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Basis for Dose and Reactor Safety Design Criteria for Army Regulation AR 50–7 and DA Pamphlet

May 2025

Peter P. Lowry
Kenneth M. Thomas

Prepared for the U.S. Department of Energy
under Contract DE-AC05-76RL01830

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Summary

This report describes the basis used to develop the radiological dose acceptance and design criteria contained in the draft updates to Army Regulation 50–7 (AR 50–7) *Army Reactor Program* and its accompanying draft Department of the Army (DA) Pamphlet (PAM), *Army Reactor Program Procedures*. These criteria will apply to Army nuclear reactors that fall under AR 50–7 and its accompanying DA PAM and ensure alignment with the overall objectives of the Army Reactor Program.

The development basis for the radiological dose and design criteria supports a modern, technology-neutral, risk-informed, and performance-based approach to Army regulation of reactors and the demonstration of "adequate protection of the public." To establish these criteria that support the Army's unique operational requirements, multiple well-known and well-established standards and their supporting documentation were reviewed to ensure consistency with existing regulatory safety levels, including guidance from U.S. and international sources. These include the U.S. Nuclear Regulatory Commission's (NRC's) regulations and policy, the Canadian Nuclear Safety Commission's (CNSC's) regulatory documents, the International Atomic Energy Agency's (IAEA's) safety standards, as well as industry input that is tailored specifically to advanced microreactors.

This report walks through the key definitions and associated references used for these criteria, which are outlined in Section 2.0. Based on these definitions, the dose acceptance criteria were established for various receptors for routine reactor operations (Section 3.2), design basis accidents (Section 3.3), and beyond design basis accidents (Section 3.4). Comparisons of multiple national and international dose limits are provided in these sections. Lastly, Section 4.0 outlines the reactor safety design criteria contained in the draft DA PAM and their associated bases.

Acronyms and Abbreviations

ALARA	as low as reasonably achievable
AOO	anticipated operational occurrence
AR	Army Regulation
ARO	Army Reactor Office
ARP	Army Reactor Program
cGy	centigray(s); preferred measurement of absorbed radiation
CLW	co-located Worker
DA PAM	Department of the Army Pamphlet
DoD	Department of Defense
DOE	Department of Energy
EPA	Environmental Protection Agency
GDC	general design criteria
IAEA	International Atomic Energy Agency
k_{eff}	effective neutron multiplication factor; the time rate of change of the neutron population
LBE	Licensing Basis Event
LCO	Limiting Condition of Operation
LD50	median lethal dose
LMP	least materialized probability
LWR	light water reactor
mrem	millirem(s)
mSv	millisievert(s); a unit that measures the amount of radiation absorbed by the body
MWe	megawatt(s) electric
MWt	megawatt(s) thermal
NDC	not design criteria
NEI	Nuclear Energy Institute
NGNP	next generation nuclear plant
NLWR	non-light-water reactor
NRC	U.S. Nuclear Regulatory Commission
OCONUS	outside the continental United States
PAG	Protective Action Guide
PIRT	Phenomena Identification and Ranking Table
PRA	probabilistic risk assessment
QHO	quantitative health objective
rem	roentgen equivalent man; a unit of measurement used to quantify the biological effects of ionizing radiation on the human body

RES	radiation exposure status
RSDC	Reactor Safety Design Criteria
SADFL	specified acceptable fuel design limits
SMR	small modular reactor
SSC	system, structure, and component
Sv	sievert(s); a unit that measures the amount of radiation absorbed by the body
TEDE	total effective dose equivalent

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1.0 Purpose

This report documents the rationale behind the development and selection of the Radiological Dose Acceptance and Design Criteria contained in the updates to Army Regulation 50–7 (AR 50–7 2025), *Army Reactor Program*, and its accompanying Department of the Army Pamphlet (DA PAM), *Army Reactor Program Procedures*. The development bases for the radiological dose and design criteria support a modern, technology-neutral, risk-informed, and performance-based approach to Army regulation of reactors and the demonstration of adequate protection of the public.

Consistent with the requirement in the draft AR 50–7 (2025), these bases underpin the development of criteria in support of the Army Reactor Program (ARP) Objectives:

> ***The fundamental objectives of the ARP are to ensure reactor safety, plant reliability, radiation safety, environmental protection, and security across all life cycle functions.***

To support a technology-neutral approach, it is necessary to define risk significance in terms of technology-inclusive risk metrics rather than light water reactor (LWR) risk metrics, such as core damage frequencies, upon which the majority of current regulation is based. As such, the development of dose criteria is based on the frequencies and consequences of events and an overall consideration of public risk (e.g., societal risk goals). However, as discussed in the bases in Section 3.0 in this document, the regulation is flexible in implementation and does not require nor does it prohibit the use of probabilistic risk assessment (PRA) to demonstrate compliance with the criteria.

This document synthesizes foundational guidance from U.S. and international sources to support the Army's unique operational context. This includes U.S. Nuclear Regulatory Commission (NRC) requirements and guidance; U.S. Department of Energy (DOE) analysis and guidance; the Canadian Nuclear Safety Commission's (CNSC's) regulatory document RD-367; International Atomic Energy Agency (IAEA) safety standards and technical reports from industry to collectively inform dose and design criteria. The goal is to maintain existing regulatory safety levels while tailoring approaches specifically for advanced microreactors. This technology-inclusive approach focuses on microreactor-specific hazards and provides a robust foundation for the Army's reactor safety framework.

1.1 Background

The draft revised AR 50–7 and associated DA PAM are predicated on being applicable to a broad spectrum of reactor designs (technology neutral) that are encompassed by the definition of a Research or Test Reactor (e.g., the Fast Burst Reactor) or advanced microreactors that are capable of producing electricity, process heat, or both. Advanced microreactors are defined as reactors up to 50 MWe in the Nuclear Energy Innovation and Modernization Act, Pub. L. 115-439 (2019) with attributes such as:

- additional inherent safety features;
- lower waste yields;
- improved fuel and material performance;
- increased tolerance to loss of fuel cooling;
- enhanced reliability or improved resilience;

- increased proliferation resistance;
- increased thermal efficiency;
- reduced consumption of cooling water and other environmental impacts;
- the ability to integrate into electric applications and nonelectric applications;
- modular sizes to allow for deployment that corresponds with the demand for electricity or process heat; and
- operational flexibility to respond to changes in demand for electricity or process heat and to complement integration with intermittent renewable energy or energy storage.

As current nuclear power reactor regulatory approaches are primarily light-water reactor (LWR) based, with some considerations typically for small modular reactors (SMRs; those with power production capacity up to 300 MWe), the development effort considered both international and U.S. approaches proposed to specifically address the unique hazards and risk profiles associated with advanced reactors.

The safety approach is designed for advanced reactors rated at or below 150 megawatt-thermal (MWth), equivalent to approximately 50 megawatt-electric (MWe) for electricity generation or combined heat and power applications based on the efficiency of the power conversion systems. Considerations behind the development included:

- ***Micro-reactors Licensing Strategies*** **(ML21235A418)**. This draft white paper outlines approaches to streamline the licensing process used by the NRC for advanced commercial nuclear power microreactors (NRC n.d.).
- ***Regulatory Review of Micro-Reactors – Initial Considerations*** **(ML20044E249).** This report suggests that the licensing approach for nonpower reactors could serve as a model for microreactor applications, given similar accident source terms. As discussed in NRC (2020b), a simplified approach based more on deterministic analyses can be adequate for microreactors. Deterministic criteria, coupled with demonstrating that the NRC's safety goals are satisfied for the rare, catastrophic event applicable to microreactors, can serve as the appropriate basis for the regulatory review and licensing of microreactors. Examples of criteria that may apply for microreactors are provided in Table 1-1.

Table 1-1.
Criteria Applicable to Microreactors

Frequency Category	Acceptance Criteria for Licensing Basis Events (LBEs)
Frequent	• No barrier failure (beyond the initiating event) • No impact on fuel integrity or lifetime and safety analysis assumptions • Redundant means of reactor shutdown, unless inherent safety features achieve shutdown • Redundant means of decay heat removal, unless inherent safety features of the design achieve the function
Infrequent	• A coolable geometry is maintained • At least one barrier remains • At least one means of reactor shutdown remains functional • At least one means of decay heat removal remains functional
Rare	• Satisfies dose limits and the Commission's Safety Goals.

Source: NRC (2020b)

- ***Micro-Reactor Regulatory Issues* (NEI 2019a).** This white paper discusses the NRC draft Non-Light-Water Reactor (NLWR) Review Strategy issued in September 2019 (ML19275F299) (NRC 2019), and acknowledges that many requirements will not be applicable to microreactors because microreactors either do not include the referenced system or cannot result in the postulated accident. In this paper, NEI presents several key points relevant to establishing the dose and safety design criteria:
    - inherent and passive safety features
    - small radionuclide inventories
    - risk-informed, performance-based approach
    - insights from non-power reactor regulations and guidance (e.g. NUREG-1537)
    - defense-in-depth philosophy.
- **INL-EXT-18-51111: *Regulatory and Licensing Strategy for Microreactor Technology* (INL 2018).** This strategy outlines additional key points relevant to establishing the dose and safety design criteria:
    - compact design and reduced emergency planning zones
    - risk-informed decision-making
    - Phenomena Identification and Ranking Table (PIRT) process
    - joint licensing approach.

2.0 Key Definitions and References for Dose Acceptance and Design Criteria Bases

To establish specific criteria that would apply to Army nuclear reactors under the draft revised AR 50–7 and its associated DA PAM, multiple well-known and well-established standards and their supporting background information were consulted.

2.1 Definitions

The following key definitions are central to developing the radiological dose acceptance and design criteria bases in this report. As there are many existing and well-known, working definitions, the definitions below were specifically tailored for Army applications. Additional remarks have been added to indicate relevant considerations for the development of these bases.

active component: A component whose functioning depends on an external input such as air actuation, mechanical movement, or supply of power. Examples of active components are pumps, fans, relays, and transistors. Note: Certain components, such as rupture discs, check valves, safety valves, injectors, and some solid-state electronic devices, have characteristics that require special consideration before designation as an active component or a passive component. See the IAEA *Nuclear Safety and Security Glossary (*hereafter IAEA *Glossary*) (IAEA 2022), NUREG-2122, the NRC *Glossary of Risk-Related Terms in Support of Risk-Informed Decisionmaking* (hereafter NRC *Risk Glossary*) (NRC 2013), and SECY-94-084 (NRC 1994).

anticipated operational occurrence: An operational event or condition deviating from normal operation that is expected to occur at least once during the operating lifetime of a reactor facility. Reactor designs are to accommodate anticipated operational occurrences without resulting in any significant damage to safety structures, systems, and components (SSCs) or leading to accident conditions. See the *Design of Small Reactor Facilities* RD-367 (CNSC 2014b), 10 CFR 50.2, and the *IAEA Glossary* "plant states considered in design" (2022).

beyond design basis accident: Accident conditions less frequent and more severe than a design basis accident. A beyond design basis accident may or may not involve significant core/fuel degradation. See also **design extension conditions** below**.** See RD-367 (CNSC 2014b) and NRC *Full-Text Glossary* "Severe accident" (NRC 2024a).

beyond design basis event: (See: **beyond design basis accident**)

confinement: The safety function aimed at preventing or controlling the release of radioactive material into the environment during normal operation or accident conditions.

- Confinement is closely related in meaning to containment, but confinement is typically used to refer to the safety function of preventing the 'escape' of radioactive material, whereas containment refers to the means for achieving that function (IAEA 2022).

containment: The methods or physical structures designed to prevent or control the release and dispersion of radioactive substances.

- Related to confinement, containment is usually used to refer to methods or structures that perform a confinement function in facilities and activities—namely, preventing or controlling the release of radioactive substances and their dispersion in the environment.

See IAEA (2022), RD-367 “Confinement Boundary” CNSC (2014b), and NRC *Full-Text Glossary* “containment building” (NRC 2024a).

control: (n) The function, power, or (v) means used to direct, regulate, or restrain a process, system, or behavior.

- In safety contexts, the term *control* often carries a stronger, more active meaning than in everyday use or in some other languages. It typically involves not only observing or monitoring conditions, but also taking corrective or enforcement actions when needed, based on the results of that monitoring.

See IAEA (2022).

defense in depth: (See: **nuclear defense in depth**)

design basis accident: Postulated accident conditions for which a nuclear facility is designed according to established design criteria and conservative methodology, and for which damage to the fuel and the release of radioactive material are kept within acceptable limits. Design basis accidents represent the set of conditions the facility must withstand without compromising the health and safety of the public or the environment, relying on the planned operation of safety systems. See RD-367 (CNSC 2014b), NRC *Full-Text Glossary* (NRC 2024a), and IAEA SSR-2/1 (2016).

Design basis event: Postulated conditions for which a nuclear facility is designed according to established design criteria and conservative methodology, including conditions for normal operation, anticipated operational occurrences, design basis accidents, external events, and natural phenomena. Design basis events establish the necessary capability of structures, systems, and components to ensure the integrity of the reactor coolant boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition, and the capability to prevent or mitigate the consequences of accidents that could result in the potential offsite exposures. Design basis events represent the set of conditions the facility must be designed to withstand while ensuring the health and safety of the public and the environment are protected. (10 CFR 50.49(b)(1)(ii))

design extension conditions: Postulated accident scenarios that are not considered as design basis accidents, but that are considered during the design process of the facility. These conditions are evaluated using best-estimate methods, with the goal of ensuring that any release of radioactive material remains within acceptable limits. For nuclear power plants, design extension conditions may include conditions ~~in events~~ without significant fuel degradation or conditions ~~in events~~ with melting of the reactor core. See IAEA *Glossary* (IAEA 2022), CNSC REGDOC-2.5.2 (CNSC 2014a), IAEA SSR-2/1 (2016), and 10 CFR Part 50.155 .

key safety functions: A set of high-level performance objectives, that, when fulfilled, support the overarching safety objective to ensure reasonable assurance that there is no undue risk to military personnel and adequate protection of the health and safety of the public and the environment. See IAEA SSR-2/1 (2016), SECY-18-0096 (NRC 2018a), and

DOE *Technology Inclusive Content of Application Project for Non-Light Water Reactors: Definition of Fundamental Safety Functions for Advanced Non-Light Water Reactors (*hereafter *TICAP) (DOE 2019).*

nuclear defense in depth: The use of multiple, independent, or redundant layers of barriers, equipment, and procedures to prevent the escalation of anticipated operational occurrences and maintain the effectiveness of barriers against accidents that release radiation or hazardous materials.

- The key is creating multiple independent and redundant layers of defense to compensate for potential human and mechanical failures so that no single layer, no matter how robust, is exclusively relied upon. Defense in depth may include the use of access controls, physical barriers, redundant and diverse key safety functions, and emergency response measures to provide comprehensive protection.

See RD-367 (CNSC 2014b), NRC *Full-Text Glossary* (NRC 2024a), IAEA *Glossary* (IAEA 2022).

passive component or system: A component or system element that performs its intended safety or operational function without the need for external input such as actuation, mechanical movement, or power. Its function relies solely on inherent physical properties such as pressure, gravity, natural circulation, material strength, or thermal conductivity. Examples include tanks, pipes, valves (that remain in a set position), heat exchangers, and structural supports. Passive components typically have no moving parts and are valued in nuclear safety design for their high reliability and low dependence on active control systems. See IAEA *Glossary* (IAEA 2022).

plant year (also: reactor year): A unit of measure used in safety analyses and *probabilistic risk assessment (PRA)* to express the frequency of events at a nuclear reactor. It represents one year of operation of a single nuclear reactor and serves as the basis for quantifying the likelihood of accidents, particularly those that could cause severe damage to the reactor core and nuclear fuel. See NUREG-2122 (NRC 2013).

receptor: Refers to the person evaluated as receiving the dose and for which specific acceptance criteria have been developed. Four receptors have been defined: Off-Base Public, On-Base Public, Installation Personnel, and Reactor Operating Staff.

Basis Considerations:

On-Base Public, **Installation Personnel**, and **Reactor Operating Staff** receptors comprise the definition of installation-related personnel from U.S. Department of Defense (DoD) 6055.09-M, Military personnel (to include family members), DoD employees, DoD contractor personnel, and other personnel having either a direct operational (military or other Federal personnel undergoing training at an installation) or logistical support (e.g., vendors) relationship with installation activities (DDESB 2024).

1. **Off-Base Public:** For purposes of protection and safety, an individual in the limiting location outside the Base Boundary used for the purpose of verifying compliance with the dose limits for public exposure.

Basis Considerations: IAEA “member of the public.” NRC “public dose” - For purposes of protection and safety, in a general sense, any individual in the population except when subject to occupational exposure or medical exposure. For the purpose of verifying compliance with the annual dose limit for public exposure, this is the representative person.

See: IAEA *Glossary* (IAEA 2022), NRC *Full-Text Glossary* “Public Dose” (NRC 2024a).

2. **On-Base Public:** Military personnel family members, DoD contractor personnel (e.g., vendors), and other personnel within the Base Boundary having a logistical support relationship with installation activities.

   This proposed designation recognizes a distinct group based on the high reliability of an effective emergency management program and prompt implementations of protective actions (e.g., sheltering, evacuation) available on military installations.

Basis Considerations: DoD Unique – (PROPOSED in section 6-3 of the draft DA PAM) a distinct class due to the high reliability of an effective (EM) program and preplanned response actions (shelter/evacuation) for military installations.

3. **Installation Personnel:** Military personnel, DoD employees, DoD contractor personnel, and other personnel (military or other Federal personnel undergoing training at an installation) located within the Base Boundary and having a direct operational relationship with installation activities, but that are not directly related to reactor operations.

   This group addresses receptors outside the reactor-controlled area. While not explicitly addressed by NRC classifications, past NRC guidance assumes that individuals not occupationally associated with the plant are considered members of the public. In contrast, the DOE refers to this group as co-located workers, and the IAEA uses the term “site personnel” for all persons working within the site area of a licensed facility.

Basis Considerations: Personnel Outside Reactor Controlled area. NRC does not address this receptor classification; from prior NRC position, all persons who are not occupationally associated with the plant are assumed public (see ML992910109) (NRC 1998). DOE defines this receptor as a Co-located Worker. IAEA defines Site Personnel as all persons working in the site area of an authorized facility, either permanently or temporarily. See also “Are Barracks and Family Housing Units on Military Bases Considered Public Receptors?” (EPA 2024).

See NRC SECY-98-038;ML992910109,(NRC 1998); DOE – Co-located Worker (DOE 2014) ; IAEA *Glossary* “Site Personnel” (IAEA 2022).

4. **Reactor Operating Staff:** Military personnel, DoD employees, DoD contractor personnel, and other personnel, typically located within the Reactor Controlled Area, having reactor operations responsibility and reactor-specific hazards and response training.

> Basis Considerations:
>
> The NRC defines dose criteria for reactor staff under Normal Operations and those with Safety Functions for accident response (Control Room -Reactor Operators, Technical Support Center).
>
> In 10 CFR Part 70, for evaluating acute worker dose high consequence events, the NRC defines a worker as an individual who receives an occupational dose.
>
> DOE defines the Worker, staff within the plant boundary, criteria qualitatively (prompt death, serious injury, significant exposure-rad/chemical).
>
> IAEA *Glossary* Operating Personnel: Individual workers engaged in the operation of an authorized facility or the conduct of an authorized activity.

See 10 CFR Part 70 , DOE-STD-3009-2014 (DOE 2014), and IAEA *Glossary* (IAEA 2022).

receptor boundaries: Receptor boundaries refer to areas for which a defined receptor is located and activities are controlled. Three receptor boundaries have been defined.

1. **Reactor Controlled Area:** A designated zone surrounding the reactor that requires authorization for access. Within this area, the Senior Reactor Leader (SRL) has full authority over all activities, including the ability to approve, restrict, or remove personnel and property to ensure safe reactor operation and security.

   This area is established to maintain control over operations and limit exposure to reactor-related hazards, ensuring that only trained and authorized individuals enter and perform tasks under controlled conditions. Personnel inside the Reactor Controlled Area are evaluated as Reactor Operating Staff.

2. **Safety Controlled Area** – The region extending outward from the Reactor Controlled Area to the nearest Inhabited Building Distance (IBD). Within this area, approved Emergency Management Plans and Procedures are in place to ensure protection and coordinated response in the event of an incident. This area is under the control of the Installation Commander. Personnel inside the Safety Controlled Area are evaluated as Installation Personnel. Outside of the Safety Controlled Area, nonmilitary personnel are evaluated as On-Base Public.

> Basis Considerations:
>
> **Inhabited Building Distance:** 27 CFR 555.11 defines the term “inhabited building” as “[a]ny building regularly occupied in whole or in part as a habitation for human beings, or any church, schoolhouse, railroad station, store, or other structure where people are

accustomed to assemble, except any building occupied in connection with the manufacture, transportation, storage, or use of explosive materials."

Structures, other than ammunition and explosives-related buildings, occupied by personnel or the general public, both within and outside DoD establishments (e.g., schools, churches, residences, quarters, Service clubs, aircraft passenger terminals, stores, shops, factories, hospitals, theaters, mess halls, post offices, or post exchanges). See the *Defense Explosives Safety Regulation*: DESR 6055.09 (DDESB 2024).

Open-air activities that are transitory in nature (e.g., walking/driving paths, some sports and other recreation spaces) are also allowed within the IBD, and as close as a Public Traffic Route Distance (PTRD; i.e., outside of the green arc), as long as these areas and functions do not require structures (e.g., bathrooms, bleachers, etc.) and do not cause people to congregate (Ross 2023).

3. **Installation Boundary:** The line beyond which land or property is not owned, leased, or otherwise controlled by the Army. This boundary defines the limit of Army jurisdiction and is used to determine the evaluation point for off-base public exposure.

Basis Considerations:

NRC/IAEA **Exclusion area:** The area surrounding the reactor where the reactor licensee has the authority to determine all activities, including the exclusion or removal of personnel and property (IAEA 2022; NRC 2013).

The term **military installation** means a base, camp, post, station, yard, center, or other activity under the jurisdiction of the Secretary of a military department or, in the case of an activity in a foreign country, under the operational control of the Secretary of a military department or the Secretary of Defense, without regard to the duration of operational control (10 USC § 2801(c)(4); 10 CFR 50.2).

safety-critical SSC: A structure, system, or component provided to ensure that a Key Safety Function remains functional and performs as analyzed during and following design basis accidents. This includes design or inherent features credited to reduce the frequency or consequence of events, and those credited to maintain design basis accident parameters within prescribed limits for the public. This category also includes SSCs required to protect or support Safety Critical functions.

Basis Considerations:

The term "safety-critical" is drawn from MIL-STD-882E; "*MIL-STD-882E: The definitions in Tables I and II, and the RACs in Table III shall be used, unless tailored alternative definitions and/or a tailored matrix are formally approved in accordance with DoD Component policy*" (DoD 2023). The definition has adapted to focus on protection of public health and safety. See IAEA *Glossary* "Plant Equipment: Safety System" (IAEA 2022); NRC *Glossary* "safety-related" (NRC 2013), 10 CFR 50.2 "safety-related."

safety-related SSC: A structure, system, or component that is not classified as safety critical but is still relied upon to remain functional during and after design basis accidents to protect military personnel. These SSCs also serve to prevent anticipated operational occurrences from escalating into accident conditions, support the mitigation of beyond design basis accidents as part of *design extension conditions*, or are identified as essential within *Level 4* of the nuclear *defense in depth* strategy.

Basis Considerations: The term "safety-related" originates from MIL-STD-882E and is adapted here to emphasize the protection of public health and safety (DoD 2023). Also see IAEA *Glossary* "Plant Equipment: Safety-Related" (IAEA 2022) and NRC *Glossary,* "Safety-Significant" (NRC 2013).

safety-significant SSC: A structure, system, or component that provides reasonable assurance that a facility can be operated without undue risk to the health and safety of military personnel and the public.

Section 6-6 of the draft DA PAM includes those SSCs that are identified as a key element of the nuclear defense in depth to prevent anticipated operational occurrences (AOOs) from progressing to accident conditions (Level 2), and those that provide monitoring needed to provide facility and installation staff and off-site emergency services with a sufficient set of reliable information in the event of an accident, including monitoring and communication means as part of the emergency response plan (Level 5).

Basis Considerations: The term "safety-significant" is used in MIL-STD-882E to encompass both safety-critical and safety-related SSCs and is adapted here to emphasize the protection of public health and safety (DoD 2023). Also see IAEA *Glossary* "Plant Equipment: Safety-Related" (IAEA 2022) and NRC *Glossary,* "Safety-Significant" (NRC 2013).

structures, systems, and components (SSCs): *A* general term that includes all physical elements of a facility or activity that contribute to protection and safety, excluding human factors.

- *Structures* are passive elements such as buildings, vessels, and shielding.
- A *system* is an assembly of components arranged to perform a specific active function.
- A *component* is an individual part of a system, such as wires, transistors, integrated circuits, motors, relays, solenoids, pipes, fittings, pumps, tanks, or valves.

See RD-367 (CNSC 2014b).

2.2 References for Key Definitions

The following references were used to establish the definitions for radiological dose acceptance and design criteria bases in Section 2.1:

- **10 CFR 50.2**, *Definitions*
- **10 CFR 50.155**, *Mitigation of beyond-design-basis events*

- **IAEA Glossary**, *IAEA Nuclear Safety and Security Glossary, Terminology Used in Nuclear Safety, Nuclear Security, Radiation Protection and Emergency Preparedness and Response* 2022 (Interim Edition), International Atomic Energy Association (IAEA 2022)
- **IAEA SSR-2/1**, *Safety of Nuclear Power Plants: Design*, International Atomic Energy Association
- **MIL-STD-882E**, *Department of Defense Standard Practice: System Safety,* CHANGE 1, 27 September 2023 (DoD 2023)
- **ML992910109**, *Hanford Tank Waste Remediation System Privatization Co-Located Worker Standards* (NRC 1998)
- **NEI TICAP (ML20021A182)** *Technology Inclusive Content of Application Project For Non-Light Water Reactors Definition of Fundamental Safety Functions for Advanced Non-Light Water Reactors* (DOE 2019)
- **NRC Full-Text Glossary** (NRC 2024a), and NUREG-0544, *Collection of Abbreviations* (NRC 2016)
- **NUREG-2122**, *Glossary of Risk-Related Terms in Support of Risk Informed Decisionmaking* (NRC 2013)
- **REGDOC-2.5.2**, *Design of Reactor Facilities*, Canadian Nuclear Safety Commission (CNSC 2014a)
- **RD-367**, *Design of Small Reactor Facilities*, Canadian Nuclear Safety Commission (CNSC 2014b)
- **SECY-18-0096**, NRC Commission Paper: *Functional Containment Performance Criteria For Non-Light-Water-Reactors* (NRC 2018a)
- **SECY-94-084**, NRC Commission Paper: *Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs* (NRC 1994).

3.0 Dose Acceptance Criteria

3.1 Dose Acceptance Criteria – Background

This section documents the rationale and consideration basis for the dose acceptance criteria proposed in the draft revised AR 50–7 and associated DA PAM. This approach was selected for the development of Army Reactor Program requirements to provide:

- A level of safety consistent with the regulation for existing commercial reactors.
- A key consideration for the Army Reactor Program for microreactors is that given the anticipated smaller mechanistic and accident source terms, and if the projected accident doses for the spectrum of credible accidents are less than 1 rem (10 mSv) at the Base Boundary, then the plume exposure pathway emergency planning zone would be confined to within the Base Boundary. Therefore, no off-site (off-installation) emergency actions would be required, but on-shift or on-site emergency response would occur according to the emergency plan for the nuclear facility.
- The factoring of risk insights as a foundation into the AR 50-7.

Basis Considerations: NUREG-0800, SRP 15.0: "If the risk of an event is defined as the product of the event's frequency of occurrence and its consequences, then the design of the plant should be such that all the AOOs and postulated accidents produce about the same level of risk (i.e., the risk is approximately constant across the spectrum of AOOs and postulated accidents). This is reflected in the general design criteria (GDC), which generally prohibit relatively frequent events (AOOs) from resulting in serious consequences but allow the relatively rare events (postulated accidents) to produce more severe consequences." (NRC 2023). SSG-2, 4.4: "Acceptance criteria should relate to the frequency of the relevant conditions. Conditions that occur more frequently, such as normal operation or anticipated operational occurrences, should have acceptance criteria that are more restrictive than those for less frequent events, such as design basis accidents or design extension conditions" (IAEA 2019).

Notable factors with respect to developing the dose acceptance criteria include:

- Many of the reference documents present a frequency–consequence curve and may be based on probabilistic risk assessment – licensing basis events, which may consider best basis estimates. The draft revised AR 50–7 is based on a neutral evaluation approach, both PRA-based (with Licensing Basis Event) and deterministic approaches are acceptable.
- The required evaluation (exposure) times vary across the reference documents. For the proposed criteria, doses have been normalized to a 30-day total effective dose equivalent (TEDE) unless otherwise noted (e.g., Normal Operations, and Emergency Planning Zone requirement of 1 rem (10 mSv) TEDE over the first 96 hours).
- Many reference approaches were developed considering applications to reactors encompassing much larger thermal power capacities than microreactors including

SMRs[1] (CNSC 2014b). A key consideration for microreactors is the source term or amount of material available for release, which is a direct function of the capacity.

- Many source documents are "Draft" or proposed approaches presenting current opinions and are not necessarily accepted positions by the respective regulators.

Documents considered in the derivation of the dose limits include:

- Army Regulations and Policies, **AR 385–10** (Army 2023)
- NRC Regulations (e.g.,**10 CFR 20, 50, 52, and 100**)
- Canadian Nuclear Safety Commission **RD-367**: *Design of Small Reactor Facilities* (CNSC 2014b)
- **DOE-STD-3009-2014**, *Preparation of Nonreactor Nuclear Facility Documented Safety Analysis* (DOE 2014)
- **EPA-400/R-17/001**, *PAG Manual: Protective Action Guides and Planning Guidance for Radiological Incidents*, January 2017 (DOE 2014)
- **IAEA SSG-2**, *Deterministic Safety Analysis for Nuclear Power Plants* (IAEA 2019)
- **IAEA SSR-2/1,** *Safety of Nuclear Power Plants: Design* (IAEA SSR-2/1 2016)[2]
- **IAEA SSR-3**, *Safety of Research Reactors* (IAEA 2016b)
- **INL/EXT-10-19521**, Next Generation Nuclear Plant [NGNP] Licensing Basis Event Selection White Paper (INL 2010)
- **INL/EXT-20-60394**, *Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors: Selection and Evaluation of Licensing Basis Events* (INL 2020)
- **NEI 18-04**, *Risk-Informed Performance-Based Technology Guidance for Non-Light Water Reactors* (NEI 2019b)
- **NRC NUREG-0800**, SRP 15.0 *Transient and Accident Analysis* (NRC 2023)
- **NRC NUREG-1860**, *Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing* (NRC 2007)
- **NRC Regulatory Guide 1.233**, *Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors* (NRC 2020a)

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[1] RD-367 sets out the requirements of the Canadian Nuclear Safety Commission (CNSC) for the design of new small reactor facilities. It establishes a set of design requirements that align with accepted national and international codes and standards. RD-367 defines a small reactor facility as a reactor facility containing a reactor with a power level of less than approximately 200 megawatts thermal (MWt) that is used for research, isotope production, steam generation, electricity production, or other applications.

[2] This publication establishes design requirements for the structures, systems, and components of a nuclear power plant, as well as for procedures and organizational processes important to safety that are required to be met for safe operation and for preventing events that could compromise safety, or for mitigating the consequences of such events, were they to occur.

- *Technology Inclusive and Risk-Informed Reviews for Advanced Reactors: Comparing the US Licensing Modernization Project with the Canadian Regulatory Approach*, Canadian Nuclear Safety Commission/NRC (ML21225A101) (NRC and CNSC 2021)

3.2 Dose Acceptance Criteria – Routine Operations and Anticipated Operational Occurrences

The evaluation of events considered within the bounds of Routine Operations and Anticipated Operational Occurrences includes events with a frequency equal to or greater than 10^{-2}/plant-yr. For this range of operating conditions, the key principle in demonstrating the overarching Safety Objective is to ensure equivalent safety to existing commercial nuclear facilities.

The Radiation Protection Programs (NRC) or Radiation Safety Program (Army) for routine operations applicable to large power reactors, research and test reactors, and nonreactor nuclear facilities are generally consistent. Program requirements are codified in the United States in 10 CFR Part 20 for commercial vendors and referenced in AR 385–10 (Army 2023), referenced in 10 CFR 835 for the Department of Energy, and discussed in IAEA standards.

3.2.1 AR 50–7: Normal Operations and Anticipated Operational Occurrences

The following requirements are outlined in the draft updated AR 50–7 (text in italics) for normal operations and anticipated operational occurrences:

> *Design features and programmatic controls are provided such that the analyses of routine operations ensure safe operation of the nuclear facility and that doses to individual members of the public and Army personnel are in accordance with AR 385–10, The Army Safety and Occupational Health Program.*
>
> *The reactor plant conforms to the requirements within technical specification and the limiting conditions of operations. When a Limiting Condition of Operation (LCO) for operation of a nuclear reactor is not met, the licensee shuts down the reactor or follows any remedial action directed by the technical specifications until the condition can be met.*
>
> *The reactor is designed to operate safely and reliably, or shutdown if necessary, during normal operations and AOOs, with an assumed availability of a minimum set of specified support features for safety systems. The response of the reactor to AOOs demonstrates the following:*
>
> - *The dose acceptance criteria in the DA PAM (associated with the draft AR 50–7) are met.*
> - *SSCs that are not involved in initiating the event remain operational and are sufficient to control the event progression without the need of actuation of systems credited in performing a key safety function to prevent damage to protection systems or prevent the occurrence of design-basis accident conditions.*

Basis Considerations:
These requirements are consistent with IAEA SSR-2/1 (IAEA 2016a) and SSR-3 (IAEA 2016b) for defense in depth Levels 1 and 2, RD-367 (CNSC 2014b), and SRP 15.0 (NRC 2023), which specifies “AOOs should not generate a postulated accident without other faults occurring independently or result in a consequential loss of function of the RCS [reactor coolant system] or reactor containment barriers.”

3.2.2 DA PAM (Associated with Draft AR 50–7): Normal Operations and Anticipated Operational Occurrences

The following dose acceptance criteria are used in the draft DA PAM (text in italics) and **apply to all receptors**:

For event frequencies of 10^{-1}/ plant year and higher, the Radiation Safety Program (AR 385–10) including consideration of ALARA provides the dose acceptance criteria (Army 2023).

Off-Base Public Receptors: *For individual AOO events with frequencies from 10^{-1}/ plant-year to 10^{-2} /plant-year, the dose limit is set at 100 mrem TEDE/per event for 30 days at the base boundary.*

On-Base Public Receptors: *For individual AOO events with frequencies from 10^{-1}/ plant-year to 10^{-2} /plant-year, the dose limit is set at 100 mrem TEDE/per event for 24 hrs at any point on the boundary of the Safety Controlled Area.*

Installation Personnel and Reactor Operating Staff Receptors: *For individual AOO events with frequencies from 10^{-1}/ plant-year to 10^{-2} /plant-year, the Radiation Safety Program (AR 385–10) including consideration of ALARA provides the Installation Personnel and Reactor Operating Staff dose acceptance criteria (Army 2023).*

Basis Considerations

Note: For normal operations, the review was based primarily on U.S. (NRC regulations), as the IAEA will typically default to the National Regulator. This approach is consistent with what is currently used by Army programs utilizing nuclear material in overseas deployment (e.g., the default is to the Army's Radiation Safety Program). Per AR 385-10, *Outside the continental United States (OCONUS) control of radiation sources will be in conjunction with host nation authorizations, Status of Forces Agreements (SOFA), Army regulations, international agreements, NRC licenses, and ARAs, as applicable (Army 2023).*

Off-Base Public

The regulations in 10 CFR 20 limits the total effective dose equivalent (TEDE) to individual members of the public from the licensed operation to no more than 0.1 rem (1 mSv) in a year. This regulation is used to establish frequency-consequence targets for Normal Operations and AOOs.

As discussed above, previous development of Normal Operations and AOO dose criteria are contained in numerous documents; however, many of these individual discussions are evaluated and summarized in:

- *Technology Inclusive and Risk-Informed Reviews for Advanced Reactors: Comparing the US Licensing Modernization Project with the Canadian Regulatory Approach* in ML21225A101 (NRC and CNSC 2021)
- INL/EXT-20-60394: *Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors: Selection and Evaluation of Licensing Basis Events* (INL 2020)

In addition, many of the documents propose a bifurcation between normal operations and AOOs based on the frequency of the event. SRP 15.0 includes reference to ANS, Plant Condition II Frequency > 10^{-1} and Plant Condition III 10^{-1}>Frequency >10^{-2} (NRC 2023).

The findings from these reports are summarized in Table 3-1 Comparison of Normal Operations and AOO Dose Limits Applicable to Public.

On-Base Public

This receptor is unique to military installations. The dose criteria for Normal Operations and AOOs for events with frequencies ≥ 10^{-1} /plant yr are set the same as for the Off-Base Public and will be the limiting criteria.

The dose criteria for the less frequent AOOs are derived from those for the Off-Base Public. The Dose is conservatively evaluated of the Safety Controlled Area Boundary. The Criteria allows for credit for an effective Emergency Management (Preparation and Response) Program being operational in the calculation of the dose, e.g., assumes a receptor is exposed for 1 day (24 hrs.) provided Emergency Response actions are identified and agreed to by the responsible Base Commander.

Installation Personnel and Reactor Operating Staff

- The Installation Personnel receptor is unique to military installations but is similar to U.S. DOE definition of a Co-Located Worker. The Dose criteria for Normal Operations down to 10^{-1} /plant yr are set consistent with the approach used for the

Public and are based on the Radiation Safety Program (AR 385–10) (Army 2023), including consideration of ALARA.

- For infrequent AOOs, the dose acceptance is set to be the maximum allowable annual occupational dose limit of 5 rem (0.05 Sv) applied on an individual event basis. This is consistent with the approach used for the public receptors.

Table 3-1.
Comparison of Normal Operations and AOO Dose Limits Applicable to Off-Base Public

Regulation	Frequency[a]	Off-Site Public Dose	Reference discussion
U.S. NRC (10 CFR 20)	≥ 1E-01	1 mSv 100 mrem	See Figure 3-4[b] (also see ML21225A101, Table 9 and Figure 9).[††] Applied to cumulative exposures for the year. Based on an ISO-risk line.
	1E-01≥ f ≥1E-02	10 mSv 1 rem	See Figure 3-4[b] (also see ML21225A101, Table 9 and Figure 9).[††]
CNSC*	≥ 1E-02	0.5 mSv 0.05 rem	RD-367: 30 day @ Site Boundary (also see ML21225A101, Table 9 & Figure 9).[††]
NUREG-1860[†]	≥1E-02	(0.05 mSv/) 5 mrem	See Figure 3-2.[b] NUREG-1860 refers to 10 CFR 50 Appendix I, Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low as is Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents.
	≥ 1E-03 *	(1 mSv) 100 mrem	See Figure 3-2.[b] *Frequency is outside proposed range of 1E-02.
NGNP (INL/EXT-10-19521)[‡]	≥ 1	ISO-risk line anchored at 1 mSv 100 mrem	See Figure 3-1[b] (also see INL/EXT-20-60394[‡]).Based on an ISO-risk line.
	1 > f ≥1E-02	1 mSv 100 mrem	See Figure 3-1[b] (also see INL/EXT-20-60394[‡]).
NEI 18-04[§] RG 1.233[#]	≥1E-01	ISO-risk line anchored at 1 mSv 100 mrem	See Figure 3-3[b] (also see INL/EXT-20-60394[‡]) Based on an ISO-risk line.
	1E-01> f ≥1E-02	10 mSv 1 rem	See Figure 3-3[b] (also see INL/EXT-20-60394[‡] and NEI 18-04).
DOE-STD-3009-2014**	≥1E-02	< 5 rem	Note: Based on unmitigated Frequency/ Consequences representing events identified as situations of minor concerns DOE-STD-3009-2014. Based on 2-8 hr. exposure at site boundary

Sources: *CNSC (2014b), [†]NRC (2007), [‡]INL (2010), [§]NEI (2019b), [#]NRC (2020a), **DOE (2014); [††]NRC and CNSC (2021); [‡]INL (2020).
[a]May be expressed in plant years or reactor years to reflect the potential for multi-reactor module facilities.
[b]All figure callouts refer to this document, unless otherwise noted.

3.3 Dose Acceptance Criteria – Design Basis Accidents

The evaluation criteria for Design Basis Accidents addresses events with a frequency lower than 10^{-2} /plant year and equal to or greater than 10^{-4} /plant year.

Note: Some regulatory constructs extend DBAs down to 10^{-5} /plant yr. However, the proposed construct is consistent with NRC policy and guidance documents, as outlined in NUREG-0800, Chapter 15 and in RG 1.174 and RG 1.233.

For this range of operating conditions, the key principles in demonstrating the overarching reactor safety objective are underpinned by the expectation of the use of passive SSCs and inherent safety features such that no operator actions or Off-Base Emergency Response Actions will be required.

3.3.1 AR 50–7: Design Basis Accidents

The following requirements are outlined in the draft AR 50–7 (text in italics) for design basis accidents:

> *Design features and controls will be provided such that analyses of design basis accidents demonstrate the following:*
>
> - *The dose acceptance criteria in DA PAM for design basis accidents are met utilizing engineered SSCs and inherent safety features without the need for operator actions to be credited for protection of the Off-Base Public.*
> - *Key plant parameters do not exceed the specified design limits.*

See also:

SECY 20-0093 (NRC 2000) and design objective from NEI 18-04 (NEI 2019b) to facilitate reduced offsite emergency response.

Regulatory Review of Micro-Reactors – Initial Considerations (NRC 2020b) and *Micro-reactors Licensing Strategies* ML21235A418 (NRC n.d.). Note: these documents provided insights into the development of the bases, although they aren't approved for regulatory licensing activities for the NRC as of the time of the publication of this report.

See SECY 20-0093 (NRC 2000) and design objective from NEI 18-04 (NEI 2019b) to facilitate reduced offsite emergency response.

3.3.2 DA PAM (associated with AR 50–7): Design Basis Accidents

For design basis accidents, the following criteria were established in the draft DA PAM (text in italics):

Off-Base Public Receptors:

An individual located at any point on the Base Boundary who is exposed to the radioactive cloud resulting from the postulated fission product release over 96 hours from the release of radioactive materials, would not receive a radiation dose in excess of 1 rem (10 mSv) total effective dose equivalent from plume exposure; and

An individual located at any point on the Base Boundary who is exposed to the radioactive cloud resulting from the postulated fission product release over 30 days from the release of radioactive materials, would not receive a radiation dose in excess of 2 rem (20 mSv) total effective dose equivalent from plume exposure.

Basis Considerations:

The dose considerations above will be below the EPA Protective Action Guides (PAGs) not requiring offsite protective actions.

The constraint of not exceeding the EPA PAGs is important because, as reflected in the discussions of a design objective in NEI 18-04 (endorsed by the NRC in RG 1.253 (2024b)), it would facilitate a reduced offsite emergency response. See also Figure 3-3 and Figure 3-4 within this document, which are from NEI (2019b), *Regulatory Review of Micro-Reactors – Initial Considerations* (NRC 2020b), *Micro-reactors Licensing Strategies* ML21235A418 (NRC n.d.), and SECY 20-0093 (NRC 2000).

See RD-367 *Design of Small Reactor Facilities*, Figure 3-4 (CNSC 2014b).

On-Base Public Receptors: *An individual located at any point on the boundary of the Safety Controlled Area who is exposed to the radioactive cloud resulting from the postulated fission product release over 8 hours from the release of radioactive materials would not receive a radiation dose in excess of 1 rem (10 mSv) total effective dose equivalent from plume exposure.*

Basis Considerations:

This basis is similar to the Off-Base Public Receptors basis presented above, but with accounting for the "effective" emergency response program for the installation, where the exposure duration is no greater than 8 hrs. (Note: A duration of 8 hrs. is the upper bound for time used in DOE evaluations of public impacts.)

Further, the evaluation of a 30-day exposure period is judged as not required due to the Army's ability to control public access and evacuate receptors

Installation Personnel Receptors: *An individual located at a distance of 100 meters from the release point who is exposed to the radioactive cloud resulting from the postulated fission product release over a 2-hour period following the onset of the postulated fission product release would not receive a radiation dose in excess of 25 rem (250 mSv) total effective dose equivalent.*

Basis Considerations:

These criteria are unique to DoD, and ensures ability to perform military duties in the event of a release, without undue risk.

These criteria are analogous to DOE Co-located Worker (CLW) criteria for Moderate Consequences. Could extend to 8 hrs.

JP 3-11 Figure D-2: Recommends maximum dose limit of 25 cGy - for Critical Tasks. Acute doses would have less than 1% of severe health effects (hospitalization required) Figure D-1 (JCS 2013).

Reactor Operating Staff Receptors: *An individual within the Limited Area Boundary or at the Control Room location, having reactor operations responsibility and reactor specific hazards and response training, who is exposed to radioactive material for a 2-hour period following the onset of the postulated release would not receive a radiation dose in excess of 100 rem (1000 mSv) total effective dose equivalent.*

Basis Considerations:

There are no “Safety-Critical” Operator Actions required for 96 hrs.

Reactor Safety Design Criterion RSDC-10 in the draft DA PAM (appendix C) addresses providing an operations control point station in which actions can be taken to operate the nuclear power plant safely under normal conditions, monitor key safety functions, and, as necessary, maintain it in a safe condition under accident and design extension conditions using well-designed human-machine interfaces.

100 rem: DOE criteria for requiring controls for the CLW. DOE-STD-3009-2014 does not provide a Facility Worker Moderate Criteria (value), but rather states: “No Distinguishable Threshold” for “Moderate Consequences” (DOE 2014)

JP 3-11, Figure D-2: Above the 75 cGy radiation exposure status radiation exposure status (RES) category 1 limit, a low incidence of acute radiation sickness can be expected and personnel should be considered for medical evaluation and evacuation upon any signs or symptoms related to acute radiation sickness (e.g., nausea, vomiting, anorexia, fatigue). 125 cGy is the limit for exceeding moderate operational risk RES-2 (JCS 2013).

Table 3-2.
Comparison of Design Basis Accidents Dose Limits Applicable to Off-Base Public

Regulation	Frequency[a]	Off-Site Public Dose	Reference discussion
CNSC*	1E-02> f ≥1E-05	2 rem (20 mSv)	See Figure 3-4.[b] RD-367: 30 day @ Site Boundary (Also see ML21225A101, Table 9 and Figure 9)[‡‡]
NUREG-1860[†]	1E-02> f ≥1E-03	100 mrem (1 mSv)	See Figure 3-2.[b]
	1E-03> f ≥1E-04	1 rem (10 mSv)	See Figure 3-2.[b]
	1E-04> f ≥1E-05	25 rem (250 mSv)	See Figure 3-2.[b] Note: Frequency is outside proposed range of 1E-04. Included for completeness and comparison to CNSC.
NGNP (INL/EXT-10-19521)[‡]	1E-02> f ≥1E-04	2.5 rem (25 mSv) 25 rem (250 mSv)	See Figure 3-1 (also see INL/EXT-20-60394[††]). Based on an ISO-risk line anchored at endpoints.
	1E-04> f ≥1E-05	Prompt Mortality Goal	See Figure 3-1.[b] Measured out to 1.6 km Note: Frequency is outside proposed range of 1E-04. Included for completeness and comparison to CNSC.
NEI 18-04[§] RG 1.233[#]	1E-02> f ≥1E-04	1 rem (10 mSv) 25 rem (250 mSv)	See Figure 3-3.[b] Based on an ISO-risk line anchored at endpoints.
	1E-04> f ≥1E-05	25 rem (250 mSv) 100 rem (1000 mSv)	See Figure 3-3.[b] Note: Frequency is outside proposed range of 1E-04. Included for completeness and comparison to CNSC. Based on an ISO-risk line anchored at endpoints.
DOE-STD-3009-2014**	1E-02> f ≥1E-04 1E-04> f ≥1E-05*	< 25 rem (250 mSv)	Note. DOE-STD-3009-2014. DOE does not define Public Dose > 25 rem. Note: Frequency is outside proposed range of 1E-04. Included for completeness and comparison to CNSC.

Sources: *CNSC (2014b), [†]NRC (2007), [‡]INL (2010), [§]NEI (2019b), [#]NRC (2020a), **DOE (2014);[††]INL (2020) [‡‡]NRC and CNSC (2021).
[a]May be expressed in plant years or reactor years to reflect the potential for multi-reactor module facilities.
[b]All figure callouts refer to this document, unless otherwise noted.

3.4 Dose Acceptance Criteria – Beyond Design Basis Accidents

The evaluation of events considered within the bounds of Design Basis includes events with a frequency lower than 10^{-4} /plant year equal to or greater than about 5×10^{-7} /plant year. Note: Some regulatory constructions may extend down to 10^{-7} /plant year.

For this range of operating conditions, the key principle in demonstrating the overarching Reactor Safety Objective is to ensure:

- no undue risk; and
- exposures would not result in prompt impacts.

3.4.1 AR 50–7: Beyond Design Basis Accidents

The following requirements are outlined in the updated draft AR 50–7 (text in italics) for beyond design basis accidents:

> *Design features and programmatic controls are be provided such that analyses of beyond-design-basis accidents (design extension events) demonstrate the following:*
>
> - *The dose acceptance criteria in the draft DA PAM (associated with the draft updated AR 50–7) for beyond design basis accidents are met*
> - *Protective actions are limited in nature and sufficient time is available such that the possibility of conditions leading to an early large radioactive release is 'practically eliminated."*

3.4.2 DA PAM (Associated with AR 50–7): Beyond Design Basis Accidents

For beyond design basis accidents, the following criteria were established in the draft DA PAM (text in italics):

<u>Off-Base Public Receptors</u>:

- *An individual located at any point on the Base Boundary who is exposed to the radioactive cloud resulting from the postulated fission product release over 96 hours from the release of radioactive materials, would not receive a radiation dose in excess of 25 rem (250 mSv) total effective dose equivalent from plume exposure;* and
- *An individual located at any point on the Base Boundary who is exposed to the radioactive cloud resulting from the postulated fission product release over 30 days from the release of radioactive materials, would not receive a radiation dose in excess of 50 rem (500 mSv) total effective dose equivalent from plume exposure*

Basis Considerations:
EPA-400/R-17/001, PAG Manual: "mandatory" evacuation versus shelter in place guideline of 5 rem increased by a factor of 5 to account for severe nature of event (EPA 2017). See discussion in EPA-400 section 2.3.4 for higher PAGs for Special Circumstances.
Fundamentally can be demonstrated to satisfy quantitative health objectives (QHOs) – NUREG-1860 Volume 2, Appendix D (NRC 2007). Conservative estimate, NUREG-1860

Table 6-1, of the possibility of deterministic effects (i.e., some early health effects are possible) at 50 rem; however it is below the threshold of early fatality (Atomic Archive 2024).

- 100 rem - threshold for radiation sickness in a few hours
- 200-300 $LD_{10-35/30}$ 10-35% of 30 days
- 400-450 is $LD_{50/30}$ received over a short time (50% in 30 days)
- Early QHO: Prompt Fatality – met (That is, the risk of prompt fatalities to average individuals in the vicinity of a nuclear power plant should not exceed 0.01% of the sum of the prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.)
- Latent QHO: Latent Cancer Fatality (as measured at Base Boundary- not distance to 10 miles); (DOE 2018).
- 50 rem * 4.1E-04 *1E-04 =2.05E-06 LCF/yr

Also see ML22287A155, *Safety Evaluation for NuScale Topical Report, TR-0915-17772, Methodology for Establishing The Technical Basis for Plume Exposure Emergency Planning Zones At NuScale Small Modular Reactor Plant Sites*, Revision 3: "The final EPZ size is the smallest distance at which the dose criteria, chosen to provide a level of protection that meets or exceeds the basis in NUREG-0396, are satisfied. These criteria essentially are:

a) total effective dose equivalent from the design basis source term is less than or equal to 1 rem;

b) the total effective dose equivalent from less severe accidents (containment intact) is less than or equal to 1 rem; or

c) a substantial reduction in early health effects from more severe accidents (containment failure or bypass), i.e., an acute whole body dose less than 200 rem" (NRC 2022).

In general bounds proposed criteria from LMP (exception minor increase [delta between 25 and 50 rem in the 1E-04 to 5E-05 range]).

IAEA NS-G-1.2 (superseded by GSR Part 4 Rev. 1, and SSG-2) "Deterministic acceptance criteria have also been specified in a number of countries, typically as follows:

- Containment failure should not occur in the short term following a severe accident,
- There should be no short-term health effects following a severe accident,
- The long-term health effects/release of ^{137}Cs should be below prescribed limits following a severe accident" (IAEA 2001).

<u>On-Base Public Receptors</u>: *An individual located at any point on the boundary of the Safety Controlled Area who is exposed to the radioactive cloud resulting from the postulated fission product release over 8 hours from the release of radioactive materials would not receive a radiation dose in excess of 50 rem (500 mSv) total effective dose equivalent.*

Basis Considerations:
Similar to Off-Base Public, but with acknowledgement of "effective" Emergency Response Program limits exposure duration to 8 hrs.; 8 hrs. is an upper time bound used for DOE evaluation of public impacts. The evaluation of radiological consequences based on a 30-day exposure is judged as not required due to the Army's ability to control public access and evacuate receptors.

Installation Personnel Receptors: *An individual located at a distance of 100 meters from the release point who is exposed to the radioactive cloud resulting from the postulated fission product release over a 2-hour period following the onset of the postulated fission product release would not receive a radiation dose in excess of 100 rem (1000 mSv) total effective dose equivalent.*

Basis Considerations:
This is unique to DoD and ensures ability to perform military duties in the event of a release, without undue risk. The lower threshold: 100-200 rem Mild radiation sickness within a few hours: vomiting, diarrhea, fatigue; reduction in resistance to infection (Atomic Archive 2024). JP 3-11 Figure D-1: Above 125 cGy - exceeds moderate operational risk. Acute doses (125 cGy) would have less than 10% of severe health effects (hospitalization required) and less than 4% probability of death from Excess Cancer (40 yr after exposure) (JCS 2013). This is analogous to the DOE CLW criteria of 100 rem requiring High Consequence Level controls (DOE 2014).

Reactor Operating Staff Receptors: *An individual within the Limited Area Boundary or at the Control Room location, having reactor operations responsibility and reactor specific hazards and response training who is exposed to radioactive material for a 2-hour period following the onset of the postulated release would not receive a radiation dose in excess of 200 rem (2000 mSv) total effective dose equivalent.*

Basis Considerations:
There are no "Safety-Critical" Operator Actions required for 96 hrs.; Reactor Safety Design Criteria for Control Location (Station) has been proposed reflecting this. Considered Prompt Fatality/Significant Radiological Exposure—DOE Consequence Threshold for the Facility Worker "High Consequence" is prompt death, serious injury, or significant radiological and chemical exposure.(DOE 2014) 100-300 rem is the threshold for early fatality (NRC 2007). Selected value is ~1/2 of LD50 associated with acute dose 410 cGy, JP 3-11 Figure D-1. Estimated to be less than a 10% probability of death from Excess Cancer (40 yr after exposure) (JCS 2013).

Table 3-3.
Comparison of Beyond Design Basis Accidents Dose Limits Applicable to Off-Base Public

Regulation	Frequency[a]	Off-Site Public Dose	Reference discussion
CNSC*	1E-04> f ≥1E-05	2 rem (20 mSv)	See Figure 3-4.[c]
	1E-05> f	Not Defined	See Figure 3-4.[c]
NUREG-1860[b†]	1E-04> f ≥1E-05	50 rem (500 mSv)	See Figure 3-2.[c] NUREG-1860, Table 6-1. Conservative value. NUREG 1860 doses provided as a range: 25 rem- 100 rem (Midpt = 62.5)
	1E-05> f ≥1E-06	200 rem (2000 mSv)	See Figure 3-2.[c] NUREG-1860, Table 6-1. Midpoint selected as doses provided as a range: 100 rem- 300 rem
	1E-06> f ≥5E-07	350 rem (3500 mSv)	See Figure 3-2.[c] NUREG-1860, Table 6-1. Midpoint selected as doses provided as a range: 300 rem- 400 rem
	5E-07> f	500 rem (5000 mSv)	
NGNP (INL/EXT-10-19521) ‡	1E-04> f ≥5E-07	300 rem (3000 mSv) 750 rem (7500 mSv)	See Figure 3-1.[c] (Also see INL/EXT-20-60394) ††. Based on an ISO-risk line anchored at endpoints. Prompt Mortality Goal Measure out to 1.6 km
NEI 18-04§ RG 1.233#	1E-04> f ≥5E-07	25 rem (250 mSv) 750 rem (7500 mSv)	See Figure 3-3.[c] Based on an ISO-risk line anchored at endpoints.
DOE-STD-3009-2014**	1E-04> f	< 25 rem (250 mSv)	Note. DOE-STD-3009-2014. DOE does not define Public Dose > 25 rem.

Sources: *NRC and CNSC (2021); †NRC (2007); ‡INL (2010); §NEI (2019b); #NRC (2020a);** DOE (2014); ††INL (2020).
[a]May be expressed in plant years or reactor years to reflect the potential for multi-reactor module facilities.
[b]The dose limits defined by the F-C curve are to be calculated consistent with the times and distances (i.e., either at the exclusion area boundary (EAB) for low doses or, for higher doses, the worst 2-hour dose at the EAB and the dose at the outer edge of the low-population zone (LPZ) for the duration of the event)
[c]All figure callouts refer to this document, unless otherwise noted.

Figures 3-1 through 3-5 illustrate frequency-consequence criteria proposed by the NRC, Next Generation Nuclear Plant (NGNP), NEI, compared to NRC and Canadian dose criteria and the proposed criteria in the DA PAM, as discussed in the comparisons in Table 3-1, Table 3-2 and Table 3-3 above.

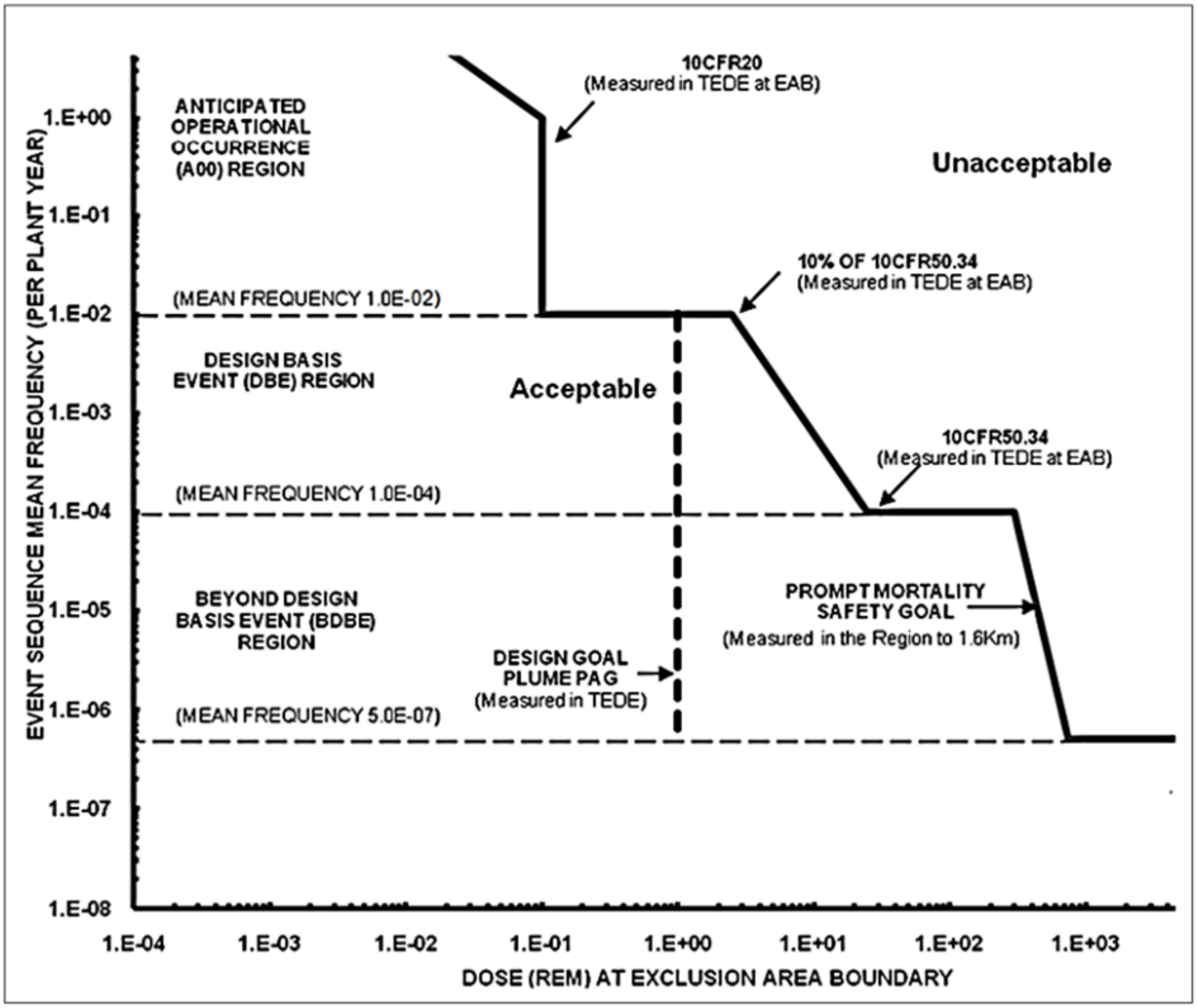

Figure 3-1. NGNP Frequency Consequence Criteria. Reproduced from Figure 3 in INL (2010).

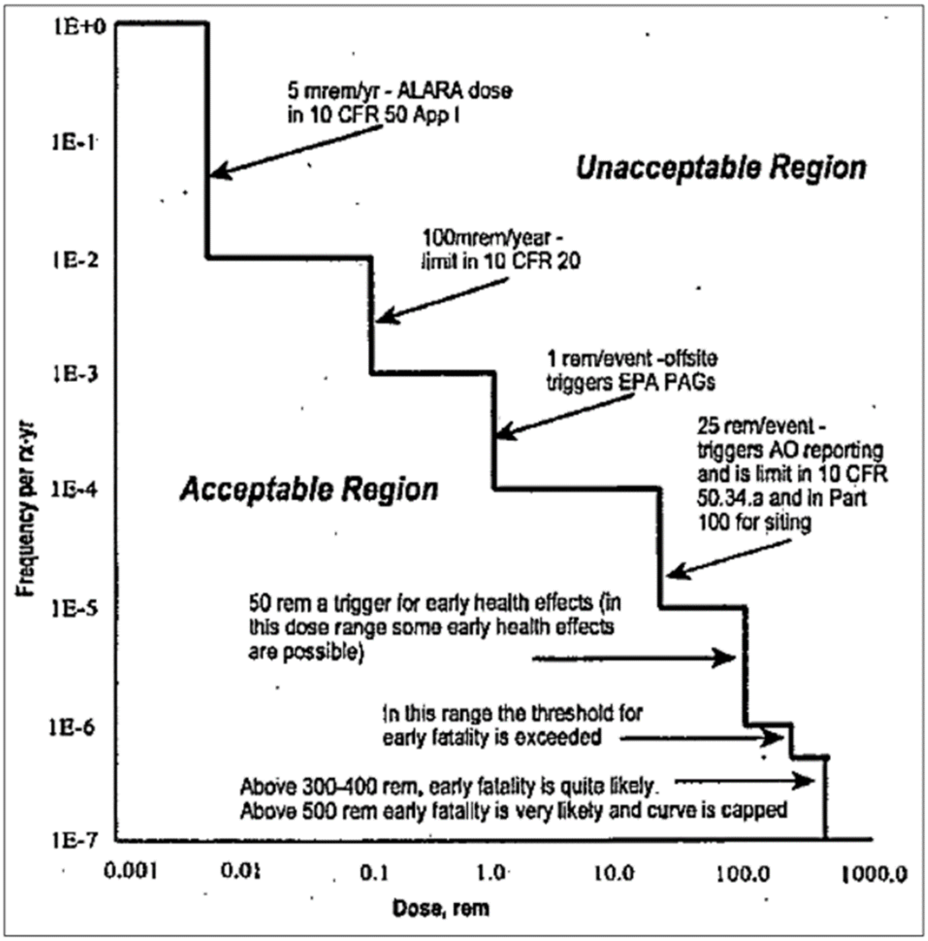

Figure 3-2. NRC NUREG-1860 Frequency-Consequence Criteria. Reproduced from Figure 3-3 in NRC (2007).

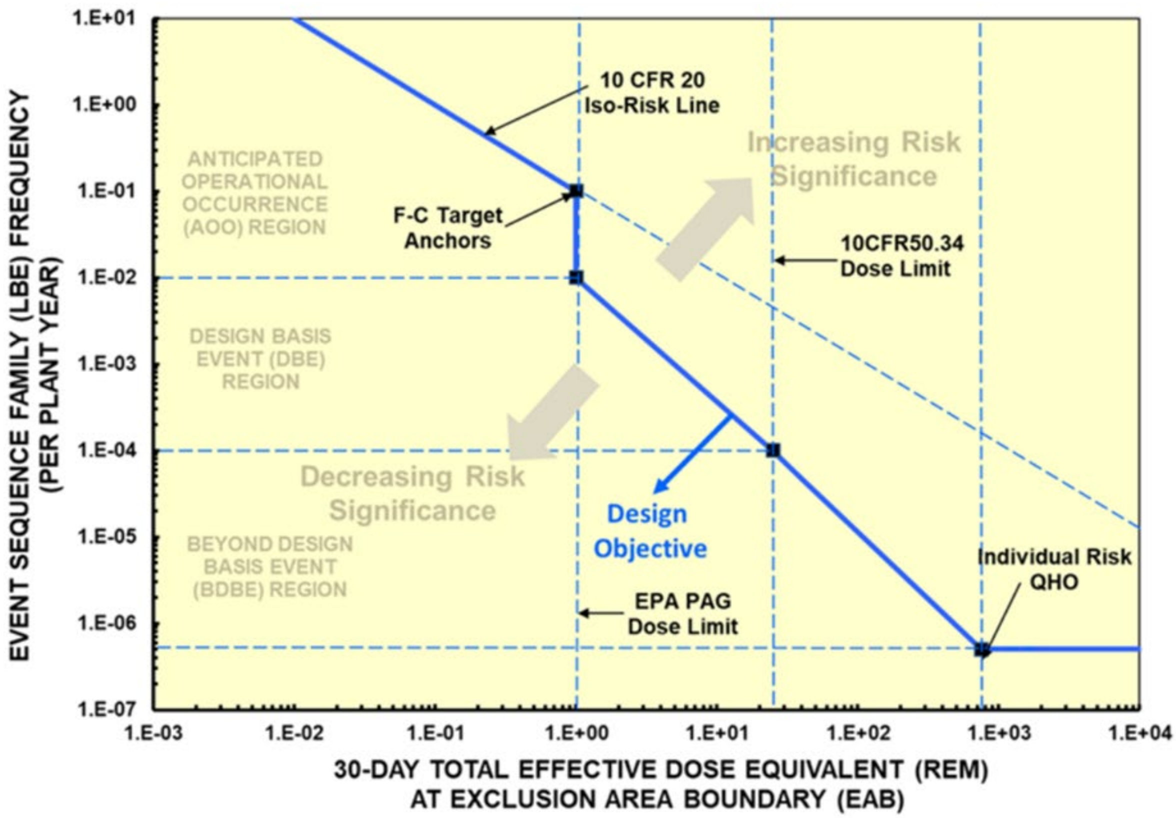

Figure 3-3. Frequency-Consequence Evaluation Criteria Proposed for LMP. Reproduced from Figure 4-5 in INL (2020).

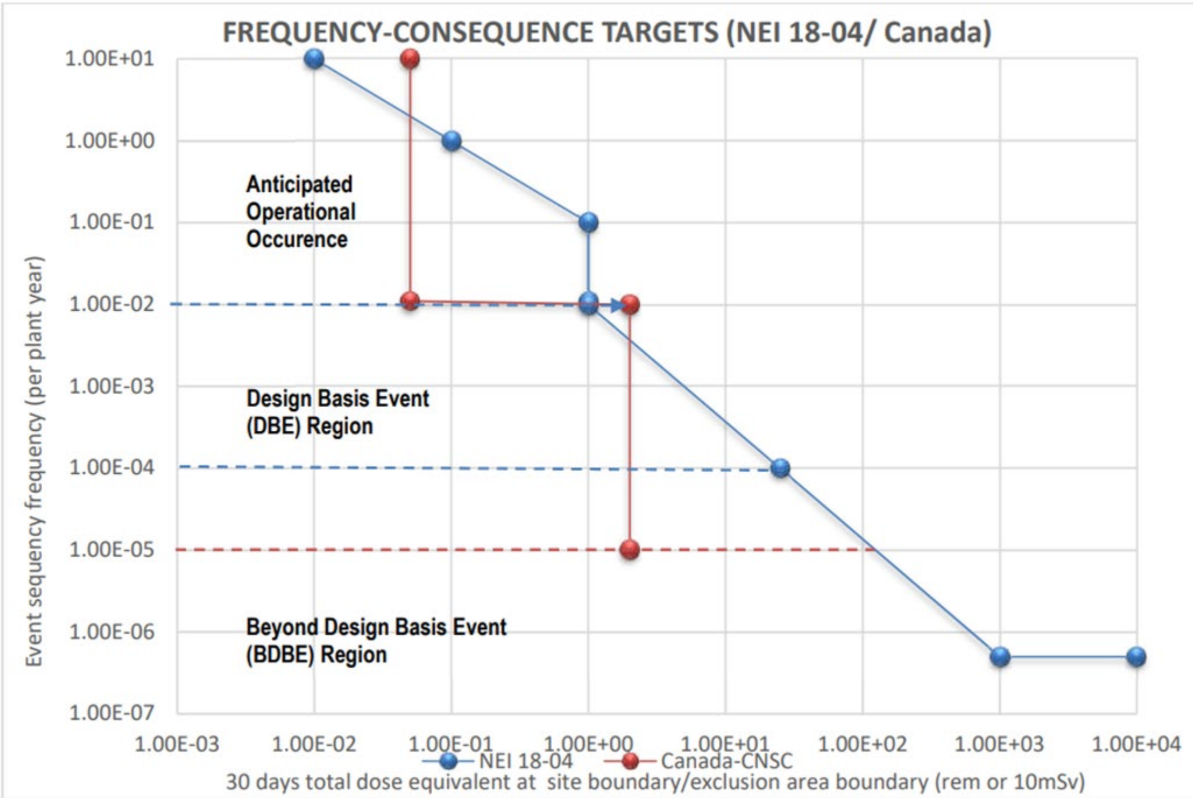

Figure 3-4. Comparison of NEI 18-04 and CNSC Frequency-Consequence Targets. Reproduced from Figure 9 in NRC and CNSC (2021).

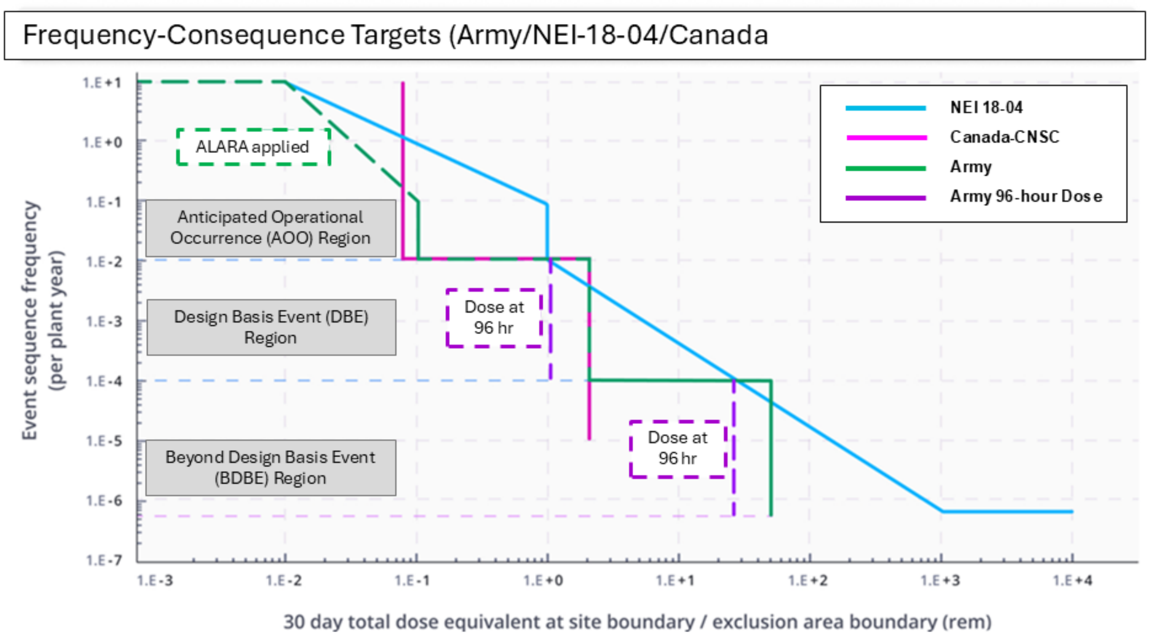

Figure 3-5. ARO DA PAM Comparison to NEI 18-04 and CNSC Frequency-Consequence Targets. Source data from NEI (2019b) and CNSC (2014b).

4.0 Reactor Safety Design Criteria

The RSDCs apply to SSCs that implement or support the key safety functions as defined in Chapter 6 the draft AR 50-7. Those functions are:

1. Control of Reactivity
2. Cooling of Radioactive Materials (Core Heat Removal)
3. Confinement of Radioactive Materials
4. Shielding against Radiation

Safety classifications within the RDSCs apply at that level and higher classifications. The criteria are written to state the minimum level of safety classification and apply to any SSC rated higher than the expressed minimum level of safety classification. For example, where a design criterion states "Safety-Significant SSCs" the criterion applies to SSCs classified as Safety-Significant, Safety-Related, and Safety-Critical.

The basis for including a reactor safety design criterion is predicated upon its support of one of the four key safety functions and informed by application of similar requirements in one or more of the following:

- NRC Regulations in Appendix A to 10 CFR Part 50, General Design Criteria (GDC)
- NRC's Regulatory Guide 1.232 (RG 1.232) applies the GDCs to Advanced Reactor Design Criteria (ARDC) and MHTGR design criteria (MHTGR-DC) (NRC 2018b)
- IAEA's Specific Safety Requirements (SSR) No. SSR-2/1, Revision 1 (IAEA 2016a)
- IAEA's Technical Document (TECDOC) 1936 (TECDOC-1936) (IAEA 2020)
- IAEA's SSR No. SSR-3 establishes requirements for the safety of research reactors (IAEA 2016b).

The approach to the development of the RSDC was informed by prior guidance and reports, including:

- INL-EXT-14-31179, *Guidance for Developing Principal Design Criteria for Advanced (Non-Light Water) Reactors* (INL 2014)
- Regulatory Guide 1.232, *Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors* (NRC 2018b)
- TECDOC-1936, *Applicability of Design Safety Requirements to Small Modular Reactor Technologies Intended for Near Term Deployment* (IAEA 2020).

Table 4-1 of this document provides the bases for the RSDC developed for the Army as contained in Appendix C of the draft DA PAM.

Table 4-1.
Reactor Safety Design Criteria

Reactor Safety Design Criteria	Title and Content	Basis or Principle
RSDC-01	Graded Approach. Safety-significant structures, systems, and components (SSCs) will be designed, manufactured, constructed, assembled, installed, erected, and tested commensurate with the importance of the safety functions to be performed.	This criterion establishes a standard for quality to have reasonable assurance that the SSCs will perform their safety functions under all operating conditions. The wording of the requirement allows the Army to use a graded approach for quality standards for the nuclear power plant. As the importance to safety increases, so do the quality and records requirements.
RSDC-02	Codes and Standards. Where the designer uses generally recognized codes and standards, the designer will identify and evaluate the codes and standards to determine their applicability, adequacy, and sufficiency to assure that Safety-significant SSCs will satisfactorily perform their safety functions. The designers will supplement or modify the codes and standards as necessary to assure a quality product in keeping with the required safety function.	The Army will establish or endorse acceptable codes to which the designer must adhere and monitor. The designer may use a graded approach to the use of codes and the extent of deviation from the code is based on the safety-significance. As the significance increases, so does the level of adherence, and the Army should allow less deviation from the code.
RSDC-03	Quality Assurance. The designer and manufacturer will establish and implement a quality assurance program to provide adequate assurance that Safety-significant SSCs will satisfactorily perform their safety functions. The reactor design authority and operating organization will maintain and control the appropriate records of the design, manufacturing, construction, assembling, installation, erection, testing and maintenance of Safety-significant SSCs throughout the life of the nuclear power plant.	In accordance with Army requirements, the designer and manufacturer will establish and maintain quality assurance and quality control programs and records ensure conformance to strict requirements, reduce the risk associated with operating nuclear power plants, and provide high confidence that Safety-Significant SSCs reliably perform their safety function throughout the life cycle.
RSDC-04	Natural Phenomena Hazards. Safety-significant SSCs shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunamis, and seiches without loss of capability to perform their safety functions. The design bases for these SSCs shall reflect: (1) The NPH design requirements of Table C-2 of this Appendix. (2) Appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena. (3) The importance of the safety functions to be performed.	This criterion establishes a design requirement to use climatological, seismic, and other natural phenomena information in the design of the SSCs to withstand the effects that could be reasonably foreseen during the life of the nuclear power plant and SSCs.

Table 4-1.
Reactor Safety Design Criteria

Reactor Safety Design Criteria	Title and Content	Basis or Principle
RSDC-05	Fire Protection. Safety-significant SSCs will be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and fire-resistant materials will be used wherever practical throughout the nuclear power plant, particularly in locations with Safety-critical SSCs. The design will provide fire detection and firefighting systems of appropriate capacity and capability to minimize the adverse effects of fires on Safety-significant SSCs, commensurate with the SSC's importance. Firefighting systems will be designed to ensure that their rupture or inadvertent operation does not significantly impair the safety capability of Safety-significant SSCs, commensurate with the SSC's importance.	Fires can significantly impair the performance of Key Safety Functions by damaging Safety-Significant SSCs and limiting the ability of response personnel from mitigating an accident. These criteria reduce the risks associated with fires and explosive events while operating nuclear power plant. They provide reasonable assurance that the Safety-Significant SSCs will perform their Key Safety Functions during and after a fire or explosive event during all modes of operation and at all points in the nuclear power plant's life cycle. The Army should consider the special or unique fire protection issues for a nuclear power plant. The design should consider the effects of sprays or chemical effects on the SSCs during fire suppression.
RSDC-06	Environmental and Dynamic Effects. Safety-critical SSCs will be designed to accommodate for the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.	The design must be sufficiently robust to adjust to changing environmental and operating conditions and withstand the dynamic effects of internal and external operating systems without challenging Safety-Critical SSCs and the performance of the Key Safety Functions. This criterion reduces the risk associated with changing environmental and operating conditions and the dynamic effects, such as changes in pressures and temperatures, while operating nuclear power plant under normal and design extension conditions. The interactions of and interfaces between two or more systems that share SSCs need to be analyzed to ensure that systems with lower operating parameters are protected from the higher operating conditions that could exceed the design specifications of the lower parameter systems. This criterion provides reasonable assurance that Safety-Critical SSCs will perform Key Safety Functions during and after an event during all modes of operation and at all points in the nuclear power plant's life cycle.

Table 4-1.
Reactor Safety Design Criteria

Reactor Safety Design Criteria	Title and Content	Basis or Principle
RSDC-07	Inspection, Testing, and Maintenance. Safety-significant SSCs will be designed to allow testing, inspecting, material surveillance, assessing, and evaluating the necessary quality, maintenance, repairs, operations, and conditions for the operability and functional performance, under conditions as close to design as practical, to ensure that the SSC is able to perform its required safety function when needed.	The design of Safety-Significant SSCs must provide for adequate inspection, testing, calibrating, and maintenance to ensure their availability and reliability to perform their function when needed. The inspections, tests, calibrating, and surveillances should not cause transients or interruptions to the safety functions or preventing the timely performance of them. The inspections, tests, and surveillances should be performed as appropriate under the conditions as close to the design as practical, test the full range of operations, interfaces, dependencies, and sequences that bring systems into operation. This criterion addresses Inspection, Testing, and Maintenance criteria found in several NRC GDC and removes the specific examples of what requires testing. The RSDC are to be implemented in conjunction with a thorough analysis of the hazards and controls associated with the reactor which will identify the SSCs and their critical characteristics requiring IT&M.
RSDC-08	Sharing SSCs. The design will not share Safety-critical SSCs between individual nuclear power plants unless it can be shown that such sharing will not significantly impair their ability to perform their KEY SAFETY FUNCTIONS, including, in the event of an accident, an orderly shutdown and cooldown of the remaining nuclear power plants.	This criterion requires that the Safety-Critical SSCs will perform Key Safety Functions without relying upon other nuclear power plants. This criterion reduces the risks associated with a common cause failure occurring in SSCs and over burdening SSCs that could impact the safe operation or shutdown of multiple nuclear power plants and their ability to perform Key Safety Functions and prevent doses to the public in excess of allowable limits. The sharing of Safety-Critical SSCs to perform Key Safety Functions for more than one nuclear power plant requires analyses and evaluations to determine the limitations and conditions under which sharing doesn't impair the Safety-Critical SSCs ability to perform Key Safety Functions during all modes of operation, design extension conditions, and at all points in the life cycle. Analysis must indicate that sufficient capacity exists within the shared Safety-Significant SSCs to perform the Key Safety Functions for simultaneous shutdowns and cooldowns of all the nuclear power plants sharing the Safety-Significant SSCs.

Table 4-1.
Reactor Safety Design Criteria

Reactor Safety Design Criteria	Title and Content	Basis or Principle
		Each nuclear power plant requires the ability to independently shut down the nuclear reactor while not relying on the Safety-Critical SSCs for other nuclear reactors. Prohibiting or limiting the sharing of safety-critical SSCs provides reasonable assurance that SSCs will perform Key Safety Functions when needed during all modes of operation and at all points in the life cycle and prevent doses to the public in excess of allowable limits.
RSDC-09	Instrumentation and Controls. Controls will be provided to maintain the variables and systems within prescribed operating ranges to perform their KEY SAFETY FUNCTIONS. These controls will be located to facilitate their availability when needed and allow for safe access for the operators under anticipated environmental conditions. Instrumentation will be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, accident conditions, as appropriate for design extension conditions, to ensure adequate safety, including those variables and systems that indicate performance of KEY SAFETY FUNCTIONS.	The instrumentation and controls system provides configuration control for Safety-Critical SSCs and reasonable assurance the SSCs will be available to perform Key Safety Functions when needed during all modes of operation, including design extension conditions, and at all points in the life cycle. This criterion provides for the monitoring of system parameters, alerting operators of adverse nuclear power plant conditions, supporting Key Safety Functions, and controlling reactivity and nuclear power plant operations to maintain sufficient margins to prevent and recover from abnormal and accident conditions or design extension conditions during all operational and accident conditions. This criterion reduces the risk associated with monitoring system performance and operations and their ability to remain within prescribed ranges with sufficient margins to account for uncertainties and to perform Key Safety Functions without exceeding design limits.
RSDC-10	Control Station. An operations control station will be provided in which actions can be taken to operate the nuclear power plant safely under normal conditions, monitor KEY SAFETY FUNCTIONS, and, as necessary, to maintain it in a safe condition under accident and design extension conditions.	This criterion allows the reactor operators and their supervisor to monitor, control, and evaluate system variables and systems to operate the nuclear power plant under all normal, design extension, and accident conditions from a single point using sound human-machine interface design principles. As used here, “as necessary” describes the time after the onset of accident conditions where the nuclear power plant’s SSC performed their Key Safety Functions, and the reactivity coefficient (k_{eff}) is less than 1. If the control station was evacuated, the control station habitability is safe to reoccupy and from which to operate or monitor the nuclear power plant conditions. This criterion allows for the immediate

Table 4-1.
Reactor Safety Design Criteria

Reactor Safety Design Criteria	Title and Content	Basis or Principle
		recognition of off normal, design extension, and accident conditions and coordination of actions to return the nuclear power plant to normal operating parameters, issue protective action orders, notify staff, and shutdown the nuclear power plant.
RSDC-11	Barrier Integrity. The design of the nuclear power plant, including structures, reactor vessel, and reactor system (including, but not limited to, the fuel, reflector, control rods, core barrel, and structural supports) will be such that their integrity is maintained during normal operations, anticipated operational occurrences, and postulated accidents: (1) To ensure the geometry for passive removal of residual heat from the reactor core to the ultimate heat sink. (2) To permit sufficient insertion of the neutron absorbers to provide for reactor shutdown. (3) To ensure credited containment barriers remain within specified parameters. (4) To ensure shielding is maintained.	This criterion requires that the design provides for robust structural integrity of the reactor vessel, reactor coolant boundary, and systems to withstand the effects of accident and post-accident conditions while: • maintaining core cooling geometry to passively transport the residual/decay heat from the core and system to the ultimate heat sink • allowing the injection of neutron absorbers (physical or chemical) to maintain $k_{eff} < 1$. Ensuring confinement function is maintained
RSDC-12	Safety Margin. The reactor core and associated coolant, control, and protection systems will be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.	This criterion provides that the design of the reactor core is sufficiently robust to withstand the effects of equipment failures, human errors, and system malfunctions, and maintain the effectiveness of the fission product barriers by avoiding damage to the nuclear power plant and barriers so that the safety limits of the fuel and reactor coolant design conditions are not exceeded, and doses remain within allowable limits for operating personnel and the public. This criterion would require the reactor coolant boundary to maintain an extremely low probability of abnormal leakage, of a rapidly propagating failure, and a gross rupture.
RSDC-13	Inherent Feedback. The reactor core and associated systems that contribute to reactivity feedback will be designed so that, in the power operating range, the net effect of the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity.	This criterion requires that the design have a strong negative feedback to changes in reactivity. This design feature contributes to stable reactor operations when parameters change, the net effect on power remains constant and suppresses oscillations. Common coefficients of reactivity are moderator temperature, fuel temperature, and void and when combined constitute a power coefficient of reactivity.
RSDC-14	Containment. SSCs performing the KEY SAFETY FUNCTION of confining radiological materials will be designed to control the release of radioactivity to the environment and to	This criterion requires functional confinement of radioactive materials from fuel, fission products, and other activated materials exists to prevent unplanned or uncontrolled releases of materials

Table 4-1.
Reactor Safety Design Criteria

Reactor Safety Design Criteria	Title and Content	Basis or Principle
	ensure that design parameters are not exceeded for as long as postulated accident conditions require.	to the environment and prevent doses to the operating personnel and the public in excess of allowable limits. This criterion requires that confinement be sufficiently robust to withstand normal operating, design extension, and accident conditions and not challenge other safety-critical SSC during all points in the life cycle and prevent doses to the public in excess of allowable limits.
RSDC-15	Electric Power. Electric power systems provided to permit functioning of Safety-significant systems and components will be designed consistent with credited performance and classification.	This criterion requires that when a safety-significant SSC requires electricity to perform its safety-significant function, the SSCs supplying the electricity must be supplied to perform the SSC's safety functions (classified as SS).
RSDC-16	Reactivity Control System. The reactivity control system will be designed to perform KEY SAFETY FUNCTIONS by: (1) Automatically (a) Detecting precursor conditions for ~~anticipated operational occurrences and~~ design-basis events. (b) Inserting negative reactivity to achieve and maintain safe shutdown conditions during anticipated operational occurrences and design-basis accidents. (2) Ensuring specified acceptable design limits (system and fuel) aren't exceeded. (3) Controlling positive reactivity addition amounts and rates during planned reactor power changes to prevent fuel damage and maintain reactor cooling capability. (4) Keeping the reactor shut down for fuel loading/unloading, inspection, repair, and during transport.	This criterion requires the reactivity control system basic protection functions of the system to prevent core damage from excessive reactivity additions and monitor core conditions and activate Key Safety Functions of the safety-significant SSCs. The system must activate the reactor protective functions in the absence of operation actions (automatically).
RSDC-17	Reactivity Control System. The reactivity control system will: (1) Assure that the effects of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the KEY SAFETY FUNCTION or the protection function or will be demonstrated to be acceptable on some other defined basis. (2) Be separated from normal operational control systems to the extent that failure of any single control system component or channel, or	This criterion requires that the design provide a protection system to prevent or reverse adverse conditions that could lead to core damage and radionuclide release to the environment. This criterion requires that reactor protection system always remain operable using one or more redundant channels. The design of the protection system must permit testing, maintenance, and conduct of operations without damaging other redundant channels, causing failure of the reactor protection system, or prevent the timely automatic actuation of the

**Table 4-1.
Reactor Safety Design Criteria**

Reactor Safety Design Criteria	Title and Content	Basis or Principle
	failure or removal from service of any single protection system component or channel that is common to the control and protection systems, leaves intact a system satisfying all reliability, redundancy, and independence requirements to perform the KEY SAFETY FUNCTION of the protection system. Interconnection of the protection and control systems will be limited to ensure that KEY SAFETY FUNCTIONS are not significantly impaired. (3) Default to a safe state or another acceptable state if issues arise.	system.
RSDC-18	Heat Transfer. SSCs performing the KEY SAFETY FUNCTION of removing heat from the reactor will be designed with sufficient margin to ensure that the system safety function will be performed when stressed under operating, maintenance, testing, and postulated accident conditions. For normal operations and anticipated operational occurrences, the system will transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable radionuclide release design limits and the design conditions of the reactor coolant boundary are not exceeded. Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to ensure that the KEY SAFETY FUNCTION can be accomplished under normal operating and accident conditions.	This criterion requires that the designers understand the characteristics of the materials for the design, fabrication, assembling, and testing of the reactor coolant boundaries and how the materials respond to changes in temperature, pressure, radiation flux, stresses, and flow; chemical interactions and properties of the various materials; foreign materials and contaminates; and the corresponding uncertainties or unknowns over the operating and accident conditions of the reactor. The design will reflect consideration of service temperatures, service degradation of material properties, creep, fatigue, stress rupture, and other conditions of the boundary material under operating, maintenance, testing, and design basis accident conditions and the uncertainties in determining 1) material properties; 2) the effects of irradiation and coolant composition, including contaminants and reaction products, on material properties; 3) residual, steady-state, and transient stresses; and 4) size of flaws. The design must reflect sufficient margin that when the reactor and reactor coolant boundary respond to the various stresses, the materials do not become brittle and are sufficiently tough to minimize the probability of a destructive propagation of cracks. The design of the heat removal system is robust to withstand a single component failure and still perform its Key Safety Function to prevent SAFDLs and design conditions of the reactor coolant boundary are not exceeded. The design of the heat removal system when combined with the reactivity control system will have sufficient margin to compensate for stuck rods during design basis accidents to cool the core and prevent exceeding SAFDLs.

Table 4-1.
Reactor Safety Design Criteria

Reactor Safety Design Criteria	Title and Content	Basis or Principle
RSDC-19	Heat Transfer. A system to transfer heat from Safety-significant SSCs to an ultimate heat sink will be provided, as necessary, to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions. Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities will be provided to ensure performance of the KEY SAFETY FUNCTIONS.	This criterion requires that the design provide for the cooling of the Safety-Significant SSCs sufficient to transfer the combined heat load from the SSCs to the ultimate heat sink sufficient to prevent the interruption of the SSCs performing their Key Safety Function.
RSDC-20	Containment System. A containment system consisting of one or multiple barriers internal or external to the reactor will be provided to control the release of radioactivity to the environment and ensure doses to the operators, Army personnel, and members of the public are below authorized limits and to as low as reasonably achievable during normal operations and anticipated operations occurrences, and perform the KEY SAFETY FUNCTIONS of radiation protection and confinement for as long as postulated accident conditions require.	This criterion requires that the design provide functional containment using multiple barriers to control the release of radionuclides to the environment, that design limits are not exceeded, and that the Key Safety Functions can be performed during and after accident conditions. This criterion incorporates the necessity of designing the nuclear power plant to ensure that doses to the operating personnel, Army Personnel (Soldiers or civilians) and members of the public do not receive doses in excess of acceptable limits and is ALARA. This criterion could be partially satisfied by the fuel if using TRISO fuel or other accident tolerant fuel. For example, the typical barriers for an HTGR using TRISO fuel could be: 1. The fuel particle kernel 2. The fuel particle coatings (silicon carbide and pyrocarbon coatings) 3. The core graphite and carbonaceous materials 4. The helium pressure boundary The reactor enclosure/building
RSDC-21	Control of Releases. Means to control the release of radioactive materials in gaseous and liquid effluents within regulatory limits and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences, will be provided.	This criterion requires that the design provide systems or the means to sufficiently control the releases of gaseous and liquid effluents produced during normal operations and anticipated operational occurrences. The criterion requires that the means to control, manipulate, store, and treat radionuclide wastes or byproducts that are produced during normal operations and anticipated operational occurrences.
RSDC-22	Fuel Handling. The fuel storage and handling, radioactive waste, and other systems that may	This criterion requires that fuel storage and handling, radioactive waste, and other

Table 4-1.
Reactor Safety Design Criteria

Reactor Safety Design Criteria	Title and Content	Basis or Principle
	contain radioactivity will be designed to assure adequate safety under normal and postulated accident conditions. These systems will be designed: (1) With a capability to permit appropriate periodic inspection and testing of components important to safety. (2) With suitable shielding for radiation protection. (3) With appropriate containment and filtering systems. (4) With a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal. (5) To prevent significant reduction in fuel storage cooling under accident conditions.	radionuclide containing systems be subject to: (1) monitoring of conditions that may result in a loss of cooling and excessive radiation levels, and (1) a systematic program of inspection and functional testing of its key components, operability, performance, shielding, cooling and leak tight integrity. The program of inspection and testing must include testing starting logic and interfaces that the system relies upon to fulfill any of its Key Safety Function.
RSDC-23	Critically Prevention. Criticality in the fuel storage and handling systems, including during transport, will be prevented by physical systems or processes, preferably by use of geometrically safe configurations.	This criterion requires that the design provide for a system, physical or geometric characteristics, or other means to prevent criticality in systems that contain or control fuel and radioactive wastes, and for fuel storage.
RSDC-24	Effluent Monitoring. Means will be provided for monitoring effluent discharge paths and plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.	This criterion requires a system of detector and indicator equipment to control and monitor effluents and the plant and environment to detect radioactivity that may be released from normal operations, anticipated operational occurrences, accident conditions, and post-accident conditions.

Table 4-2 provides a crosswalk from the ARP RSDC contained in Appendix C of the draft DA PAM to the key NRC General Design Criteria and IAEA's Safety Standard Requirements SSR 2/1 requirements. Both considered in the development of the RSDC.

Table 4-2.
Crosswalk ARP RSDC to NRC GDC and IAEA SSR 2/1

Reactor Safety Design Criteria (RSDC)	NRC General Design Criteria	SSR-2/1
RSDC-01	1, 30	2, 3, 22, 47, 72, 75, 76
RSDC-02	1, 30	2, 9, 15, 18, 23, 24, 25, 26, 47, 48, 63
RSDC-03	1	2, 3, 14, 47
RSDC-04	2	17
RSDC-05	3	17, 74
RSDC-06	4	17, 20, 30, 31, 40, 73

Table 4-2.
Crosswalk ARP RSDC to NRC GDC and IAEA SSR 2/1

Reactor Safety Design Criteria (RSDC)	NRC General Design Criteria	SSR-2/1
RSDC-07	18, 32, 36, 37,39,40, 41, 42, 43, 44, 45, 46, 52, 53	6, 29, 30, 47, 68
RSDC-08	5	20, 31, 33
RSDC-09	13	16, 20, 31, 45, 46, 59, 60, 61, 66
RSDC-10	19, 33	16, 20, 32, 37, 65, 67
RSDC-11	10, 14, 15, 16, 31, 34, 38, 50, 51	34, 43, 44, 51, 52, 53, 54
RSDC-12	10, 14	15, 45, 47, 49
RSDC-13	11, 12	45, 46
RSDC-14	16, 41, 50, 51	20, 31, 34, 43, 54, 55, 58
RSDC-15	17	41, 68
RSDC-16	20, 23, 25, 26, 27, 28, 29	20, 21, 24, 25, 46, 60, 61
RSDC-17	21, 22, 24	21, 24, 25, 40, 60, 61, 64
RSDC-18	14, 30, 31, 33, 34, 35, 51, 54	21, 24, 25, 40, 47, 51, 52, 53, 77
RSDC-19	38, 39, 44, 55, 56, 57	21, 24, 34, 40, 51, 53, 70, 73
RSDC-20	16	5, 20, 54, 55, 81
RSDC-21	60	34, 50, 78, 79
RSDC-22	61, 63	34, 80
RSDC-23	62	17, 34, 80
RSDC-24	64	34, 35, 56, 82

Table 4-3 provides a crosswalk between the NRC General Design Criteria, RG 1.232 (Advanced Reactor Criteria), and the ARP RSDC in Appendix C of the draft DA PAM.

Table 4-3.
Crosswalk of NRC GDC and RG 1.232 to ARP RSDC

Nuclear Regulatory Commission NRC General Design Criteria	Nuclear Regulatory Commission PDC for Non-light-water Reactors	Reactor Safety Design Criteria (RSDC)
1	GDC 1	RSDC-01
1	GDC 1	RSDC-02
1	GDC 1	RSDC-03
2	GDC 2	RSDC-04
3*	ARDC 3	RSDC-05
4	ARDC 4 SFR-DC 4 MHTGR-DC 4	RSDC-06
5	GDC 5	RSDC-08
10	ARDC 10 MHTGR-DC 10	RSDC-11 RSDC-12
11	ARDC 11	RSDC-13
12	ARDC 12	RSDC-13
13*	ARDC 13 SFR-DC 13 MHTGR-DC 13	RSDC-09
14	ARDC 14 SFR-DC 14 MHTGR-DC 14	RSDC-11 RSDC-12 RSDC-18
15	ARDC 15 SFR-DC 15 MHTGR-DC 15	RSDC-11 RSDC-12

Table 4-3.
Crosswalk of NRC GDC and RG 1.232 to ARP RSDC

Nuclear Regulatory Commission NRC General Design Criteria	Nuclear Regulatory Commission PDC for Non-light-water Reactors	Reactor Safety Design Criteria (RSDC)
16	GDC 16 SFR-DC 16 MHTGR-DC 16	RSDC-11 RSDC-14 RSDC-20
17	ARDC 17 MHTGR-DC 17	RSDC-15
18	ARDC 18	RSDC-07
19	ARDC 19 SFR DC 19	RSDC-10
20	GDC 20 MHTGR-20	RSDC-16
21	GDC 21	RSDC-17
22	GDC 22	RSDC-17
23	GDC 23 SFR DC 23	RSDC-16
24	GDC 24	RSDC-17
25	ARDC 25 MHTGR-DC 25	RSDC-16
26	ARDC 26 MHTGR-DC 26	RSDC-16
27	See GDC 26	
28	ARDC 28 SFR-DC 28 MHTGR-DC 28	RSDC-16
29	GDC 29	RSDC-16
30	ARDC 30 SFR-DC 30 MHTGR-DC 30	RSDC-01 RSDC-02 RSDC-18
31	ARDC 31 SFR-DC 31 MHTGR-DC 31	RSDC-11 RSDC-18
32	ARDC 32 SFR-DC 32 MHTGR-DC 32	RSDC-07
33	ARDC 33 SFR-DC 33	RSDC-18
34	ARDC 34 SFR-DC 34 MHTGR-DC 34	RSDC-11 RSDC-18
35	ARDC 35	RSDC-18
36	ARDC 36 MHTGR-DC 36	RSDC-07
37	ARDC 37 MHTGR-DC 37	RSDC-07
38	ARDC 38	RSDC-11 RSDC-19
39	ARDC 39	RSDC-07 RSDC-19
40	ARDC 40	RSDC-07
41	ARDC 41	RSDC-07 RSDC-14
42	GDC 42	RSDC-07
43	ARDC 43	RSDC-07
44	ARDC 44 MHTGR-DC 44	RSDC-07 RSDC-19

Table 4-3.
Crosswalk of NRC GDC and RG 1.232 to ARP RSDC

Nuclear Regulatory Commission NRC General Design Criteria	Nuclear Regulatory Commission PDC for Non-light-water Reactors	Reactor Safety Design Criteria (RSDC)
45	ARDC 45	RSDC-07
46	ARDC 46	RSDC-07
50	ARDC 50 SFR-DC 50	RSDC-11 RSDC-14
51	ARDC 51 SFR-DC 51	RSDC-11 RSDC-14 RSDC-18
52	ARDC 52 SFR-DC 52	RSDC-07
53	ARDC 53 SFR-DC 53	RSDC-07
54	ARDC 54 SFR-DC 54	RSDC-18
55	ARDC 55 SFR-DC 55	RSDC-19
56	ARDC 56 SFR-DC 56	RSDC-19
57	ARDC 57 SFR-DC 57	RSDC-19
60	GDC 60	RSDC-21
61	ARDC 61	RSDC-22
62	GDC 62	RSDC-23
63	GDC 63	RSDC-22
64	ARDC 64 SFR-DC 64 MHTGR-DC 64	RSDC-24
None†	MHTGR-DC 70	RSDC-11
None†	MHTGR-DC 71	RSDC-11
None	MHTGR-DC 72	RSDC-07

**Appendix R Requirement added*
† 10 CFR 50.34(f) Post-Three Mile Island Requirement

Table 4-4 provides crosswalks between the IAEA's Safety Standard Requirements and the ARP RSDC in Appendix C of the draft DA PAM. Note that a number of the Specific Safety Requirements are process-related and are not directly related to established Reactor Safety Design Criteria. These requirements are noted as Not Design Criterion (NDC).

Table 4-4.
Crosswalk of IAEA SSR 2/1 and SSR 3 to ARP RSDC

IAEA		ARP
SSR-2/1	**SSR-3**	**Reactor Safety Design Criteria**
1	2	NDC
2	4	RSDC-01 RSDC-02 RSDC-03
3	4	RSDC-01 RSDC-02 RSDC-03
4	7	NDC

Table 4-4.
Crosswalk of IAEA SSR 2/1 and SSR 3 to ARP RSDC

IAEA		ARP
SSR-2/1	**SSR-3**	**Reactor Safety Design Criteria**
5	8	RSDC-20
6	9	RSDC-07
7	10	NDC
8	11	NDC
9	13	RSDC-02
10		NDC
11	14	RSDC-01
12	15	NDC
13		NDC
14	17	RSDC-03
15	21	RSDC-02 RSDC-12
16	18	RSDC-09 RSDC-10
17	19	RSDC-04 RSDC-05 RSDC-06 RSDC-23
18	13 [6.20, 6.24]	RSDC-02
19	20	NDC
20	22	RSDC-06 RSDC-08 RSDC-09 RSDC-10 RSDC-14 RSDC-16 RSDC-20
21	27	RSDC-16 RSDC-17 RSDC-18 RSDC-19
22	16	NDC
23	24	RSDC-02
24	26	RSDC-02 RSDC-16 RSDC-17 RSDC-18 RSDC-19
25	25	RSDC-02 RSDC-16 RSDC-17 RSDC-18
26	28	RSDC-02
27		NDC
28	71	NDC
29	31	RSDC-07
30	29	RSDC-06 RSDC-07
31	37	RSDC-06 RSDC-08 RSDC-09 RSDC-14
32	35	RSDC-10
33	--	RSDC-08
34	43	RSDC-11

Table 4-4.
Crosswalk of IAEA SSR 2/1 and SSR 3 to ARP RSDC

IAEA		ARP
SSR-2/1	**SSR-3**	**Reactor Safety Design Criteria**
		RSDC-14 RSDC-19 RSDC-21 RSDC-22 RSDC-23 RSDC-24
35		RSDC-24
36	32	NDC
37	32	RSDC-10
38	--	NDC
39	39	NDC
40	40	RSDC-06 RSDC-17 RSDC-18 RSDC-19
41	--	RSDC-15
42	41	NDC
43	44	RSDC-11 RSDC-14
44	44	RSDC-11
45	45	RSDC-09 RSDC-12 RSDC-13
46	46	RSDC-09 RSDC-13 RSDC-16
47	47	RSDC-01 RSDC-02 RSDC-03 RSDC-07 RSDC-12 RSDC-18
48		RSDC-02
49		RSDC-12
50		RSDC-21
51	47 [6.158]	RSDC-11 RSDC-18 RSDC-19
52	48	RSDC-11 RSDC-18
53		RSDC-11 RSDC-18 RSDC-19
54	42 43	RSDC-11 RSDC-14 RSDC-20
55	43	RSDC-14 RSDC-20
56		RSDC-24
57		NDC
58		RSDC-14
59	49	RSDC-09
60	49	RSDC-09 RSDC-16 RSDC-17

Table 4-4.
Crosswalk of IAEA SSR 2/1 and SSR 3 to ARP RSDC

IAEA		ARP
SSR-2/1	**SSR-3**	**Reactor Safety Design Criteria**
61	50	RSDC-09 RSDC-16 RSDC-17
62	51	RSDC-17
63	52	RSDC-02
64	27	RSDC-17
65	53	RSDC-10
66	54	RSDC-09
67	55	RSDC-10
68	56	RSDC-07 RSDC-15
69	60	RSDC-01
70		RSDC-19
71		RDSC-01
72	65	RSDC-01
73	64	RSDC-06 RSDC-19
74	61	RSDC-05
75	62	RSDC-01
76	63	RSDC-01
77		RSDC-18
78	59	RSDC-21
79	59	RSDC-21
80	58	RSDC-22 RSDC-23
81	8 34	RSDS-12 RSDS-14 RSDC-20
82	57	RSDC-24

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