10A RADIATION PROTECTION EVALUATION FOR DRY STORAGE FACILITIES (SL)

10A.1 Review Objective

The objective of the U.S. Nuclear Regulatory Commission's (NRC's) radiation protection evaluation is to determine the following:

- The applicant has proposed a functional radiation protection program that will effectively manage, monitor, and control radiation exposures and doses to facility workers and members of the public from a dry storage facility (DSF) that is either an independent spent fuel storage installation (ISFSI) or a monitored retrievable storage installation (MRS) in compliance with NRC regulations and acceptance criteria.

- The proposed DSF radiation protection features meet the NRC design criteria for direct radiation and effluent controls.

- The applicant has proposed engineering features and operating procedures for the DSF that will ensure that occupational exposures will remain as low as reasonably achievable (ALARA).

- Occupational radiation doses will not exceed the limits specified in the NRC's radiation protection standards.

- Radiation doses to the public will meet regulatory standards during both normal conditions and anticipated occurrences and will meet the regulatory dose limits for accident conditions.

- Radiation exposures and radioactive effluent releases will be maintained at levels that meet ALARA objectives and comply with the NRC limits.

For the purposes of this standard review plan (SRP) chapter, radiation protection refers to organizational, design, and operational elements that are relied upon to limit radiation exposures from normal operations, anticipated occurrences (that is, off-normal conditions), and accidents and natural phenomenon events (collectively referred to as accident conditions or design-basis accidents). This includes those design and other elements that may have a different primary function but are nonetheless credited or considered in the applicant's radiation protection evaluation.

10A.2 Applicability

This chapter applies to the review of applications for specific licenses for ISFSIs and MRSs, referred to as DSFs. Thus, the chapter title is denoted with **(SL)**.

10A.3 Areas of Review

The areas of review include means and methods used to protect workers and members of the public, facility design features, dose assessments and dose assessment methods, radiation monitoring instrumentation, sampling and analytical equipment, and operational elements and procedures.

This chapter addresses the following areas of review:

- ALARA objectives
  - policies and programs
  - design considerations
  - operational considerations
- radiation protection design features
  - installation design features
  - access control
  - radiation shielding
  - confinement and ventilation
  - area radiation and effluent monitoring and instrumentation
  - radiological environmental monitoring program
- radiation exposures and dose assessment
  - basis and assumptions of dose assessment
  - onsite dose
  - offsite dose
- health physics program
  - organization and staffing
  - equipment, instrumentation, and facilities
  - policies and procedures

10A.4 Requirements and Acceptance Criteria

This section summarizes those parts of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste and Reactor-Related Greater Than Class C Waste," and 10 CFR Part 20, "Standards for Protection Against Radiation," that are relevant to the review areas this chapter addresses. The reviewer should refer to the exact language in the regulations.

This section describes the acceptance criteria used to guide the review of radiation protection features and programs. The safety analysis report (SAR) should address these acceptance criteria. The acceptance criteria are organized according to the areas of review specified in Section 10A.3 above. The reviewer should consider the applicability and implementation of NRC and industry guidance against that presented in the SAR.

The radiation protection review also requires coordination with other reviews under this SRP related to site characteristics (Chapter 2), principal design criteria (Chapter 3), shielding (Chapter 6), confinement (Chapter 9), operation procedures and systems (Chapter 11), conduct of operations (Chapter 12), waste management (Chapter 13), accident analysis (Chapter 16), and technical specifications (Chapter 17). A complete evaluation of the facility's radiation protection program, as outlined in this chapter, is also dependent on accurate and adequate evaluations of these other aspects of the facility's design and operation.

This guidance recognizes that applicants have various options on how to demonstrate compliance with NRC regulations and NRC guidance (e.g., rely only on NRC guidance or use alternative methods). In general, the acceptance criteria listed in the SAR should adopt, by reference, appropriate NRC guidance or, alternatively, cite relevant and appropriate industry codes and standards. The SAR should identify and justify alternative approaches used to demonstrate compliance with applicable NRC guidance and industry codes and standards. Use of a code or standard in lieu of NRC guidance may require the applicant to discuss the applicability of the code or standard and the basis for its selection and use. Section 10A.5, "Review Procedures," of this SRP provides more specific guidance on the conduct of reviews whenever the SAR cites industry codes and standards.

With respect to the implementation of NRC guidance, the SAR should identify whether the applicant has adopted the NRC guidance in whole or in part. The SAR should identify any differences between this SRP chapter and design features, analytical techniques, exposure and dose assessment codes, and procedural measures proposed for the facility and discuss how the proposed alternatives to this SRP acceptance criteria provide acceptable methods of complying with regulations. In any case, the SAR should provide sufficient information and data for the staff to conduct an independent evaluation in confirming compliance with regulatory requirements and SRP acceptance criteria. The reviewer will confirm that the applicant has adequately addressed these things in the SAR.

If there are multiple versions of a guidance document, such as a regulatory guide or an industry standard, the SAR should describe which version of the guidance document the applicant used, whether it is the most current revision and the basis for using the selected version. In the case of an industry standard, the applicant should consider what, if any, staff position exists with respect to the acceptability of the standard and its different revisions as part of that selection. An applicant may propose to use a particular revision because the proposed DSF is co-located with the applicant's facility licensed under 10 CFR Part 50 "Domestic Licensing of Production and Utilization Facilities," or 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," which used that revision of the guidance as part of its approved licensing basis. As a result, the reviewer will identify the guidance documents the applicant used and assess whether the version of each document the applicant adopted is adequate for demonstrating compliance with NRC requirements.

Table 10A-1 matches the relevant regulatory requirements to the areas of review covered in this chapter. While Table 10A-1 includes specific 10 CFR Part 20 requirements, additional requirements in 10 CFR Part 20 may also apply. Accordingly, the reviewer should consult 10 CFR Part 20 to identify relevant requirements and ensure that the SAR addresses them. Moreover, the applicant and reviewer should be aware of and consider the relevant requirements in 10 CFR Part 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations," including the requirements in 10 CFR 19.11, "Posting of Notices to Workers," 10 CFR 19.12, "Instruction to Workers," and 10 CFR 19.13, Notifications and Reports to Individuals."

The reviewer should also be aware that the Environmental Protection Agency (EPA) has established annual dose limits, which apply to DSFs, in 40 CFR Part 191, "Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes," particularly in 40 CFR 191.03(a). These limits are the same as the limits in 10 CFR 72.104(a). Thus, compliance with the limits in 10 CFR 72.104(a) ensures compliance with the EPA's limits.

As the guidance in this chapter indicates, evaluations and information concerning 10 CFR Part 20 include demonstrations that the facility design and operations are adequate to ensure compliance with the 10 CFR Part 20 requirements, including dose limits. These evaluations include dose assessments. These dose assessments provide an indication as to whether regulatory dose limits will be met and whether the facility design and operations have adequately considered radiation protection, including the dose limits and as low as is reasonably achievable (ALARA) principles. For locations at or beyond the 10 CFR Part 72 controlled area boundary, the dose estimates should show the limits will not be exceeded. For locations within that controlled area boundary, dose estimates should be fairly low versus limits. For instances where that is not the case, the applicant should provide additional information, such as any administrative controls or measures or physical design features to be used, to show how the applicant as a licensee will ensure limits are met and ALARA principles are followed.

Table 10A-1 Relationship of Regulations and Areas of Review

Areas of Review	10 CFR Part 20 Regulations									
	20.1101	20.1201 (a)	20.1301	20.1302	20.1406 (a)(c)	20.1501	20.1601 (a(b)(c)(d)(e)	20.1602	20.1701	20.1702
ALARA Objectives	(a)(b)(c)(d)		(d)		•	(a)(1)				•
Radiation Protection Design Features	(a)(b)(d)		(e)	(a)	•	(a)(1), (c)(d)	•	•	•	•
Radiation Exposures and Dose Assessment	(a)(b)(d)	•	(a)(b)(d) (e)(f)	(a)(b)						
Health Physics Program	(a)(b)(c)(d)	•	(b)(d)	(a)	•	(a)(1), (c)(d)				•

Areas of Review	10 CFR Part 72 Regulations									
	72.24	72.40	72.44 (c)(d)	72.100	72.104	72.106	72.120 (a)(b)(c)	72.122 (b)(4), (e)(h), (3), (4), (5)	72.126	72.128 (a)(2), (3)
ALARA Objectives	(b)(c)(d)(e)(l)	(a)(1), (5)(13)	•		(b)		•		(a) (d)	
Radiation Protection Design Features	(b)(c)(d)(e)(l)	(a)(1), (2)(5) (13)	•		(a)(b)(c)	(a)(b)(c)	•	•	(a)(b)(c) (d)	•
Radiation Exposures and Dose Assessment	(d)(e)(m)(l)	(a)(1), (2)(5), (13)		•	(a)(c)	(a)(b)		•	(d)	
Health Physics Program	(e)(l)	(a)(1), (5)(13)	•						(a) (d)	

10A.4.1 ALARA Objectives

In evaluating the elements of the ALARA program, the applicant should describe a functional program (including a management policy and organizational structure), proposed engineering design features, activities conducted by individuals having responsibility for radiation protection, and operating procedures that will ensure that occupational exposures and doses to members of the public will be maintained ALARA objectives and meet regulatory standards during normal conditions and anticipated occurrences. The applicant should demonstrate that releases of radioactive materials in liquid and gaseous effluents will be ALARA and describe how the applicant will ensure that releases will be maintained at levels that are ALARA and comply with NRC regulations.

10A.4.1.1 *Policies and Programs*

As a minimum, the policy, program, and activities for ensuring that radiation exposures will be ALARA should include the elements described below in this section. Regulatory Guide (RG) 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," and RG 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable," provide acceptable guidance on the development and implementation of an ALARA program. Additionally, International Commission of Radiological Protection (ICRP) Publication 27, "Problems Involved in Developing an Index of Harm," issued in 1997, and Publication 55, "Optimization and Decision-Making in Radiological Protection," issued in 1990, and National Council on Radiation Protection and Measurements (NCRP) Report No. 116, "Limitation of Exposure to Ionizing Radiation," issued in 1993, provide useful information for developing an ALARA policy and program.

Policy Statement

The SAR should include a written policy that states management's commitment to maintain exposures to workers and the public at ALARA levels and addresses both facility design and operations. The policy should include the following provisions:

- No practice involving radiation exposure will be undertaken unless evaluation of the practice demonstrates that its use will produce a net benefit to society.
- All exposures will be kept ALARA, with technological, economic, and social factors considered.
- Individual dose limits will be established that are appropriate for practices involving radiation exposure, and exposures to individuals will not exceed these limits.
- Supervisors will integrate appropriate radiation protection controls into all work activities.
- Workers will be appropriately instructed in the objectives and implementation of the ALARA program, with this information included in training modules.
- There will be strict compliance with all regulatory requirements and license conditions regarding procedures, radiation exposures, and releases of radioactive materials.

- A comprehensive program will be maintained, and periodically evaluated, to ensure that both individual and collective doses meet ALARA objectives and do not exceed acceptable levels.

Program Organization

This element should include an organizational structure of the ALARA program, describe the functional responsibilities at all staff levels with respect to its implementation, provide adequate staffing, and include the duties of the personnel directly responsible for the implementation of the ALARA program and policies. The health physics manager and facility health physics staff should have the authority to supervise, monitor, and halt any facility operations and procedures that could result in unnecessary radiation exposures to workers and members of the public or lead to doses in excess of administrative limits and NRC regulations.

Program Elements

The SAR should document how the implementation of the ALARA program will ensure that ALARA objectives are achieved for both onsite and offsite radiation exposures and for monitoring and controlling effluent releases. ALARA program elements should include the use of the following:

- procedures and engineering controls to minimize doses to site personnel, radiation workers, and members of the public allowed access into controlled areas
- tracking of individual doses to identify trends and causes and use of such data in developing alternative procedures that would yield lower doses
- periodic training and exercises for management, radiation workers, health physics staff, and other site workers in radiation protection, ALARA, operating procedures, and emergency response, and then periodic evaluation of the effectiveness of such training and exercises
- periodic reviews and evaluations of the other program elements to ensure their continued effectiveness, making improvements where beneficial, such as revising training programs, drills, and exercises to keep up to date with current radiation protection practices, operations practices, and emergency response plans and procedures
- procedures and controls to monitor, process, and treat radioactive effluents before release into the environment to minimize discharges of radioactive materials and minimize doses to members of the public
- administrative controls and radiation monitoring equipment to prevent unmonitored and uncontrolled releases of radioactive materials

10A.4.1.2 *Design Considerations*

The applicant's discussion of the facility's design, including design of facility features and structures, systems, and components (SSCs), and the facility's layout, including overall layout and layout within facility structures, should demonstrate consideration of ALARA principles and operational knowledge. The design criteria for the facility's features and SSCs, described in the

SAR's principle design criteria chapter, should include ALARA criteria, and the SAR should identify choices between otherwise comparable alternatives affected by ALARA considerations and the basis for the selected alternative(s). Applicants should use RG 8.8 for ALARA design guidance, although they may use specific alternative approaches if clearly indicated in the SAR. Examples of ALARA design considerations include the following:

- engineered design features that minimize radiation levels and the total amount of time that maintenance, health physics, or inspection personnel must stay in restricted areas while performing their duties
- engineered design features that minimize the need for maintenance
- provisions for the use of remotely operated or robotic equipment, such as automated welders, wrenches, and remote radiation monitors
- use of closed-circuit television to monitor for possible blockage of air cooling passages, to perform inspections and other activities
- provisions for remote placement and use of temporary shielding
- incorporation of materials, design features, and operational practices that minimize the potential for accumulation of radioactive materials or surface contamination, and facilitate decontamination and decommissioning of facilities and equipment
- incorporation of design experience from other ISFSIs, MRSs, or waste management facilities using ALARA design alternatives that are similar to or are improvements of those used at these other facilities
- use of relevant operations experience from other ISFSIs, MRSs, or waste management facilities
- placement of occupiable areas (e.g., office, security stations, access and egress control points, or health physics and laboratory facilities) away from sources of radiation and radioactivity
- ALARA provisions built into health physics training facilities and equipment

10A.4.1.3 *Operational Considerations*

Operational procedures, methods of operation, and methods to develop detailed plans and procedures should incorporate ALARA principles and objectives to ensure personnel exposures and contamination levels are ALARA. The SAR description of these methods and procedures should include the criteria or conditions under which various procedures or techniques are implemented to ensure personnel exposures and residual contamination levels for all facility SSCs that handle radioactive materials are ALARA. The associated operational requirements should be reflected in facility design, as described in Sections 6.4.1 and 6.5.1 of this SRP as well as this chapter. Detailed plans and procedures should be developed in accordance with RG 1.33, "Quality Assurance Program Requirements (Operation)," RG 8.8, and RG 8.10, and should consider the following to the extent practical:

- tradeoffs between requirements for increased monitoring or more frequent maintenance activities (and the resulting increases in radiation exposures) and potential hazards (e.g., premature failures or reduced effectiveness of SSCs) associated with reduced frequency of these activities

- performance of storage container (e.g., cask) preparation efforts (for loading) away from the spent nuclear fuel (SNF) pool or dry transfer facility

- sequencing the placement of SNF, reactor-related greater-than-Class-C waste (GTCC), or high-level radioactive waste (HLW), as appropriate, in a manner that maximizes the shielding effectiveness of storage containers and structures

- conducting of dry runs to develop proficiency in procedures involving radiation exposures; determination of exposures likely to be associated with specific procedures; identification of conditions likely to be associated with specific operational evolutions leading to potentially higher exposures; and consideration, development, and implementation of more efficient alternative procedures in order to control and minimize exposures and doses

- consideration and inclusion of tested and proven contingency plans and procedures in responding to potential anticipated occurrences

- consideration and incorporation of ALARA operational alternatives based on related industry experience at other ISFSIs, MRSs, or similar types of waste management facilities

- research, evaluation, and development of improved operational procedures, types of tools and instruments, and use of personal protective equipment to minimize radiation exposures, releases of radioactive materials, and duration of exposures and reduce risks associated with exposures

10A.4.2 Radiation Protection Design Features

This element addresses the adequacy of the incorporation of radiation protection considerations into the facility design, including meeting regulatory requirements and ALARA objectives. For this element, the SAR should provide information on facility design features, access control, provisions for and effective use of shielding, confinement and ventilation, and means and methods in monitoring external radiation exposure rates and airborne radioactivity concentrations. RG 8.8 includes guidance that, where applicable, may be useful for ensuring adequate incorporation of radiation protection considerations into the facility design.

The SAR descriptions should include facility features and SSCs used for facility operations, including package receipt; package decontamination and unloading; package loading and preparation; waste (SNF, reactor-related GTCC waste, HLW) transfer between package and storage container; storage container preparation, loading, movement, and use; storage container array(s); and site-generated waste treatment packaging, storage, and shipment. The SAR should also describe the provisions made for personnel protective measures, particularly for areas where radioactive materials may become airborne. This information may be referenced from other sections of the SAR as appropriate. The SAR should include scaled layout and arrangement drawings for the facility. These drawings should include locations where SNF, reactor-related GTCC waste, HLW waste, and site-generated wastes will be stored. The SAR should also

include information on definition of work areas, designation of radiologically controlled areas and their boundaries (e.g., radiation areas, restricted areas, controlled area), shield wall thicknesses, individual and equipment decontamination areas, contamination control areas and types of controls, personnel and vehicular traffic patterns, health physics facility locations, area radiation monitoring and airborne radioactivity monitoring locations, locations of onsite analytical laboratories (for chemical and radioactive sample analyses) and counting room facilities, and other pertinent facility features and SSCs relevant for radiation protection.

10A.4.2.1 *Installation Design Features*

Installation design features for radiation protection can minimize either offsite or onsite exposures. Features that specifically minimize offsite exposures include the following:

- Siting Considerations—The facility is located away from population centers to the extent feasible, consistent with other factors.

- Controlled Area or Perimeter Distance—The DSF controlled area is located to maintain sufficient distances to the perimeter of the site and locations of public occupancy.

- Transfer Route—Transfer routes for DSF containers are located to maintain sufficient distances from the site perimeter.

- Effluent Discharges and Impacts—Natural and manmade contours, existing or planned rerouting of natural surface water, and points at which surface water exits the site relative to residences and public use areas are considered and incorporated. Cutoffs, drains, well points, or other means are used to control surface water flow into uncontrolled areas.

- Engineered Features—Berms, shield walls, or other engineered features are used as needed to reduce direct radiation exposures and levels beyond the DSF storage area(s).

Features that minimize onsite exposures include the following:

- Transfer Route—Transfer routes for DSF containers to or from the storage area and the handling areas (intermodal transfer points, or wet or dry transfer facility) are located to minimize the route between the handling and storage facilities, minimize other traffic on the route, remain within controlled areas, and maintain appropriate distances between radioactive materials and other site functions and work stations.

- Multiple Restricted Areas—The controlled area contains multiple restricted areas to limit access to areas with elevated radiation levels that would pose unacceptable risks or exposures to workers within those areas.

- Controlled Area and Perimeter Distance—Radioactive material-handling and storage functions are separated from other functions on the site. Distances are maximized, to the extent practical, between radioactive material and both the boundary of the controlled area and the adjacent onsite work stations outside the restricted area.

10A.4.2.2 *Access Control*

Access to controlled and restricted areas is controlled for the purposes of radiation protection as well as safeguards and security. This section addresses the control of access for purposes of limiting exposure to external radiation and radiological contamination hazards.

In consideration of the provisions of 10 CFR 73.21(b) on information to be protected, the description of the DSF design should include the following access control elements:

- site layout to scale showing the DSF controlled area and its boundary (given 10 CFR 72.106, "Controlled area of an ISFSI or MRS," criteria) and any traversing right(s) of way
- description of the barrier(s) used to preclude ready access to the controlled area
- location and summary description of individual and vehicular access gates and security overlook stations

The SAR should identify the criteria used to designate restricted areas (or zones within restricted areas). It should describe all protective features designed to limit access to restricted areas, including physical barriers, locked entryways, and audible or visible alarm signals. The SAR should also describe continuous direct or electronic surveillance used to prevent unauthorized entry.

Restricted areas may require further designation as high or very high radiation areas (per the definitions in 10 CFR 20.1003, "Definitions") and be controlled according to 10 CFR 20.1601, "Control of access to high radiation areas," and 10 CFR 20.1602, "Control of access to very high radiation areas," respectively, and the requisite postings in accordance with requirements in 10 CFR Part 20, Subpart J, "Precautionary Procedures." RG 8.38, "Control of Access to High and Very High Radiation Areas of Nuclear Power Plants," provides guidance on access control features applicable to these areas.

Restricted areas may be further divided to identify areas where the potential for contamination exists. The SAR should identify criteria used to designate contamination control areas (including airborne radioactivity areas). Such criteria, and facility features and operational considerations used to meet them, should be developed, designed, and implemented in compliance with the requirements in 10 CFR 20.1406 "Minimization of contamination." Access control features applicable to contamination control areas may include the following:

- incorporation of access control features and equipment into the designs of the facility's buildings or provisions to use temporary or mobile-type access control features and equipment immediately adjacent to the confinement barrier of the potentially contaminated area
- gender-designated change rooms, including lavatories and showers; provisions for personal protective equipment; stations for detecting and monitoring hands, feet, and whole body for contamination; and locations of designated stepoff pads or threshold stations used for the removal of personal protective equipment upon leaving controlled areas

- shower and lavatory water collection and storage, and provisions for routing of potentially contaminated water to treatment, storage, and monitoring systems

Useful information to consider for assessing compliance with 10 CFR 20.1406 in minimizing contamination appears in RG 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," and NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 12.3–12.4, "Radiation Protection Design Features."

The SAR should document that appropriate measures are provided for the collection of possibly contaminated wash water and that leakage of possibly contaminated liquid onto or into the ground is precluded. The SAR should explain in detail the systems or design features (including their functions) included in the facility design to fulfill these measures, with drawings showing the locations of these systems or features in the design. Wash water may include liquids temporarily stored pending sampling and sample analysis before being released to the sanitary sewer (in accordance with 10 CFR 20.2003, "Disposal by release into sanitary sewerage"); collected, treated, monitored, and held as radioactive waste in designated tanks; or treated, monitored, and released in surface bodies under the provisions of 10 CFR 20.1301, "Dose limits for individual members of the public," 10 CFR 20.1302, "Compliance with dose limits for individual members of the public," Table 2, "Effluent Concentration," Column 2, "Water," of Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," to 10 CFR Part 20, and in Footnote 4 of Appendix B to 10 CFR Part 20 in applying the sum of the ratios for radionuclide mixtures.

10A.4.2.3 *Radiation Shielding*

The DSF design should incorporate, and the SAR should describe, provisions for effective shielding as an integral part of the ALARA and radiation protection programs to protect the public and workers against direct radiation. The SAR descriptions should include special protective features that use shielding, geometric arrangement (including separation), or remote handling to ensure exposures will meet ALARA objectives. The SAR should describe the materials of construction and penetrations of facility SSCs and features relied upon for shielding. The SAR should include descriptions of the use of portable shielding, berms, or special buildings at the site that are used for shielding, if applicable.

SRP Chapter 6, "Shielding Evaluation," provides guidance for conducting detailed engineering evaluations aimed at determining the performance and effectiveness of the proposed shield design. However, Section 10A.4.1 above provides criteria for determining whether the proposed shielding and installation designs satisfy dose and ALARA requirements. The radiation protection review uses dose rate estimates from the shielding review in combination with estimates of radionuclide release rates or doses from effluents (from Chapter 9, "Confinement Evaluation," and Chapter 13, "Waste Management Evaluation" of this SRP) to ensure that combined doses (i.e., from all sources and exposure pathways) meet the acceptance criteria, as described in Sections 10A.4.3.2 and 10A.4.3.3 below.

10A.4.2.4 *Confinement and Ventilation*

Confinement refers to the ability of the DSF to prevent the release of radioactive materials from controlled areas (e.g., fuel handling, loading, and unloading areas) and SSCs (e.g., containers), in which these materials are contained, into other areas of the facility and the surrounding

environment. Confinement barrier systems may be sealed, as in the case of the facility's storage containers, or vented with off-gas treatment systems, as in the case of the facility's waste management systems. For the latter, intake and exhaust filters and dampers, as well as portions of ducts and stacks of ventilation systems, function as elements of the confinement system. Together, confinement and ventilation function to protect personnel and the public against radiation exposures associated with releases of radioactive materials under normal conditions, anticipated occurrences, and accidents.

Chapters 9 and 13 of this SRP address the evaluation of the confinement and ventilation systems' performance and effectiveness and resulting radionuclide release rates and doses from effluents. These considerations are included in the evaluation of compliance with regulatory dose requirements, including maintaining exposures and releases ALARA.

Area Monitoring and Effluent Monitoring Instrumentation

The SAR should describe the locations, types, capabilities, and operational parameters of fixed-area radiation monitors and equipment, such as continuous airborne monitoring instrumentation, used to control and monitor releases of radioactive materials in liquid and gaseous effluents. The SAR descriptions should include appropriate details in the drawings and specifications defining the DSF design. The operational parameter descriptions should include the range, sensitivity, reliability, accuracy, performance testing, energy dependence, calibration methods and frequency, alarms and alarm setpoints (including criteria and methods for determining those setpoints), limits for action, readouts, release paths to be monitored, sampling frequency, and locations for sampling line pumps and obtaining samples from effluent monitors. The SAR should describe the operational personnel's intended responses to alarms and emergency conditions.

For a DSF, the NRC accepts, to the extent applicable, the criteria and guidance for such equipment and monitoring that are described in the following documents:

- American National Standards Institute (ANSI)/Health Physics Society (HPS) N13.1, "Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities," as it relates to principles for obtaining valid samples of airborne radioactive materials and acceptable methods and materials for gas and particle sampling

- ANSI/America Nuclear Society (ANS)/HPS Standards Committee 6.8.1-1981, "Location and Design Criteria for Area Radiation Monitoring Systems for Light Water Nuclear Reactors," as it relates to the criteria for locating fixed continuous area gamma radiation monitors and for design features and ranges of measurement

- NUREG-0800, Section 11.5, "Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems"

- RG 1.13, "Spent Fuel Storage Facility Design Basis"

- RG 1.21, "Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste"

- RG 4.1, "Radiological Environmental Monitoring for Nuclear Power Plants"

- RG 8.25, “Air Sampling in the Workplace,” as it relates to use of fixed and portable air samplers in the workplace

The following documents contain criteria and guidance that also may be useful in relation to monitoring and monitoring equipment for a DSF:

- NCRP Report No. 57, “Instrumentation and Monitoring Methods for Radiation Protection”
- NCRP Report No. 112, “Calibration of Survey Instruments Used in Radiation Protection for the Assessment of Ionizing Radiation Fields and Radioactive Surface Contamination”
- NCRP Report No. 169, “Design of Effective Radiological Effluent Monitoring and Environmental Surveillance Programs”
- NUREG-0800, Section 11.2, “Liquid Waste Management System”
- NUREG-0800, Section 11.3, “Gaseous Waste Management System”
- RG 4.15, “Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination)—Effluent Streams and the Environment”

Classification of auxiliary power sources for monitoring instrumentation as “emergency” (for SSCs important to safety) or “standby” (for SSCs not important to safety) should correspond to the classification of the instrumentation itself. The following describes discriminators in classifying instrumentation and auxiliary power sources as important to safety:

- If data provided by the monitoring system can have an immediate and determining effect on personnel actions and operations to maintain compliance with established safety criteria and limits, including prevention of unacceptable doses to workers, then monitoring instrumentation should be classified as emergency.
- If any of the following is true, then the instrumentation and its auxiliary power source may not be important to safety:
  - Instrumentation data are not provided in real time to a central control room or, if provided, do not trigger an alarm that results in actions that should preclude or mitigate unacceptable consequences.
  - Instrumentation does not trigger an alarm necessary to avoid unacceptable worker exposures at its location when a setpoint threshold is reached.
  - Data are collected only periodically.
  - No normal, off-normal, or accident events or conditions can result in changes in the monitored phenomena that can jeopardize satisfaction of safety criteria and limits.

10A.4.2.5 *Radiological Environmental Monitoring Program*

The SAR should describe the radiological environmental monitoring program for the facility. A licensee uses the program to verify compliance with the 10 CFR 72.104(a) dose limits during DSF

operations. The program employs a combination of methods, as appropriate, including direct radiation measurements (such as thermoluminescent or optically stimulated luminescent dosimeters) and sampling and analyses of gaseous and liquid effluents and environmental samples.

The SAR description of the radiological environmental monitoring program should include information regarding the exposure pathways that will be monitored. The monitored pathways should include the pathways that lead to the highest potential external and internal radiation exposures of individuals that result from DSF operations. The programs should be designed to provide data on exposures and radionuclide concentration levels for those exposure pathways. The SAR should identify the sample types (e.g., water, soil, vegetation), number of samples, sample locations, collection frequency, and sample analysis to be performed along with its frequency. The SAR should include a map of suitable scale that identifies the sampling locations to show distance and direction of monitoring stations, with release points and relevant boundaries (e.g., controlled area boundary, site boundary) also indicated on the map. The SAR description should include the program for continuing meteorological data collection and evaluation to supplement the estimates of individuals' external and internal radiation exposures developed in accordance with Section 10A.4.3.3 below. Additionally, the SAR description of the radiological environmental monitoring program should also include the approach for determining background levels and the contribution of the facility's incremental releases to background levels. The SAR should include the results of the background level determination.

10A.4.3 Radiation Exposures and Dose Assessment

The SAR should provide dose estimates and describe the methods and means, including all assumptions and bases, used to derive dose estimates for occupational workers, members of the public located at or beyond the controlled area boundary, members of the public using public access facilities (e.g., highways, railways, waterways) that traverse the controlled area, and nonradiation worker facility personnel and others that may access the site (e.g., carriers involved in shipments of materials to/from the facility, construction workers brought onsite for building additional storage pads). The dose estimates should include individual and collective doses from direct radiation exposures and effluent releases.

It should be noted that there is considerable overlap in the information presented in the SAR between this section and the section describing radiation protection design features. The overlap offers a dual purpose and benefits. From the shielding evaluation (SRP Chapter 6), estimated dose rates for direct radiation should be provided for representative points within the controlled area (as defined in 10 CFR Part 72) and any restricted areas (as defined in 10 CFR Part 20), as well as on and beyond the boundary of the controlled area. Additionally, the confinement evaluation (SRP Chapter 9) and site-generated waste management evaluation (SRP Chapter 13) should have produced estimates of radioactive materials (radionuclide concentrations) present in effluents and dose (rate) estimates from effluents. Accordingly, the radiation protection evaluation includes a dose assessment that incorporates results of each of these evaluations, as applicable. The major elements of the dose assessment and the applicable acceptance criteria are described below.

10A.4.3.1 *Basis and Assumptions of Dose Assessment*

The applicant should provide sufficient information describing and justifying the bases, models, and assumptions applied in estimating all doses. This description should identify all exposure pathways, locations and occupancy (or residence) times with their bases, essential parameters

and their selected values, sources of the data for these values (site specific, default from the NRC, or industry guidance), computer codes and software version, and dose results. For any codes used, the SAR should provide information to demonstrate the validation of the codes in a manner similar to what is described in Sections 6.4.4.1 and 6.5.4.1 of this SRP. The discussion of dose results should address the degree of conservatism applied in all assumptions and parameters, whether any part of the dose assessment was modified in light of the results of separate sensitivity analyses, and conclusions in demonstrating compliance with the NRC regulations and acceptance criteria. If results for individuals located at or beyond the 10 CFR Part 72 controlled area boundary are marginally close but still in compliance with the NRC dose criteria in 10 CFR Part 20 and 10 CFR Part 72, the SAR should describe the direction and magnitude of underlying uncertainties, given all assumptions, in providing reasonable assurance that such doses represent conservative bounding estimates. If results approach or exceed the appropriate limits for individuals onsite (within the 10 CFR Part 72 controlled area boundary), the SAR should describe the controls, conditions or other means by which the licensee will ensure individual doses will not exceed the appropriate limits. Chapter 9 of this SRP contains additional guidance regarding the information the SAR should contain related to analyses of doses from effluents or releases from the storage containers. That guidance may also be useful for identifying the information that the SAR should contain related to analyses of doses from effluents or releases from the site-generated waste management systems (discussed in Chapter 13 of this SRP).

10A.4.3.2 *Onsite Dose*

The SAR should provide the objectives and criteria for design dose rates for the various areas of the facility. Individual and collective doses should be calculated for all onsite areas at which workers will be exposed to elevated radiation levels (e.g., greater than 2 millirem per hour (mrem/hr) (0.02 millisieverts per hour (mSv/hr)) or airborne radioactivity concentrations during normal operation and anticipated occurrences. The dose estimates should be based on direct exposure and inhalation of airborne radioactivity and should be derived for workers performing specific DSF functions, including routine, contingency, maintenance, or repair procedures or other activities that can occur in areas with elevated dose rates. Individual and collective doses should also be determined for onsite functions outside the DSF restricted areas associated with package receipt and with package preparation and transfer to conveyance for shipment of the radioactive materials to be stored at the facility.

The SAR should include estimates of occupancy times for personnel involved in these functions, including the maximum expected total hours per year for any individual and total person-hours per year for all personnel. The annual collective doses associated with each major function and each radiation area should be estimated. Individual doses to workers should be well below the dose limits specified in 10 CFR 20.1201, “Occupational dose limits for adults.”

Collective doses should be consistent with the objectives contained in the applicant’s ALARA program. The information provided by the applicant should allow for the determination of compliance with these criteria. In general, the following information will allow for such a determination:

- The SAR should identify and list collective and individual doses associated with all operations involved with placing one full storage container in the storage position according to the associated function.

- The SAR should provide estimates of the annual collective and individual doses by multiplying the single-storage container dose by the maximum annual placement rate of

containers into storage. This estimation assumes that the same personnel will be involved in the same operations for each container. If the doses exceed those allowed by 10 CFR 20.1201(a), the planned conduct of operations (SRP Chapter 12) should include conditions (e.g., staffing plan, monitoring) that will ensure that 10 CFR 20.1201(a) dose limits are not exceeded.

- The SAR should provide estimates of annual doses for operation of the DSF for material in storage and material in wet holding or wet storage for comparison with maximum allowable doses given in 10 CFR 20.1201.

- The SAR should include a discussion of sensitivity of dose results to assumptions and uncertainties, including the use of conservative parameters.

Depending on the applicant's proposed conduct of operations (see SRP Chapter 12), not all facility personnel may necessarily be radiation workers. This may include administrative staff among others. In addition, carrier personnel involved in the shipments of materials to or from the site may have access to the controlled area but not be radiation workers. For these individuals, the 10 CFR Part 20 dose limits for members of the public apply, and the onsite dose evaluations should address the ability to meet those limits, which are given in 10 CFR 20.1301. The applicant should describe and justify the bases and any assumptions used in the evaluation. The SAR should include a description of any administrative controls the applicant will use to ensure that the bases of the evaluation and assumptions remain valid during facility operation.

10A.4.3.3 *Offsite Dose*

Dose rates and doses should be controlled so that doses in any unrestricted areas, which include areas beyond the controlled area boundary, do not exceed the 10 CFR 20.1301(a)(2) limit of 2 mrem (0.02 mSv) in any single hour from external sources from all licensed activities at the site and the 10 CFR 20.1101(d) constraint on airborne radioactive material emissions of 10 mrem (0.1 mSv) total effective dose equivalent (TEDE) per year.

For normal operations and anticipated occurrences, the estimated dose to any real individual located at or beyond the controlled area boundary may not exceed the limits of 10 CFR 72.104(a). Note that the 10 CFR 72.104(a) dose limits are expressed as annual dose equivalent to the whole body, the thyroid, and any other critical organ.

Calculated doses must include both direct radiation and associated exposures to airborne radioactivity, such as from planned discharges of radioactive materials, if applicable (see 10 CFR 72.104(a)). The doses must also include the radiation (direct and effluent) from other activities (e.g., reactor, enrichment facility radioactive waste storage facility) in the region (see 10 CFR 72.104(a)). Assessments of doses should consider all sources of radiation and radioactivity (including effluents) and exposure pathways (external and internal) as potential contributors to doses to members of the public from all onsite facilities. Since anticipated occurrences are expected to occur at a frequency of once per year, the sum of the doses from normal operations and the bounding anticipated occurrence (that is, off-normal condition) must comply with the limits in 10 CFR 72.104(a).

Applicants may demonstrate compliance with 10 CFR 72.104(a) in one of two ways.

1. Show that an individual's dose at the controlled area boundary with full-time occupancy will not exceed the regulatory dose limits.

– OR –

2. Identify individuals within the geographical location of the DSF and estimate their maximum radiological exposures. Use this information to identify a maximally exposed real individual. Calculations may involve site-specific information, such as the number of storage containers; the container array configuration(s); the characteristics of the actual SNF, HLW, or reactor-related GTCC waste (or any combination of the three) to be stored at the facility; the site characteristics; and the surrounding topography features. Alternatively, the calculations may involve bounding parameters for each of these items. This approach should consider the current as well as potential changes in population and water and land use based upon projections of these aspects described as part of the site evaluation (SRP Chapter 2). Calculations may estimate the amount of time that a real individual spends near the facility, the distance the real individual is from the facility, and other factors that may mitigate radiological exposure to the real individual.

If the second approach is taken, then the applicant should establish measures in the radiological protection program, environmental monitoring program, and operating procedures, as applicable, to identify and periodically reevaluate potential increases in exposure to the real individual during the term of the license.

For exposures occurring under accident conditions, including design-basis accidents and natural phenomenon events, the estimated doses to any individual located on or beyond the nearest boundary of the controlled area may not exceed the limits specified in 10 CFR 72.106(b). The estimated doses should include the contributions from direct radiation and any releases that occur as a result of the accident.

If radioactive effluents from the DSF are anticipated, the applicant should provide the estimated annual collective dose (in person-rem or person-Sievert) related to the DSF. The SAR should present details on estimated radioactive effluents and models and equations used to determine doses. The applicant should also provide estimated collective doses resulting from releases under accident conditions. Doses should be based on all important exposure pathways (e.g., airborne releases) and modes of exposure (e.g., external exposure, inhalation) and should be specified as whole-body, or effective dose equivalent. In addition, the SAR should identify the organs, including critical organs, receiving the highest doses and provide their doses.

The applicant should apply a methodology that the NRC accepts, as described in applicable NRC guidance. If an application uses alternative methods and assumptions in deriving doses, the SAR should contain sufficient information for the staff to independently confirm the results presented in the SAR. This information is used to evaluate the facility's impacts in accordance with 10 CFR 72.100(a). The SAR should include appropriate justification for why these estimated collective doses are ALARA. For these analyses, the applicant should consider current, and potential changes in, population and land and water use.

The following considerations also apply to the offsite dose assessments:

- The applicant should calculate dose rates from direct radiation on the basis of the maximum quantity or inventory of radioactive materials permitted by the DSF license.

- The dose assessment should assume that radioactive materials are distributed in such a manner as to produce the highest perimeter dose rate, unless such arrangements are specifically precluded by operational considerations, license conditions, or technical specifications.

- RG 4.20, "Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees other than Power Reactors," provides guidance on methods the NRC considers acceptable for meeting the airborne emissions constraint in 10 CFR 20.1101(d).

- (Proposed) license conditions or technical specifications regarding facility design or operations that affect offsite doses (e.g., stored materials quantities and specifications, dose rate limits, contamination limits).

Additional engineering features, such as berms or shield walls, may be used to mitigate doses to real individuals near the site. However, if these features are relied upon to comply with the dose limits in 10 CFR 72.104(a) or in 10 CFR 72.106(b) for any individual, the applicant should adequately describe and analyze such features in the SAR and classify them as important to safety (at the appropriate category).

10A.4.4 Health Physics Program

The SAR should include a description of the health physics program for the proposed facility. The program's scope should be sufficiently broad to support all expected operational events, including normal operations, anticipated occurrences, and accident conditions, and demonstrate compliance with the applicable requirements of 10 CFR Parts 19, 20, and 72. Table 10A-2 lists major program elements, along with the parameters and applicable regulatory criteria and guidance documents, for each element that the applicant should describe in the SAR.

This section addresses the health physics program's organization, staffing, lines of authority, facilities (including equipment and instrumentation), and administrative policies and procedures used in implementing radiation protection functions.

The management and functions of the health physics program should be commensurate with expected radiological conditions and ranges of radiation exposure rates and doses. The DSF should have the facilities, equipment, and instrumentation necessary to ensure that the health physics program can be properly carried out and the health physics staff can adequately discharge its functions and responsibilities. In part, the evaluations described in this SRP chapter and results of evaluations described in other SRP chapters (e.g., Chapters 6, 9, 13, and 16) provide supporting information in bracketing the range of expected radiological conditions.

Table 10A-2 Program Elements of the Health Physics Program

Item	Description	Criteria
Radiation surveys	Method, frequency, and plans for conducting radiation surveys, records of surveys	10 CFR 20.1501(a) and 10 CFR 20.2103
ALARA plans	Plans developed to ensure occupational exposures will be ALARA	10 CFR 20.1101(b) RG 8.8 and RG 8.10
Access control and postings	Physical and administrative functions and measures (e.g., personnel monitoring) for controlling access to and limiting stay times in restricted and controlled areas	10 CFR 20.1601, 10 CFR 20.1602, 10 CFR 20.1702, and 10 CFR 20.1902; 10 CFR 72.126(a)(3) and 72.126(b) RG 8.38
External exposure monitoring	Monitoring criteria, types of dosimeters, collection frequency, processing, review of results (including how results are used for operational planning)	10 CFR 20.1502 RG 8.2, “Administrative Practices in Radiation Surveys and Monitoring” RG 8.4, “Personnel Monitoring Device—Direct-Reading Pocket Dosimeters” RG 8.28, “Audible-Alarm Dosimeters” RG 8.34, “Monitoring Criteria and Methods to Calculate Occupational Radiation Doses”
Internal exposure monitoring	Types of monitoring (e.g., whole-body counts, lung counts, urinalysis), monitoring criteria, procedures for estimating dose from bioassay results, and review of results	10 CFR 20.1204, 10 CFR 20.1502, 10 CFR 20.1703(c)(2), (c)(4)(i), and 20.1703(i) RG 8.9, “Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program” RG 8.26, “Applications of Bioassay for Fission and Activation Products” RG 8.34
Air sampling and analysis	Methods and procedures for air sampling and analysis, evaluation and control of airborne radioactivity, requirements and procedures for special air sampling	10 CFR 20.1204(a)(1), 10 CFR 20.1501(a)(2)(ii), 10 CFR 20.1502, 10 CFR 20.1701, 10 CFR 20.1702(a), and 10 CFR 20.1703(c)(1), (c)(4)(i) RG 8.25
Effluent releases and monitoring	Means, methods, procedures and equipment to sample, analyze, monitor, and control airborne and liquid effluents from facility systems and buildings	10 CFR Part 72.126(c) and (d)
Minimization of contamination and waste generation	Methods and procedures to monitor, control, and reduce contamination levels in facilities (including personnel, equipment, and surfaces) and waste generation	10 CFR 20.1406, 10 CFR 72.24(f), and 10 CFR 72.126(a)(1),(2),(4) RG 4.21
Respiratory protection program	Policy statement on respirator usage; respirator certification, fit-testing, and usage; medical surveillance of respirator users	10 CFR 20.1702 and 10 CFR 20.1703 RG 8.15, “Acceptable Programs for Respiratory Protection”

Item	Description	Criteria
Radiation protection	Requirements for initial and refresher training, contents (topics), health physics-related qualification of workers	RG 1.8, "Qualification and Training of Personnel for Nuclear Power Plants" RG 8.2 RG 8.27, "Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants" RG 8.29, "Instruction Concerning Risks from Occupational Radiation Exposure"
Pregnant worker protection	Provisions to inform female workers of fetal protection requirements, to monitor fetal dose, and to provide alternatives to minimize fetal dose	10 CFR 20.1208 RG 8.13, "Instruction Concerning Prenatal Radiation Exposure"
Instrument QA	Requirements and procedures for calibration, maintenance, and care of radiation detection, monitoring, and dosimetry instruments and records	10 CFR 20.1501(b), 10 CFR 20.1501(c),(d) and 10 CFR 20.2103
Recordkeeping and reports	Preparing of reports and records for health physics program contents and audits, surveys, calibrations, personnel monitoring results	10 CFR Part 20, Subpart L, "Records," and Subpart M, "Reports"

10A.4.4.1 *Organization and Staffing*

RG 8.8, RG 8.10, and NUREG-0800, Section 12.5, "Operational Radiation Protection Program," include guidance applicable to the organization and planning for health physics (radiation protection) activities at a DSF. The DSF management organization should identify an individual with clearly designated responsibilities for health physics. To avoid the potential for conflict of interest, this individual's reporting line should not include facility managers responsible for the operation of the DSF. The health physics manager and facility health physics staff should have the authority to supervise, monitor, and halt facility operations and procedures that could result in unnecessary radiation exposures to workers and members of the public or lead to doses in excess of administrative limits and NRC regulations. The health physics manager position (or its equivalent title) should be maintained for the operational life of the facility, including all decontamination and decommissioning operations. The health physics organization should include adequate staffing with appropriate experience, training, and qualifications. RG 8.2 and RG 8.8 describe acceptable programs and methods for complying with NRC requirements. RG 1.8 provides guidance for reactors that may also be useful for DSFs.

10A.4.4.2 *Equipment, Instrumentation, and Facilities*

The SAR should describe health physics program equipment, instrumentation, and facilities. The need for specific health physics equipment and facilities depends on the nature of the installation and its operations, such as whether specific laboratory functions are performed at offsite facilities.

In all cases, the program should include adequate means to properly monitor all expected operational evolutions and associated radiological conditions. The equipment should include portable and laboratory equipment, such as the following:

- personal radiation monitoring devices for external dosimetry, including provisions for dosimeter processing by a dosimetry service accredited by the National Voluntary Laboratory Accreditation Program

- an appropriate number of handheld and portable radiation survey meters and detectors for performing radiation and contamination surveys for each type of survey to be performed (e.g., Geiger-Mueller survey instruments for contamination surveys and personnel "frisking," ionization chambers for exposure rate surveys, neutron detectors for conducting neutron flux or dose rate surveys)

- methods and equipment, including radioactive sources and standards (National Institute of Standards and Technology-traceable primary and secondary), used to check the operation and to calibrate fixed and portable radiation monitoring survey equipment and laboratory radioanalytical equipment

- methods and equipment used to calibrate flow rates of air sampling equipment, including ambient air portable and fixed sampling stations and airborne effluent release points (e.g., facility stacks or building vents)

- portable air sampling equipment and airborne radioactivity monitors

- facilities for internal radiation monitoring, including whole-body counters, thyroid counters, bioassay sample collection and analytical equipment

- personal protective equipment (including anticontamination clothing and respirators certified by the National Institute for Occupational Safety and Health, Mine Safety and Health Administration)

- designated areas and facilities to inspect, maintain, clean, and store equipment and the means to test personnel for respiratory qualification and fitness

- decontamination equipment and facilities, including spill control materials, shower, eyewash, changing facilities

- area radiation monitoring equipment

- laboratory facilities and equipment for radioactive materials and sample analyses

- contamination control and monitoring equipment and areas

The SAR should describe the types of radiation detectors and monitors, numbers, locations, operational sensitivity and range, and frequency and methods of calibration for all of the equipment and instrumentation identified above.

Health physics facilities can be set up in permanent structures, temporary buildings, or trailers. Facilities should be located outside restricted areas and, if practicable, away from areas with elevated external dose rates and potential sources of airborne radioactivity. Exceptions can

include facilities for storing items that need to be readily available within restricted or elevated dose rate areas, as well as personnel decontamination, shower, and changing facilities. The site plot drawings of the installation should identify and describe the health physics facilities to sufficiently demonstrate the applicant's understanding of the associated requirements and operational functions.

The following regulatory guides and industry standards provide information, recommendations, and guidance on various aspects of health physics equipment, instrumentation, and facilities. The NRC considers these sources as acceptable guidance for describing the basis for implementing activities to comply with applicable regulatory requirements:

- ANSI/HPS N13.1
- RG 8.2
- RG 8.4
- RG 8.25
- RG 8.28
- NUREG-0800, Sections 11.5 and 12.5
- NCRP Report No. 57
- NCRP Report No. 112

10A.4.4.3 *Policies and Procedures*

Under 10 CFR 20.1101, "Radiation protection programs," licensees are required to "develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities." The SAR should describe the radiation protection program, including details of all health physics-related policies and procedures to be implemented at the DSF, including an annual review of the program content and implementation.

In addition to the regulatory guides identified in Table 10A-2, the following documents contain applicable guidance and criteria for health physics procedures relevant to DSF operations:

- ANSI/HPS N13.6, "Practice for Occupational Radiation Exposure Record Systems"
- American Society for Testing and Materials (ASTM) E1167, "Standard Guide for Radiation Protection Program for Decommissioning Operations"
- ASTM E1168, "Standard Guide for Radiological Protection Training for Nuclear Facility Workers"
- ANSI/HPS N13.30, "Performance Criteria for Radiobioassay"
- ANSI/HPS N13.32, "Performance Testing of Extremity Dosimeters"
- ANSI/HPS N13.41, "Criteria for Performing Multiple Dosimetry"
- ANSI/HPS N13.42, "Internal Dosimetry for Mixed Fission and Activation Products"
- NCRP Report No. 87, "Use of Bioassay Procedures for Assessment of Internal Radionuclide Deposition," issued 1987

- NCRP Report No. 112, “Calibration of Survey Instruments Used in Radiation Protection for the Assessment of Ionizing Radiation Fields and Radioactive Surface Contamination,” issued 1991

- NCRP Report No. 116, “Limitation of Exposure to Ionizing Radiation,” issued 1993

- NCRP Report No. 127, “Operational Radiation Safety Program,” issued June 1998

- NCRP Report No. 134, “Operational Radiation Safety Training,” issued 2000

- National Safety Council, “Accident Prevention Manual: Engineering and Technology,” 14th edition, 2015.

- NUREG-0800, Section 12.5

10A.5 Review Procedures

This section describes review procedures used to evaluate (1) compliance of facility design and operations with regulatory requirements for radiation protection, (2) implementation of design and operations features and programs to ensure that exposures (public and personnel) will be ALARA and comply with regulatory dose limits, and (3) the adequacy of the applicant’s radiation protection and health physics programs for the proposed DSF. Figure 10A-1 shows the interrelationship between the radiation protection evaluation and the other areas of review described in this SRP.

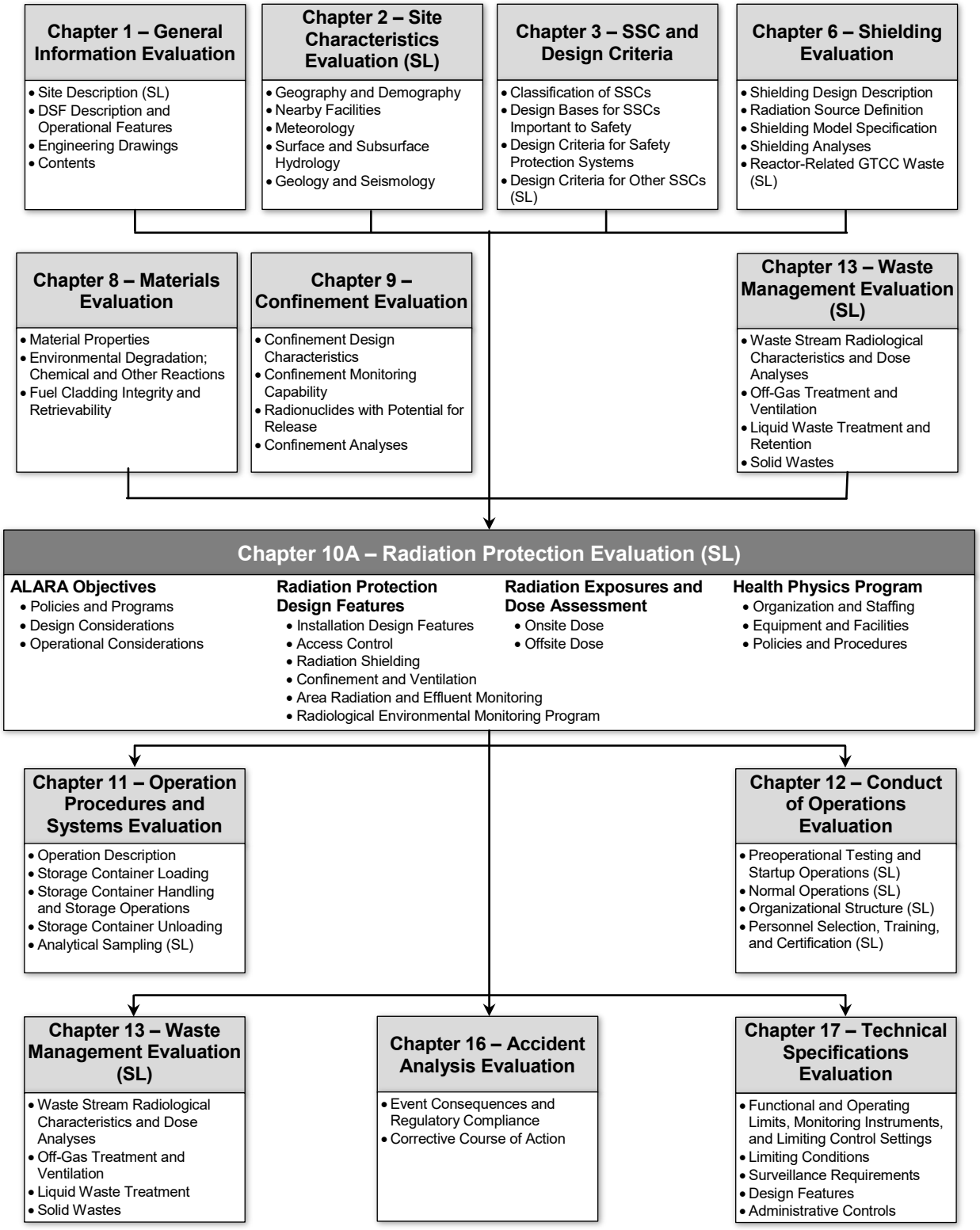

Figure 10A-1 Overview of Radiation Protection Evaluation

The radiation protection review includes evaluation of compliance with all regulatory requirements and acceptance criteria given in this SRP and other applicable NRC documents and accepted codes and standards. Always assume that such a comprehensive scope of the review applies, even though it is not further detailed or repeated in this section. Coordinate with the conduct of operations (SRP Chapter 12) reviewer to ensure that preoperational testing includes testing of design features and procedures that are significant to radiation protection and that ensure doses are ALARA.

10A.5.1 ALARA Objectives

This section provides procedures for reviewing the scope and objectives of the ALARA program in protecting workers and members of the public. The review of the ALARA program addresses policies, procedures, and facility design features that reduce radiation exposures, dose rates and doses and minimize release of radioactive materials in the environment. The review includes evaluations of compliance with all regulatory requirements and acceptance criteria given in this SRP and other applicable NRC documents and accepted codes and standards. Section 10A.5.2 also provides review guidance related to ALARA because of the significant overlap between that section and this section.

10A.5.1.1 *Policies and Programs*

Determine that an effective ALARA program and objectives will be applied to most functions associated with construction, operation, and eventual decontamination and decommissioning of the DSF. Verify that ALARA philosophies and program goals are evident throughout the SAR in the description of equipment, facility designs, and operational procedures. In addition, through discussions with reviewers of other topics, verify that other topic areas of the SAR appropriately reflect the ALARA policies (e.g., facility design and operations descriptions).

Ensure that the applicant's ALARA policy and program includes a written policy statement that expresses management's commitment to maintain exposures to workers and the public ALARA and addresses both facility design and operations. Review the proposed ALARA program organization and ensure that it identifies the organizational structure, including descriptions of responsibilities and activities of ALARA personnel. Review the ALARA policy and program content and ensure that the policy includes the elements identified in Section 10A.4.1.1 above and that the program content includes provisions for those items described in Section 10A.4.1, including the program organization and programmatic elements listed in Section 10A.4.1.1. In addition, consider the guidance of NUREG-0800, Section 12.1, "Assuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable," and RG 8.8, as they provide guidance that may be applicable to the review of an applicant's proposed ALARA program for a DSF.

10A.5.1.2 *Design Considerations*

Ensure that the facility design and layout demonstrate consideration of ALARA principles. Ensure that the design criteria (see SRP Chapter 3, "Principal Design Criteria Evaluation") also incorporate ALARA criteria in facility features. Determine whether the SAR identifies choices between otherwise comparable alternatives affected by ALARA considerations and provides sufficient bases for the selected option(s) as the most appropriate. Evaluate the design and layout for consideration of the factors identified in Sections 10A.4.1.2 and 10A.4.2 above.

10A.5.1.3 *Operational Considerations*

Determine that the descriptions of proposed operations adequately demonstrate that the applicant has incorporated ALARA principles into operational procedures. Ensure that the applicant has developed plans, methods of operation, and procedures in accordance with applicable guidance and that these items adequately address considerations detailed in Section 10A.4.1.3 above.

10A.5.2 Radiation Protection Design Features

This section addresses review procedures that apply to installation design, access control, shielding, confinement and ventilation, area radiation and effluent monitoring (including the instrumentation), and the radiological environmental monitoring program. In support of this process, NUREG-0800, Sections 12.3–12.4, also provide guidance that the NRC finds acceptable to use to review DSF radiation protection design features.

In reviewing the DSF design features and dose analyses, as described in Section 10A.5.3 below, consider whether the license (in the technical specifications) should include dose rate limits for some of the facility SSCs and features, such as the SNF, reactor-related GTCC waste, or HLW storage containers. In determining the need for such limits, consider factors such as the dose rates for different operational configurations, the nature of the DSF design, potential dose impacts of design changes, and the need for such limits to ensure continued compliance with 10 CFR Part 72 and 10 CFR Part 20 dose limits. Ensure that any dose rate limits are derived from the applicant's dose rate and dose analyses for normal (and off-normal) conditions. The limits should be developed for appropriate configurations of appropriate facility SSCs and features and aspects of these SSCs and features that are important for personnel or public doses. The limits should be compared against the maximum measured dose rates. Ensure that the license (technical specification) condition that specifies dose rate limits also specifies an appropriate number of measurements at appropriate locations on facility SSC or feature surfaces. The specified measurements (numbers, locations, SSC, or feature surfaces) should be sufficient to ensure compliance with the dose rate limits. Also consider whether the license technical specifications should include any limits and measurement requirements for (removable) contamination for appropriate facility SSCs (e.g., SNF, reactor-related GTCC waste, or HLW storage containers). Considerations should include impacts to dose estimates. Appropriate technical specifications should result in contamination levels that contribute negligibly to doses and dose rates at or beyond the controlled area boundary (that is, off site).

10A.5.2.1 *Installation Design Features*

Review the SAR installation design features and ensure that the site and facility drawings and diagrams clearly identify facility features that affect the radiation protection analyses. Ensure that the radiation protection analyses are consistent with, or are bounding for, the design of the facility and the site as described in the site and facility drawings and diagrams. The facility should be constructed in accordance with the design drawings and diagrams. Ensure that the drawings and diagrams also clearly identify any public access facilities that traverse the controlled area (e.g., as allowed by 10 CFR 72.106(c)).

For systems used to treat liquid and gaseous effluents, coordinate with the waste management (SRP Chapter 13) reviewer to review piping and instrumentation diagrams and system process flow diagrams and verify that the applicant has adequately characterized and included in its analyses the aspects of these systems relevant to the DSF radiation protection design and analyses. These aspects include all sources and volumes of liquid process and effluent streams;

points of collection of liquid wastes; flow-paths of process streams through each system, including potential bypasses; the treatment provided and expected decontamination factors or removal efficiencies for radionuclides and holdup or decay time; and points of release of liquid and gaseous effluents to the environment. With respect to potential bypasses, ensure that the applicant adequately considered improper connection to nonradioactive systems and the possibility of uncontrolled and unmonitored liquid and gaseous effluent releases.

10A.5.2.2 *Access Control*

Review the SAR description and provisions for access control and verify that (1) the facility and operational planning incorporate the necessary and desirable personnel protective measures, (2) the facility's design provisions reflect a radiological and engineering appreciation of potential dose rates and contamination levels in the transfer facilities (dry or wet (e.g., a pool)) and waste management facilities, (3) the descriptions of ALARA and other radiological protection features as well as the planning for implementation of physical protection incorporate provisions for access control, and (4) the facility design and operations incorporate the necessary means and methods (e.g., barriers, arrangements with appropriate enforcement agencies) for controlling access to controlled areas in order to ensure public health and safety.

10A.5.2.3 *Radiation Shielding*

Examine the applicant's evaluation of the facility shielding design; coordinate this review with the shielding (SRP Chapter 6) reviewer. Confirm that the applicant has identified facility design and site features that have a bearing on occupational and public doses and dose rates. These features include aspects of the facility and site that result in increased dose rates (e.g., streaming paths) as well as those that help to reduce dose rates (e.g., shield walls). Confirm that the applicant's evaluation treats these features in a manner that is consistent with, or bounding for, the facility design and site features descriptions, including the drawings and diagrams, in the chapters of the SAR that provide general information, site characteristics, and principal design criteria. Confirm that the applicant's evaluations for the different design-basis conditions (i.e., normal conditions, anticipated occurrences, and accident conditions) account for the effects of those conditions on the facility design and site features.

Also ensure that the applicant's evaluations adequately address the different configurations of the facility's features and SSCs consistent with the variety of facility operations, including those that may only be temporary. This includes, for example, construction work to expand a storage array that removes or exposes materials relied on for shielding that are not otherwise removed or exposed during normal operations. Depending upon the applicant's analyses, consider the need for any license or technical specification conditions regarding these configurations and operations. ANSI/ANS 6.4.2, "Specification for Radiation Shielding Materials," includes information that may be useful to consider as part of this review. Also confirm that the applicant's evaluations account for facility layout and the maximum quantities of SNF, reactor-related GTCC waste, and HLW that will be stored at the facility.

Examine the dose rates the applicant derived from its shielding analysis, coordinating with the shielding reviewer (SRP Chapter 6). Confirm that the evaluations produce dose rates for a sufficient number of locations to support the evaluation of the occupational doses and public doses. These locations should include surfaces of facility features and SSCs that are used to handle or store SNF, reactor-related GTCC waste, or HLW; locations of personnel conducting operations (e.g., during storage container loading, transfer, surveillance, maintenance activities); other locations on site that will be occupied by facility personnel in restricted areas and outside of

restricted areas, including both radiation worker and nonradiation worker personnel (e.g., administrative staff); locations on site of public access facilities (e.g., roads and waterways); and locations at the controlled area boundary and beyond the controlled area boundary that are needed to determine doses to real individuals around the facility. The applicant's analysis should provide sufficient dose rate information to support the evaluations for demonstrating the facility design and operations meet, or will meet, the 10 CFR Part 20 dose limits for facility personnel and the public and the limits in 10 CFR 72.104(a) and 10 CFR 72.106(b).

The shielding (SRP Chapter 6) reviewer is responsible for reviewing the applicant's shielding analysis, including the computer codes and models and calculation of dose rates, and the performance of any confirmatory shielding calculations. However, since the radiation protection review is based, in part, on the outcome of the shielding analysis, coordinate the review of this SRP chapter with the shielding reviewer to determine the adequacy and acceptability of the applicant's shielding analysis. This coordination includes confirming with the shielding reviewer that the applicant's analyses, including model parameters and assumptions, are appropriate and that the calculated dose rates are reasonable. This coordination may also include identification of the need for confirmatory calculations and determining the level of effort that should be expended in performing calculations. In addition to the considerations described in Section 6.5.4.4 of this SRP, determination of the level of effort should include consideration of the margins in estimated doses to dose limits, such as the limits in 10 CFR 72.104(a) and 10 CFR 72.106(b).

Evaluate whether the proposed shielding use is consistent with the applicant's design objectives relative to keeping radiation doses ALARA. As part of this evaluation, consider the applicant's descriptions, if any, in the SAR of the use of temporary or portable shielding, remote handling, or other protective features. Ensure that use of these features or other actions is included in appropriate sections of the SAR, such as in the descriptions of facility operations for the handling, receipt, transfer, and storage of materials. Descriptions of the facility may include placement of barriers between occupied areas and radioactive materials; use of these barriers may be for ALARA purposes.

Consider scenarios and designs where the use of additional features or SSCs or certain kinds of remote operations may not be in keeping with ALARA objectives. Such scenarios and designs include those where significant extra shielding or the use of remote operations (e.g., container movements directed from a separate area using lasers and cameras) is needed to ensure adequate protection of personnel. Designs with such features or modes of operations introduce the possibility of scenarios leading to potential significant personnel doses in the event they need to perform specific actions to recover from anticipated occurrences (e.g., crane, laser, or camera malfunction) occurring when containers are not located within the extra shielding. For such designs, ensure that the SAR includes additional information to justify that the facility design and operations are consistent with ALARA objectives. This information may include descriptions of actions taken to minimize the likelihood of such occurrences (e.g., equipment failure) or other kinds of features or operations that the applicant will use to minimize doses in such instances. Ensure that the design and operations adequately follow ALARA principles. Also consider whether conditions regarding the design or operations may be needed in the license technical specifications to ensure adequate protection, compliance with regulatory requirements (including dose limits), and adequate consideration of ALARA. These technical specifications may include verifications of correct operations, monitoring the condition of SSCs when the extra shielding is used, specifications (e.g., thicknesses) of the extra shielding features and remote operations equipment, requirements for use of these features and equipment, requirements for recovery

actions for off-normal events, preoperational testing of remote operations and equipment, and limits on the duration of high dose-rate configurations.[1]

10A.5.2.4 *Confinement and Ventilation*

The confinement evaluation (SRP Chapter 9) includes an assessment of the applicant's estimates of radionuclide releases to the environment from the SNF, HLW, and reactor-related GTCC waste containers. The waste management evaluation (SRP Chapter 13) addresses radionuclide releases from site-generated wastes. Those analyses include confinement and ventilation aspects applicable to sealed storage containers (for which releases are usually minimal) and to systems and components that are not designed to be sealed.

The radiation protection review of confinement and ventilation has two components. The first is to evaluate information from the confinement and waste management evaluations and to determine if estimates of radionuclide release rates and other site-specific information or estimates of release/effluent doses or dose rates are adequate for estimating onsite and offsite doses, as described in Section 10A.4.3 above. The second is to evaluate the protection features of the waste management facility's and other DSF facility's (e.g., pool) ventilation systems. This part of the review should identify how the confinement and ventilation system components and controls function to do the following:

- Maintain all radiation exposures and doses ALARA.
- Prevent the spread of radioactive materials and contamination between and among areas, including the possibility of unmonitored and uncontrolled releases.
- Limit the spread of radioactive materials within ventilation system(s) beyond installed filtration components (e.g., high-efficiency particulate air and charcoal filters).
- Handle process off-gases (e.g., waste treatment, venting of storage containers, venting of liquid waste collection tanks).
- Monitor off-gases, including through sample collection and analysis, in complying with effluent release limits in unrestricted areas.

Ensure that confinement and ventilation systems conform to the applicable guidance of NUREG-0800. Section 11.3, RG 1.140, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," and RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants."

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[1] The shielding design features are important for ensuring compliance with regulatory dose limits, including the limits in 10 CFR 72.104(a) and 10 CFR 72.106(b). These limits apply to all dry storage operations, including loading and unloading operations as well as storage at the ISFSI pad. Thus, for canister-based storage container designs, the limits apply to operations with the transfer cask as well. This would be true even for a DSF that is co-located with a 10 CFR Part 50 or 10 CFR Part 52 facility and loading and unloading operations occur in that Part 50 or Part 52 facility's spent fuel building. This position is consistent with the November 16, 2006, rulemaking's definition between 10 CFR Part 50 and 10 CFR Part 72 for criticality safety (see Volume 71 of the Federal Register, page 66648).

10A.5.2.5 Area Monitoring and Effluent Monitoring Instrumentation

Evaluate the applicant's description of fixed area radiation monitoring instrumentation and continuous airborne and liquid (as applicable) monitoring instrumentation, placement of such monitors, and whether the equipment includes automatic control features (such as terminating or diverting effluent releases as warranted by safety classification). Review the criteria and methods used for determining alarm setpoints. Review the information provided on the auxiliary and emergency power supply. Evaluate information and specifications on instrument range, sensitivity, accuracy, energy dependence, calibration methods and frequency, recording devices, readouts, and alarms. NUREG-0800, Section 11.5, describes acceptable guidance for conducting reviews of DSF area radiation and effluent monitoring instrumentation. NUREG-0800 Sections 11.2, 11.3, and 12.3–12.4 also include guidance that may be useful for these reviews.

The documents referenced here and in Section 10A.4.2.5 above include criteria and guidance that the NRC accepts to the extent it is applicable to the equipment and monitoring for a DSF. Confirm that the SAR demonstrates that the equipment and its placement are adequate to ensure that the DSF design and operations will meet regulatory requirements, including those in 10 CFR 72.104(b), 72.104(c), 72.126(b), 72.126(c), 72.126(d), and 10 CFR 20.1406, as well as any technical specifications regarding monitoring and effluents. The equipment should have sufficient sensitivity and response capabilities to detect the expected area dose rates and nuclide concentrations in effluents as well as changes to these parameters that would indicate a problem and require personnel actions. The equipment should be adequate for the functions for which monitoring is to be performed, detecting the types and spectra of radiation to be monitored, detecting radionuclides (considering the chemical and physical properties of the nuclides in the effluents), and performing under all required conditions. The equipment should also be adequate to ensure prompt detection of a problem (includes appropriate alarm setpoints) and enable prompt personnel response to address the problem and avoid unmonitored or uncontrolled releases and spreading of contamination to non-radiation systems and areas. In addition, the reviewer should ensure that the SAR describes the intended personnel responses to alarms and emergency conditions. Coordinate, as required, with other reviewers (e.g., conduct of operations, operating procedures, and accident analysis). Ensure that the personnel responses are reasonable to enable or ensure that the DSF design and operations will meet the relevant regulatory requirements and limits.

10A.5.2.6 Radiological Environmental Monitoring Program

Review the description and scope of the effluent and environmental monitoring program. Ensure that it considers all potential exposure pathways and provides the necessary data to identify and assess those pathways that would lead to the highest potential external and internal exposures of the offsite population (both collectively and for the maximally exposed real individual(s)). Also confirm that the program is designed to yield information and results that can be used to (1) estimate collective doses with reasonable accuracy, (2) estimate doses to offsite individuals (to ensure compliance with 10 CFR 72.104(a) and other applicable regulatory limits), and (3) assess the effectiveness of radiological controls applied to minimize effluent releases and maintain releases and offsite doses ALARA. Confirm that the program is also capable of verifying that the assumptions and bases used in the SAR dose assessments are valid and maintained during facility operations. NUREG-0800, Section 11.5, and RG 4.1 present useful guidance on the development and implementation of a radiological environmental monitoring program for DSFs. Finally, ensure that the license technical specifications include appropriate program information in accordance with 10 CFR 72.44(d) and that the program, as described in the SAR and technical specifications, is adequate to fulfill the purposes identified in 10 CFR 72.44(d).

10A.5.3 Radiation Exposures and Dose Assessment

This section addresses the review of dose assessment methods and results presented for evaluating doses to individuals and collective doses on site (i.e., within the controlled area) and off site (i.e., at or beyond the controlled area boundary) for compliance with applicable regulatory criteria. For onsite dose evaluations, ensure that the results are adequate to support evaluations for facility personnel that are occupational workers and facility personnel that are non-radiation workers (e.g., administrative staff) as well as evaluations for members of the public for facilities that include public access areas within the controlled area boundary (e.g., as allowed, in accordance with 10 CFR 72.106(c)).

Coordinate with the shielding (SRP Chapter 6), confinement (SRP Chapter 9), and waste management (SRP Chapter 13) reviewers to understand the bases for estimates of doses and dose rates and radionuclide concentrations in effluents. Coordinate with these reviewers to also ensure that the analyses adequately and appropriately consider the effects of the facility's design features and SSCs and facility operations as well as site characteristics, including layout and features, as described in the SAR. Ensure that the analyses address the effects of potential configuration changes of the storage container contents (e.g., reconfiguration of damaged fuel) under different conditions.

Ensure that the applicant considered design and operations effects that may result in configurations and conditions that only exist for limited durations, and not for the life of the facility, and are not traditionally considered or evaluated in normal facility configurations. Such configurations include scenarios where construction at a facility to expand the storage array removes materials relied on for shielding or exposes those materials to the impacts of normal, off-normal, and accident conditions that may occur during that period of time. Such configurations may also necessitate consideration and evaluation of off-normal and accident conditions that are not typically considered in DSF SARs.

Consider whether the facility SSC designs and operations could result in significant dose impacts to personnel or members of the public for anticipated occurrences and ensure that the applicant's analyses adequately account for those effects, including during recovery from the anticipated occurrences. For example, the design of an SSC may necessitate that operations be conducted remotely under normal conditions (due to significantly high dose rates), but recovery for an anticipated occurrence may require that personnel perform recovery actions near the SSC. The extended time that this configuration exists, compared with the duration under normal operations, may also impact doses to members of the public, including those evaluated for 10 CFR 72.104(a).[2]

The results of the confinement and waste management evaluations include doses and dose rates for effluents and releases from the storage containers and the facility's waste management systems, respectively. Thus, evaluation of those analyses is the purview of those reviewers. Coordinate with those reviewers to ensure that the results are sufficient to support the dose assessments described in this section.

Ensure that the dose analyses include contributions from direct radiation, effluents, and, as appropriate, surface contamination (at the levels allowed by the technical specifications). Since the evaluation of doses from surface contamination would be similar to that for effluents,

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[2] See Footnote 1 on page 10A-30 regarding applicability of regulatory dose limits to all dry storage operations.

coordinate as needed with the confinement reviewer to evaluate any contamination contributions, which technical specifications limits should make negligible for offsite doses. The guidance below regarding effluent doses and dose rates should also be understood to include, where appropriate, contributions from surface contamination.

10A.5.3.1 *Basis and Assumptions of Dose Assessment*

Review the calculation of dose rates and doses associated with radioactive releases or effluents, as needed, including applications that use computer codes to calculate these dose rates and doses, cases where the confinement or waste management reviewers need assistance, and when there's a need to evaluate effluent release rates or dose rates and doses at locations in addition to those considered in the confinement or waste management evaluations. The NRC recognizes that various computer codes are available for analyzing radiological impacts associated with releases of radioactive materials. The considerations and guidance described in Chapter 6 of this SRP for computer codes used for shielding analyses (assessment of direct radiation dose) apply to the use of these computer codes and review of their use.

As part of this review, the staff should carefully evaluate the applicability of the codes described in the application and the reasonableness of all assumptions and parameters forming the basis of each set of dose results. Examples of these codes include the following:

- MARC-1—a suite of linked computer codes used for calculating the radiological effects of releases of radionuclides to the environment developed by the United Kingdom's National Radiological Protection Board

- LINGAP and HMARC—modules of MARC-1 used to calculate the effects of an atmospheric release

- NRCDose—a code that implements the method described in RG 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I." This code expresses doses as whole body and critical organ doses

In reviewing the results of the dose assessment, confirm that the applicant has provided sufficient information describing the bases and assumptions of the dose assessment in demonstrating compliance with the NRC dose limits. As part this review, confirm the appropriateness of all selected exposure pathways, applied values for all essential parameters, sources of the data supporting the use of these values (site specific, default from NRC guidance, or from industry guidance), and computer codes and software versions. For any codes used, ensure that the SAR demonstrates the code has been properly validated for its use, in a manner similar to that described for shielding codes (see SRP Sections 6.4.4.1 and 6.5.4.1). Also, review a sample input file to verify proper entry of facility information into the code and that the applicant used proper input parameters and code features. Consider the levels of conservatism applied in all assumptions and selection of parameters, and the extent to which some of the analyses were modified in light of the results of separate sensitivity analyses, as identified by the applicant. If dose results for individuals located at or beyond the 10 CFR Part 72 controlled area boundary are marginally close but in compliance with the NRC dose criteria in 10 CFR Part 20 and 10 CFR Part 72, independently assess the direction and magnitude of underlying uncertainties to confirm that dose results represent conservative upper bounding estimates and still comply with the NRC limits and criteria when the impacts of uncertainties are taken into account.

10A.5.3.2 *Onsite Dose*

Use all relevant information to estimate total individual and collective doses DSF workers receive and determine whether applicable dose and ALARA criteria have been met. This onsite dose evaluation includes the following steps:

- Review the estimated annual occupancy times, including the maximum expected total number of hours per year for any individual and total person-hours per year for all personnel for each radiation area, including storage areas during normal operations and anticipated occurrences and ensure these times are reasonable.

- Ensure that estimates of annual doses are based on the maximum number of storage containers placed into storage in 1 year and include both direct radiation and inhalation of airborne radioactive materials, as warranted by operations.

- Ensure that descriptions of procedures that involve exposures to workers are compatible with the occupancy times and proximities assumed in the bases of dose estimates.

- Ensure that estimates of individual and collective doses are based on reasonable assumptions regarding presumption of skill levels and training, extent of care taken in managing and conducting facility operations (including nuclear safety related), presence of proper supervision and quality control, and other factors that might tend to increase doses.

- Ensure that dose calculation methods are appropriate and correctly implemented, and confirm that there is sufficient information in the SAR for the staff to conduct an independent evaluation of dose results.

Perform independent estimates of onsite collective doses. The level of effort for these estimates may depend upon various factors, such as indications that the SAR estimates may not be bounding, reasonableness of assumptions and parameters used in the analyses, and applicable considerations discussed in SRP Chapter 6, Section 6.5.4.4, "Confirmatory Analysis," regarding level of effort for confirmatory shielding analyses. Clearly identify assumptions or models that differ from those in the SAR, and discuss whether the staff's assessment of collective dose estimates support the applicant's considerations related to maintaining occupational exposures ALARA.

Compare the estimated annual individual occupational doses with the dose limits in 10 CFR 20.1201(a). If the estimated doses approach or exceed these limits, confirm that the planned conduct of operations (SRP Chapter 12) includes conditions (e.g., staffing plan, monitoring) that ensure that individual doses will be controlled and that all dose limits will not be exceeded.

Consider all relevant information presented in the applicant's evaluation for meeting the 10 CFR 20.1301 limits for members of the public, which apply to personnel that are not radiation workers (i.e., are not receiving an occupational dose as defined in 10 CFR 20.1003) and to members of the public when access to controlled areas is allowed (e.g., facilities such as those in 10 CFR 72.106(c) that traverse a controlled area). Consider the assumptions and bases of the applicant's evaluation, including actions to be taken or administrative controls to be instituted by the applicant to ensure that doses meet regulatory limits.

10A.5.3.3 *Offsite Dose*

For offsite doses, evaluate the following four principal sets of doses and the calculation methods against the relevant acceptance criteria: (1) annual collective (person-rem) dose to the surrounding population if effluents are anticipated from the facility, (2) annual dose to the maximally exposed real individual, (3) maximum hourly dose in unrestricted areas, and (4) maximum dose from any design-basis accident to any individual located on or beyond the controlled area boundary. If effluent releases are anticipated from accidents, then ensure that the applicant calculated the collective dose to the surrounding population for accidents as well for evaluations for 10 CFR 72.100(a). For each of these determinations, ensure that the applicant's analysis considers all potential exposure pathways; identifies the pathways the applicant determined to lead to the highest external and internal doses; describes the methods and data applied in assessing doses (e.g., estimated radionuclide concentrations, atmospheric dispersion and deposition parameters (both long and short term)); and provides the bases for all selected data, methods, and exposure pathways assessment. Ensure that dose contributions from other activities in the surrounding area (i.e., within the surrounding region) are also addressed in analyses of compliance with 10 CFR 72.104(a) limits, where applicable.

Consult with the confinement (SRP Chapter 9) and waste management (SRP Chapter 13) reviewers to obtain dose or dose rate estimates for effluents or releases from the storage containers and the facility's waste management systems. Obtain doses or dose rates from effluents or releases for normal, off-normal, and accident conditions (from the confinement and waste management reviewers). Ensure that the total doses from both direct radiation and effluents or releases do not exceed the relevant acceptance criteria. For the annual dose to the maximally exposed real individual, the total of the annual doses from normal conditions and bounding doses from anticipated occurrences, together with doses from other facilities in the region, should not exceed the limits in 10 CFR 72.104(a). If the confinement or waste management analyses only provide effluent dose results at 100 meters (328 feet) or for only a single storage container (confinement only), coordinate with these reviewers to (1) evaluate how effluent releases may contribute to doses at additional distances and for the full array(s) of storage containers to be allowed by the proposed license and (2) determine what additional analyses may be needed in the SAR. For analyses where the applicant chooses to demonstrate compliance with the limits in 10 CFR 20.1301, using the option described in 10 CFR 20.1302(b)(2), coordinate review of the analysis results with the confinement and waste management reviewers to ensure all criteria for that option are met.

Collective Dose to Surrounding Population

In reviewing annual collective doses attributable to direct radiation and facility effluents, ensure that the models, assumptions, and parameters that were used to estimate doses have duly considered the site's and surrounding region's characteristics (SRP Chapter 2) and facility shielding, confinement, and waste management design features (SRP Chapters 6, 9, and 13, respectively). These characteristics and features include the following:

- site layout and location of all onsite facilities and sources of radiation exposures and radioactive effluents
- land and water use, topography, and population data, both current and projected distributions, in each sector and radial distances from the site

- direct radiation exposure and dose rates as a function of sector and radial distances from the site and dose receptor locations

- meteorological data for the site and its surroundings in each sector and radial distances from the site

- radioactive material release rates, downwind dispersion, and deposition in site surroundings and at locations of identified offsite dose receptors

- engineered design features such as berms and shield walls and their configurations

Ensure that the applicant has determined a collective dose for the surrounding population and that the dose considers all important exposure pathways (e.g., direct radiation, airborne releases) and modes of exposure (e.g., external exposure, inhalation). Assess the increment by which the collective dose would be increased by the presence of any other (existing or projected) activities (e.g., fuel cycle facility) within the surrounding area or region of the proposed DSF. Ensure that the computational models or equations and assumptions used are acceptable and consistent with NRC guidance. Ensure that the data used in computer models or equations are appropriate and accurate.

Confirm that there is sufficient information in the SAR for the NRC staff to conduct an independent, confirmatory evaluation of collective doses. In performing an independent evaluation of doses, the level of effort may vary depending on several factors (e.g., large uncertainties in results, analyses use methods that are not consistent with those described in the SRP). Determination of the necessary level of effort may involve coordination with the shielding (SRP Chapter 6), confinement (SRP Chapter 9), and waste management (SRP Chapter 13) reviewers. If the SAR methods and assumptions are deemed acceptable, perform an appropriate number of confirmatory or spotcheck calculations. For all independent, confirmatory calculations, clearly identify any assumptions or models that differ from those in the SAR.

Evaluate collective dose estimates for accidents in a similar manner as that for annual collective dose estimates for normal operations, applying appropriate considerations regarding the nature of such events. A primary consideration is the limited duration of an accident event and recovery operations. These considerations should influence selection of modeling parameters and values for site characteristics used in the analysis (e.g., meteorological conditions that result in bounding doses and impacted sectors).

Determine whether the annual collective dose estimates support the applicant's considerations and conclusions in maintaining radioactive effluent releases and offsite doses ALARA for normal and off-normal conditions. Determine whether these annual collective dose estimates and the accident collective dose estimates sufficiently characterize the radiological impacts of the facility on populations in the surrounding region, in compliance with 10 CFR 72.100(a).

Dose to Maximally Exposed Real Individual

Determine whether the highest offsite dose received by a real individual is less than the limits specified in 10 CFR 72.104(a). Many of the same factors considered in the collective dose assessment are applicable to this review. Refer to the two approaches discussed in Section 10A.4.3.3 above to demonstrate compliance with dose limits and assess the implications of the approach the applicant used for its site. Ensure that the methods, including any computational models or equations and assumptions, used are acceptable and appropriate for

this analysis. Confirm that there is sufficient information in the SAR for the staff to conduct an independent, confirmatory evaluation of doses.

Evaluate the applicant's assessment of direct dose rates and radioactive material concentrations in effluents or dose rates from the effluents at locations beyond the controlled area boundary for normal operations and anticipated occurrences. Identify the location of offsite individual(s) likely to receive the highest dose from direct radiation and doses associated with releases of facility effluents. Confirm that the applicant's dose estimates include the contributions from the relevant exposure pathways for the individual(s). Confirm that the applicant has adequately and correctly identified the location(s) and individual(s) that are likely to receive the highest doses.

Assess the annual whole-body dose equivalent, as well as the dose equivalent to the thyroid and any other critical organs (other than the thyroid). The NRC has accepted the use of TEDE as a surrogate for whole-body dose equivalent. Confirm that the applicant has described the methods used for these dose calculations.

As in the collective dose assessment, perform independent, confirmatory calculations, as necessary, to verify the applicant's results and adequacy of assumptions. Clearly identify any assumptions or models that differ from those in the SAR and confirm that such assumptions and associated parameters are adequate and conservatively bounding.

Assess the increment by which the whole-body dose would be increased by the presence of other (existing or projected) activities (e.g., a radioactive waste facility) within the area or region surrounding the proposed DSF. Ensure that the combined annual dose equivalent from the DSF and the other activities does not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other critical organ.

Assess the TEDE resulting only from airborne effluent releases, and ensure that this dose does not exceed the 10-mrem (0.1-mSv) per year ALARA constraint level in 10 CFR 20.1101(d) for an individual member of the public likely to receive the highest dose from effluents. Confirm that all supporting assumptions and associated parameters are adequate and conservatively bounding.

Review the applicant's determination that the maximum dose in any unrestricted areas resulting from external sources does not exceed 0.002 rem (0.02 mSv) in any single hour (10 CFR 20.1301(a)(2)), and determine whether the distance to the nearest boundary of the controlled area is sufficient to ensure compliance with this dose standard. Confirm that all supporting assumptions and associated parameters are adequate and conservatively bounding.

Determine whether the highest offsite dose from accident conditions is less than the limits in 10 CFR 72.106(b). Confirm that the applicant has calculated the dose consequences for each accident condition. Also confirm that the applicant calculated the doses for locations on the nearest boundary of the controlled area. Ensure that the doses include the contributions from direct radiation and releases resulting from the impacts of the accident conditions on the affected facility SSCs and features (e.g., SNF, reactor-related GTCC waste, or HLW waste containers, and waste management systems). Finally, ensure that the doses for each accident condition are based on a reasonably bounding or conservative time duration that includes recovery from the accident condition's impacts (e.g., moving SNF from a damaged container to a configuration consistent with normal operations). The NRC has accepted 30 days as sufficient for previous applications. Ensure that the applicant's selected timeframe is appropriate based on considerations unique to the facility design and operations.

10A.5.4 Health Physics Program

This section addresses procedures to review the scope, functions, and capabilities of the health physics program. NUREG-0800, Sections 11.5, 12.1, 12.5, 13.2.2, "Non-Licensed Plant Staff Training," and 13.4, "Operational Programs," provide guidance that may be useful and applicable in reviewing DSF health physics programs. Moreover, NUREG-1736, "Consolidated Guidance: 10 CFR Part 20 – Standards for Protection Against Radiation," provides additional NRC guidance on the implementation of a health physics program and radiation protection.

10A.5.4.1 *Organization and Staffing*

Evaluate the administrative organization of the applicant's health physics program. As part of the review, confirm that the program describes the authority, responsibility, experience, and qualifications of the personnel responsible for the health physics program and that the program is sufficiently staffed to conduct all program operations. Confirm that the health physics manager and health physics staff have the authority to supervise, monitor, and halt any facility operations and procedures that could result in unnecessary radiation exposures to workers and members of the public or lead to doses in excess of administrative limits and NRC regulations. Ensure that the organization and staffing description satisfactorily addresses the other criteria provided in Section 10A.4.4.1 above. Some of this information may be described in the SAR chapter on conduct of operations; thus, coordinate with the conduct of operations reviewer (SRP Chapter 12, particularly Sections 12.4.1.1–2, 12.4.6.1–2, 12.5.1.1–2, and 12.5.6).

10A.5.4.2 *Equipment Instrumentation and Facilities*

Review the applicant's description of the portable, fixed, and laboratory equipment and instrumentation for performing radiation and contamination surveys, sampling airborne radioactive materials in ambient facility areas and release points (e.g., building vents or liquid discharges into onsite or offsite surface water bodies), monitoring area radiation, and monitoring personnel exposures during normal operations, anticipated occurrences, and accident conditions.

With respect to operational descriptions and functions of radiation monitoring equipment, confirm the types and locations of annunciations and alarms and actions each type of instrumentation initiates. Confirm that once tripped by an alarm setpoint, the instrumentation system properly initiates and completes the expected action, such as providing local and remote audio and visual warnings, and, if so equipped, terminating or diverting a release or process stream to appropriate systems.

Confirm that the SAR indicates that an appropriate number of survey instruments will be available for all facility radiation monitoring functions and types of radiation surveys to be performed (e.g., Geiger-Mueller survey instruments for contamination surveys, release of equipment and tools from controlled areas, personnel "frisking;" ionization chambers used in external radiation exposure rate surveys; neutron detectors used to determine neutron flux or dose rates).

In supporting the implementation of surveys requiring sample collection, confirm that sampling and analytical equipment will be provided to collect and analyze the spectrum of expected samples, including gases, water vapors, water, wastes (e.g., dry, solid, and wet), wipes or smears, filters and absorption media, bioassays, and environmental media (e.g., soil, sediment, air, water, and biota, as described in the radiological environmental monitoring program). RG 4.1 provides supporting details for assessing compliance.

The guidance in Section 10A.5.2.5 above lists criteria to consider in this evaluation, as applicable, in addition to the criteria presented here and in Section 10A.4.4.2.

10A.5.4.3 *Policies and Procedures*

Review the applicant's plans and procedures to ensure that provisions have been made for the following:

- controlling, storing, securing, and moving radioactive materials on site, including radioactive wastes, contaminated equipment and tools, and calibration sources and standards the health physics program uses

- physical and administrative measures aimed at ensuring that occupational doses are ALARA and are within administrative limits and NRC requirements and criteria

- radiation monitoring equipment calibration and maintenance, including systems used at fixed monitoring locations, portable radiation survey equipment, fixed and portable air sampling equipment, liquid and gaseous effluent monitoring (process and release points), analytical laboratory equipment (operational samples and bioassays), and personal dosimetry devices, including documentation of National Voluntary Laboratory Accreditation Program certification when using third-party commercial dosimetry services

- personal protective equipment maintenance, inspection, and issuance and qualification and testing of fitness for the use of respiratory equipment

- records of waste management activities, including compilation of inventories of radioactive materials by physical and chemical forms, quantities (volumes or weight), radioactivity levels (according to radionuclide distributions and concentrations present in such wastes, materials, or calibration sources), and disposition (onsite or offsite storage, processing by waste brokers, shipped for offsite disposal, equipment sent out for refurbishment, or returned to manufacturer for disposition)

- retention of records for personnel dosimetry results, bioassays, radiation surveys, personnel qualification and training, personal qualification of respiratory fitness, data on radioactive sources and standards (National Institute of Standards and Technology traceable primary and secondary) used in implementation of health physics program, instrument and sampling equipment calibration methods and results, and data on radiological events that would support the planning and decommissioning of the facility, whenever initiated

Measurement Methods and Analyses

Review the applicant's methods to convert raw instrumentation readings into meaningful radiological results to use in assessing radioactivity levels, concentrations, exposure rates, and doses to confirm compliance with the criteria identified in this chapter and in the regulatory requirements. These methods may include reliance on the use of easy-to-detect surrogate radionuclides to identify the presence and determine the concentrations of hard-to-detect radionuclides. The methods may also include radiological determinations using gross

beta-gamma or alpha concentrations to infer the concentrations of specific radionuclides. Ensure that the selected methods are appropriate for the analyses for which they will be used and that they are based on sound principles.

Equipment Calibration and Maintenance

Review the program descriptions for calibrating and maintaining survey equipment, area radiation monitors, continuous airborne monitors, effluent monitors, and laboratory equipment. Consider descriptions of instrumentation calibration methods and procedures in confirming instrumentation response characteristics, sensitivity levels and detection limits, and detection ranges for facility-derived radionuclides expected during normal operation, anticipated occurrences, and accident conditions. Compare the types, levels, energy spectra, and, for radioactive materials, concentrations described as the design basis of the facility to the methods described for specifying the types and ranges of radiation monitoring instrumentation. When two or more radiation-monitoring systems are used for routine operations or accident monitoring in a single system (e.g., area radiation monitoring or an effluent release point), ensure that the SAR describes the differences in instrumentation response characteristics over their overlapping operational ranges and expected radionuclide distributions and concentrations. Confirm that the calibration and maintenance methods and program are adequate and appropriate to ensure that monitoring and laboratory equipment will perform properly for the characteristics of the radiation and radioactive materials they are used to detect, measure, and analyze.

10A.6 Evaluation Findings

The NRC reviewer should prepare evaluation findings upon satisfaction of compliance with the regulatory requirements in Section 10A.4 of this SRP. Such a review includes coordination with other reviewers to make determinations on aspects such as radiation exposure rates, doses, and releases of airborne radioactive materials. If the documentation submitted with the application fully supports positive findings for each of the regulatory requirements, the statements of finding should be similar to the following, as applicable:

F10A.1 The SAR includes adequately detailed descriptions of the [DSF designation] SSCs' design and operation characteristics, including design criteria and design bases for the radiation protection evaluation and the radioactive materials to be stored at the facility, in compliance with 10 CFR 72.24(b), 10 CFR 72.24(c), 10 CFR 72.24(l), 10 CFR 72.120(a), 10 CFR 72.120(b), and 10 CFR 72.120(c). The SAR also includes evaluations of the performance of the facility's SSCs important to safety with respect to radiation protection, in compliance with 10 CFR 72.24(d).

F10A.2 The SAR includes descriptions that establish the owner controlled area and the controlled area boundary for the [DSF designation] in accordance with 10 CFR 72.106(a). The descriptions show the boundary meets the minimum distance requirements in 10 CFR 72.106(b). The SAR also describes effective and appropriate arrangements to adequately protect public health and safety and adequately control traffic on public access facilities (e.g., highways, railroads, or waterways) that traverse the controlled area in compliance with 10 CFR 72.106(c).

F10A.3 The design and operating procedures of the [DSF designation] provide acceptable means for controlling and limiting occupational radiation

exposures within the limits given in 10 CFR Part 20 and for meeting the objective of maintaining exposures to meet ALARA objectives, in compliance with 10 CFR 72.24(e).

F10A.4 The SAR provides reasonable assurance that the activities authorized by the license can be conducted without endangering the health and safety of the public and that the operations procedures are adequate to protect health and minimize danger to life or property in compliance with 10 CFR 72.40(a)(5) and 10 CFR 72.40(a)(13).

F10A.5 [If appropriate] The proposed [DSF designation] is to [be on the same site as/near other, specify] nuclear facilities, [identify]. The cumulative effects of the combined operations of these facilities will not constitute an unreasonable risk to the health and safety of the public, in compliance with 10 CFR 72.122(e).

F10A.6 The SAR provides analyses showing that the cumulative effects of the combined operations of these facilities will be within the dose limits given in 10 CFR 72.104(a). These analyses include both direct radiation and effluent releases from the [DSF designation] to the general environment during normal operations and anticipated occurrences. The SAR also includes appropriate and adequate operational restrictions and limits to meet the limits in 10 CFR 72.104(a) and ALARA objectives in compliance with 10 CFR 72.104(b) and 10 CFR 72.104(c).

F10A.7 The SAR provides analyses of the doses from accident conditions at the facility in accordance with 10 CFR 72.24(m), and these analyses show these doses will not exceed the limits in 10 CFR 72.106(b).

F10A.8 The SAR provides analyses that show that the doses to members of the public will not exceed the limits in 10 CFR Part 20.

F10A.9 The SAR provides adequate evaluations that show the effects of the proposed site and facility, including effects due to operation and releases under normal and accident conditions on the regional environment and populations in the region in accordance with 10 CFR 72.100.

F10A.10 The SAR describes adequate measures that will preclude transport of radioactive materials to the environment through an aquifer over which the facility is located that serves as a major water resource in accordance with 10 CFR 72.122(b)(4).

F10A.11 The design of the [DSF designation] provides suitable shielding for radiation protection and confinement of radioactive materials under normal, off-normal (that is, anticipated occurrences), and accident conditions, in compliance with 10 CFR 72.128(a)(2) and 10 CFR 72.128(3). This includes ventilation systems and off-gas systems, continuous monitoring capability for the storage confinement systems, and HLW and reactor-related GTCC waste packaging that allows handling and retrievability without releases or exposures in excess of regulatory limits in accordance with 10 CFR 72.122(h)(3~5).

F10A.12 The facility design and operations include adequate means for controlling personnel exposures and for controlling and monitoring effluents and direct radiation, in compliance with 10 CFR 72.126.

F10A.13 The facility operations include programs, such as the health physics program, environmental monitoring program, and other pertinent programs that are needed to ensure compliance with the requirements in 10 CFR Part 20 and 10 CFR Part 72. These programs include the necessary elements to perform their intended functions, including the policies for and management commitments to the programs and their objectives.

F10A.14 The proposed license technical specifications include those items necessary to ensure adequate radiation protection in the design, fabrication, construction, and operation of the DSF SSCs in accordance with the requirements in 10 CFR 72.44(c) and to meet the requirements in 10 CFR 72.44(d).

F10A.15 The facility design and operations will, to the extent practicable, minimize contamination of the facility and the environment and generation of radioactive wastes in accordance with 10 CFR 20.1406(a) and 10 CFR 20.1406(c).

The reviewer should provide a summary statement similar to the following:

> The staff finds, with reasonable assurance, that the radiation protection design and program for the [DSF designation] meet the requirements in 10 CFR Part 20 and 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The staff also finds, with reasonable assurance, that the facility design, operations, and programs are adequate to ensure compliance with the regulatory dose limits and ALARA requirements in 10 CFR Part 20 and 10 CFR Part 72 for personnel and the public. The evaluation of the radiation protection program, facility design features, ALARA objectives, and health physics program provide reasonable assurance that the [DSF designation] will allow safe storage of SNF, HLW [applies to MRS only], and reactor-related GTCC waste. The staff reached this finding based on a review that considered applicable NRC regulations and regulatory guides, codes and standards, accepted health physics practices, the statements and representations contained in the SAR, and the staff's confirmatory analyses.

10A.7 References

10 CFR Part 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations."

10 CFR Part 20, "Standards for Protection Against Radiation."

10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste."

40 CFR Part 191, “Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes.”
Subpart A - Environmental Standards for Management and Storage

American National Standards Institute (ANSI)/American Nuclear Society (ANS) 6.4.2, “Specification for Radiation Shielding Materials.”

ANSI/ANS-Health Physics Society Standards Committee-6.8.1, “Location and Design Criteria for Area Radiation Monitoring Systems for Light Water Nuclear Reactors.”

ANSI/Health Physics Society (HPS) N13.1, “Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities.”

ANSI/HPS N13.6, “Practice for Occupational Radiation Exposure Record Systems.”

ANSI/HPS N13.30, “Performance Criteria for Radiobioassay.”

ANSI/HPS N13.32, “Performance Testing of Extremity Dosimeters.”

ANSI/HPS N13.41, “Criteria for Performing Multiple Dosimetry.”

ANSI/HPS N13.42, “Internal Dosimetry for Mixed Fission and Activation Products.”

American Society for Testing and Materials (ASTM) E1167, “Standard Guide for Radiation Protection Program for Decommissioning Operations.”

ASTM E1168, “Standard Guide for Radiological Protection Training for Nuclear Facility Workers.”

International Commission on Radiological Protection (ICRP) Publication 27, “Problems Involved in Developing an Index of Harm,” *Annals of the ICRP*, Vol. 1, Issue 4, 1977.

ICRP Publication 55, “Optimization and Decision-Making in Radiological Protection,” *Annals of the ICRP*, Vol. 20, Issue 1, 1990.

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Jones J.A. and D. Charles, “AD-MARC: The atmospheric dispersion module in the methodology for assessing the radiological consequences of accidental releases,” Chilton, NRPB-M72, (1982)

Charles D., M.J. Crick, T.P. Fell, and J.R. Greenhalgh, “DOSE-MARC: The dosimetric module in the methodology for assessing the radiological consequences of accidental releases,” Chilton NRPB-M74 (1982).

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Jones, J.A. and D. Charles, “AD-MARC: The Atmospheric Dispersion Model in the Methodology for Assessing the Radiological Consequences in Accidental Releases,” NRPB-M72. National Radiological Protection Board, Chilton, U.K

National Council on Radiation Protection and Measurements (NCRP) Report No. 57, “Instrumentation and Monitoring Methods for Radiation Protection,” 1978.

NCRP Report No. 59, “Operational Radiation Safety -Training,” 1978.

NCRP Report No. 71, “Operational Radiation Safety Training,” 1983.

NCRP Report No. 87, “Use of Bioassay Procedures for Assessment of Internal Radionuclide Deposition,” 1987.

NCRP Report No. 112, “Calibration of Survey Instruments Used in Radiation Protection for the Assessment of Ionizing Radiation Fields and Radioactive Surface Contamination,” 1991.

NCRP Report No. 116, “Limitation of Exposure to Ionizing Radiation,” 1993.

NCRP Report No. 127, “Operational Radiation Safety Program,” June 1998.

NCRP Report No. 134, “Operational Radiation Safety Training,” 2000.

NCRP Report No. 169, “Design of Effective Radiological Effluent Monitoring and Environmental Surveillance Programs,” 2010.

National Safety Council, “Accident Prevention Manual: Engineering and Technology,” 14th edition, 2015.

NRCDose, “Code System for Evaluating Routine Radioactive Effluents from Nuclear Power Plants with Windows Interface,” Version 2.3.20, Tape list C00684, PC586 14, Radiation Safety Information Computational Center, U.S. Department of Energy, Oak Ridge National Laboratory.

NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition.”

NUREG-1736, “Consolidated Guidance: 10 CFR Part 20—Standards for Protection Against Radiation.”

Regulatory Guide 1.8, “Qualification and Training of Personnel for Nuclear Power Plants.”

Regulatory Guide 1.13, “Spent Fuel Storage Facility Design Basis.”

Regulatory Guide 1.21, “Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste.”

Regulatory Guide 1.33, “Quality Assurance Program Requirements (Operations).”

Regulatory Guide 1.109, “Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I.”

Regulatory Guide 1.140, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants.”

Regulatory Guide 1.143, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants.”

Regulatory Guide 4.1, “Radiological Environmental Monitoring for Nuclear Power Plants.”

Regulatory Guide 4.15, “Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination)—Effluent Streams and the Environment.”

Regulatory Guide 4.20, “Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees other than Power Reactors.”

Regulatory Guide 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning.”

Regulatory Guide 8.2, “Administrative Practices in Radiation Surveys and Monitoring.”

Regulatory Guide 8.4, “Personnel Monitoring Device—Direct-Reading Pocket Dosimeters.”

Regulatory Guide 8.8, “Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low Is Reasonably Achievable.”

Regulatory Guide 8.9, “Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program.”

Regulatory Guide 8.10, “Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Reasonably Achievable.”

Regulatory Guide 8.13, “Instruction Concerning Prenatal Radiation Exposure.”

Regulatory Guide 8.15, “Acceptable Programs for Respiratory Protection.”

Regulatory Guide 8.25, “Air Sampling in the Workplace.”

Regulatory Guide 8.26, “Applications of Bioassay for Fission and Activation Products.”

Regulatory Guide 8.27, “Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants.”

Regulatory Guide 8.28, “Audible-Alarm Dosimeters.”

Regulatory Guide 8.29, “Instruction Concerning Risks from Occupational Radiation Exposure.”

Regulatory Guide 8.34, “Monitoring Criteria and Methods to Calculate Occupational Radiation Doses.”

Regulatory Guide 8.38, “Control of Access to High and Very High Radiation Areas of Nuclear Power Plants.”

U.S. Nuclear Regulatory Commission, “Criticality Control of Fuel Within Dry Storage Casks or Transportation Packages in a Spent Fuel Pool,” *Federal Register*, Vol. 71, No. 221, November 16, 2006, pp 66648–66657.