,
deposition and committed effective dose
- organ or tissue equivalent doses and committed effective dose
- records related to accreditation by DOELAP, if applicable.

In addition to the records maintained for the personnel dosimetry program, records
should be maintained of the surveys of airborne radionuclides that are performed including the
information that is detailed in Section 5.9.5.
12.3. **Radiological Surveillance Program Records**

Documentation of the measurements related to radiation fields or contamination at a facility is an essential component of the monitoring program. This documentation includes the data collected, assumptions made when estimating unmeasured parameters, the name of the person collecting the data, the date of collection, the instrumentation used, data related to background levels for the instrumentation, and any unusual circumstances (e.g., instrumentation malfunction). In addition, it is equally important to maintain records of adjunct measurements or information important to dose assessment, such as flow rate measurements for air exhausts; instrument calibrations; direct measurement, sampling and analytical techniques employed; and quality assurance procedures for the various components of the measurement program.

The contents of radiological surveillance records are discussed in Section 5.9 and should be maintained as radiation safety program records. NCRP Report No. 114 (NCRP 1992) provides additional guidance on radiological surveillance record maintenance and retention.

12.4. **Environmental Monitoring Records**

Documentation of the measurements of effluents from the facility is an essential component of the monitoring program. This documentation should include the data collected, information about unusual occurrences (e.g., sampler malfunction) and assumptions made when estimating unmeasured releases. In addition, it is equally important to maintain records of effluent flow rate measurements, instrument calibrations, analytical techniques employed and the modifications of those techniques with time, and quality assurance procedures for the various components of the measurement program.

The environmental monitoring program must be similarly well documented. In addition to documentation of measurement techniques, as discussed above, the meteorological and hydrological measurement programs must also be documented, including records of equipment calibration and testing and appropriate quality assurance measures that were employed in data collection and analysis.
The capability for long-term safe retention of these records must be developed and maintained. Further recommendations regarding such records are provided in NCRP Report No. 114 (NCRP 1992).

12.5. **Radiation Safety Instrument Program Records**

Detailed guidelines for records that should be maintained as part of a radiation safety instrument program are specified in NCRP Report No. 114 (NCRP 1992). It is important that adequate records be kept documenting the specification of instrumentation for each specific purpose, the calibration of the instrumentation, the maintenance and repair of the instrumentation, as well as an instrument inventory. For institutions or facilities that have large numbers of instruments, these records should be kept in a computer database and linked in such a way that all the information concerning a particular instrument can be retrieved easily.

Specification records should include the type of detector, energy range, dynamic range, sensitivity, accuracy and precision, response time, directional response, mixed radiation field response, and any restrictions on the operating environment of the instrument.

Calibration records should include information about the calibration facility, such as a general description, standard instrumentation, radiation sources, calibration information pertaining to the sources, and the results of any studies of background or radiation scattering that are applicable to the facility. Instrument calibration records should include the instrument identification, the date of the calibration, the sources used, the individual or commercial calibration facility who performed the calibration, the environmental conditions, and the results including any adjustments that were made.

Maintenance records should include the instrument identification, the date of repair, a statement of the problem and corrective action, the individual who made the repair, and whether a recalibration was indicated. Information about instrument reliability is also useful to provide guidance for replacing instruments.

Inventory records should include the instrument identification, the dates the instrument was placed into or removed from service, and the location, department/section, or individual to whom the instrument is assigned.
12.6. Radioactive Materials and Waste Disposal Records

Records of the receipt of orders for radioactive materials, the shipments of radioactive materials to other persons, as well as surveys on certain incoming material shipments (in accordance with 10 CFR 20.1906 (NRC 2020a) (or equivalent Agreement State regulations)) should be maintained, in conformance with the applicable and appropriate standards of the regulatory agency. Note, however, that in addition to the regulations of the applicable regulatory authority, shipments of radioactive material are to conform to the appropriate regulations of the US Department of Transportation’s requirements (DOT 2015).

Radioactive material records should include inventories of sealed radioactive sources (hereafter referred to as “sealed sources,” or “sources”) and their leak tests, in accordance with the NRC (NRC 2020a) and DOE (DOE 2011a) requirements. For more guidance, see NCRP Report No. 182 (NCRP 2019). If the inventory of sealed radioactive sources notes an unresolved discrepancy, the notification to the regulatory agency, the subsequent investigation, and the resolution to that discrepancy constitute records that should also be maintained.

Records of the disposal of radioactive and mixed wastes should be maintained in a retrievable and legible form for the lifetime of the facility involved. Guidance for this record keeping is found in NCRP Report No. 114 (NCRP 1992).

12.7. Other Records

Other records that should be maintained at a facility include the technical basis documents for the radiation safety program, including area and air monitoring programs, internal and external dosimetry programs, radiological laboratories, periodic audits and self assessments of the program (see Section 11.2), training records, and regulatory mandated inspections.

Audit and self-assessment records should include what was reviewed within the program, any items found as noncompliant or not adhering to currently accepted business practice and the corrective actions for each. These documents should include the date and identity of the person or persons performing the audit or self assessment.

Training records should include the topics covered, the date, the instructor’s qualifications and the names of all individuals who attended. The records should also include
course descriptions, lesson plans for classroom, computer-based training, and on the job training.

Training record maintenance requirements may vary, depending on regulatory agency requirements, but should be able to demonstrate the adequacy of training.

Inspection documentation should also be maintained for review by the appropriate regulatory agency. This may include quarterly documentation (certain RGDs), six-month documentation (inventory for many programs), or annual documentation (certain RGDs).

Certain medical devices (e.g., accelerators, teletherapy units, mammographic x-ray units) require periodic calibrations to ensure that the radiation outputs are accurate and that the doses delivered to patients are in accordance with applicable standards. These calibration records must also be maintained.
Appendix A: Suggested Topics for Radiation Safety Training (NCRP 2000b)

Table A.1 lists topics that should be considered for inclusion in the preparation of a general training program for occupationally exposed employees.

“Essential elements” are those topics that should be included in all radiation safety training programs. The extent of instruction and its technical content should be adjusted depending on the work assignments of the staff being trained.

“Optional elements” are those topics that should be considered for inclusion in the training program. These depend on the types of operations conducted at the facility and the work assignments of the staff.

Radiation safety technicians need additional detailed training in the subjects listed in the third column of Table A.1.
### Table A.1—Topics for a general training program.

<table>
<thead>
<tr>
<th>Essential Elements</th>
<th>Optional Elements</th>
<th>Additional Elements for Radiation Safety Technicians</th>
</tr>
</thead>
<tbody>
<tr>
<td>Risks related to exposure to radiation and to other workplace hazards</td>
<td>Waste management/waste minimization and pollution prevention</td>
<td>Inspections/audits</td>
</tr>
<tr>
<td>Dose limits and site-specific administrative controls</td>
<td>Institutional radiation safety manual</td>
<td>Packaging/shipping labeling for transportation</td>
</tr>
<tr>
<td>Mode of exposure</td>
<td>Contamination control</td>
<td></td>
</tr>
<tr>
<td>• External radiation sources</td>
<td>• protective clothing and equipment</td>
<td></td>
</tr>
<tr>
<td>• Internal radioactive materials</td>
<td>• surveying for radiation and contamination</td>
<td></td>
</tr>
<tr>
<td>– Absorption</td>
<td>• decontamination</td>
<td></td>
</tr>
<tr>
<td>– Inhalation</td>
<td></td>
<td></td>
</tr>
<tr>
<td>– Ingestion</td>
<td></td>
<td></td>
</tr>
<tr>
<td>– Wounds</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Basic protective measures including engineering and administrative controls</td>
<td>Work area decontamination</td>
<td>Use and calibration of survey and monitoring equipment</td>
</tr>
<tr>
<td>Security including securing materials and facilities and the individual’s role</td>
<td>License/regulations</td>
<td>Recording information and survey data and keeping records</td>
</tr>
<tr>
<td>Emergency notification procedures and response</td>
<td>Characteristics of ionizing and nonionizing radiation</td>
<td>Principles and practices for reducing radiation exposure</td>
</tr>
<tr>
<td>Warning signs, postings, labeling and alarms</td>
<td>Radioactive material and decay</td>
<td>Radiation and radioactive contamination survey techniques</td>
</tr>
<tr>
<td>Responsibility of employees and organizations</td>
<td>Radiation-producing equipment</td>
<td>Emergency response and personal decontamination</td>
</tr>
<tr>
<td></td>
<td>Natural and manufactured sources</td>
<td></td>
</tr>
<tr>
<td>Interaction with radiation safety staff</td>
<td>Acute effects of exposure</td>
<td>Chronic effects of low-level exposure</td>
</tr>
<tr>
<td>----------------------------------------</td>
<td>--------------------------</td>
<td>--------------------------------------</td>
</tr>
<tr>
<td>Overall safety</td>
<td>Determination of dose</td>
<td>Bioassay requirements</td>
</tr>
<tr>
<td>Description of and requirements related to specific facility hazards</td>
<td>Basic radiation survey instrumentation</td>
<td>Radiation accident control techniques</td>
</tr>
<tr>
<td>Special requirements for women of reproductive age</td>
<td>Radiation monitoring program and procedures</td>
<td>Proper selection and use of personal protective equipment</td>
</tr>
<tr>
<td>Basic monitoring for radiation exposure</td>
<td>Identification and control of radiation sources</td>
<td></td>
</tr>
<tr>
<td>Procedures for maintaining doses ALARA</td>
<td>Selection and use of appropriate personal radiation dosimeters</td>
<td></td>
</tr>
</tbody>
</table>
Appendix B: Example of the use of the HotSpot Code for Workplace Selection

The HotSpot codes (Homann and Aluzzi 2014) include a “Radionuclides in the Workplace’ module which was developed to aid in the selection of a suitable workplace for handling dispersible radioactive material. It should be emphasized that this Code is a preliminary screening technique and as with any of the workplace selection techniques discussed in Section 5.3.1, the final selection of a workplace should be made by a qualified expert.

The HotSpot code calculates a hazard guide value ($H$) (which corresponds to a workplace type), such that an accidental inhalation intake would not likely exceed a committed effective dose of 5 mSv (500 mrem) or a committed equivalent dose to an organ or tissue of 50 mSv (5 rem). That is, if an accidental inhalation dose from a benchtop operation would result in a committed effective dose $>5$ mSv or a committed equivalent dose of 50 mSv, the code recommends use of a fume hood; if an accidental inhalation dose from a fume hood operation would result in $>5$ mSv or a committed equivalent dose of 50 mSv, the code recommends use of a glovebox.

The basic equation for $H$ is:

$$H = QTU / \sqrt{m} \quad (B.1)$$

Where:

$H$ = hazard guide value (dimensionless)

$Q$ = quantity of radionuclide (Bq)

$T$ = toxicity factor (dimensionless)

$U$ = use factor (dimensionless)

$m$ = mass of the material, including the radioactive and nonradioactive constituents (g)

Note: If $m < 1$ g, assume $m = 1$ g. The square root of “$m$” is applied because, in general, as the mass of the material increases, it is assumed that the dispersible fraction of the total mass decreases in the event of an incident.
Workplace determination as a function of H

<table>
<thead>
<tr>
<th>Hazard Guide Value (for input in Bq)</th>
<th>Hazard Guide Value (for input in µCi)</th>
<th>Recommended Workplace</th>
</tr>
</thead>
<tbody>
<tr>
<td>H ≤ 3.7 E+11</td>
<td>H ≤ 1 E+7</td>
<td>Type 1: Bench Top</td>
</tr>
<tr>
<td>3.7 E+11 &lt; H &lt; 3.7 E+14</td>
<td>1 E+7 &lt; H &lt; 1E+10</td>
<td>Type 2: Chemical Fume Hood</td>
</tr>
<tr>
<td>H ≥ 3.7 E+14</td>
<td>H ≥ 1 E+10</td>
<td>Type 3: Glovebox</td>
</tr>
</tbody>
</table>

Toxicity Factor

The toxicity factor, T, is defined as the ratio of the DAC for HTO to the DAC of the radionuclide(s) being handled as shown in equation B3:

\[ T = \frac{DAC_{HTO}}{DAC_{\text{radionuclide(s) under consideration}}} \]  

Because of its common usage, HTO was arbitrarily selected as the reference material on which to base the relative toxicity. The user has the option of selecting SI or traditional units. The DACs are calculated using the dose conversion factors (DCFs) from DC_PAK3.02, (Eckerman, and Leggett 2008).

The equation for the DAC is: (Equation B.3)

\[ DAC_{(Bq \ m^{-3})} = \frac{CED_{(SV)}}{2,400 \ m^3} \]  

Where:

- \( DAC \) = derived air concentration (Bq m\(^{-3}\))
- \( Q \) = quantity of radionuclide (Bq)
The use factor represents the estimated respirable fraction of the radioactive material being handled and is a function of the work being performed. For normal activities the inhaled fraction is assumed to be $1 \times 10^{-5}$. This inhalation fraction is based on actual incidents involving the inhalation of radioactive material and is conservative by factors of 10 to 1,000 with respect to actual observed values (Franke et al. 1966; Brodsky 1980).

Compared to normal operations, the use factor is assumed to be higher for activities that have a greater potential for dispersing material (e.g., evaporation to dryness) and lower for activities where the potential for dispersing material is less (e.g., washing precipitates). The use factors for various activities are shown in B.1.

**Example of Using Radionuclides in the Workplace**

This example uses DC_PAK3.02 DCFs (Eckerman and Leggett 2008) for calculating the DACs (workers).

If an experimenter plans to pipette a solution containing 10 MBq of $^{239}$Pu (Class M). The activity concentration of the solution is 740 kBq g$^{-1}$, therefore total mass is 13.5 g. The DCF for HTO is $1.83 \times 10^{-11}$ Sv Bq$^{-1}$ and the DCF for $^{239}$Pu (Class M) is $1.14 \times 10^{-3}$ Sv Bq$^{-1}$ (the tissue-reaction committed equivalent dose to bone surface).
Table B.1—Use factors for various types of operation.

<table>
<thead>
<tr>
<th>Type of Operation</th>
<th>Use Factor</th>
</tr>
</thead>
<tbody>
<tr>
<td>Storage</td>
<td>0.01</td>
</tr>
<tr>
<td>Very simple, wet</td>
<td>0.1</td>
</tr>
<tr>
<td>Diluting stock solutions</td>
<td></td>
</tr>
<tr>
<td>Washing precipitates</td>
<td></td>
</tr>
<tr>
<td>Normal</td>
<td>1</td>
</tr>
<tr>
<td>No production of dry material</td>
<td></td>
</tr>
<tr>
<td>No vigorous chemical reactions</td>
<td></td>
</tr>
<tr>
<td>Precipitation</td>
<td></td>
</tr>
<tr>
<td>Filtration or centrifuging</td>
<td></td>
</tr>
<tr>
<td>Solvent extraction</td>
<td></td>
</tr>
<tr>
<td>Chromatography</td>
<td></td>
</tr>
<tr>
<td>Pipetting or titration</td>
<td></td>
</tr>
<tr>
<td>Ambient temperatures</td>
<td></td>
</tr>
<tr>
<td>Simple dry</td>
<td>5</td>
</tr>
<tr>
<td>Fusion reactions</td>
<td></td>
</tr>
<tr>
<td>Fluorination</td>
<td></td>
</tr>
<tr>
<td>Transfer of dry precipitates</td>
<td></td>
</tr>
<tr>
<td>Complex wet</td>
<td>50</td>
</tr>
<tr>
<td>Distillation</td>
<td></td>
</tr>
<tr>
<td>Evaporation to dryness</td>
<td></td>
</tr>
<tr>
<td>Elevated temperatures</td>
<td></td>
</tr>
<tr>
<td>Dry and dusty</td>
<td>500</td>
</tr>
<tr>
<td>Machine or hand crushing</td>
<td></td>
</tr>
<tr>
<td>Machine sawing</td>
<td></td>
</tr>
<tr>
<td>Sieving</td>
<td></td>
</tr>
<tr>
<td>Mixing</td>
<td></td>
</tr>
</tbody>
</table>
Therefore,

\[ \text{DAC (HTO)} = \frac{0.05 \text{ Sv} / 1.83 \times 10^{-11} \text{ Sv Bq}^{-1}}{2,400 \text{ m}^3} = 1.14 \times 10^6 \text{ Bq m}^{-3} \]

\[ \text{DAC (}^{239}\text{Pu)} = \frac{0.5 \text{ Sv} / 1.14 \times 10^{-3} \text{ Sv Bq}^{-1}}{2,400 \text{ m}^3} = 1.83 \times 10^1 \text{ Bq m}^{-3} \]

\[ T = \frac{1.14 \times 10^6 \text{ Bq m}^{-3}}{1.83 \times 10^{-1} \text{ Bq m}^{-3}} = 6.23 \times 10^6 \]

\[ H = \frac{1 \times 10^7 \times 6.23 \times 10^6 \times 1}{\sqrt{13.5}} = 1.7 \times 10^{13} \]

\[ U = 1 \text{ (from Table B.1 for pipetting a solution)} \]

Entering these values in the HotSpot code or manually in the above equations, the value of H is $1.7 \times 10^{13}$. Thus, for this scenario the HotSpot Workplace preliminary screening technique recommends a Type 2 Workplace \((i.e., a \text{ chemical fume hood})\).
Appendix C: Supplemental Information Specific to Different Operational Settings

The main body of this Report provides guidance for establishing and implementing a comprehensive radiation safety program. This guidance includes the essential components, basic principles, and practices that are generally applicable to radiation safety programs across the wide spectrum of operational settings. Individual operational settings have unique characteristics that impact the radiation safety program. These characteristics include the types and quantities of radioactive materials handled, the types and levels of radiation produced by radiation generating equipment, the magnitude of the potential radiological hazards, and the operational setting.

This appendix was developed to provide radiation safety information to supplement the basic guidance in the main body of this Report. Each subsection focuses on the unique aspects of the operational radiation safety programs for different operational settings relating to facility design, facility operations, and decommissioning.

C1. Medical Facilities

The scope of a radiation safety program at a medical facility will depend on the variety of radiation sources that are used at the facility (Vetter 2005). A basic facility may be limited to diagnostic radiology equipment, while a complex facility may incorporate the use of diagnostic radiology equipment, interventional fluoroscopic equipment, therapeutic radiation producing machines, diagnostic nuclear medicine, radiopharmaceutical therapy, and brachytherapy. As the complexity of the radiation safety program increases, the level of radiation safety support and expertise increases. Basic facilities may only need an RSO, while more complex programs require additional radiation safety staff and a Radiation Safety Committee.

Many medical establishments must operate under a dual system whereby the NRC or State issues a radioactive materials license and the State issues a separate x ray/accelerator registration. Even in Agreement States, these can be administered by different departments within the same agency. Some large health care institutions operate across state boundaries, further complicating the management of the radiation safety program.
Academic medical centers may also be involved in research that involves the use of radiation sources for *in vitro* or *in vivo* procedures. Any protocols involving the research use of radiation on humans requires Institutional Review Board (IRB) review and approval. Similarly, any protocols involving the research use of radiation on animals requires Institutional Animal Care and Use Committee (IACUC) review and approval. Both committees will need radiation safety review and comment as part of their approval process.

**C.1.1 Fluoroscopy Guided Interventional (FGI) Procedures**

Fluoroscopic x-ray equipment is routinely used to guide physicians in minimally invasive procedures in interventional radiology, interventional cardiology, vascular surgery, and other medical specialties. Recommended reading for staff engaged in these services includes NCRP Report No. 168 (NCRP 2010b) and ICRP Publications 117 (ICRP 2010) and 139 (ICRP 2018). Key elements of concern with FGI procedures include:

- **Patient skin dose.** The potential for serious skin injury exists in complex FGI procedures. It is not unusual for these procedures to deliver greater than 2 Gy peak skin doses; the threshold for deterministic skin effects. As the complexity of the procedure increases, the peak skin dose can also increase; therefore, it is incumbent upon the radiation safety staff, in conjunction with medical physics staff, to develop an awareness program to reduce patient skin doses and mitigate injurious patient skin doses (more detail is available in NCRP Report No. 168 (NCRP 2010b)).

- **Staff occupational dose.** Scatter radiation from the patient is the primary source of occupational radiation dose in the fluoroscopy environment. Any effort to reduce patient skin dose will also reduce occupational staff dose. The collaboration between radiation safety and medical physics staff is beneficial to minimizing occupational exposure – the combination of time, distance, and shielding (ceiling mounted or rolling) with the optimization of fluoroscopy equipment operation is a powerful combination to reduce radiation levels in the procedure room.

FGI operators and ancillary staff tend to have the highest dosimeter readings in the medical facility. However, these results are not representative of their actual effective dose (ED) or effective dose equivalent (EDE) if a single dosimeter is worn outside the protective apron, because so much of their body is shielded by protective aprons. A correction factor is needed to calculate the ED and EDE to evaluate the risk to FGI staff on the same spectrum as the risk to
Nuclear Medicine staff who do not wear protective aprons. NCRP Report No. 122 (NCRP 1995) addresses the derivation of a correction factor for fluoroscopy users.

Fluoroscopy users typically wear a whole body dosimeter at the collar, outside the protective apron. In this location, the dosimeter can be used as an analog for the thyroid dose (if a thyroid shield is not employed) and the lens of the eye dose. The NCRP (2018) has recommended that the absorbed dose to the lens of the eye should not exceed 50 mGy y\(^{-1}\). The same procedures used to reduce patient and staff radiation dose are effective in reducing dose to the lens of the eye. If necessary, the operator can use leaded protective eyewear. However, the effectiveness of the eyewear is only about a factor of two to three due to the incident angle of the scatter radiation, which does not all enter the head and eyes through the eyewear.

C.1.2 Positron Emission Tomography (PET)

Historically, diagnostic Nuclear Medicine has been predominately performed using \(^{99m}\)Tc labelled pharmaceuticals. Technitium-99m has relatively low gamma ray energy (140 keV) and is relatively easy to attenuate using thin lead shielding and lead syringe shields. The addition of PET radionuclides (e.g., \(^{18}\)F, \(^{11}\)C, and \(^{68}\)Ga) to the diagnostic Nuclear Medicine practice has, with the associated 511 keV gamma ray energy, clearly impacted shielding requirements; for example, lead’s half value layer is 4.95 mm for \(^{18}\)F and is 0.234 mm for \(^{99m}\)Tc (Smith and Stabin 2012).

Fluorine-18 radiopharmaceuticals have a shorter physical half-life than \(^{99m}\)Tc radiopharmaceuticals (~2 h vs. 6 h), but, because of the higher energy, the whole body and extremity radiation doses to the nuclear medicine technologists are approximately three times greater when handling PET pharmaceuticals (Sudbrock et al. 2011; Carnicer et al. 2011). Methods to reduce the occupational exposure include using tungsten instead of lead for syringe and vial shielding, as well as the development of automated delivery systems to minimize handling. Staff rotation is another means of controlling dose to specific individuals. For example, some medical facilities with dedicated PET departments frequently rotate the technologists between PET and Tc-99m work so that no one person is exclusively exposed to the higher PET dose environment.
Radioactive Material Licensing and Use

Radioactive material regulations (NRC 2002) are relatively prescriptive regarding licensing of the standard procedures to be performed. In addition, the authorization of the Authorized User – the physician permitted to administer radioactive materials to the patient – is very prescriptive. Interestingly, the greatest flexibility is provided in the area of emerging technologies where regulations have not been promulgated and licensing is through guidance documents which allow some flexibility to the licensee and reviewer.

Following is a summary of the standard medical procedures covered in the regulations, or in the case of emerging technologies, guidance documents:

- **Nuclear Medicine**
  - Use of unsealed byproduct material for uptake, dilution, and excretion studies,
  - Use of unsealed byproduct material for imaging and localization studies,
  - Use of unsealed byproduct material for which a written directive is required.

- **Radiation Oncology**
  - Use of sealed sources for manual brachytherapy,
  - Use of a sealed source in a remote afterloader unit, teletherapy unit, or gamma stereotactic radiosurgery unit.

- **Emerging Technologies**
  - Other medical uses of byproduct material or radiation from byproduct material
  - Low Activity Radioactive Seeds Used for Localization of Non-Palpable Lesions and Lymph Nodes Licensing Guidance,
  - TheraSphere and SIRSpheres Yttrium-90 Microspheres.

Radiopharmaceutical Therapy

One of the most challenging aspects of medical health physics is the support of the radiopharmaceutical therapy program (NCRP 2006). The use of radiopharmaceuticals to treat
disease involves the administration of 10 GBq or more of the material to the patient. Following are a few of the issues that warrant consideration.

**New therapy.** All of the radiation safety issues surrounding radiopharmaceutical therapy procedures require knowledge of the decay scheme, metabolism, and excretion of the administered material. However, in the case of a new therapy in research trials, the availability of this information is problematic to obtain and may require using information from similar studies to develop a conservative radiation safety program.

**Outpatient vs. inpatient.** The decision as to whether a therapy procedure will be performed as an outpatient or inpatient procedure is based on the administered activity, radionuclide, and the patient circumstances. If the licensee can demonstrate that a caregiver or member of the public will not receive a radiation dose greater than 5 mSv, the therapy may be performed as an outpatient procedure. The patient-specific calculations used to demonstrate the estimated dose to others may include temporary restrictions such as separate sleeping and bathroom facilities, to ensure the dose to others is less than 5 mSv.

**Patient instructions.** For both inpatient and outpatient procedures, the patient receives instructions on how to minimize the potential radiation dose to others. These instructions, covering time lengths up to a few days, include maintaining at least 1 m from others, limiting time with pregnant woman and children, double flushing the toilet, and others.

**Room preparation, decontamination, and release.** The radiopharmaceutical metabolism and excretion pathways determined during the therapy procedure review and approval process will drive the need for room preparation to limit the contamination of surfaces and items. In the classic case of radioiodine therapy, with excretion through urine, saliva and sweat, it is imperative to cover the surfaces and items that may be touched by the patient, including wrapping the toilet fixture and papering the surrounding floor and walls. After the patient is released from the hospital, the room must be surveyed for radioactive contamination and decontaminated as needed. Prior to releasing the room for cleaning by housekeeping staff, a release survey must be completed and documented to confirm the measurements are below the contamination action levels.

**Shielded room.** For inpatient procedures, it is likely that the room will need to be shielded, or adjacent rooms vacated. The shielding calculations should be based on the highest
activity likely to be administered, the longest stay time for a patient, and the expected caseload for a year.

Staff occupational dose. The patient and the administration delivery system are the primary sources of occupational radiation dose in the radiotherapy environment. Time, distance, shielding, and contamination control – standard radiation safety practice – are employed to reduce staff doses. Special attention must be paid to the delivery system during infusion procedures – the administration vial and delivery tubing can expose staff to significant radiation fields. Since radiotherapy infusions can take up to 30 minutes, it is essential that the delivery system be shielded to reduce unnecessary radiation exposure to staff members (and the patient).

C.1.5 Medical Events

A medical event (10 CFR 35.3045 (NRC 2002)) occurs when the prescribed radiation dose is not properly delivered to the patient and certain dose and practice criteria thresholds are crossed. In some but not all regulations, there is an exception for cases where patient intervention caused the deviation. A medical event does not necessarily include patient injury; it may be an indicator of a potential quality assurance issue at the facility. An abridged listing of the requirements for a medical event includes:

- A dose that exceeds 0.05 Sv effective dose equivalent, 0.5 Sv to an organ or tissue, or 0.5 Sv shallow dose equivalent to the skin from any of the following:
  - An administration of an incorrect radioactive drug containing byproduct material or an incorrect radionuclide for a brachytherapy procedure;
  - An administration of a radioactive drug containing byproduct material by an incorrect route of administration;
  - An administration of a dose or dosage to an incorrect individual or human research subject;
  - An administration of a dose or dosage delivered by an incorrect mode of treatment;
  - A leaking sealed source.
A medical event must be reported to the regulators within 24 h of discovery and a written report is required to be submitted. The report must include a discussion of what happened, where and why it happened, how is the patient likely to be impacted, and what corrective actions have been implemented, as well as a few additional requirements. If proposed corrective actions are known, these may also be included.

C.1.6 Institutional Review Board

Radiation sources are commonly used for research purposes at large academic medical centers. Department of Health and Human Services regulations (HHS 2018), as well as the applicable federal and state radiation protection regulations, apply to these institutions. The research protocol must be reviewed and approved by an Institutional Review Board (IRB) prior to the initiation of the research protocol. Typically, an ancillary IRB committee reviews the radiation safety aspects of the research protocol, focusing on the following elements:

Justification. The radiation use is justified and no other methods – possibly nonradiation involving modalities – are available to obtain the same diagnostic information. The American College of Radiology’s appropriateness guidelines (ACR 2020) provide a good tool for these evaluations.

Optimization. The minimum amount of radiation will be used to collect the diagnostic information required for the study. This review includes an examination of the frequency and quantity of radiation used throughout the study.

Informed Consent. The language in the informed consent clearly explains the use of radiation or radioactivity in the study. For example, the consent should state that a CT scan involves the use of x rays to study structures in the body. It is also important to ensure that the informed consent adequately explains the potential risk involved from the research use of ionizing radiation.

Licensing. The proposed radiation use is authorized by the facility’s radioactive materials license and an authorized user is involved in the research protocol.
C2. University/Research Facilities

Universities and research facilities may have unique operations requiring complex planning. These facilities may include the use of cyclotrons or linear accelerators, the production of investigational radiopharmaceuticals, animal research, human subjects research, hot cell or glovebox operations, ultrafast lasers, and research reactors. Due to the breadth of their operations, they may require licensing or registration by multiple agencies, as well as enhanced emergency plans and a decommissioning funding plan.

The potential for unique or high-level radioactive sources and novel uses of radiation in an academic environment is high, and cannot necessarily be anticipated in advance, nor is it practical to attempt to enumerate the possible sources or uses, as the very nature of research is innovation. Thus, radiation safety programs must be robust, flexible, and vigilant to identify and provide controls for new potential uses.

C.2.1 Organization and Administration of the Radiation Safety Program

The more complex the operations, the greater the need for professional level health physicists and medical physicists. Since the radiation safety program generally serves multiple and diverse units, the reporting structure should clearly reflect the lines of communication, and the chancellor’s office or senior executive official should unambiguously delegate the authority to halt unsafe operations.

To ensure a coherent and comprehensive safety program, the radiation safety program should be part of a larger overall safety program or department, independent from operational units, and integrated with the emergency response command system. In addition, radiation safety considerations should be built into auxiliary departments (e.g., planning, facilities and purchasing), such that when new facilities are being planned, existing facilities are being reconfigured, or new equipment is being purchased (or leased, rented or borrowed), there is clear direction as to when the radiation safety staff must be consulted.

The radiation safety committee will require expertise from all principal areas of research, and should include the ability to add members or use consultants as necessary to address novel uses of radioactive material or RGDs. Where human subjects research is contemplated, it is critical to include physicians practicing in the given specialty. The radiation safety committee
should be integrated into the overall research protection program, including a defined role for interacting with other safety committees, such as animal use committees, institutional review boards, radioactive drug research committees, institutional biosafety committees, and scientific review committees.

C.2.2 Facility and System Design Considerations

Certain challenges may be unique to university and research operations. For example, while open floor-plan laboratory space is often conducive to collaborative research, it may also create challenges for radiation safety. In shared spaces, where radioactive materials may be used by one or more researchers, and not by others who share the space, consideration should be given to:

- security of radioactive material. Simply controlling access to the laboratory to prevent access by the general public does not ensure unauthorized access to the radioactive material if the space is shared with researchers not authorized for radioactive material use.

- control of storage areas, including in shared refrigerators, freezers, or waste storage compartments. Similar to the control of the whole of the laboratory space, special additional security may be necessary in shared storage areas, such as built-in, lockable containers within a freezer, or other storage area to segregate radioactive materials or samples.

- control of potentially contaminated work benches, fume hoods, gloveboxes, and radioactive wastes, which may require more frequent surveys and decontamination, or additional security measures to maintain access control to these areas.

- ensuring office space, meeting space, lunchrooms, and other nonresearch space is clearly separated from and outside of the radioactive material use space.

Where RGDs are used, consideration should be given to:

- securing the RGDs from unauthorized use or removal;

- periodically verifying use (workload), including technical parameters for exposures, and laboratory occupancy of nonradiation workers to ensure the configuration of the
primary beam, and shielding for it and the scatter radiation are protective for all users in and adjacent to the space;

- communicating the need to report new uses, new RGDs, development of new RGDs, or plans for substantial changes to existing RGDs to the radiation safety staff.

Another important consideration for universities is the fact that they will generally have on-site residences. Consideration must be given to the placement of new laboratory, research, or radioactive waste facilities relative to on-site housing, including placement of fume hood, glovebox or hot cell exhaust relative to air intakes on on-site housing. For example, the NRC constraint rule in 10 CFR 20.1101 (NRC 2020a) was designed to comport with the National Emission Standards for Hazardous Air Pollutants and applies to releases at the site boundary, unless there are residential quarters on site, in which case it applies at the threshold of the residence.

Many regulatory agencies apply DOT regulations even on quasi-private roads within a site, if those roads are also accessible by the public. For universities and large research complexes, internal roads may be considered public roads for purposes of regulation by NRC or the Agreement States, and DOT regulations (DOT 2011) may apply (e.g., when transporting radioactive waste from a laboratory area to a common radioactive waste storage facility at the campus or complex). In the planning stages, minimizing the need for motorized vehicles to travel on publicly-accessible roads, or including limited access roads for transportation of radioactive material within the site can reduce the regulatory burden of on-site transport of radioactive materials.

C.2.3 Work Planning, Audits, and Training

Work planning is critical, since by its nature, research involves nonroutine operations. In contrast to well-established uses of radioactive material and RGDs, such as for industrial radiography, research uses may be quite novel and require early integration of radiation safety staff in the planning stages, as well as mock-ups or dry-runs of the experimental setup to identify critical safety gaps, prior to introducing the radioactive material or energizing the RGD.

In contrast with standard, routine operations, audit requirements for novel uses may need to be more intense in the early stages of a project, and could taper off as the safety of the
operations is established. Defined benchmarks in the radiation use authorizations can help establish when oversight may be reduced, such as when area radiation monitoring demonstrates that dose in unrestricted areas is well below the benchmark, over a pre-defined period.

Particularly at universities, the population of potential trainees will be large and the audience will be diverse in both previous education and training, and in the potential for exposure. There will be a variety of training opportunities, and it may be more efficient to incorporate radiation safety training into other established training programs. For example, in a medical school setting, radiation safety training may be incorporated into medical school seminars, grand rounds, or the graduate medical education programs. Radiation safety training for nursing staff may be tailored to address specific radiation safety topics appropriate for the nursing units (e.g., oncology units). Some units may only require basic radiation safety awareness training. Academic staff training may be individualized, and may include student researchers. In general, the development of multiple training products, including computer-based training, in-person (or remote) individual or small-group training for specific uses, and broad-based radiation safety awareness training for ancillary services (e.g., housekeeping and security staff) will provide flexibility to meet regulatory requirements and address the needs of the diverse research and ancillary populations.

At universities, there is a higher potential for minors to work with or around radiation (e.g., in physics or chemistry labs) than in other types of facilities. Minors are subject to different radiation dose limits than adults, and the radiation use authorization or work planning process should include an ability to identify when minors may be exposed, and whether university policy requires some type of parental notification or acknowledgement under these circumstances.

C.2.4 Emergency Planning

Planning for emergencies involving radioactive material should be incorporated into a site’s overall emergency response planning, and include how the technical experts in radiation safety will be integrated into the site’s incident command operations for large events. Many universities and research complexes will have their own on-site law enforcement, security, and fire response agencies, and training for these entities should be incorporated in the overall emergency response training structure to ensure their understanding of the potential hazards, as
well as the precedence emergency medical measures take over decontamination. Pre-planned messaging is critical, especially in the university environment where media interest in a radiological incident may be heightened.

C.2.5 Decommissioning

While many universities and research facilities may operate under a “broad scope” license, such authorization does not grant unlimited autonomy, and it is important to clearly delineate in radioactive material license applications the operations that are considered routine for the purposes of releasing areas from radiological controls. The routine release of small laboratories for continued laboratory use will usually be within a broad scope licensee’s authorization, however a major reconfiguration of laboratory space, which includes removal of exhaust systems or plumbing may not be, and may require express authorization from the regulatory agency; similarly, the total deconstruction of a building may require the submission of a full decommissioning plan and its approval prior to demolition. This is especially true where long-lived radioactive materials have been used, and where there is a potential for activation of building materials due to the use of large accelerators.

C3. Biotechnology Facilities

Biotechnology fabrication facilities are formed to develop medicines that are used to treat serious, life-threatening diseases, the effects caused by cancer treatment, as well as other medical conditions. Biotech workers discover and develop compounds and perform pre-clinical trials both in vivo (in animals, not humans) and in vitro. The clinical trials involving humans are performed at other facilities and are outside the scope of this Report.

The biotech industry uses radioactive material for labeling and as tracers in chemical and biological studies. Large quantities (GBq levels) of $^{3}\text{H}$, $^{14}\text{C}$, and $^{32}\text{P}$ are used in radiochemical synthesis facilities. Iodination is performed with both $^{125}\text{I}$ and $^{131}\text{I}$ using MBq quantities, with up to 37 GBq on site.

Biotech facilities generally include a pre-clinical imaging facility authorized for isotopes used in PET/CT (e.g., $^{18}\text{F}$, $^{64}\text{Cu}$, $^{68}\text{Ga}$, $^{89}\text{Zr}$, $^{124}\text{I}$) and isotopes used in SPECT/CT (e.g., $^{67}\text{Ga}$, $^{99m}\text{Tc}$, $^{123}\text{I}$, $^{177}\text{Lu}$, $^{201}\text{Th}$). The activities used are typically in the 3.7 GBq range, except the short-
lived $^{18}\text{F}$, which requires up to 200 GBq. A $^{57}\text{Co}$ flood source and dose calibrator for calibration of PET cameras are usually included as part of the imaging facility. Most facilities will also have gamma irradiators with up to 100 TBq of $^{137}\text{Cs}$.

In addition to radioactive materials, many different types of RGDs are used, as shown in Table C.3.1.

C.3.1 Principal Radiological Hazards

The radiological hazards at biotech facilities are similar to those found in research settings, except the amounts of radioactivity may be higher.

C.3.2 Facility Design Requirements

The majority of biotech work is conducted in standard labs. However, special use facilities require additional facility design considerations. For instance, fume hoods used for volatile materials such as iodinations, should have dedicated duct work, filter banks that include HEPA and carbon filters, and appropriate radiation shielding. The pre-clinical imaging facility, where $in \, vivo$ research is conducted with animals injected with PET radionuclides, should have sufficient lead shielding integrated into the design. Shielding is needed to protect workers locally during work with injected animals and also to protect nonradiological workers or members of the public who may have access to the adjacent hallways during PET scanning and fluoroscopy.

The radiochemical synthesis facility should have the following design elements:

- fire suppression in all rooms.
- fume hoods and associated duct work constructed of stainless steel.
- a strobic fan on the exhaust stack to increase the effective stack height
- an airlock to control primary access to all rooms.
- stainless steel counter tops and fume hoods in labeling rooms.
- rooms with a negative pressure relative to the hallways and nonlab areas.
### Table C.3.1—Typical operating parameters for Research RGDs.

<table>
<thead>
<tr>
<th>Research RGD Type</th>
<th>Radiation</th>
<th>kVp (max)</th>
<th>mA (max)</th>
</tr>
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<tbody>
<tr>
<td>Electron microscopes</td>
<td>x ray</td>
<td>15 – 30</td>
<td>0.004 – 0.1</td>
</tr>
<tr>
<td>Industrial cabinet x ray</td>
<td>x ray</td>
<td>80 – 160</td>
<td>0.3 – 1</td>
</tr>
<tr>
<td>Veterinary cabinet x ray</td>
<td>x ray</td>
<td>40 - 100</td>
<td>0.1 – 30</td>
</tr>
<tr>
<td>Veterinary bone densitometers</td>
<td>x ray</td>
<td>80 - 160</td>
<td>0.5 – 10</td>
</tr>
<tr>
<td>Veterinary C-arm fluoroscope</td>
<td>x ray</td>
<td>160</td>
<td>20 – 40</td>
</tr>
<tr>
<td>Veterinary micro-CT</td>
<td>x ray</td>
<td>80 - 160</td>
<td>0.1 – 0.6</td>
</tr>
<tr>
<td>Veterinary radiographic</td>
<td>x ray</td>
<td>60 - 350</td>
<td>45 – 400</td>
</tr>
<tr>
<td>X-ray diffraction</td>
<td>x ray</td>
<td>40 - 45</td>
<td>40 – 50</td>
</tr>
<tr>
<td>X-ray fluorescence</td>
<td>x ray</td>
<td>40</td>
<td>20 – 100</td>
</tr>
</tbody>
</table>
• in special use areas utilizing powders, a glovebox and powders weighing fume hood, with in-line HEPA filter on the exhaust so that 100 % of the air is exhausted to the outside of the building.

• a decontamination shower equipped with a drain collection tank.

C.3.3 Radioactive Waste handling and Disposal

Waste handling and disposal methods used at Biotechnology facilities are similar to those employed at a large laboratory. Separating waste by form and radionuclides, compacting waste, and holding short-lived isotopes for decay-in-storage are typical waste handling and minimization methods used at these facilities.

There are exceptions, however. For instance, large amounts of liquid waste could be disposed directly from the experimental area using dedicated stainless steel piping to a waste collection system. Dry waste could be collected into dedicated drums in a room equipped with fire suppression and room access controlled by mechanical and administrative means. To assure no spillage occurs, the collection system could have liquid level detectors that alarm when a drum reaches 90 % capacity and a notification system to tell radiation safety staff when drums are full.

C.4. National Laboratories

The DOE has many different missions which they manage through different “Offices” (e.g., Science, Energy Efficiency and Renewable Energy, Environmental Management, Fossil Energy, Nuclear Energy), as well as the National Nuclear Security Administration (NNSA). As shown in Table C.4.1, each of these Offices and the NNSA manage one or more National Laboratories, each with individual missions that, collectively, support the goals of the Office.

C.4.1 Radioactive Material and RGD Use

Some of the National Laboratories have narrower missions than others, and consequently have radiological operations that are more succinctly defined. For example, the Thomas Jefferson National Accelerator Facility in Newport News, VA, the SLAC National Accelerator Laboratory in Stanford, CA, and the Fermi National Accelerator Laboratory in Batavia, IL are all
## Table C.4.1—Management\(^a\) of the DOE national laboratories.

<table>
<thead>
<tr>
<th>Name</th>
<th>Location</th>
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<tr>
<td><strong>Office of Science</strong></td>
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<td>Monsanto (ca. 1945–1947)[2]</td>
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<td>Union Carbide (1947–1984)[2]</td>
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<td>UT–Battelle (since April 2000)</td>
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<td>Battelle Memorial Institute (since 1965)</td>
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<td>Fermi Research Alliance (since 2007)</td>
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**National Nuclear Security Administration**

<table>
<thead>
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<th>Name</th>
<th>Location</th>
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<td>Los Alamos National Laboratory</td>
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<td>University of California (1943–2007)</td>
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**Office of Energy Efficiency and Renewable Energy**

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**Office of Environmental Management**

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**Office of Nuclear Energy**

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\(^a\) Note that the Operating Contractor is subject to change. Source:

large accelerator facilities, whose radiological concerns are well described in section B2.2. Other National Laboratories (e.g., Brookhaven in NY, Oak Ridge in TN; Los Alamos in NM; Lawrence Livermore and Lawrence Berkeley in CA) are multi-program science and technology laboratories with broad uses of radioactive material and radiation generating devices. The radioactive materials present at a given multi-program National Laboratory may range from tritium ($^3$H) to the transuranics (e.g., plutonium, curium, americium) and with programs that utilize from a few Bq to tens of PBq, while the RGDs may range from those that produce low-levels of radiation incidentally, to accelerators that produce very high radiation fields.

C.4.2 Administration of the Radiation Safety Program

The radiation safety program must function within the parameters established by federal rules (e.g., DOE 2011a, 2011d); DOE Orders (e.g., DOE 2011b, 2011c, 2011e, 2011f, 1999); policies of the NNSA or Office that manages the site; policies of the site Operator; and sometimes, requirements of the State in which the site is located.

An important situation to understand is whether more than one DOE Office or site operator is managing operations at a given site; in such cases, there may be more than one radiation safety program on site, run by separate Radiation Safety organizations. In addition, the DOE, NRC, and many state governments have different views of their respective jurisdictional boundaries, so administrative care must be exercised when conducting off-site DOE operations (e.g., when using sealed sources to train local first responders or to test radiation detection systems at airports or other ports of entry).

If there is more than one radiation safety program on site, it is important to understand the respective roles, responsibilities, and authorities, how the radiation safety programs interface, and the conflict resolution process.

C.4.3 Principal Radiological Hazards and Facility Design Considerations

Both internal and external radiation hazards are present at many of the National Laboratories and consequently, extensive radiological controls are integrated into facility and equipment designs, as needed to protect the workforce, public, and environment. For example, hot cells are used for handling radioactive materials with very high external dose rates; glovebox
lines are used for handling dispersible transuranics such as plutonium and americium; and
extensive shielding is used to protect against high levels of radiation from RGDs. There are also
many other less hazardous radiological operations that are safely conducted in fume hoods and
on benchtops. In addition to research and development operations, facility and equipment
maintenance sometimes requires work on contaminated systems, in contaminated areas, or on
activated components.

The operations conducted at a National Lab may be relatively short lived (e.g., days to
months) or long-standing (years to decades); even with long-standing operations, changes or
upgrades to the equipment and facility may result in new or changing radiological hazards.

C.4.4 Radiological Monitoring Equipment

There is nothing particularly unique about the radiological equipment used at National
Laboratories. However, the security requirements in some facilities can prevent the use of off-
the-shelf equipment (e.g., those with telemetry capabilities), so it is important to understand the
security limitations before purchasing radiological equipment for use in areas where classified
work is conducted.

C.4.5 Internal and External Dosimetry

Both internal and external dosimetry are used extensively at National Laboratories and
both must be accredited by the Department of Energy Laboratory Accreditation Program
(DOELAP). Some sites use vendor-provided services, while others have in-house services;
however, regardless of who provides the dosimeter service, it is the site’s responsibility to
maintain their DOELAP accreditation.

Internal dosimetry for transuranic radionuclides is considerably more difficult than for
the more common radionuclides (e.g., $^3$H, $^{14}$C, $^{60}$Co, $^{90}$Sr/$^{90}$Y, $^{125}$I, $^{137}$Cs) as the transuranic
radionuclides must be chemically separated from urine or fecal samples, electroplated, and then
counted in an alpha spectrometer for several days. For routine bioassay samples (where the
postulated date of intake may have been 6 months earlier), it is not always possible to detect an
intake equivalent to the screening threshold of 1 mSv. Consequently, the bioassay program must
be carefully designed to minimize the likelihood of an undetected intake. Internal dose
assessments following a suspected uptake benefit from prompt and ongoing bioassay sampling, but the data analysis often takes months to perform, especially in cases where DTPA is administered, as there is about a 100 d wait after DPTA administration to obtain samples unaffected by the DPTA and upon which the mathematical excretion function will be determined. With or without a suspected transuranic intake, it is important to have an experienced internal dosimetrist available.

The challenges for external dosimetry programs at National Laboratories are neutron dosimetry and being able to detect a wide variety radiation types/energy using a single whole body dosimeter. It is important to have an experienced external dosimetrist available to ensure the chosen dosimeter remains well matched to the (potentially) changing radiological environment, and to interpret raw data from the dosimetry systems.

C.4.6 Radiological training

Most National Laboratories have mature radiation safety training programs that meet the regulatory requirements. With web-based courses now largely available, delivery of training to larger and sometimes mobile populations has become relatively efficient.

C.4.7 Effluent Monitoring and Radioactive Waste Handling

National Laboratories often conduct effluent monitoring of both air and water, and have extensive environmental monitoring programs. The environmental program is typically a large operation conducted by groups outside the radiation safety program.

As shown in Table C.4.1, most of the National Laboratories were established more than 65 y ago. Aged facilities can present unique waste-management challenges, as the types of operations conducted in a given facility may have changed many times over the course of history. However, once waste has been characterized and packaged to meet the waste acceptance criteria of the disposal site, disposal is reasonably accomplished as DOE has waste disposal sites for both low-level and high-level radioactive waste. Waste programs are also typically managed by groups outside the radiation safety program.
C5. Commercial Nuclear Power Reactors

There are two types of commercial nuclear power reactors in the United States:

Pressurized Water Reactors (PWR) and Boiling Water Reactors (BWR). The radiation safety program established for each are similar, but the actual radiological conditions can be quite different. The radioactivity in a PWR is on the primary side of the plants (shown in pink in Figure C.5.1). The heat generated in the reactor core by nuclear fission is transferred to the reactor coolant, which in turn is transferred to make steam in the steam generator. The steam generated is then used to generate electricity in the turbine generator. The secondary side of the plant (shown in blue in Figure C.5.1) will not have any radiological impact unless there is leakage from the reactor coolant into the steam generator.

In a BWR, the radioactivity is on both the inside and outside of the primary containment structure. As shown in Figure C.5.2, the heat generated from fission in the reactor core transfers heat to the reactor coolant and exits the reactor vessel as steam. This steam is then transferred to the turbine and generator to make electricity. As a result, the secondary side of the plant (i.e., the portion that is outside the containment structure) needs radiological controls. The radioactive material on the secondary side of a BWR is less than on the primary side.

The sources of radiation at a nuclear power plant (inclusive of PWRs and BWRs) include the following:

- fission products, such as cesium and krypton,
- coolant activation products such as $^{16}$N,
- activation of corrosion and wear products such as $^{58}$Co and $^{60}$Co, and
- uranium contamination of the in-core portion of the primary heat-transport system, (sometimes referred to as “tramp uranium”).

The radionuclides that tend to contribute the most to personnel dose are $^{58}$Co and $^{60}$Co.
**Fig. C.5.1.** Typical PWR (Wikipedia 2020a).
Fig. C.5.2. Typical BWR (Wikipedia 2020b).
C.5.1 Radiation Safety Program

In the U.S., nuclear power plants are regulated by the NRC and the radiation safety programs are required to meet 10 CFR 20 (NRC 2020a); they should also meet the requirements developed by the Institute of Nuclear Power Operations (INPO). INPO was formed after the 1979 accident at Three Mile Island and has the sole purpose of driving excellence in all aspects of Nuclear Power Operations. Even though rules, standards and Regulatory Guides are periodically updated, licensees will be held to what was committed to during the licensing process and may therefore be bound to implementing earlier versions of a given document.

The Radiation Protection Manager (RPM) is typically responsible to implement the radiation safety program at a nuclear power plant and typically reports to the Plant Manager; mid-level managers and supervisors typically report to the RPM and are responsible for the ALARA, dosimetry, and instrument programs, and supervision of the radiation protection technicians.

Under current regulatory standards the RPM is required to meet the knowledge and experience requirements of Regulatory Guide 1.8 (NRC 2019, or predecessor document), which endorses, with exceptions, ANSI standards for training and qualification of nuclear power plant personnel. Each nuclear power plant will commit to a specific standard for qualification of the plant staff, such as ANSI/ANS 3.1 (ANSI/ANS 2014), or an earlier document depending upon when the facility was licensed. In some cases, the RPM may have a degree, but is not required to have a degree. As stated by the NRC, a bachelor’s degree is not considered germane to fulfilling the role of the RPM at a nuclear power plant, as the attributes of a good RPM are considered to be gained almost exclusively by on-the-job, practical and supervisory experience (NRC 1994c).

At nuclear power plants, the radiation safety committee function is typically performed by the Station ALARA Committee, which is chaired by the Site Vice President or the Plant Manager. The committee’s primary function is related to managing and maintaining station exposures ALARA. In addition, the committee may review and approve high risk radiological.

Dose management is typically performed by the Radiation Protection Department. INPO has established an administrative guideline of 20 mSv y⁻¹ for all nuclear power plants and the process to extend individual dose administrative guidelines requires various senior management
approvals, with higher levels of management being required to approve higher administrative guidelines. For example, to approve an individual’s annual dose guideline above 20 mSv y\(^{-1}\) requires the RPM approval, to approve above 25 mSv y\(^{-1}\) requires plant manager approval, and to approve above 30 mSv y\(^{-1}\) requires the site vice president approval. Dose extensions above 30 mSv y\(^{-1}\) rarely occur.

**C.5.2 Facility Design**

Nuclear power plant design includes three barriers to the release of fission products during operation and emergencies. The first barrier is the fuel pellet and fuel rods, which are designed and constructed to contain the radioactivity. The second barrier is the primary system, which will contain in the reactor coolant any radioactivity released from the fuel and rod. The third barrier is the primary containment, which is designed to contain any radioactivity that escapes the reactor coolant system. The containment structures are made of reinforced concrete and, in addition to providing a barrier to the release of radioactive material, are an effective shield to reduce the personnel dose.

At a BWR, radioactive material is carried into the secondary side of the plant. At both a BWR and PWR, support systems (e.g., clean-up and purification) will also transport radioactive material outside of containment into shielded rooms constructed of reinforced concrete. In addition to shielded rooms, nuclear power plants have developed robust temporary shielding control programs using lead and other materials (e.g., tungsten).

The layout of most nuclear power plants includes an owner-controlled area containing the restricted area and the Radiologically Controlled Area (RCA). Access to the restricted area is controlled with robust fences and a security force; the main RCA is inside the restricted area.

**C.5.3 Radioactive Material and Controls**

Radioactive material at nuclear power plants is largely the result of the fission process. These materials include activated reactor components, used nuclear fuel, and control rod blades. However, the largest volume of radioactive materials generated is a result of operations, which includes resins used to filter and treat reactor water and feed water systems, and waste generated by maintenance activities.
The highly radioactive used fuel and in-core components are stored in specially designed pools. These pools include shielding to reduce personnel exposure, and systems to maintain the water temperature and chemistry to ensure the integrity of these components. Used fuel may also be stored in special designed canisters on the facility, called Independent Spent Fuel Storage Installations. Other radioactive materials such as resins and dry active waste are stored in vaults and other secure areas.

Personnel monitoring devices are installed at the exits to both the restricted area and the RCA. Personnel contamination monitors sensitive to beta radiation and portal monitors sensitive to gamma rays are located at the exit from the RCA; portal monitors are also located at the exit to the restricted area. Materials and personal items are generally surveyed with automated Tool and Equipment monitors.

C.5.4 Work Planning

Work planning follows a rigorous process that may start as early as 26 weeks before the work is to be initiated. Most nuclear power plants have a specific department, the work management department, to develop work schedules, coordinate the reviews and oversee the execution of the process. Radiation protection departments will typically have a dedicated planner to interface with the work management department. Radiation protection involvement will start at approximately 13 weeks before execution. During this time frame the radiological controls and process will be factored into ALARA planning documents, procedures, work orders, and the associated radiation work permits.

The radiological controls are based upon the radiological risk associated with the work. In conjunction with INPO, the industry has developed common guidance to classify the radiological risk to ensure consistent implementation and drive high standards for radiological protection in the industry. High radiological risk work activities are required to be approved by the RPM at a minimum, and in many cases will be approved by the Station ALARA Committee/radiation safety committee.
C.5.5 Training

General Employee training is conducted for all personnel who enter the facility. The training content has been standardized by the industry and the National Academy of Nuclear Training. There are typically three components to the training: facility access training, RCA access training, and practical factors training. Facility access training is provided to any permanent employee working at the site who does not require access into the RCA. The training consists of a basic overview of radiation protection principles. Anyone who accesses the RCA is required to take the radiation worker training, which may include practical factor training dependent on the individual’s experience level.

Radiation protection technician training adheres to the requirements of ANSI 3.1 (ANSI/ANS 2014) or its predecessor documents, depending on when the facility was licensed. In addition, INPO has developed a set of guidelines and a process to accredit the radiation protection technician training program. Each training program will have an initial review and a periodic review to obtain and maintain this accreditation. The time frame required for a technician to be fully trained and qualified is 2 to 3 y depending on the specific ANSI standard the site committed to implement in the licensing basis documents.

C.5.6 Personnel Monitoring

Personnel are typically monitored with both a passive and active dosimeter. The passive dosimeter is issued to everyone that is expected to receive external dose in the course of their work or enter the RCA. The active dosimeter is issued at the main access points to the RCA. The individual will interface with the electronic access control system which performs checks for training, authorization to enter the RCA, and ensures the individual is listed on an RWP. The system then sets the dose rate and dose rate alarms on the active dosimeter to the appropriate values for the work to be conducted. Upon exit, the individuals once again interface with the access control system where dose for the entry will be recorded and the system will check for any dose and dose rate alarms. If either alarm is received, the individual is locked out of the RCA until an investigation can be conducted.
C.5.7 Posting and Labeling

The US nuclear industry has implemented standard postings and controls for various radiological hazards (INPO 2017). (These values are provided in traditional units because INPO specifies traditional units. These areas include:

- Very High Radiation Area (> 500 rads h\(^{-1}\) at 1 m from any source or surface)
- Locked High Radiation areas (≥ 800 mR h\(^{-1}\) at 30 cm)
- High Radiation Areas (≥ 80 mR h\(^{-1}\) and < 800 mR h\(^{-1}\) at 30 cm)
- Radiation areas (≥ 4 mR h\(^{-1}\) and < 80 mR h\(^{-1}\) at 30 cm)
- Contaminated areas (Beta-gamma—Removable: 1,000 dpm/100 cm\(^2\), Total (fixed plus removable): 5,000 dpm/100 cm\(^2\); Alpha—Removable: 20 dpm/100 cm\(^2\), Total (fixed plus removable): 300 dpm/100 cm\(^2\)) and
- Airborne radioactivity areas (>0.30 DAC).

The standard controls specify when to post an area and the posting color and verbiage, to ensure posting consistency across all U.S. nuclear power plant programs.

A typical U.S. nuclear power plant will have one to two posted “Grave Danger” areas. At a BWR, the Drywell at 100 % power is generally posted as Grave Danger’ area. At both a PWR and a BWR, these areas may include the rooms that are used to store the detectors that traverse the reactor core at power to calibrate the installed instruments for reactor power. Access to these rooms is controlled by locking the rooms in combination with administrative controls such as requiring the approval of the Site Vice President to enter the area.

The areas controlled as a Locked High Radiation Area (LHRA) vary between BWRs and PWRs. At a BWR, the areas that carry the radioactive steam are normally controlled as a LHRA due to the carryover of \(^{16}\)N. At both BWRs and PWRs, the areas that store reactor coolant resins, and other rooms that contain reactor coolant clean-up systems, are generally controlled as a LHRA. The total number of individual rooms controlled as an LHRA typically range from 20 to 100.
C.5.8 ALARA

The ALARA function is typically performed by staff dedicated to this function who review the work and, based upon the expected total dose for the activity, implement various planning documents and radiological controls. Work that is classified as medium or high radiological risk will typically also have formal radiological planning as documented in an ALARA planning document. Based upon the expected total dose and the radiological risk, the work requires approvals prior to execution, as indicated in the Table C.2.5.1.

C6. Nuclear Gauging Devices

Nuclear gauging devices, commonly referred to as “nuclear gauges,” use radiation from sealed radioactive sources or RGDs to measure density, thickness, flow rates, and composition of materials. They can be designed as either transmission gauges or backscatter gauges, and can be either portable or fixed at a specific location.

A backscatter gauge consists of a radiation source on the same side as the detector. The detector provides information on the surface being assessed based on the magnitude of radiation that scatters off the surface and into the detector. They are typically portable and are widely used in industry for purposes such as measuring moisture/density in soil and analyzing the composition of paint or coatings.

A transmission gauge consists of a radiation source on one side of the item to be evaluated and a detector on the other side. Transmission gauges are frequently installed (i.e., fixed) on manufacturing process lines to monitor material densities or thicknesses, levels of liquids or solids in bottles or tanks, and the presence, volume, or flow of material in a pipe or a vessel.

Portable and fixed gauges use a variety of radionuclides and activities. Most gauges contain sources that are in IAEA Category 4, although some fixed gauges can contain Category 3 sources (IAEA 2005, 2006).

Photon radiation is frequently used due to its ability to penetrate higher density materials, including the material being measured, and the container or vessel itself. Typical radionuclides used for these applications include $^{137}$Cs, $^{60}$Co, $^{241}$Am, and $^{192}$Ir.
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<td>RPM</td>
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Beta radiation is used for measuring lower density materials (e.g., paper, fiberglass, thin sheets of aluminum). The most common beta emitters used in industrial gauging applications are $^{147}\text{Pr}$, $^{85}\text{Kr}$ and, more rarely, $^{90}\text{Sr}$.

Alpha radiation is rarely used in gauges, except where $^{241}\text{Am}$ is combined with beryllium to generate neutrons. Neutron emitters (e.g., $^{252}\text{Cf}$, $^{241}\text{Am-Be}$) are used for moisture detection and for chemical analysis of materials.

The chemical and physical forms of the sealed radioactive sources used in nuclear gauges vary widely by manufacturer, application, and radionuclide. The sources are normally encapsulated, except where beta radiation is used for the measurement (e.g., $^{85}\text{Kr}$ gas or $^{147}\text{Pm}$ sealed in a capsule with a very thin metal foil window). Most $^{137}\text{Cs}$ gauges contain ceramic pellets; however, some older units may contain cesium chloride. In most gauges using $^{60}\text{Co}$ and $^{241}\text{Am}$, the radionuclides are in metal form.

RGD-based fixed and portable gauges also used in specific industrial applications as their designs improve to better handle the environment in which they must operate. Currently, portable hand-held x-ray spectrometers are used for chemical composition analysis, replacing those using $^{109}\text{Cd}$ and $^{55}\text{Fe}$ sources. In addition to chemical composition analyses, RGDs are also used extensively for fixed gauge applications such as density and thickness measurements.

The NRC’s NUREG-1556 series publications provide useful information on radiation safety programs for both portable and fixed gauges. Volume 1 (NRC 2016a) addresses portable gauges and Volume 4 (NRC 2016b) addresses fixed gauges. NCRP Report No. 182 (NCRP 2019) provides an in-depth discussion on the radiation safety of sealed sources and their application in nuclear gauges. A summary discussion is presented here.

C.6.1 Radiological Hazards

The two main hazards associated with fixed and portable gauges are inadvertent external exposures, and personnel and area contamination in the event of a leaking sealed source. The likelihood of these events is quite small when the devices are used as specified by the manufacturer but if a gauge is damaged and the source is exposed, high dose rates may be present.
Procedures and training should address the proper management of “credible” emergencies. Most organizations that use portable and fixed gauges typically do not have the appropriate staff to respond to a significant event (e.g., a breached sealed source or one that is stuck in the exposed position), and thus need to rely on consultants to properly decontaminate and survey the affected facility to prevent inadvertent exposures and the spread of contamination.

Most fixed industrial gauges have built-in protective devices (shutters) to limit any potential exposures to workers who might work in close proximity to the gauge. Exposures are typically very minimal (e.g., <0.02 mSv h\(^{-1}\) at one foot with shutter open) with few exceptions which may require access restriction.

**C.6.2 Radiation Safety Program**

A critical part of the radiation safety program for portable and fixed gauge users is explicitly stating what workers can and cannot do, particularly as it pertains to routine and nonroutine maintenance and emergency response.

The radiation safety program should cover the information discussed below.

- Uses, requirements and limitations on handling or working in the vicinity of fixed and portable gauges, including confirmation of the shutter position and Lock Out/Tag Out of the device.
- Storage requirements and limitations (e.g., ambient radiation both inside and outside the storage area that may limit the number of gauges stored).
- Security and control of the gauges while in storage, use, and “on-the-road” (e.g., details on storage of the gauges during multi-day operations away from the home location.)
- Radiation monitoring requirements during storage and use.
- External dosimetry requirements.
- Protocols to be followed if a leak test indicates that the sealed source may be leaking.

The radiation safety program should be audited periodically, as specified in the applicable regulations.
C.6.3 Facility Design

The primary consideration for facility design for portable gauges is to make sure the dose rate in and around the storage location is acceptable, and that the storage area is adequately secured. Design considerations for fixed gauges involve providing adequate access to the gauges for required leak testing or maintenance.

C.6.4 Radiation Monitoring Equipment

Two types of survey instruments are typically needed for portable and fixed-gauge users: hand-held radiation instruments for measuring the ambient radiation levels, and either a swipe counter to analyze a leak test swipe, or a contamination detector for screening leak tests prior to mailing them to a company specifically licensed to analyze leak test swipes.

Gauges containing sealed sources must be surveyed initially and before and after any changes, such as for nonroutine maintenance (e.g., moving a fixed gauge). RGD gauges must be surveyed at a frequency established by the applicable regulatory authority, which varies by state and type of equipment.

Frequently, organizations that use fixed and/or portable gauges do not have the equipment necessary to analyze a leak test and therefore rely on contractors to perform this service. If the organization wishes to perform their own leak tests, they must obtain the necessary license to perform and analyze the swipe in-house, or include swipe counting in their radioactive material license. Radiation monitoring instrumentation is discussed in detail in Section 7.

Most regulatory agencies require a documented semi-annual inventory and leak test for fixed and portable gauges.

C.6.5 Physical security monitoring and surveillance

Fixed and portable gauges must be properly secured to prevent loss or theft. The NRC requires that portable gauges be kept under two separate, independent locks. When in use, fixed gauges must be monitored by an operator or be secured. Fixed gauges in storage should be subject to increased surveillance (e.g., quarterly inventory) and the amount of time in storage limited before re-use, disposal, or transfer to another licensed user. Access to storage areas
should be limited to those who actually require access and the access list should be periodically
reviewed and updated.

C.6.6 Dosimetry

Individual external dosimeters are required by the NRC (NRC 2020a) and Agreement States if workers are likely to receive 5 mSv or more in a calendar year. If an organization is able
to document that workers are not likely to receive 5 mSv or more in a year (e.g., using radiation
surveys and stay-time calculations), individual dosimeters are not required, but the basis for this
determination must be documented and retained. DOE requires external dosimetry if workers are
likely to exceed 1 mSv 1 mSv y⁻¹). In addition, both the NRC and DOE have lower thresholds for
external dosimetry for certain types of workers (e.g., declared pregnant workers and minors).

Internal dosimetry is generally not needed for normal operations using portable and/or fixed gauges. If a significant off-normal contamination event occurs, outside technical may be needed. Contractors that are skilled in conducting internal dose assessments should be identified in advance if the organization does not have the capability to perform analyses in-house.

C.6.8 Training

Workers using portable and/or fixed gauges need to be trained on radiation safety fundamentals, operating procedures, limitations on the use of the gauges, use of radiation detection instrumentation and how to identify and respond to off-normal conditions. Additional information on training can be found in Section 5.3 of this report. NUREG-1556 Vol. 1 (NRC 2016a) and Vol. 4 (NRC 2016b) contain suggested training topics for portable and fixed gauge users, respectively.

C.6.9 Radioactive Waste Handling and Disposal

The use and storage of fixed and portable gauges do not produce any radioactive waste streams during normal operations. When a gauge is no longer needed or useable, the manufacturer should be contacted for return and disposal. If the manufacturer is no longer in business or this option isn’t available, a radioactive waste service company should be contacted to determine disposal options. In addition, many fixed and portable gauges contain EPA-listed
hazardous compounds, and the manufacturer may be able to provide guidance on separation of components or materials to facilitate disposal. (Note that with newer RGDs, the tube housing may contain the EPA-listed hazardous compounds.)

C7. Industrial Radiography

Gamma ray industrial radiography (hereafter referred to as “radiography”) uses gamma radiation from sealed sources to “see” inside manufactured items like pipe welds and castings to find defects. When conducted outside of rooms specifically designed for radiography, it is often called “field radiography”.

Radiography is performed using a radiographic “camera” containing a sealed radioactive source on a cable inside an S-shaped tube, which minimizes steaming of the radiation out the camera penetrations. The position of the source cable is locked in the shielded position when the camera is not in use, thus shielding the user from the sealed source during transport and setup. To minimize camera size, part of the camera housing is typically made of depleted uranium. Figure C.7.1 shows one configuration of an industrial radiography camera.

To operate this type of camera, the outlet port cover is opened and the guide tube is attached. Then, on the other end of the camera, the source connector is unlocked and the drive cable connected, coupling it with the sealed source cable. The far end of the guide tube is secured at the exposure location and the far end of the drive cable (where the operator will stand during the exposure) is placed distant from the camera to minimize the operator’s radiation exposure during source use. To initiate the exposure, the operator turns a crank on the drive cable reel which pushes the source to the far end of the guide tube. When the exposure is complete, the operator cranks the reel in the opposite direction and retracts the source back into the camera.
Fig. C.7.1 Sentinel 880 Delta Industrial Radiography Camera.
C.7.1 Radiographic Sources

Most radiography uses $^{192}$Ir as it is readily available. The main disadvantage of using $^{192}$Ir is its 74 day half-life, which necessitates frequent source changes as the activity decays to the point requiring excessively long exposure times. Other radionuclides used for radiography include $^{60}$Co, $^{75}$Se, $^{137}$Cs, and $^{169}$Yb. Source activities typically range from 3.7 – 14.8 TBq; Table C.7.1 shows the dose rates for a variety of radionuclides with a nominal activity of 5.55 TBq.

C.7.2 Hazards

The major radiological hazard from radiography is the very high dose rate close to the unshielded source, as shown in Table C.2.4.1. Worldwide, exposure to unshielded radiography sources has resulted in overexposures, radiation burns, acute radiation syndrome, amputations, and death.

Inadvertent exposures in radiography are usually caused by a source not retracting into the shielded location of the camera, coupled with an inadequate or missing survey to verify the source position. The IAEA (1998) classifies the causes of radiographic accidents as:

- Inadequate regulatory control
- Failure to follow operational procedures
- Inadequate training
- Inadequate maintenance
- Human error
- Equipment malfunction or defect
- Design flaws
- Willful violation
Table C.7.1—Dose rates\textsuperscript{a} from various industrial radiography sources with a nominal activity of 5.55 TBq source.

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>Half life</th>
<th>Gamma constant\textsuperscript{b} (mSv h\textsuperscript{-1} at 1 m from a 1 MBq point source)</th>
<th>Dose rate at 1 m from a 5.55 TBq source (mSv h\textsuperscript{-1})</th>
<th>Gamma constant\textsuperscript{c} (rem h\textsuperscript{-1} at 1 m from a 1 Ci point source)</th>
<th>Dose rate at 1 m from a 150 Ci source (rem h\textsuperscript{-1})</th>
</tr>
</thead>
<tbody>
<tr>
<td>(^{192})Ir</td>
<td>74 d</td>
<td>1.60E-04</td>
<td>8.87E+02</td>
<td>0.59163</td>
<td>8.87E+01</td>
</tr>
<tr>
<td>(^{75})Se</td>
<td>120 d</td>
<td>2.32E-04</td>
<td>1.29E+03</td>
<td>0.85951</td>
<td>1.29E+02</td>
</tr>
<tr>
<td>(^{169})Yb</td>
<td>32 d</td>
<td>8.84E-05</td>
<td>4.90E+02</td>
<td>0.32697</td>
<td>4.90E+01</td>
</tr>
<tr>
<td>(^{137})Cs</td>
<td>30 y</td>
<td>1.02E-04</td>
<td>5.64E+02</td>
<td>0.3761</td>
<td>5.64E+01</td>
</tr>
<tr>
<td>(^{60})Co</td>
<td>5.3 y</td>
<td>3.70E-04</td>
<td>2.06E+03</td>
<td>1.37</td>
<td>2.06E+02</td>
</tr>
</tbody>
</table>

\textsuperscript{a} Traditional units are included as they are commonly used by radiographers.

\textsuperscript{b} Unger and Trubey (1982)

\textsuperscript{c} Calculated from Unger and Trubey (1982).
C.7.3 Radiation Safety Controls

In the United States, radiography is regulated by the NRC (1997) and its Agreement States. The following subsections summarize the elements of a good radiation safety program based on these regulations.

C.7.4 Equipment

- Each camera is of robust design and able to withstand potential mechanical shocks in use and have the source remain in the shielded position (ANSI/HPS 2015).
- Each camera must have a lock or an outer locked container preventing accidental or unauthorized movement of the source from its shielded position.
- Users have calibrated survey instruments capable of detecting and measuring dose rates of the shielded and unshielded source.
- Sealed sources are encapsulated and leak tested at least every six months.
- Sealed sources are inventoried quarterly.
- The camera or sealed source container has a durable warning label that says “caution” or “danger,” if it contains radioactive material, and the point of contact if it is found uncontrolled.

C.7.5 Training

- Radiography is performed with at least two qualified individuals, one operating the camera and the other providing oversight to prevent unauthorized access to the high radiation area.
- The radiography program has an RSO who is trained and qualified to ensure operations are conducted in accordance with the requirements and good radiation safety practices.
- Users are trained and qualified to safely operate radiographic equipment using operating procedures.
● Users are trained on the hazards associated with radiography, fundamentals of radiation safety, and the history of accidents in radiography.

● Users receive annual refresher training.

C.7.6 RSO Duties

● Establish and oversee operating, emergency, and ALARA procedures; review them periodically to ensure that the procedures conform to current regulations and to the license conditions.

● Oversee and approve the training program for radiographic personnel, ensuring the training includes appropriate and effective radiation safety practices.

● Ensure that required radiation surveys and sealed source leak tests are performed and documented in accordance with the regulations.

● Ensure that personnel monitoring devices (e.g., dosimeters, pocket ionization chambers, personal electronic dosimeters) are calibrated and used properly; records are kept of the monitoring results; and that timely notifications are made, as required.

● Periodically evaluate operations to ensure they are conducted safely; institute corrective actions, including stopping of operations, when necessary.

C.7.7 Operating and Emergency Procedures

● Ensure radiation dose are ALARA.

● Require dose rate surveys to ensure the source is in the shielded position after use and before transport of the camera.

● Require posting the area and controlling access during source use to prevent accidental exposure of radiographers and other persons.

● Require inspection, maintenance, operability checks on radiographic equipment, and survey meter use.

● Cover response, notification, and minimization of exposure in the event of an accident, off-scale dosimeter, or dosimeter alarm.
C.7.8 Personnel Monitoring

- Personnel wear a dosimeter processed and evaluated by an accredited NVLAP-processor (NIST 2001) and a direct-reading dosimeter or electronic alarming ratemeter dosimeter on the trunk of the body.

- Direct-reading and electronic dosimeters are checked annually for correct response to radiation.

- NVLAP-accredited dosimeters are exchanged and read at least quarterly.

- Electronic alarming ratemeter dosimeters are checked for operability before the start of each shift.

- If an individual's pocket ionization chamber is found to be off-scale, or if their electronic personal dosimeter reads greater than 2 mSv, the individual cannot perform radiography until his or her dose is evaluated by the RSO.

- The results of dose evaluations are documented.

C.7.9 Surveys

- Monitoring is performed while approaching the camera guide tube and the guide tube itself after each exposure to verify the source has returned to the shielded position.

- The camera is monitored each time the source is exchanged and whenever a radiographic camera is placed in a storage area to ensure the source is in the shielded position.

In summary, radiography can be safely performed with proper and well-maintained equipment, adequate operating and emergency procedures, verbatim compliance with these procedures, well-trained users, and the operating philosophy that the source is in an unsafe condition until it is verified via radiation surveys to be in a safe condition.

C8. Irradiator Facilities

In irradiator facilities, objects and materials are exposed to IAEA Category 1 and 2 sealed radioactive sources (IAEA 2005, 2006) that contain hundreds of PBq and produce dose rates...
exceeding 5 Gy per hour 1 m from the source in air or water, as applicable. The sources are stored either in shields made of solid material (e.g., a panoramic dry-source-storage) or in water (e.g., self-contained wet-source storage, panoramic wet-source-storage). A “pool irradiator” is one in which the sources are stored or used in a pool of water, including panoramic wet-source-storage irradiators and underwater irradiators (NRC 2018a).

The NRC regulates irradiator facilities under 10 CFR Part 36 (NRC 1993a) and provides useful implementing information in NUREG-1556 Vol 6 (NRC 2018a). NUREG-1556 Vol 5 (NRC 2018d) provides licensing information for self-shielded irradiators.

Irradiators typically consist of an array of doubly-encapsulated $^{137}$Cs or $^{60}$Co sources that provide a wide exposure field, relative to the material being irradiated. They come in many different designs and may contain different sources based, in part, on their intended use (e.g., in research, industry, medicine, and other fields) (NRC 2018a).

The orientation of the sources within the facility varies with the irradiator design, resulting in a variety of radiation safety challenges. When regulations and safety protocols specified in guidance documents are followed, there are minimal hazards associated with irradiator operations. However, failure to adhere to the regulations and safety protocols can result in potentially dangerous or lethal exposures.

C.8.1 Facility Design

The regulating authority must approve the design of irradiator facilities, including the:

- shielding calculations and thicknesses (which are based on the maximum activity proposed) for both the irradiator system and its enclosing walls;
- pool and source racks;
- facility layout, to ensure doses to people in adjacent areas are acceptable;
- radiation monitoring systems;
- interlocks and access control systems that ensure exclusion of people during irradiations;
- water handling and purification systems (as applicable);
• a fire suppression system, as ignition of the product is possible should a malfunction (e.g., of the conveyer system) occur; and
• structural integrity, as the facility must be able to support the heavy (e.g., thousands of kg) shielding load.

C.8.2 Monitoring Requirements

Upon entry to the irradiation area, workers must monitor with a calibrated and operable survey instrument to ensure the sources are in their shielded position. Area monitoring is also required, including an exit monitor on conveyor systems (in case a source fell into or onto a product box) and an irradiation room monitor to warn of unexpectedly high levels of radiation (NRC 1993a). The instruments must be periodically calibrated and checked for operability.

Irradiator sources should be leak tested semi-annually, or as specified by the regulations and the manufacturer’s recommendations. The sources in pool irradiators can be directly leak tested, however the detection efficiency may be low and uncertainties in the method may be high, so installed pool monitors can be used to continuously measure radioactivity in the pool water. The water filter can be surveyed with an instrument that can detect contamination in the pool water as close to the leak test limits as possible, and otherwise meet the performance requirements specified in local regulations.

Additionally, pool irradiators may have issues with conductivity in the water, and the water conductivity must be periodically tested (NRC 2018a).

C.8.3 Access Controls

Access controls are specified in 10 CFR 36 (NRC 1993a) and include the typical controls (e.g., interlocks, alarms, and notifications) used to prevent access to very high radiation areas.

C.8.4 Security monitoring and surveillance

Given that irradiator facilities contain IAEA Category 1 and 2 level sealed sources, physical protection measures (above those required for radiation safety) are required by 10 CFR 37 (NRC 2013). Examples of enhanced protections include constant security monitoring of the irradiator and motion alarms whenever an authorized irradiator operator is not physically present.
C.8.5 Dosimetry

Individuals working with irradiators must wear either a whole body dosimeter, a direct reading dosimeter, or an electronic dosimeter. Alarming dosimeters should also be available for workers and visitors to warn quickly of unexpected dose rates.

C.8.6 Training

Training requirements for irradiators are comparable to other aspects within radiation safety (e.g. basic radiation safety information, regulations, procedures, use of required instruments) and should include past accidents and case histories, as required for NRC and Agreement state licensees. In addition, irradiator facilities have prescribed procedures that must be followed precisely, so the prospective operator should receive in-depth training on these procedures and then work under the supervision of a qualified, experienced worker before demonstrating proficiency and being permitted to work independently. Irradiator facility operators should be subject to safety reviews and tested on their knowledge and understanding.

C.8.7 Effluent Monitoring

Ozone may be produced in irradiator facilities (due to the extremely high level of radiation) and may need to be monitored.

C.8.8 Radioactive Waste Handling and Disposal

Decommissioning and disposal of the sealed sources requires extreme caution and diligent preparation. Due to the very high activities contained in irradiators, source disposal can only be handled by the manufacturer or another appropriately licensed entity.

C9. Large Accelerator Facilities

Large accelerator facilities are found throughout the world and comprise important venues for scientific progress in a variety of fields of study including nuclear physics, elementary particle physics, basic energy science, and materials science. Applications of large accelerator facilities can be found in medicine, veterinary science, engineering, and agriculture. The radiological aspects of particle accelerators have been described in great detail by a number of
authors and organizations, including NCRP Report No. 144 (NCRP 2003b), Patterson and Thomas (1973), Cossairt and Quinn (2019), and ANSI/HPS (2011d). The DOE has published DOE Order 420.2C, Safety of Accelerator Facilities (DOE 2011e), which provides a framework for an acceptable accelerator radiation safety program, and an associated guidance document (DOE 2014) which comprehensively addresses operational issues at such installations in a useful manner.

C.9.1 Types of Accelerators and Types of Radiation Produced

Present day facilities accelerate and use a variety of particle types such as protons, electrons, and ions. In general, ions with mass numbers > 4 are commonly called heavy ions, while ions with mass numbers \( \leq 4 \) are classified as light ions. At accelerators, energies of accelerated particles are universally expressed in units of electron volts (eV) and multiples thereof (e.g., keV, MeV, GeV, TeV) rather than SI units. For most purposes, the kinetic energy of the accelerated particle is implied by a specified particle energy (e.g., 1.0 MeV). The domain of energies reached by modern large accelerators is from upward of 1.0 GeV, culminating in a maximum design energy of 7 TeV in each beam of the Large Hadron Collider at the European Center for Nuclear Research (CERN) headquartered in Geneva, Switzerland. Along with the beam energy, the beam intensities are also important. The product of particle energies and their intensities is embodied in the beam power, with currently available beam powers at many installations now in excess of several megawatts (MW). Many radiological impacts (e.g., prompt radiation fields, the production of induced radioactivity, radioactivity that might enter the public environment) are directly linked to the delivered beam energy, the integral over time of the beam power.

Along with the acceleration of so-called “primary” beams of particles, modern accelerators also produce a wide variety of secondary particles such as photons, neutrons, and a multitude of other particles, notably in some cases, muons. Energies of these secondary particles can reach from near zero up to nearly the kinetic energies of the accelerated particles. In some situations, secondary particles have been collected and developed into beams that have characteristics similar to those of the primary beams. These secondary beams are then delivered to target stations for use by the research scientists. Some of the secondary particles (e.g., muons...
can be of unique importance. The radiation fields can be very complex, with mixed fields of different particles with a large range of coefficients relating dose to particle fluence.

C.9.2 Specialized Categories of Large Research Facilities

Large research accelerator facilities usually are complex installations that vary somewhat in design reflective of their intended purposes. Those focused on high energy nuclear and particle physics using protons, ions, and electrons commonly have a significant number of experimental beam lines that direct secondary beams of selected types of charged particles, photons, and electrons toward experiments and their associated targets. The primary particles from the accelerator are used to create these secondary beams by means of their interactions in specifically designed targets and associated systems such as deflection and focusing magnets, radiofrequency systems, and sometimes lasers.

The secondary beams can be comprised of any particles that can be produced by the primary ones extracted from the accelerator with high intensities possible and kinetic energies up to nearly that of the accelerated primary beams. At some facilities, radioactive nuclear fragments of short half-lives are produced, accelerated, and transported. Such beams can be inclusive of nuclear fission fragments. Even beams of neutral electrical charge are sometimes employed. Even with proton and ion accelerators, high energy electron and positron fields can be produced by electromagnetic cascades. Neutrons are possible at all such accelerators and can have energies ranging from thermal up to nearly the energy of the accelerated beam.

Worldwide, there are several dozen synchrotron radiation sources specifically designed to produce photons, notably in the “x ray” spectral domain; these facilities are called “light sources”. The resultant photons are used by scientists and engineers in a multitude of technological disciplines. In these facilities, energetic electrons are produced by an accelerator, and then stored in a storage ring where they radiate x rays because of momentum changes forced by magnetic fields. An adaptation of this technology is that of the free electron laser (FEL) in which unbound (i.e., “free”) electrons generated by passing a high-energy (relativistic) electron beam through a linear undulator (alternating magnetic fields) are used to produce coherent x-ray beams analogous to that of visible light lasers.
In synchrotron light sources, there are often a very large number of extracted beamlines having different properties as determined by electromagnetic insertion devices placed in the storage ring at an associated extraction point. These insertion devices are of two basic types, called “wigglers” and “undulators,” that define the energy spectrum and the intensity (i.e., the “brilliance”) of the extracted photon beam. The specialized experimental apparatus or instruments are placed in separate experimental enclosures, sometimes called “hutches”. The instrumentation is either provided by the host institution or supplied by the experimental users.

Spallation neutrons sources are specialized proton accelerators that are used to produce neutrons from the nuclear spallation reaction process. The main accelerator typically accelerates protons to energies in the GeV range and at very high beam intensity. Additionally, these facilities have many portals, commonly called “instruments,” that provide a varied set of neutron beams and instrumentation designed to have specified properties. These instruments are used by a multitude of users in manner somewhat analogous to that found at the light sources, with a number of experiments of relatively brief duration. The experiments may supply additional instrumentation designed to address their scientific objectives.

**C.9.3 Types of Radioactive Materials Present**

A clear distinction arises between *prompt* radiation fields (i.e., those present when the accelerator is operational), and *residual* radiation fields (i.e., those resulting from radioactivity induced by the primary and secondary particles). Most of the radioactive materials at accelerator facilities results from volume activation, with removable surface contamination of much less relative importance than at other types of radiological facilities. However, when handling targets bombarded by high energy particles, removable radioactive contamination is a concern that must be addressed with appropriate PPE and monitoring equipment.

While the bulk shielding can be massive (e.g., many meters in all dimensions corresponding to multiple metric tons), generally only the portions near the accelerated beam become detectably radioactive. At proton and ion accelerators, the scaling parameters (e.g., the mean free path and, most prominently the radiation length) for protons, neutrons, ions, and other particles subject to the “strong” or nuclear interaction are much larger than those applicable to
electron beams. Thus, the measurably activated volumes of shielding material are much larger for proton and ion accelerators than for electron accelerators.

The beamline components at larger accelerators can nearly always be presumed to be volume-activated due to the incident fluence of hadrons and photons ubiquitous at such facilities. That is, the entire mass of a given volume is more or less uniformly activated. Many references (ANSI/HPS 2011d; Cossairt and Quinn 2019; DOE 2011e, 2014, 2016; Patterson and Thomas 1973) give guidance as to the radionuclides produced and their production cross sections, which may, especially at the lower energies, be energy-dependent. Most of these radionuclides are beta-gamma emitters readily detected by ordinary survey instruments. The production of alpha-emitting radioactive material is virtually nonexistent but should be considered when materials of high atomic number are encountered (e.g., in the materials being studied by the researchers, or in the shielding). In general, high atomic number (e.g., lead, uranium) material is excluded from the shielding materials to the extent possible to preclude the presence of alpha-particle emitters.

While not “prompt radiation” per se, the production of radioactive gases during accelerator operations can be important. The radioactive gases produced at accelerators are dominated by the relatively short-lived positron emitters (e.g., $^{11}\text{C}$, $^{13}\text{N}$, $^{15}\text{O}$) along with, in some cases, $^{41}\text{Ar}$. Other short-lived radionuclides may also be present depending on the facility configuration. In terms of exposure pathways, these radionuclides represent immersion rather than inhalation hazards. Often, due to the short half-lives, the hazard is largely mitigated by ventilation and by requiring personnel to allow time after the beam is disabled for decay to occur before accessing enclosures where the gases are present. These gases also represent a source of radiation exposure to the public and must be managed and reported in accordance with applicable regulatory requirements.

**C.9.4 Principal Radiological Hazards**

The greatest radiological hazard at large accelerator facilities is exposure to the direct beam, where even a short exposure could be lethal. From that level, hazards are reduced until they reach those associated with the natural background levels encountered outside the shielding.

Due to the effectiveness the bulk shielding and interlock systems, doses received by personnel at most large accelerator facilities are predominantly, and in most cases completely,
due to exposures to residual radioactivity. The radiological hazards from the residual activity
range from negligible, near background levels up to those that represent very high radiation
fields (i.e., those greater than 5 Gy in an hour). At the higher dose rates, the likelihood of
encountering removable radioactive contamination rises correspondingly.

C.9.5 Facility Design Considerations

Because of the energetic nature of the primary and secondary particles, very large shields
must be used to restrict dose rates and integrated doses to acceptable levels. A typical large
proton or ion accelerator may well require lateral shielding of approximately 7.0 m of earth
shielding. If muons are present, their ionization ranges may be quite large. For example, the
mean range of a 1.0 GeV muon is about 23 meters while that of a 100 GeV muon, a muon energy
readily available by modern accelerators, is about 200 meters of earth of density 2.0 g cm$^{-3}$
(Cossairt and Quinn 2019). Since the production of muons is generally straight ahead (forward
peaked) from their point of origin, careful attention must be paid in designing shielding in the
forward direction. The overall need for large amounts of shielding material leads to the use of
relatively low cost, bulk materials that are readily available (e.g., earth, concrete, and iron).
Modular shielding that might be found at smaller accelerators such as concrete blocks are used at
large accelerators on a limited scale where specific needs for portability justifies the large cost
increment. The nature of the radiation fields has led most large accelerators to extensively use
earth as shielding, with accelerator enclosures commonly placed underground, thus limiting the
number of access points. The provision for safe enclosure access and egress thus can require a
collaborative effort including radiation safety, industrial hygiene, and fire protection expertise.

An appropriate methodology for screening bulk shielding materials for radioactivity is an
important component of the radiation safety program and becomes of forefront importance as a
part of civil construction activities when the bulk shielding must be modified to accommodate
accelerator improvements.

At all accelerators, personnel access to the direct beam, as well as other hazardous levels
of radiation, is prevented by a well-designed system of interlocks. Excellent design standards for
such interlock systems have been published (ANSI/HPS 2011d; DOE 2014). The combination of
well-designed interlocks and the bulk shielding mitigates the prompt radiation hazard.
Radiation monitoring instrumentation must be designed to address the radiation fields present, which may include neutrons, muons, electrons, photons, and other energetic particles subject to the nuclear (strong) interaction. The beam time structure can range from very short pulses (~nanoseconds) to a continuous beam. Beams of short duration pulses can produce very high instantaneous dose rates, resulting in instrumental dead-times that may, with certain instrumentation, render measured results misleadingly low. It is also common for static magnetic fields and radio-frequency (RF) electromagnetic fields to be present with strengths capable of giving false readings on some radiation detectors. Instruments based on photo-multiplier tubes are especially susceptible to static magnetic fields, while ionization chambers can be sensitive to RF.

While workplace monitoring and surveillance programs may be quite similar to those found at other types of accelerator facilities, the large expanse of these facilities motivate innovative approaches. Remote readouts of monitoring equipment from a central location, perhaps available online, is likely to be a major facility need, along with the ability to electronically track and record the results.

In general, scientific collaborations for nuclear and particle physics experiments tend to be large, ranging from dozens to thousands of members. A very large range of educational levels are often found, along with a multitude of primary languages. Furthermore, personnel arrive at a facility often on an unpredictable schedule. Also, it is quite common for specific beams to be used by a multitude of different groups of experimenters who conduct their research work during periods of relatively short duration; perhaps days or weeks.

Given these factors, training should ideally be modular, efficiently delivered, and available on-line. The technical content of the training should be delivered in a concise, but scientifically correct manner. Furthermore, the training for international scientific visitors should include an orientation to the traditional system of radiological units (e.g., rem instead of Sieverts, Curies instead of Becquerels) used at United States installations for posting and dosimetry reports, as opposed to the SI system in use elsewhere.
Effluent monitoring and radioactive waste management

Effluent monitoring programs can best meet applicable requirements by following the methodologies recommended in NCRP Report No. 169 (NCRP 2010c) and ANSI/HPS (2011a). Attention needs to be devoted to evaluating the unique nature of the airborne radionuclides produced by accelerators which are dominated by the relatively short-lived positron emitters (e.g., $^{11}$C, $^{13}$N, $^{15}$O, and $^{41}$Ar). Since these gases represent a potential source of radiation exposure to the public, they need to be monitored and reported to insure compliance with applicable regulatory requirements.

Management of activated cooling water and lubricating oils is of great importance, as the production of tritium ($^3$H) in the form of tritiated water (HTO) is a near certainty. Another consideration is the production of HTO in groundwater in the vicinity of the accelerator. Given that tritium cannot be directly monitored, extra precautions are warranted where this is a possibility.

When high energy accelerators are decommissioned, there are typically large quantities of activated material, including accelerator components and shielding, which will have to be repurposed, disposed of, or allowed to decay in place.

C10. Oil and Gas Industry

Radioactive sources are used in energy exploration and recovery, exemplified by the use of sealed sources for well logging in oil and gas (and coal) exploration and the use of unsealed radioactive materials either as tracers or in hydraulic fracturing of strata to enhance production. NCRP Report No. 182 (NCRP 2019) provides an in-depth discussion on the radiation safety of sealed radioactive sources and their application in well-logging devices. A summary discussion is presented here.

Well-logging devices typically contain a sealed $^{241}$Am-Be source. The source generates neutrons for analyzing strata for the presence of carbonaceous materials (i.e., the hydrocarbons in oil and natural gas), and also for measuring the porosity of the strata. These sources are generally GBq quantities. The emitted neutrons interact with the hydrogen or hydrocarbons, which reduce their energy and are backscattered to the detector. Other sources that may be used include $^{137}$Cs
and $^{241}$Am (also in GBq ) quantities) for measurement of the density of the strata. The detector(s) in the device measure photons backscattered from the source. Rarely, other neutron sources such as $^{252}$Cf or a D-T generator are used, although their use does not have the wealth of data that has been obtained through the use of the Am-Be sources. The source should have a certificate, issued by the US NRC or Agreement State Licensing Board, designating that the source is suitable for well logging.

The sealed sources are generally IAEA Category 2 or 3 (IAEA 2005, 2006) and as such, their use is subject to the increased security requirements of the NRC or equivalent Agreement State regulations. For accelerator-produced neutrons, a radioactive material license is needed for the possession and use of the $^3$H target, and the appropriate state regulatory agency must authorize use of the accelerator.

The well-logging device is a 1m - 2 m long, cylindrical item that contains both the sealed source, the detector, and associated electronics for sending the signals from the detector in its location in the well back up to the surface where the data is analyzed. A display ("log") of the signals gives the desired information to the logging team (typically one or two operators).

The well-logging device operators are considered radiation workers and require basic radiation safety training, as well as specific training on the use, storage, transportation and security of the Category 2 or 3 sealed sources. The operators also must be trained in overall workplace safety in accordance with the OSHA regulations, since they are working in and around heavy equipment, the drill platform and rig, high pressure, and electrical and mechanical equipment. Although there is a potential for radiation exposure, the dominant concern is industrial safety.

Well-logging devices are portable and are transported on trucks to the drilling location (a temporary job site). The trucks have a storage compartment for the transport and storage of the logging device, which also provides protection to the operators during transport to and from the job sites. The operators should possess and use radiation survey instruments for measurement of radiation fields in or around the job site, and also for transportation requirements (DOT 2015).

For well logging, the NRC (2017, 2018b) requires the following:

- Use of operator logs.
Personnel dosimetry for workers conducting well-logging.

- Copies of operating and emergency procedures to be carried on the trucks for use as a reference as needed by the operators, and in case of an unexpected event.
- completed shipping papers available in the truck.
- specific documents and records that the regulating authority requires to be available at jobsites or field stations.

For hydraulic fracturing (a.k.a. “fracking”), authorization from the regulatory agency is required, and the licensee must implement procedures for the receipt, possession and use (and any disposal of unused quantities) of the radioactive materials. The typical radioactive materials used are $^{46}$Sc, $^{82}$Br, $^{110m}$Ag, $^{124}$Sb and $^{192}$Ir, in MBq quantities.

For field flood or enhanced recovery studies, either liquid or gaseous unsealed radioactive materials, such as $^3$H, $^{14}$C, $^{131}$I, $^{140}$La, $^{192}$Ir and others may be used, in MBq quantities. In these studies, the radioactive material, either by itself or incorporated in another substance, is injected into a single well (“tracer study”) or into several wells (“field flood”), to determine the flow into the well or see how much more can be produced from the well(s).

In addition to the requirement for personnel dosimetry, there may be license conditions that require the evaluation of internal exposure, especially for unexpected events involving the rupture of the containers of the unsealed radioactive materials. The use of other PPE, such as nonporous gloves, is specified for the operators. Similarly, the operating and emergency procedures need to address normal handling, radioactive waste generation and handling, and the disposal of the radioactive materials.

For additional information, see NCRP Report No. 182 (NCRP 2019), which contains extensive information on sealed sources, and IAEA (2003), which contains information regarding the management of waste in the oil and gas industry.

C11. Decommissioning/Cleanup Sites

This section provides general radiation safety program information unique to the support of decontamination/cleanup and decommissioning of sites where radioactive material has been
produced, processed, stored, or used, but does not address the design and implementation of the
decommissioning activity. Detailed guidance regarding decommissioning planning can be found
in NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)*
(NRC/EPA/DOE/DOD 2000), NUREG-1575, Supplement 1, *Multi-Agency Radiation Survey
and Investigation of Material and Equipment Manual (MARSAME)* (NRC/EPA/DOE/DOD
2009) and NUREG-1757, *Consolidated Decommissioning Guidance*, Volume 1 and 2 (NRC
2006a, 2006b).

The section does not discuss radiation safety program elements required to support site
cleanup following a large-scale nuclear or radiological incident. A discussion and program
recommendations on this topic can be found in NCRP Report No. 175 (NCRP 2014).

**C.11.1 Regulatory Guidance and Cleanup Criteria**

MARSSIM (NRC/EPA/DOE/DOD 2000) was developed jointly by NRC, EPA, DOE and
DOD to provide consistent guidance for planning and conducting site surveys, analyzing the
results and verifying compliance with established cleanup criteria. MARSSIM provides guidance
for cleanup sites using Derived Concentration Guideline Levels (DCGLs) that correspond to the
release criteria. MARSAME (NRC/EPA/DOE/DOD 2009) addresses release of equipment and
materials generated during decommissioning activities.

It is important to note that the federal agencies can establish either dose-based or risk-
based cleanup criteria. The site cleanup criteria established by the NRC is a dose-based criterion
and is provided in 10 CFR 20.1402 (NRC 2020a). The NRC criterion is 25 mrem y\(^{-1}\) [0.25 Sv y\(^{-1}\)] to the critical group; residual radioactivity must be reduced to ALARA. The cleanup or release
criterion for EPA-governed sites is risk-based and establishes the criterion of an acceptable
lifetime excess cancer risk of \(10^{-6}\) to \(10^{-4}\). DOE has established a dose limit of 100 mrem y\(^{-1}\) to a
member of the public from all sources and has accepted NRC’s dose limit of 25 mrem y\(^{-1}\) [0.25
Sv y\(^{-1}\)] for a single decommissioning site.

Some states have established restrictive (sometimes referred to as “conditional”) release
criterion when the site requires a restricted rather than an unrestricted release. A restricted
release includes some number of conditions – or limitations – for future use of the site. As an
example, at the Rocky Flats Superfund site, DOE/EPA used a risk-based end state for a
conditional release approach to the cleanup. The Rocky Flats site end-state was designated a
restricted use wildlife refuge only. No other uses of the site are authorized. Additional
(expensive) cleanup would have been required if unrestricted release of the site had been the
goal.

NCRP Report No. 146 (NCRP 2004) provides additional details and specifically includes
“an analysis of current regulatory guidance and practice used by NRC and EPA in remediation of
radioactively contaminated sites.”

ANSI N13.12 (ANSI/HPS 2013) provides risk-based release criteria and survey
methodologies for unrestricted release of items or material that may contain residual levels of
radioactivity. ANSI N13.12 includes a dose criterion 0.01 mSv 0.01 mSv y\(^{-1}\) and derived
screening levels for groups of radionuclides for both surface and volume contamination. DOE O
458.1 (DOE 2011c), also includes a clearance criterion of 1 mrem y\(^{-1}\) [0.01 mSv y\(^{-1}\)] for personal
property (i.e., property other than land or buildings).

**C.11.2 Types of Cleanup Sites**

The radionuclides and level of residual contamination at cleanup sites varies greatly
based on the facility type, previous operation and time elapsed since closure. Nuclear power
reactor sites, uranium sites (fabrication, mills, and gaseous diffusion plants), thorium and radium
manufacturing sites, former U.S. Naval bases, university and research facilities, and DOE
laboratories each present unique contaminants, quantities, and challenges.

**C.11.3 Phases of Decommissioning**

The Radiation Safety Program is responsible for radiation safety of the workers and
environment throughout initial planning, decontamination/cleanup, and through final site closure.
The radiation safety program generally functions independently from the site closure team
responsible for planning the decommissioning and provides the radiation safety support for
decommissioning activities. A site decommissioning project is typically divided into four phases:
assessment, development, operations, and closeout (DOE 1994). During the assessment phase,
the Site Historical Assessment (HSA) is developed and scoping and characterization surveys are
performed. The development phase includes preparation of the cleanup, decommissioning, or
license termination plan. The operations phase includes the dismantlement, decontamination of plant systems and equipment, demolition, soil excavation and waste disposal and may be the most challenging phase (Abelquist 2014). The final closeout phase begins when it is anticipated that the site conditions meet the established release criteria. The closeout phase includes the independent verification by regulatory agencies or independent third parties to certify that the site meets the cleanup or decommissioning criteria.

C.11.4 Facility Historical Site Assessment and Characterization Surveys

To ensure proper radiation safety controls are implemented and will adequately address the hazards, the radiation safety program procedures should be designed based on the information provided in a sufficiently detailed HSA and subsequent scoping and characterization surveys. The HSA report should include historical information such as a description of facility operations, facility drawings and historical photographs, operating time frame, licenses and permits, known accidents/incidents, environmental monitoring reports and radionuclide uses and storage. The HSA is used to identify potential sources, levels of contamination in each area of the site and to plan scoping and characterization surveys. Results from the scoping and characterization surveys provide the basis for design of the successful radiation safety program at a cleanup/decommissioning site.

C.11.5 Work control

The radiation safety program controls should include established, proven procedures to ensure worker safety, properly manage waste, and minimize environmental releases. Ensuring radiation safety during the dismantlement of plant systems and equipment, soil excavation and waste handling requires implementation of a robust radiation safety program that addresses most of the elements discussed previously in this report (Section 5). Remediation activities should be carefully planned, reviewed and approved prior to authorizing work on any new activity. Due to the likelihood of many nonradioactive hazards during deconstruction activities, the required radiation safety precautions should be integrated with other safety procedures and coordinated with other safety professionals. In contrast to most operating facilities, the activities undertaken during facility deconstruction, decontamination and cleanup are not routine and will likely present some number of unanticipated hazards. A Radiological Work Permit or other work
control/authorization detailing the task scope, hold points, stop work limits, PPE, dosimetry, training, and survey requirements should be reviewed often to ensure that the controls remain appropriate for a changing work site.

As work moves from deconstruction/decontamination to decommissioning activities, the radiation safety program requirements should be modified to address the reduced worker and environmental hazards, while ensuring all required program elements are met.

C.11.6 Background Reference Areas and Instrumentation

A decommissioning License Termination Plan (LTP) or cleanup plan may include a site-wide background level or provide multiple reference background areas and the requirement that material background reference areas be established prior to designing and conducting field surveys. The background reference areas are designated areas of the site that were not impacted by previous site activities, that will remain free of contamination throughout the cleanup work and are representative of the area to be surveyed (survey unit) (NRC/EPA/DOE/DOD 2000). At some sites, multiple background areas may be required due to the variation in naturally occurring radionuclides, $^{137}\text{Cs}$ fallout present in the soil, or previous landscaping and import soil variability. The background levels are used to establish survey investigation levels, which prompt additional sampling and surveys if exceeded.

Surface material background reference areas are also established for each of the materials to be surveyed (e.g., asphalt, poured concrete, concrete block, metal or wood). The radiation safety program may be responsible for establishing and tracking the multiple soil and material reference background levels and ensuing the appropriate instruments used and calibrated for the radionuclides of concern (see Section 7). NUREG-1501 (NRC 1994b) is a useful reference addressing site background.

C.11.7 Remedial Action Survey Support

Remedial action surveys and sampling are used to assess the success of the decontamination and remediation efforts and may be the assigned responsibility of the radiation safety program, rather than the license termination or site closure team. Separating responsibility for conducting these remedial action surveys from the final status survey effort allows
5786 independent verification of the effectiveness of the remediation and provides the information
5787 necessary for the radiation safety program to provide appropriate controls during throughout the
5788 decommissioning effort.

5789 The radiation safety program should include procedures to address measurement and
5790 characterization of radioactivity in liquid and airborne effluents, soils, groundwater and any
5791 subsurface structures to support remedial action surveys. (See Section 7.2 for information on
5792 analytical laboratories.)

5793 **C.11.8 Waste Management**

5794 Waste generated during cleanup activities may include building material, metal scrap,
5795 equipment and tools, soil, and liquids. A large facility generally has a decommissioning or
5796 cleanup plan that provides the types and volumes of waste generation anticipated. Smaller
5797 facility cleanup sites may determine the appropriate disposition of the waste as characterization
5798 data becomes available. Designating the waste material as potential reuse, recycle, or evaluating
5799 various disposal options relies on detailed waste characterization survey data and/or process
5800 knowledge. The radiation safety program may be responsible for implementing the radioactive
5801 waste management activities and ensuring that the waste is properly controlled, packaged,
5802 inventoried and shipped (see Section 6).

5803 Several resources are available to the Radiation Safety professional responsible for
5804 managing waste generated from cleanup and decommissioning activities. NUREG-1640 (NRC
5805 2003) provides dose factors for scrap iron and steel, copper, aluminum, and concrete rubble for
5806 many radionuclides. These dose factors address surface contamination and volumetric
5807 contamination of the scrap materials and provide various exposure scenarios for assessing risk of
5808 each pathway. MARSAME (NRC/EPA/DOE/DOD 2009) provides various disposition criteria
5809 and action levels for clearance of materials and equipment.

5810 NCRP Report No. 143 (NCRP 2003) provides guidance for minimizing waste from
5811 decommissioning activities at smaller facilities and includes the following points for
5812 considerations during the planning stage:

5813 • Determine the optimum scope of decommissioning—should the facility be fully or
5814 partially decommissioned?
Consider the future use of the facility to establish decommissioning objectives. The future use may not require full decontamination, thereby reducing costs and waste generation from decontamination procedures.

- Evaluate options for decontamination of contaminated equipment and debris versus disposal.
- Maximize recovery of recyclable materials. Opportunities for the recycle of decontaminated material should also be considered in the planning process.
- Select the optimal technologies for each decommissioning activity.

At DOE facilities, recycling metals from decommissioning activities has presented additional challenges. DOE suspended the release of scrap metal from DOE Radiological Control Areas in a memorandum (DOE 2001). In 2012, DOE provided guidance in a draft programmatic environmental assessment (DOE 2012). The assessment "evaluates alternatives for the management of scrap metal originating from DOE radiological control areas, including the proposed action to allow for the recycle of uncontaminated scrap metal that meets the requirements of DOE Order 458.1” (DOE 2011c). Metals with volumetric radioactive contamination were included in the scope the assessment. Some DOE sites have met additional requirements and have received approval to release scrap metal for recycling. NCRP Report No.141 (NCRP 2002) provides some additional information.

The requirements that must be met for designation of soil as waste for offsite disposal or as potential reuse material onsite is often included in the cleanup plan or a License Termination Plan (LTP). The LTP is approved by the NRC. The radiation safety program should be designed to ensure that the waste management program meets the expectations defined in the LTP. NUREG-1700, Rev 2, (NRC 2018c) is a useful resource addressing expectations for license termination, especially regarding waste disposal options.
### Abbreviations, Acronyms, and Symbols

<table>
<thead>
<tr>
<th>Abbreviation</th>
<th>Description</th>
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<tbody>
<tr>
<td>AAPM</td>
<td>American Association of Physicists in Medicine</td>
</tr>
<tr>
<td>ACGIH</td>
<td>American Conference of Governmental Industrial Hygienists</td>
</tr>
<tr>
<td>ACR</td>
<td>American College of Radiology</td>
</tr>
<tr>
<td>ADG</td>
<td>Administrative Dose Guideline</td>
</tr>
<tr>
<td>AGS</td>
<td>American Glovebox Society</td>
</tr>
<tr>
<td>AIHA</td>
<td>American Industrial Hygiene Association</td>
</tr>
<tr>
<td>a.k.a</td>
<td>also known as</td>
</tr>
<tr>
<td>ALARA</td>
<td>As Low as Reasonably Achievable</td>
</tr>
<tr>
<td>ALI</td>
<td>annual limit on intake</td>
</tr>
<tr>
<td>ANL</td>
<td>Argonne National Laboratory</td>
</tr>
<tr>
<td>ANS</td>
<td>American Nuclear Society</td>
</tr>
<tr>
<td>ANSI</td>
<td>American National Standards Institute</td>
</tr>
<tr>
<td>ASHRAE</td>
<td>American Society of Heating, Refrigerating and Air-Conditioning Engineers</td>
</tr>
<tr>
<td>ASSP</td>
<td>American Association of Safety Professionals</td>
</tr>
<tr>
<td>ASTM</td>
<td>American Society for Testing and Materials</td>
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<tr>
<td>BWR</td>
<td>Boiling Water Reactor</td>
</tr>
<tr>
<td>CAM</td>
<td>Continuous Air Monitor</td>
</tr>
<tr>
<td>CERN</td>
<td>Collider at the European Center for Nuclear Research</td>
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<tr>
<td>COMPLY</td>
<td>An EPA computer code for evaluating radiation exposure from atmospheric releases of radionuclides.</td>
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<tr>
<td>Abbreviation</td>
<td>Description</td>
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<tr>
<td>CZT</td>
<td>Cadmium-Zinc-Telluride</td>
</tr>
<tr>
<td>D</td>
<td>Dose</td>
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<tr>
<td>D-D</td>
<td>deuterium-deuterium</td>
</tr>
<tr>
<td>D-T</td>
<td>deuterium-tritium</td>
</tr>
<tr>
<td>DAC</td>
<td>Derived Air Concentration</td>
</tr>
<tr>
<td>DAC-h</td>
<td>Derived Air Concentration-hours</td>
</tr>
<tr>
<td>DCFPAK 2.0</td>
<td>A computer code of dose conversion factors</td>
</tr>
<tr>
<td>DCGL</td>
<td>Derived Concentration Guideline Level</td>
</tr>
<tr>
<td>$d\mathcal{E}, , dm$</td>
<td>the mean energy ($d\mathcal{E}$) imparted by ionizing radiation to matter of mass ($dm$)</td>
</tr>
<tr>
<td>DIS</td>
<td>Direct Ion Storage</td>
</tr>
<tr>
<td>DOD</td>
<td>Department of Defense</td>
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<tr>
<td>DOE</td>
<td>Department of Energy</td>
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<tr>
<td>DOELAP</td>
<td>Department of Energy Laboratory Accreditation Program</td>
</tr>
<tr>
<td>DOT</td>
<td>Department of Transportation</td>
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<tr>
<td>DSL</td>
<td>derived screening level</td>
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<tr>
<td>ED</td>
<td>effective dose</td>
</tr>
<tr>
<td>EDE</td>
<td>effective dose equivalent</td>
</tr>
<tr>
<td>EGS</td>
<td>Electron Gamma Shower (computer code)</td>
</tr>
<tr>
<td>EML</td>
<td>Environmental Measurements Laboratory</td>
</tr>
<tr>
<td>EPA</td>
<td>Environmental Protection Agency</td>
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<tr>
<td>EPD</td>
<td>electronic personal dosimeter</td>
</tr>
<tr>
<td>EPIPs</td>
<td>emergency plan implementing procedures</td>
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<tr>
<td>eV</td>
<td>electron volt</td>
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<tr>
<td>Acronym</td>
<td>Description</td>
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<tr>
<td>FDA</td>
<td>Food and Drug Administration</td>
</tr>
<tr>
<td>FEL</td>
<td>free electron laser</td>
</tr>
<tr>
<td>FGI</td>
<td>Fluoroscopy Guided Interventional [procedures]</td>
</tr>
<tr>
<td>FIDLER</td>
<td>Field Instrument for the Detection of Low Energy Radiation</td>
</tr>
<tr>
<td>FWHM</td>
<td>Full Width at Half of the Maximum height of the peak</td>
</tr>
<tr>
<td>GM</td>
<td>Geiger Mueller</td>
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<tr>
<td>H</td>
<td>Hazard Guide Value</td>
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<tr>
<td>HEPA</td>
<td>High-efficiency particulate air [filter]</td>
</tr>
<tr>
<td>HF</td>
<td>Hydrofluoric acid</td>
</tr>
<tr>
<td>HHS</td>
<td>Health and Human Services</td>
</tr>
<tr>
<td>HotSpot</td>
<td>Health physics computer codes for assessing a variety of radiological conditions</td>
</tr>
<tr>
<td>HPS</td>
<td>Health Physics Society</td>
</tr>
<tr>
<td>HRA</td>
<td>High Radiation Area</td>
</tr>
<tr>
<td>HSA</td>
<td>Site Historical Assessment</td>
</tr>
<tr>
<td>$H_T$</td>
<td>is the equivalent dose to tissue T</td>
</tr>
<tr>
<td>HTO</td>
<td>Hydrogen Tritium Oxygen; “tritiated water”</td>
</tr>
<tr>
<td>IACUC</td>
<td>Institutional Animal Care and Use Committee</td>
</tr>
<tr>
<td>IAEA</td>
<td>International Atomic Energy Agency</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>INPO</td>
<td>Institute of Nuclear Power Operations</td>
</tr>
<tr>
<td>IRB</td>
<td>Institutional Review Board</td>
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<tr>
<td>IRPA</td>
<td>International Radiation Protection Association</td>
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</table>
ISO  International Organization for Standardization
JCAHO  Joint Commission on Accreditation of Healthcare Organizations
LHRA  Locked High Radiation Area
LN  Liquid Nitrogen
LSC  Liquid Scintillation Counter
LTP  License Termination Plan
m  Mass of the material
MARSAME  Multi-Agency Radiation Survey and Investigation of Material and Equipment Manual
MARSSIM  Multi-Agency Radiation Survey and Site Investigation Manual
MCA  Multi-Channel Analyzer
MCNP  Monte Carlo N-Particle (MCNP) transport computer code
MSHA  Mine Safety and Health Administration
MW  megawatts
NCRP  National Council on Radiation Protection and Measurements
NEMA  National Electrical Manufacturers Association
NEMA  National Electrical Manufacturers Association
NEPA  National Environmental Policy Act of 1969
NEPA  National Environmental Policy Act
NID  Negligible Individual Dose
NIOSH  National Institute for Occupational Safety and Health
NIST  National Institute of Standards and Technology
NNSA  National Nuclear Security Administration
NORM  naturally occurring radioactive material
<table>
<thead>
<tr>
<th>Abbreviation</th>
<th>Full Form</th>
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<tbody>
<tr>
<td>NRC</td>
<td>Nuclear Regulatory Commission</td>
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<tr>
<td>NUREG</td>
<td>Nuclear Regulatory (Commission guides)</td>
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<tr>
<td>NVLAP</td>
<td>National Voluntary Laboratory Accreditation Program</td>
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<tr>
<td>OSHA</td>
<td>Occupational Safety and Health Administration</td>
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<tr>
<td>OSHMS</td>
<td>Occupational Safety and Health Management System</td>
</tr>
<tr>
<td>OSLD</td>
<td>Optically Stimulated Luminescent Dosimeters</td>
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<tr>
<td>PC</td>
<td>Protective clothing</td>
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<tr>
<td>PCM</td>
<td>personnel contamination monitor</td>
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<tr>
<td>PET</td>
<td>Positron Emission Tomography</td>
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<tr>
<td>PMT</td>
<td>photomultiplier tube</td>
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<tr>
<td>PPE</td>
<td>personal protective equipment</td>
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<tr>
<td>PWR</td>
<td>Pressurized Water Reactor</td>
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<tr>
<td>Q</td>
<td>Quantity of radionuclide</td>
</tr>
<tr>
<td>QMP</td>
<td>Qualified Medical Physicist</td>
</tr>
<tr>
<td>RA</td>
<td>Radiation Area</td>
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<tr>
<td>RBA</td>
<td>Radiological Buffer Area</td>
</tr>
<tr>
<td>REAC/TS</td>
<td>Radiation Emergency Assistance Center/Training Site</td>
</tr>
<tr>
<td>RCA</td>
<td>Radiologically Controlled Area</td>
</tr>
<tr>
<td>RCO</td>
<td>radiological control organization</td>
</tr>
<tr>
<td>RCRA</td>
<td>Resource Conservation and Recovery Act</td>
</tr>
<tr>
<td>RCT</td>
<td>radiological control technician</td>
</tr>
<tr>
<td>RE</td>
<td>Relative Efficiency (as in 40 % RE detector)</td>
</tr>
<tr>
<td>RESRAD</td>
<td>RESidual RADioactive materials. An EPA-sponsored computer code</td>
</tr>
<tr>
<td>RF</td>
<td>radio-frequency</td>
</tr>
<tr>
<td>Abbreviation</td>
<td>Definition</td>
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<tr>
<td>RGD</td>
<td>radiation generating device</td>
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<tr>
<td>RMA</td>
<td>Radioactive Material Areas</td>
</tr>
<tr>
<td>RPM</td>
<td>Radiation Protection Manager</td>
</tr>
<tr>
<td>RPS</td>
<td>Radiation Protection Supervisor</td>
</tr>
<tr>
<td>RPT</td>
<td>Radiological Protection Technologist</td>
</tr>
<tr>
<td>RSC</td>
<td>Radiation Safety Committee</td>
</tr>
<tr>
<td>RSO</td>
<td>Radiation Safety Officer</td>
</tr>
<tr>
<td>RWP</td>
<td>Radiation work permit</td>
</tr>
<tr>
<td>T</td>
<td>Toxicity Factor (when used in the HotSpot codes)</td>
</tr>
<tr>
<td>T</td>
<td>equivalent dose in a tissue</td>
</tr>
<tr>
<td>TENORM</td>
<td>technologically enhanced NORM</td>
</tr>
<tr>
<td>TLD</td>
<td>thermoluminescent detector</td>
</tr>
<tr>
<td>U</td>
<td>Use Factor (when used in the HotSpot codes)</td>
</tr>
<tr>
<td>U.S.</td>
<td>United States</td>
</tr>
<tr>
<td>VHRA</td>
<td>Very High Radiation Area</td>
</tr>
<tr>
<td>WGS</td>
<td>Work Group Supervisor</td>
</tr>
<tr>
<td>wT</td>
<td>weighting factor for tissue</td>
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</tbody>
</table>
Glossary

**absorbed dose**: The mean energy $d\varepsilon$ imparted to matter of mass $dm$ by ionizing radiation at the point of interest. In the Systeme Internationale (SI), the unit for absorbed dose is J kg$^{-1}$ with the special name gray (Gy), formerly the unit was the rad (1 Gy = 100 rad).

**accuracy**: The accuracy of a measurement system is the degree of closeness of measurements of a quantity to that quantity's true value.

**activation**: The process of inducing radioactivity.

**administrative dose guidelines**: The predetermined value of radiation dose to workers, below the dose limit, which triggers a specific course of action when the value is exceeded, or is expected to be exceeded.

**air monitoring**: A method of determining the amount of airborne radioactivity using an instrument with real-time alarm capability.

**air sampling**: A method of determining the amount of airborne radioactivity by collecting a sample (e.g., on a filter paper or silica gel column) which is subsequently sent into a laboratory for retrospective analysis.

**ALARA** (as low as reasonably achievable): A principle of radiation safety philosophy that requires that exposures to ionizing radiation should be kept as low as reasonably achievable, economic and social factors being taken into account. The ALARA principle is equivalent to the principle of optimization defined by the NCRP and ICRP which states that protection from radiation exposure is optimum when the expenditure of further resources would be unwarranted by the reduction in exposure that would be achieved.

**alpha particles**: A charged particle having a mass and charge equal to a helium nucleus (i.e., two protons and two neutrons) emitted from the nucleus of an atom. Synonymous with “alpha radiation”.

**annual limit on intake (ALI)**: The derived limit for the amount of radioactive material taken into the body of an adult worker by inhalation or ingestion in a year. ALI is the smaller value of intake of a given radionuclide in a year by the reference man (ICRP 1975) that would result in a committed effective dose of 50 mSv or a committed equivalent dose of 500 mSv to any individual organ or tissue. ALI is expressed in becquerels (Bq).
**area monitor**: A radiation detector designed to measure the radiation levels in a specified location.

**beta particle**: A negatively charged particle indistinguishable from an electron that is emitted from the nucleus of an atom as a consequence of radioactive decay. Synonymous with “beta radiation”.

**bioassay**: A technique used to identify, quantify and/or specify the location of radionuclides in the body by direct (*in vivo*) or indirect (*in vitro*) analysis of tissues or excretions from the body.

**cascade summing**: The process that can occur when a radionuclide emits two photons simultaneously in a cascade (e.g., $^{60}\text{Co}$). If the detector absorbs both photons completely, a characteristic sum peak will be produced (referred to as “summing in”). If the detector absorbs one of the photons completely and the other photon partially, a random energy peak will be produced which will in effect decrease the apparent efficiency of the detector for that radionuclide (referred to as “summing out”). This effect is primarily a function of the size of the detector and the source-to-detector geometry.

**contamination**: Radioactive material suspended in air or deposited in any area or on any surface where its presence is unwanted or unexpected.

**committed dose**: The total dose [mean absorbed dose in an organ or tissue (organ dose), equivalent dose, effective dose] expected to be delivered by a radionuclide incorporated in the body within a specified time period. For the purpose of radiation protection, the specified time period is 50 y for workers; for members of the public the time periods are 50 y for adults and 70 y for infants and children.

**continuous air monitor (CAM)**: A device with real-time alarm capability used to measure airborne radioactivity.

**criticality**: The point at which a nuclear fission reaction becomes capable of sustaining a chain reaction.

**decommission**: Close a facility and prepare its buildings and land for release to unrestricted use.

**derived air concentration (DAC)**: the airborne concentration equal to the ALI divided by the volume of air breathed by an average worker for a working year of 2,000 h (assuming a breathing volume of 2400 m$^3$). The unit of the DAC is Bq m$^{-3}$. The DAC is defined for each radionuclide (NRC 2020a, Appendix B; DOE 2011a).

**Derived air concentration-hour (DAC-hour)**: the product of the concentration of radioactive material in air (expressed as a fraction or multiple of the DAC for each radionuclide) and the time of exposure to that radionuclide, in hours.

**deterministic effect (tissue reaction)**: Injury in a population of cells, characterized by a threshold dose and an increase in the severity of the reaction as the dose is increased further.


**detriment:** The total harm that is expected to be experienced by an exposed group and its descendants as a result of the group’s exposure to radiation. The expected harm includes fatal and nonfatal cancer, severe hereditary effects, and life shortening.

**dose** (ionizing radiation): A general term used when the context is not specific to a particular ionizing radiation dose quantity. When the context is specific, the name or symbol for the quantity is used (e.g., organ dose, effective dose).

**dosimeter:** A device, instrument or system usually worn by an individual to determine his or her personal dose equivalent.

**effective dose** ($E$): The sum over specified organs and tissues of the products of the equivalent dose in a tissue ($H_T$) and the tissue weighting factor for that tissue or organ ($w_T$):

$$E = \sum_T w_T H_T,$$

Effective dose applies only to stochastic effects. The SI unit for $E$ is joule per kilogram ($J \, kg^{-1}$) with the special name sievert (Sv).

**effluent monitoring:** The measurement of radioactivity in air, liquids and solid material leaving a facility.

**electron gamma shower (EGS):** a computer code for transporting electrons and the resultant photons produced.

**electron volt (eV):** the kinetic energy gained by a particle carrying the charge of one electron falling through an electric potential drop of 1.0 Volt. $1.0 \, eV = 1.602 \times 10^{-19} \, J$.

**emergency:** A sudden, urgent, usually unforeseen occurrence or occasion requiring immediate action.

**engineered controls:** A system of design features that are intended to prevent or limit exposures to radiation or releases of radioactive materials.

**environmental monitoring:** The measurement of external dose or the amount of radioactivity in air, soil, water, plant and animal matter in areas outside the control or boundary of a facility.

**equivalent dose** ($H_T$): Mean absorbed dose in a tissue or organ (organ dose) due to radiation type R ($D_{T,R}$) weighted by the radiation weighting factor ($w_R$):
The SI unit of equivalent dose is joule per kilogram (J kg\(^{-1}\)) with the special name sievert (Sv). 1 Sv = 1 J kg\(^{-1}\).

exposure (general): A general term used to express the act of being exposed to ionizing radiation.

external dose: The dose received from a source of radiation outside of the body.

fail-safe: A design feature or practice that in the event of a specific type of failure, inherently responds in a way that will cause no or minimal harm to other equipment, to the environment, or to people.

fissile: A descriptor of a nuclide that is capable of undergoing nuclear fission, usually following the absorption of a slow neutron.

Geiger-Muller (GM) counter: A gas-filled radiation detector most often used to detect the presence of low dose rate beta particles, x rays, or gamma rays. The detector is not appropriate for use with pulsed radiation sources or when the type or energy of the radiation is to be determined.

gamma rays: Short wavelength electromagnetic radiation emitted from an atomic nucleus as a result of radioactive decay.

glovebox: A sealed container with controlled access and ventilation, usually having a viewing window and ports covered by gloves to provide access for work inside the box.

grab sample: A sample of limited volume taken at random or at preselected frequencies.

gray (Gy): The special name for the SI unit J kg\(^{-1}\) (i.e., energy imparted per unit mass of a material). 1Gy = 1J kg\(^{-1}\).

hot cell: A heavily shielded enclosure for handling, processing or storing highly radioactive sources.

internal dose: The dose received from a source of radiation inside of the body

justification: The part of the decision-making process in which the options that are expected to do more good than harm are identified.

kerma (kinetic energy released per mass): The sum of the initial kinetic energies of all the charged particles liberated by uncharged particles in a mass of material. The unit for kerma is J kg\(^{-1}\), with the
special name gray (Gy). Kerma can be quoted for any specified material at a point in free space or in an absorbing medium (e.g., air kerma).

**limit:** In radiation safety, the level of dose established by authoritative or consensus bodies above which the consequences to an individual would be regarded as unacceptable.

**member of the public:** A member of the public is any person at, near, or otherwise impacted by a facility’s operations, and who is not receiving occupational radiation exposure.

**monitoring:** Continuous or periodic determination of the amount of radiation or radioactivity present in a specified area or volume of effluent. In this Report, when used in the context of workers monitoring themselves or areas, documentation is not required.

**negligible individual dose (NID):** A level of effective dose to an individual per source or practice that may be ignored. This term was defined in NCRP Report No. 127 and its recommended value is 0.01 mSv.

**occupancy factor:** The factor used in dose projection calculations for shielding design or other purposes to account for the fraction of time that a space will be occupied by any single individual.

**occupational exposure:** Exposures to individuals that are incurred in the workplace as the result of situations which can reasonably be regarded as being the responsibility of management (exposures associated with medical diagnosis or treatment are excluded).

**performance check:** A before-use and after-use check to ensure an instrument is properly functioning. It typically includes a battery check, a response to radiation check, and a visual inspection.

**personal dose equivalent [HP(d)]:** The equivalent dose determined at an appropriate depth, \(d\), in the body. This depth is usually taken to be 10 mm for penetrating radiation and 0.07 mm if the individual is expected to be exposed to nonpenetrating radiation. HP(10) and HP(0.07) are also referred to as the “deep” dose and “shallow” dose, respectively. The unit for the personal dose equivalent is sievert (Sv); formerly the unit was the rem (1 Sv = 100 rem).

**protective clothing (PC):** In this Report, a general term for garb worn for radiological control purposes, inclusive of a lab coat, coveralls, shoe covers, and gloves. PC are a subset of PPE.

**personal protective equipment (PPE):** In this Report, a general term inclusive of any item placed on the worker for the purpose of protection (e.g., lab coat, coveralls, shoe covers, gloves, leaded vest, leaded gloves, respirator, hard hat, safety glasses, face shield).

**planned special exposure:** A radiation dose that is authorized for a worker that will exceed occupational dose limits. The planned special exposure is considered only in exceptional circumstances when it can...
be justified and when alternatives that might prevent a worker from exceeding limits are unavailable or impractical.

**precision:** The precision of a measurement system is the degree to which repeated measurements under unchanged conditions show the same results.

**process knowledge:** in the context of waste management, information that includes, but is not limited to, records of materials present at the facility, analyses of process stream composition, engineering calculations based on material balances, process stoichiometry, or previous test results (provided the results are still relevant).

**public:** All persons who are not already considered occupationally exposed by a source or practice under consideration.

**qualified expert:** In radiation safety, a person having the knowledge and training to provide advice regarding radiation safety principles, standards and measurements. Persons having relevant certification from the American Board of Health Physics, the American Board of Medical Physics, the American Board of Radiology, or the American Board of Industrial Hygiene may be considered qualified experts.

**qualified medical physicist (QMP):** An individual who is competent to independently provide clinical professional services in one or more of the subfields of medical physics. A QMP is qualified to practice only in the subfield(s) in which they are certified. The subfields of medical physics are: Therapeutic Medical Physics; Diagnostic Medical Physics; Nuclear Medical Physics; Magnetic Resonance Imaging Physics. (AAPM 2018).

**radiation generating device (RGD):** An electric or battery-powered device (including accelerators and certain high-power lasers) that generates ionizing radiation either incidentally or intentionally.

**radiation weighting factor** ($w_R$): A factor used to allow for differences in the biological effectiveness between different radiations when calculating equivalent dose. The set of $w_R$ values are general and are independent of the tissue or organ irradiated. They are selected by judgment after review of a broad range of experimental biological effectiveness data that are relevant to stochastic effects.

**radiation safety:** A system of controls directed at utilizing the benefits of radiation while keeping people and the environment safe from its potential deleterious effects. In this report, used synonymously with “radiation protection” and “health physics”.

**radiation safety technician:** A general term referring to a technician who is specifically trained (and usually qualified) in matters pertaining to radiation safety. This individual may have any number of
titles, including radiological control technician (RCT), health physics technician (HP tech), radiological protection technologist (RPT).

**radiation work permit (RWP):** With regard to radiation safety, an authorization to perform a specific procedure that will involve the exposure of persons to radiation or uncontained radioactive material in a specified area of a facility.

**radionuclide:** A radioactive species of atom characterized by its mass number, atomic number and sometimes its nuclear energy state (provided that the mean lifetime of that state is sufficiently long for the species to be observed).

**REAC/TS:** The Radiation Emergency Assistance Center/Training Site (REAC/TS) is a leader in emergency medical response to radiological/nuclear incidents, providing emergency response and subject matter expertise on the medical management of radiation incidents.

**reference man:** A person with the anatomical and physiological characteristics defined in the report of the ICRP Task Group on Reference Man (ICRP 1975).

**respirator or respiratory protection device:** A device worn on the face or head to prevent the inhalation of toxic or radioactive materials.

**run-safe switch:** A switch that, when toggled to “safe” ensures that a specific piece of equipment (e.g., an accelerator) cannot be operated and it is safe to be in the area. When toggled to “run,” the specific piece of equipment is capable of running and potentially causing harm to those in the area.

**sealed radioactive source:** A manufactured item containing a discrete quantity of radioactive material to be used as an ionizing radiation source that is encapsulated, plated or bonded in a matrix to prevent the dispersal of the radioactive material under the conditions of use and normal wear for which it was designed.

**stochastic effects:** Effects, the probability of which, rather than their severity, is assumed to be a function of dose without a threshold. For example, cancer and heritable effects of ionizing radiation are regarded as being stochastic effects.

**surveys:** Periodic determination of the amount of radiation or the amount of radioactive material present in an area. By convention in this Report, surveys are documented.

**tissue weighting factor (wT):** The dimensionless factor by which equivalent dose is weighted to represent the relative contribution of that tissue or organ to the total radiation detriment resulting from uniform irradiation of the body. The wT's are judgment values grouped by organs and tissues in the interest of simplicity and rounded to sum to 1.0.
user: As used in this Report, a person, group of people, company, or corporation that uses radioactive material or radiation generating devices.
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