National Reactor Innovation Center

Pacific Northwest NATIONAL LABORATORY

Advanced Nuclear Reactor Plant Parameter Envelope and Guidance

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NRIC-21-ENG-0001; PNNL-30992 | 2.18.2021

Summary

Pacific Northwest National Laboratory (PNNL) is supporting the National Reactor Innovation Center (NRIC) at Idaho National Laboratory (INL) by developing advanced nuclear reactor plant parameter envelopes (PPEs) to facilitate environmental reviews of potential future advanced reactor demonstration projects at INL and elsewhere in the United States. Two PPEs are developed in this report for two size ranges: (1) microreactors, which are defined for this PPE as single units with outputs of 60 MWt or less, and (2) small- to medium-sized advanced reactors with outputs above 60 MWt up to 1,000 MWt.

This report describes the methodology for developing the PPEs, including reactor vendor responses to NRIC questionnaires, input from INL staff, independent assessments by subject matter experts, and a review of regulatory requirements a vendor would have to meet during construction and operation. This report presents the compiled PPEs for surrogate plants derived from these inputs, lists documentation supporting the PPE, and provides recommendations for its use when developing an environmental impact assessment.

Acronyms and Abbreviations

AC	alternating current
ALARA	as low as reasonably achievable
ANR	advanced nuclear reactor
BWR	boiling-water reactor
BWXT	BWX Technologies
CFR	Code of Federal Regulations
CITRC	Critical Infrastructure Test Range Complex
CT	combustion turbine
dBA	A-weighted decibel
DOE	U.S. Department of Energy
DoD	U.S. Department of Defense
EBR-II	Experimental Breeder Reactor II
EIS	environmental impact statement
ER	Environmental Report
ESP	early site permit
ESRP	Environmental Standard Review Plan
FLIBE	fluorine-lithium-beryllium
FHR	fluoride salt-cooled high-temperature reactor
FPR	Fuel Processing Restoration
FSAR	Final Safety Analysis Report
FTE	full-time equivalent employee
GAIN	Gateway for Accelerated Innovation in Nuclear
GDC	General Design Criteria
GE	General Electric
GEIS	generic environmental impact statement
GHG	greenhouse gas
HALEU	high-assay low-enriched uranium
HTGR	high-temperature gas-cooled reactor
HTTR	high-temperature engineering test reactor
INL	Idaho National Laboratory
LFR	lead-cooled fast reactor
LFTR	liquid fluoride thorium reactor
LMR	liquid metal-cooled reactor
LRWS	liquid radioactive waste management system
LWR	light-water reactor
MFC	Materials and Fuels Complex
MSR	molten salt reactor
MSRE	Molten Salt Reactor Experiment
MWe	megawatt(s) electric
MWt	megawatt(s) thermal
NAAQS	National Ambient Air Quality Standards

NEI	Nuclear Energy Institute
NEPA	National Environmental Policy Act
NGCC	natural gas combined cycle
NRC	U.S. Nuclear Regulatory Commission
NRIC	National Reactor Innovation Center
ORIGEN	Oak Ridge Isotope Generation code
PBF	Power Burst Facility
PBR	pebble bed reactor
PCS	power conversion system
PNNL	Pacific Northwest National Laboratory
PPE	plant parameter envelope
PRISM	Power Reactor Innovative Small Module
PSEG	Public Service Enterprise Group
PWR	pressurized water reactor
RCCWS	reactor component cooling-water system
RCRA	Resource Conservation and Recovery Act
RD&D	research, design, and development
RIC	Regulatory Information Conference
ROW	right-of-way
RWDS	radwaste drain system
SCALE	Standardized Computer Analyses for Licensing Evaluation
SC-HTGR	steam cycle high-temperature gas-cooled reactor
SFR	sodium fast reactor
SME	subject matter expert
SMR	small modular reactor
SPE	site parameter envelope
TRISO	tri-structural isotropic
TRITON	Transport Rigor Implemented with Time-dependent Operation for Neutronic depletion (software)
TVA	Tennessee Valley Authority
UWS	utility water system
VTR	Versatile Test Reactor
ZPPR	Zero Power Physics Reactor

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1.0 Introduction

The National Reactor Innovation Center (NRIC) at Idaho National Laboratory (INL) in and near Idaho Falls, Idaho, was authorized under the Nuclear Energy Innovation Capabilities Act (Public Law 115–248) to provide innovators with resources and infrastructure for testing, demonstration, and performance assessment to accelerate demonstration and deployment of new advanced reactor technology concepts. NRIC plans to offer existing buildings and multiple undeveloped and previously developed sites at INL to advanced reactor developers for use in demonstrating a wide range of reactor technologies, designs, and sizes.

A National Environmental Policy Act (NEPA; 42 U.S.C. § 4321 *et seq*.) environmental review may be initiated to facilitate NRIC's mission before any commitments by individual reactor vendors are made. To evaluate the potential environmental impacts without knowing which particular designs may be deployed, NRIC has developed plant parameter envelopes (PPEs) based on potential advanced reactor demonstrations. These PPEs could be used by the U.S. Department of Energy, other federal agencies, and/or others to evaluate the largest or "bounding" environmental impacts of the deployment of any particular advanced reactor with design parameters falling within the envelopes. In this way, a NEPA review can be initiated prior to final design selection, thereby streamlining the regulatory review process for several buildings, sites, and reactor designs.

1.1 NRIC Background

NRIC accelerates the deployment of advanced nuclear energy through its mission to **inspire** stakeholders and the public, **empower** innovators, and **deliver** successful outcomes. NRIC is a national program led by INL, allowing collaborators to harness the world-class capabilities of the U.S. National Laboratory System. NRIC is charged with and committed to demonstrating advanced reactors.

NRIC accelerates technology from proof of concept to proof of operation by allowing innovators to leverage the U.S. government's investment in nuclear energy research, development, demonstration, and deployment. By bridging world-leading laboratory infrastructure and expertise with the promise of visionaries working to commercialize new nuclear energy systems, NRIC is enabling a new era of clean, affordable, reliable energy.

NRIC intends to provide existing facilities and other undeveloped and previously developed sites at INL to advanced reactor vendors for prototype technology testing and deployment. The existing Materials and Fuels Complex (MFC)-767 (Experimental Breeder Reactor II [EBR-II]), Zero Power Physics Reactor (ZPPR), and Power Burst Facility Building 613 (PBF-613; Critical Infrastructure Test Range Complex [CITRC]) Communications Research Facility buildings will be modified for use as parts of a technology "test bed" (INL 2020a). Vendors will be able to install, test, and operate prototype reactor technology, then remove the prototype upon completion of testing. In addition, undeveloped and previously developed outdoor sites have been identified as potential locations for construction and operation of full-scale reactor prototypes (INL 2020a).

The national deployment of advanced reactors will require not only technical innovations, but innovations in the regulatory processes for siting and construction. To facilitate the U.S. Department of Energy's (DOE's) NEPA review for reactor demonstration and deployment, NRIC has developed an approach adapted from the U.S. Nuclear Regulatory Commission's (NRC's) early site permit

(ESP) process under Title 10 of the *Code of Federal Regulations* Part 52 (10 CFR Part 52). The NRC developed the ESP process to give applicants the option to evaluate environmental impacts at development sites even before a specific reactor design is chosen, thereby allowing for early resolution of many environmental issues. After issuance of an ESP, a construction permit and operating license or a combined license (combined construction permit and operating license) would need to be completed before operation of any nuclear power plant could occur.

In 2003, to facilitate NRC's ESP process, the Nuclear Energy Institute (NEI) developed an approach to an ESP application based on a PPE and site parameter envelope (SPE) approach (NEI 2012). This report describes the adaptation of NEI's PPE/SPE approach to NRC's ESP process to possible deployment of advanced reactor prototypes at INL and potentially for reactor demonstrations and/or deployments elsewhere in the United States.

1.2 Advanced Reactor Types and Descriptions

Many different non-light-water reactor (LWR) technologies are in development. In addition to these general reactor types, there are several design-specific variations in materials, coolants, and geometries within each type and/or hybrids within/across these technologies. Sodium-cooled designs are more mature technologies that involve operational experience (e.g., the Fast Flux Test Facility at the Hanford Site in southeastern Washington).

A brief description of potential types of advanced reactors is provided below for context in developing a PPE. This list of advanced reactor types is intended to provide familiarity with and an overview for reviewers of the potential types of reactors, but it is not intended to be entirely comprehensive nor to exclude any specific reactor designs that are adequately represented by the parameters established in the PPE.

- High-temperature gas-cooled reactors (HTGRs) refer to graphite-moderated, typically helium-cooled systems that use tri-structural isotropic fuel micro particles. The particles are packed into a graphite matrix to form either spherical or cylindrical fuel elements. The pebble bed version of the HTGR uses spherical billiard ball-sized fuel elements that flow continuously through the reactor. The prismatic version of the HTGR uses the cylindrical fuel compacts in hexagonal blocks in a fixed geometry. HTGRs may be used for electricity production and/or process heat applications.
- Fluoride salt-cooled high-temperature reactors (FHRs) refer to a hybrid design that uses pebble fuel elements (like pebble bed HTGRs) and a fluoride salt coolant (like salt-cooled molten salt reactors). Some fixed-fuel FHR designs (like prismatic HTGRs) have been proposed, but none is currently under commercial consideration.
- Molten salt reactors (MSRs) come in several varieties. Some designs use molten fluoride salt, while others use chloride salts as the coolant. Some designs have stationary fuel rods or plates, while others have moving fuel pebbles or fissile material dissolved within the flowing coolant. In addition, some MSRs use a fast neutron spectrum, while others use a thermal spectrum.
- Liquid metal-cooled reactors (LMRs) are an advanced type of nuclear reactor in which the primary coolant is a liquid metal. LMRs are classified based on the liquid metal coolant used, such as sodium, lead-bismuth eutectic alloy, and lead-bismuth.
- Heat pipe reactors typically consist of a solid block core with the fuel in holes inside the solid block. Heat pipes are built into the block in a lattice configuration and remove the heat from the block as the liquid in the heat pipe is vaporized.

- Integral pressurized water reactors are an advancement upon historical pressurized water reactor designs that use coolant and fuels similar to existing LWRs, but that have the primary coolant circuit components placed within the reactor pressure vessel, thereby eliminating the need for primary circuit pipework with the intention of enhancing safety and reliability.

1.3 Report Content and Organization

This report describes the methodology for developing two PPEs for surrogate nuclear plants, one for microreactors, which are defined for this PPE as single units with outputs of 60 MWt or less, and one for small- to medium-sized advanced reactors[1] with outputs above 60 MWt up to 1,000 MWt.

Section 2.0 describes the background of the PPE and SPE, how the NRC has used the PPE and SPE approach, and how this approach was adapted for NRIC. Section 3.0 describes the various sources of information used to inform the development of the NRIC PPEs. Section 4.0 provides the summary PPEs, one for microreactors and one for small- to medium-sized advanced reactors. Section 5.0 is intended to give the reader guidance for the use of the PPEs, including limitations and environmental impact assessment considerations. Sources and documentation cited in the narrative are listed in Section 6.0. The appendices contain supplemental detail, including a vendor questionnaire (Appendix A), INL supporting information (Appendix B), advanced reactor parameter value assessments (Appendix C), NRC advanced nuclear reactor (ANR) generic environmental impact statement (GEIS) values (Appendix D), and the PPE Data Sources and Methodology (Appendix E).

[1] The IAEA defines “small” as anything below 300 MWe and “medium” as below 700 MWe. (IAEA 2001).

2.0 PPE/SPE Background and Adaptation

The concept of the PPE and how it has been used in previous ESP applications to evaluate environmental impacts is described below. Also described is the SPE, which has been used to bound environmental impacts prior to selection of a site, and how it has been adapted to NRIC's needs.

2.1 PPE Approach and NEPA Compliance

NRC's Environmental Standard Review Plans (ESRPs) provide guidance for developing Environmental Impact Statements for several different licensing actions (NRC 2000). While the ESRPs were originally prepared to guide staff in the review of construction permits and operating licenses only, in 2000 the NRC broadened the ESRPs to cover additional options including ESPs. The benefits of the ESP approach are that site safety, environmental protection, and emergency preparedness issues are resolve independent of a specific nuclear plant design. The development of the PPE associated with a surrogate reactor was originally developed and published in NEI 10-01 [Revision 1], Industry Guideline for Developing a Plant Parameter Envelope in Support of an Early Site Permit, which states that "[the PPE] approach provides an equivalent level of finality to that achieved through an ESP based on a specific reactor design (NEI 2012). In 2002, NRC staff conducted an internal ESRP workshop to consider the implications of ESP reviews employing the PPE approach rather than a specific nuclear plant design. NRC concluded that the PPE can serve as the foundation for the environmental report, and that the PPE values would be provided as a surrogate for the design information identified in the ESRP. NRC confirmed this in a letter to NEI in 2003, stating that "ESP applicants may use the PPE approach as a surrogate for actual facility information to support required safety and environmental reviews." (NRC 2003).

Use of the PPE approach to streamline NEPA compliance is consistent with the Council on Environmental Quality's (CEQ's) Final Guidance for Effective Use of Programmatic NEPA Reviews, which states that "Programmatic NEPA reviews assess the environmental impacts of proposed policies, plans, programs, or projects for which subsequent actions will be implemented either based on the PEA or PEIS, or based on subsequent NEPA reviews tiered to the programmatic review (e.g., a site- or project-specific document)" (CEQ 2014). Per CEQ's guidance, "in the absence of certainty regarding the environmental consequences of future proposed actions, agencies may be able to make broad program decisions and establish parameters for subsequent analyses based on a programmatic review that adequately examines the reasonably foreseeable consequences of a proposed program, policy, plan, or suite of projects." The CEQ Programmatic NEPA guidance provides an example of a programmatic EIS where the location, type, and timing of specific facilities was unknown. Therefore, the programmatic EIS appropriately "focused on a bounded range of potential activities and their impacts." The development of PPE parameters facilitates the review of impacts associated with development of future plants by allowing analysis of broad impacts of the surrogate plant; at the time of the project-specific review, these impacts can be compared to the impacts of the surrogate plant and supplemented as necessary.

2.2 Use of PPEs in Early Site Permit Applications

A PPE is a set of reactor and owner-engineered parameters that are expected to bound the characteristics of a reactor that might later be deployed at the ESP site. A PPE sets forth postulated values of parameters that provide sufficient details to support the NRC staff's review of an ESP application (NEI 2012). Using the PPE approach, the applicant for an ESP need not provide a detailed design—but should provide sufficient bounding parameters and characteristics—of a reactor or reactors and the associated facilities so that an assessment of site suitability can be made. Consequently, the ESP application may refer to a PPE as a "surrogate" for a nuclear power plant and its associated facilities.

The ESP application and review process uses the PPE to evaluate and resolve safety and environmental issues related to siting before a specific reactor design is chosen, allowing the applicant to "bank" the site and rely on this analysis for up to 20 years of future reactor siting. Analysis of environmental impacts based on the PPE for a surrogate plant permits an ESP applicant to defer the selection of a specific reactor design until the construction permit or combined construction permit and operating license stage.

The NRC has issued EISs for six early site permits (ESPs) to date, five of which (all but the Vogtle ESP) used the PPE approach to define a surrogate reactor (NRC 2020b).

Site	Applicant
Clinton ESP Site	Exelon Generation Company, LLC
Grand Gulf ESP Site	System Energy Resources Inc.
North Anna ESP Site	Dominion Nuclear North Anna, LLC
Vogtle ESP Site	Southern Nuclear Operating Company
PSEG Site	PSEG Power, LLC, and PSEG Nuclear, LLC (PSEG)
Clinch River Nuclear Site	Tennessee Valley Authority (TVA)

It has used the PPE concept most recently in its review of the ESP application for the Clinch River Nuclear Site in Tennessee (NRC 2019c). The Tennessee Valley Authority (TVA) proposed an ESP at the Clinch River site for two or more small modular reactors (SMRs) using a bounding PPE developed by TVA based on four different light-water SMRs—the BWX Technologies (BWXT) mPower™, Holtec SMR-160, NuScale, and Westinghouse SMRs with a total installed capacity of 800 MWe. The environmental impact statement (EIS) for the Clinch River ESP was completed in April 2019 (NRC 2019a).

The NRC is currently applying the ESP process and associated PPE approach in the development of its ANR GEIS (NRC 2020a). The NRC published a *Notice of Intent* initiating development of the ANR GEIS on April 30, 2020 (85 FR 24040).

2.3 Adaptation of the PPE/SPE Approach to NRIC at INL

2.3.1 PPE Approach

Both the NRC ESP process and the preparation for deployment of advanced reactor prototypes in existing buildings and sites at INL are intended to streamline NEPA reviews. Similar to the NRC ESP process, NRIC anticipates DOE will prepare a NEPA review of potential deployment of advanced reactor prototypes at INL prior to NRIC receiving commitments from any specific advanced reactor vendors. Although the buildings and sites at INL are known, the key features of advanced reactor prototypes (from an environmental review perspective) may vary considerably across reactor sizes and designs. As described below, NRIC has adapted the NRC's ESP process as one reasonable approach to an early NEPA review.

The NRC's use of a PPE in the ESP process has generally been for ranges of potential reactors that are of relatively similar size and designs. For instance, the four reactors used to develop the Clinch River ESP were all pressurized water reactors with multiple modules ranging from 60 MWe (NRC 2020c) to 225 MWe (NRC 2019b). Because the reactors were of similar design, choosing bounding parameters from the individual reactor parameters for the Clinch River project and other previous projects was relatively straightforward.

Choosing bounding parameters for potential reactor prototype deployments at INL, on the other hand, is more complicated because of the wide range of reactor designs. For example, the potential reactor prototype deployments at INL could range from less than 1 MWe to over 500 MWe. In addition, coolants could include a wide variety of types, including liquid metal, high-temperature gas, and molten salt. Some reactors may be constructed in a factory and delivered to the site in modules. Some may produce electricity, and some only heat. Different types of fuel are envisioned, from TRISO to fuel incorporated directly into a molten salt.

Because of the potential for a wide range of reactor designs, NRIC developed a modified approach to identifying the PPE for a surrogate plant. As was true for the NRC approach, the advanced reactor PPEs developed for NRIC are intended to identify the characteristics of the range of anticipated reactor designs that in turn provide a set of parameters associated with the construction and operation of a surrogate plant. Under NRIC's approach, surrogate plants are defined by PPEs that are generally based on reasonable values for a wide range of anticipated designs as opposed to easily identifiable bounding values for generally similar designs. In some cases, e.g., the plant footprint, these parameters could be the largest parameter values of the potential reactors that could be deployed. In other cases, there may not be a "bounding" value because of the wide range of potential designs, as would be the case for the nature of the fuel, coolant, or cooling technology. In those cases, the range of potential parameter values would be presented, and a reasonable value would be chosen. Furthermore, some prototype reactor demonstrations may not be designed to produce electricity; therefore, the size of these demonstrations in this NRIC PPE is described in terms of thermal output, or megawatts-thermal (MWt).

The NEI approach relies on obtaining plant design information by surveying vendors that could be chosen by an applicant for reactor deployment (NEI 2012). Because of uncertainties about demonstration projects and the early stage of the advanced reactor designs that could be sited at INL, this information may not be readily available for all reactor designs. As a result, NRIC has also adapted NEI's approach by supplementing vendor surveys with additional data sources, including INL site documents, subject matter expert assessments, PPE estimates in the NRC ANR GEIS, and any regulatory requirements a vendor would need to meet during construction and operation. In addition, the Versatile Test Reactor design provided an additional data source for advanced reactor plant parameters (DOE 2020c). These sources are described in Section 3.0. The combination of these sources informed the development of a robust PPE for a surrogate plant that can be analyzed in future NEPA processes.

NRIC has further adapted the PPE approach previously used by the NRC in ESP reviews based on reactor size. Two PPEs are developed in this report for two size ranges: (1) microreactors, which are defined for this PPE as single units with outputs of 60 MWt or less, and (2) small- to medium-sized advanced reactors with outputs above 60 MWt up to 1,000 MWt. These two PPEs are described in Section 4.0.

The PPE values using NRIC's adapted approach can be used as inputs to a NEPA review that would evaluate the environmental impacts of the surrogate plant or plants, assuming that these impacts would be reasonable estimates of the impacts of any particular design chosen. During the NEPA review for a specific design application, the impacts of the proposed reactor would be compared to the impacts of the surrogate reactor; assuming that the impacts are bounded by the analysis of the surrogate reactor, no additional analysis would be needed. For specific parameters exceeding the values identified in the PPE, additional reactor and/or site-specific documentation or analysis would be necessary. The benefit of two PPEs (microreactors and small- to medium-sized advanced reactors) is that the environmental impacts of two very different types and sizes of reactors can be assessed and disclosed as part of DOE's NEPA review.

Operation of advanced reactors will generate spent fuel. The ultimate disposition of spent fuel has not been identified by DOE and is not included in the PPE. Spent fuel management at INL pending disposition would include monitoring and storage in accordance with applicable DOE and other legal requirements. However, the PPE does not include parameters for construction, modification and operation of spent fuel storage facilities.

2.3.2 SPE Approach

Site parameters are usually specified by a reactor vendor independent of the proposed site, and they represent postulated physical, environmental, and demographic features of an assumed site that is used as a basis for the design analysis. In NEI's approach, site parameters are provided as part of an applicant's standard design certification and allow the NRC to evaluate the safety and environmental impacts of the specific reactor design on a postulated or "typical" site (NEI 2012). The SPE is used to identify the characteristics of suitable sites that would allow for deployment of the surrogate advanced reactor plant, minimizing and mitigating adverse environmental impacts to the extent possible. The SPE characteristics follow from the PPE characteristics; that is, the amount of resources necessary for siting the reactor is proportional to the resource impacts associated with the reactor design itself. The SPE is a valuable tool when a site has not yet been selected; once the site is known, the environmental needs and characteristics of that site can be specifically considered, and the PPE can be applied.

The NRC's ANR GEIS is also being developed with both a PPE approach and an SPE approach to streamline potential advanced reactor deployments without needing to identify specific sites until an application is received (NRC 2020a; 85 FR 24040).

Adapting the SPE approach to evaluating the impacts of prototype reactors at INL, however, has two challenges. First, because many of the anticipated reactor designs are in the early stages of development, limited information is available regarding the owner-engineered parameters. Second, the SPE is not as relevant in cases where the buildings and sites are known, and specific information is either available or being developed to characterize the environmental setting. At INL, the *Evaluation of Sites for Advanced Reactor Demonstrations at Idaho National Laboratory* published in March 2020 (INL 2020a) identified a list of candidate site locations and areas within INL boundaries that would be suitable for onsite demonstration of advanced reactors. INL identified, ranked, and recommended suitable sites based on this evaluation. Based on this analysis, INL indicated that advanced reactor demonstrations would be most suitable at the EBR-II, ZPPR, and Fuel Processing

Restoration (FPR) facilities, as well as at four undeveloped areas and two previously developed areas within the INL boundary.

To address these challenges, NRIC has included the site parameters that typically represent the postulated physical, environmental, and demographic features relevant to an environmental review and that may have been considered as site parameters in other reviews.

3.0 NRIC PPE Development

Two PPEs are developed in this report for two size ranges: (1) microreactors, which are defined for this PPE as single units with outputs of 60 MWt or less, and (2) small- to medium-sized advanced reactors with outputs above 60 MWt up to 1,000 MWt. The reactor sizes described in this PPE are expressed in terms of megawatts-thermal because not all demonstration reactors would produce electricity. Within the PPEs, certain parameters may have more than one value, depending on the anticipated water use associated with the design. These PPEs were developed using the following sources of information as described in the ensuing sections and appendices:

- Advanced Reactor Vendor Questionnaire (Appendix A)
- INL Supporting Information (Appendix B)
- Subject Matter Expertise (Appendix C
- NRC Advanced Reactor GEIS PPE values (Appendix D)
- parameters from the Versatile Test Reactor Draft EIS
- limits on construction and operation activities imposed by regulations.

PPE values without identified sources were developed using professional judgment and the bases for these decisions are provided. References and sources for the remaining values are also provided in this report and within the individual appendices. Appendix E (PPE Data Sources and Methodology) summarizes the information evaluated and the values chosen for the PPEs.

3.1 Advanced Reactor Vendor Questionnaire Responses

NRIC issued a questionnaire in June 2020 seeking vendor input on designs, features, and plant and site requirements of advanced reactor technologies to facilitate analysis of potential future deployment and implementation of these reactors. The questions were adapted from the NEI 10-01 PPE template (NEI 2012) and focused on the plant and site parameters that are most relevant to the potential analysis of environmental impacts.

The questionnaire was sent to vendors that may have interest in deploying advanced reactors at INL. The list of recipients was developed using sources such as Third Way (2020), the DOE (2019b), DOE’s Office of Advanced Reactor Technologies (2020a), DOE’s Gateway for Accelerated Innovation in Nuclear Program (GAIN 2020), NRIC’s industry engagement efforts, and others. As of October 2020, NRIC received a total of 11 responses: 6 responses were for designs in the microreactor size range, and 5 responses were for small- to medium-sized advanced reactor designs.

While multiple questions were included in the questionnaire, it was recognized that many vendors would not have answers to all of the questions, either because the design has not been finalized, the characteristics of a site location have not been determined, or the question is inapplicable to their design. In those cases, instead of providing a bounding plant parameter, staff used any input received from the vendors to inform and ground-truth the PPE values developed using other sources.

3.2 INL Input

As part of the capabilities provided by NRIC, INL may provide support services to vendors during construction, operation, and decommissioning. For example, INL provided information about the site infrastructure available for reactor deployment, e.g., types of transmission lines at proposed sites (INL 2020a). Because INL might prepare DOE-owned high-assay, low-enriched uranium (HALEU) as feed stock for reactor fuel vendors, INL provided a report, the *Isotopic Characterization of HALEU from EBR-II Driver Fuel Processing* (INL 2020b; TEV3537, Rev. 1). INL staff provided information about these support services to inform the PPE because the full scope of services was not available to reactor vendors at the time of this report (see Appendix B).

3.3 Subject Matter Expert Input

Resource subject matter experts (SMEs) at the Pacific Northwest National Laboratory (PNNL) involved in the development of this report previously conducted many of the resource assessments in the EISs for the six NRC ESPs, including all of the assessments for the recent Clinch River EIS (NRC 2019a). Knowledge of how a PPE is applied to specific sites in past reviews was used to recommend appropriate and reasonably bounding PPE values for key environmental resource issues that should be considered in DOE's NEPA review, particularly for the deployment of reactor technologies in the existing buildings and surrogate plants at the sites identified in *Evaluation of Sites for Advanced Reactor Demonstrations at Idaho National Laboratory* (INL 2020a).

PNNL SMEs provided independent assessments of key parameters for a surrogate plant in the following technical areas:

- land use
- water demand
- transportation
- workforce
- waste
- other air emissions
- fission product inventory.

The SMEs reviewed literature provided by INL, including the 2020 site evaluation report (INL 2020a), publicly available information regarding advanced reactor technologies and development, and previous NRC ESP NEPA reviews, including the Clinch River EIS (NRC 2019a), to develop and inform parameter values and assumptions.

The approaches for developing PPE values differed by technical area. The approaches and assumptions are listed in Appendix C.

3.4 NRC Advanced Reactor GEIS

Preliminary NRC PPE and SPE values presented during the ANR GEIS public scoping meeting are included in Appendix D. The PPE and SPE in the ANR GEIS were, in general, downscaled from impacts associated with previous environmental reviews for LWRs, rather than incorporating plant parameter values associated with specific advanced reactor designs. These values are subject to change during the development of the draft and final ANR GEIS.

While the NRC's preliminary PPE and SPE values should be considered when developing the NRIC PPE and SPE, the two processes differ in multiple ways, including that the intent of the NRC's PPE and SPE is to identify thresholds for "Category 1" issues (for which a generic analysis of environmental impacts is possible, provided that relevant assumptions in the PPE and SPE are met) and "Category 2" issues (for which a meaningful generic analysis of environmental impacts is not possible without consideration of site-specific information). Therefore, for Category 1 issues, the NRC has developed the PPE and SPE to identify the largest possible resource values to be able to conclude that impacts are small. The NRIC PPE and SPE, on the other hand, are simply intended to identify the characteristics associated with various advanced reactor technologies and resource needs without consideration of the relative magnitude of the effects of any given design. The impacts resulting from deployment of a surrogate reactor at INL would be assessed by DOE in a future NEPA review.

3.5 Versatile Test Reactor Draft EIS

Relevant values from the Versatile Test Reactor (VTR) Draft EIS published in December 2020 (DOE 2020c) are included in the Small- to Medium-Sized Advanced Reactor PPE summarized in Section 4.2 and described in Appendix E, Table E.2. The intent of the VTR is to provide a capability for large-scale testing, accelerated testing, and qualification of advanced nuclear fuels, materials, instrumentation, and sensors. The Draft EIS evaluates the environmental impacts of alternatives for constructing and operating the VTR, which would be an approximately 300 MWt sodium-cooled, pool-type, metal-fueled reactor constructed at INL. While not all of the values associated with the VTR are relevant to a PPE that focuses on advanced reactor demonstrations, these parameters were added and considered as data sources in certain cases to help define and refine the bounding values in the Small- to Medium-Sized Advanced Reactor PPE.

3.6 Regulatory Limits

Preliminary design information for advanced reactors may not be available for every plant parameter that may result in environmental impacts. In the absence of specific design information or other sources, regulatory limits, safety goals, or thresholds were evaluated as a set of parameters that would not be exceeded by the surrogate plant during construction or operation. Accordingly, environmental impacts resulting from releases have been evaluated and addressed, assuming regulatory limits as bounding values. For instance, the Clean Air Act (42 U.S.C. § 7401 *et seq.*) establishes limits for criteria pollutants, including, for example, those that could be released during operation of emergency diesel generators. Regulatory limits used in the PPE are specified in Appendix E.

4.0 Summary PPEs

Two separate PPEs, one for microreactors and one for small- to medium-sized advanced reactors, are included in the sections below.

4.1 Microreactor PPE

Table 4.1 lists plant parameter values for a surrogate microreactor plant, which is defined as a single unit with an output of less than 60 MWt, plus any associated support facilities. The parameters listed are those that have a nexus to the environment and that are important for determining the impacts of prototype construction and operation. Note that the values listed in the PPE are not meant to imply that an actual reactor with these parameters could be constructed and operated. See Appendix E for the data sources and methodology used to develop this table.

Table 4.1. Microreactor PPE

Plant Design Parameter	Definition	PPE Value
1.0 Structure and Layout		
Structure height	Vertical height from finished grade to the top of the tallest power-block structure.	28 ft
Stack height	Vertical height from finished grade to the top of the tallest exhaust stack.	50 ft
Foundation embedment	Depth from finished grade to the bottom of the basemat or the most deeply embedded power-block structure (excavation depth is the same elevation as embedment depth).	20 ft
Permanent disturbed acreage to support plant operations	Land area required to provide space for plant facilities, including any support facilities, switchyards, spent fuel management, and cooling towers.	8 ac
2.0 Construction		
Temporary disturbance during construction	Land area required to provide space for construction support facilities.	18 ac
Duration of construction	Duration of construction activities onsite.	24 mo
Construction workforce	Maximum number of people onsite during construction.	150
Construction noise	Maximum expected sound level due to construction activities, measured at 50 ft from the noise source	101 dB at 50 ft [(a)]
3.0 Operations		
Design type	Design of the plant, primarily the cooling system.	Five different design types: high-temperature gas, molten salt, liquid metal, heat pipe, and nuclear battery.

Table 4.1. Microreactor PPE (continued)

Plant Design Parameter	Definition	PPE Value
Megawatts-thermal	Thermal power generated by one plant module	60 MWt
Megawatts-electric	Best estimate of maximum megawatt electric generator output	20 MWe (based on estimated 33% thermal efficiency)
Plant operational life	Operational life for which the plant is designed or anticipated to be operated at INL.	30 yr
Planned modules	Number of modules that would be installed and operated.	One
Stationary or mobile	Planned design future use of the plant following demonstration	Stationary
Offsite power	Power from utility systems essential to support safety class structures, systems, and components (SSCs), such as electrical power supply and water supply	Required per General Design Criterion 17 (10 CFR Part 50, Appendix A, Criterion 17 - Electric Power Systems)
Normal plant heat sink	Technology (or technologies) for the normal plant heat sink	Mechanical draft cooling towers
Support facilities	Support facilities such as switchyards, spent fuel management, and cooling towers.	Multiple support facilities, including: cooling-water system; switchyard/ transformers; chemical/gas/fuel storage, potable water supply; wastewater system, including retention basins and associated discharge equipment; liquid radwaste system; fire protection and emergency response buildings; Administration/Maintenance Building(s); Security Facility; Chemistry and Meteorology Facility; Radioactive Waste Storage Facility (Region/Country Dependent); spent fuel management facilities, various offsite facilities
Operations staff	Number of total permanent staff to support operations of the plant.	50
Refueling/major maintenance staff	Additional number of temporary staff required to conduct refueling and major maintenance activities.	100

Table 4.1. Microreactor PPE (continued)

Plant Design Parameter	Definition	PPE Value
Consumption of raw water	Average short-term consumptive use of water by the cooling-water systems (evaporation and drift losses) and service water systems, including potable and sanitary water use (if required).	450 gpm, for air-cooled reactors, 25 gpm
Water discharge	Average characteristics of plant water discharges.	400 gpm, for air-cooled reactors, 25 gpm
Water source	Source of any potable water or cooling water.	Groundwater[(b)]
Water discharge constituents	Chemical and radionuclide constituents of the plant discharges, and maximum and expected concentrations/activities in the discharge.	See Appendix C.
Waste streams	Volume of radioactive and nonradioactive wastes generated during routine plant operations.	See Appendix C. It includes bounding values of radioactive solid waste generation, hazardous waste, nonhazardous waste, and gaseous waste.
Air emissions	Routine and periodic releases of criteria pollutants and greenhouse gases.	See Appendix C.
Stack exit velocity	Exit velocity of the stack for dispersion calculations.	10 ft/s
Auxiliary systems	Fuel source and size of auxiliary boilers, emergency power systems and standby power systems	Two diesel 50–150 kW standby power generators
Operation noise	Maximum expected sound level produced by operation of cooling towers, measured at 1,000 ft from the noise source.	65 dBA
4.0 Fuel		
Fuel form	Form of fuel associated with the plant design.	Fuel types could include UO_2, MOX, Metal (U, U alloys, Pu-containing alloys), TRISO, molten salt, uranium nitride, uranium carbide, QUADRISO, cermet, accident-tolerant fuel. Impacts associated with fuel form would differ depending on type; therefore fuel type is not a bounding PPE parameter.
Annual fuel requirements	Annual average fuel requirement (metric tons) per module.	0.5 MT (5 MT initial fuel loading)
Fuel source	Source location of the fuel or fuel feedstock.	Offsite commercial source

Table 4.1. Microreactor PPE (continued)

Plant Design Parameter	Definition	PPE Value
Radionuclide inventory	Radionuclide inventory for irradiated fuel at time of shipment (curies/metric ton of uranium [Ci/MTU] by radionuclide).	See Appendix C
Refueling	Refueling frequency (yr) and MTU per refueling.	5 MTU (full core refueling), online and continuous refueling
Fission product inventory	Annual activity, by radionuclide, contained in routine plant airborne effluent streams, excluding tritium.	See Appendix C
5.0 Transportation		
Shipments of unirradiated fuel	Total number of shipments and MTU for unirradiated fuel shipped to reactor or site.	10 shipments over the 30 yr life of the plant. 45 MTU total
Shipments of radioactive waste	Total number of shipments and volume of radioactive waste shipments from reactor/site.	49 shipments over the 30 yr life of the plant. Volume of each shipment is 2.34 m^3.
Transport method	Method of transporting reactor, fresh fuel, and other large components to the site.	Truck
Spent fuel disposition	Ultimate disposition of spent fuel.	89 irradiated fuel shipments over the 30 yr life of the plant. Offsite storage or disposal. Treatment, storage, and disposal in accordance with applicable legal requirements.
6.0 Decommissioning		
Prototype removal	Vendor plans to remove prototype from the INL site following demonstration.	Yes
Workforce	Estimated number of temporary staff required to conduct decommissioning activities.	150
Duration	Duration of decommissioning activities onsite.	18 mo
Waste generation	Amount of waste generated during decommissioning activities	Bounded by the waste streams evaluated in NUREG-0586

(a) Parameter value not included in vendor questionnaire. Construction noise value derived from Clinch River EIS (NRC 2019c).
(b) Site parameter value not included in vendor questionnaire. Groundwater source is assumed to bound obtaining water from INL plant services. Also assumed a new surface water intake and connection would not be required.

4.2 Small- to Medium-Sized Advanced Reactor PPE

Table 4.2 lists plant parameter values for a surrogate small- to medium-sized advanced reactor, which is defined as a single unit with an output of 1,000 MWt or less, plus any associated support facilities. The parameters listed are those that have a nexus to the environment and that are important for determining the impacts of prototype construction and operation. Note that the values listed in the PPE are not meant to imply that an actual reactor with these parameters could be constructed and operated. See Appendix E for the data sources and methodology used to develop this table.

Table 4.2. Small- to Medium-Sized Advanced Reactor PPE

Plant Design Parameter	Definition	PPE Value
1.0 Structure and Layout		
Structure height	Vertical height from finished grade to the top of the tallest power-block structure.	75 ft
Stack height	Vertical height from finished grade to the top of the tallest exhaust stack	87 ft
Foundation embedment	Depth from finished grade to the bottom of the basemat or the most deeply embedded power-block structure (excavation depth is the same elevation as embedment depth).	155 ft
Permanent disturbed acreage to support plant operations	Land area required to provide space for plant facilities, including any support facilities, switchyards, fuel management and cooling towers.	50 ac
2.0 Construction		
Temporary disturbance during construction	Land area required to provide space for construction support facilities.	100 ac
Duration of construction	Duration of construction activities onsite.	54 mo

Table 4.2. Small- to Medium-Sized Advanced Reactor PPE (continued)

Plant Design Parameter	Definition	PPE Value
Construction workforce	Maximum number of people onsite during construction.	1,400
Construction noise	Maximum expected sound level due to construction activities, measured at 50 ft from the noise source	101 dB at 50 ft[(a)]
3.0 Operations		
Design type	Design of the plant, primarily the cooling system.	High-temperature gas, molten salt, boiling water, liquid metal.
Megawatts-thermal	Thermal power generated by one plant module	1,000 MWt
Megawatts-electric	Best estimate of maximum megawatt electric generator output	333 MWe (based on estimated 33% thermal efficiency)
Plant operational life	Operational life for which the plant is designed or anticipated to be operated at INL	80 yr
Planned modules	Number of modules that would be installed and operated	One
Stationary or mobile	Planned design future use of the plant following demonstration	Stationary
Offsite power	Power from utility systems essential to support safety class structures, systems, and components (SSCs), such as electrical power supply and water supply	Two 230 kV transmission lines required. Offsite ROW 1,000 ft x 100 ft (new) or within or adjacent to existing ROW
Normal plant heat sink	Technology (or technologies) for the normal plant heat sink	Mechanical Draft Cooling Towers
Support facilities	Support facilities such as switchyards, fuel management, and cooling towers	Multiple support facilities, including: cooling-water system; switchyard/ transformers; chemical/gas/fuel storage, potable water supply; wastewater system, including retention basins and associated discharge equipment; liquid radwaste system; fire protection and emergency response buildings; Administration/Maintenance Building(s); Security Facility; Chemistry and Meteorology Facility; Radioactive Waste Storage Facility (Region/Country Dependent); various offsite facilities
Operations staff	Number of total permanent staff to support operations of the plant.	207

Table 4.2. Small- to Medium-Sized Advanced Reactor PPE (continued)

Plant Design Parameter	Definition	PPE Value
Refueling/major maintenance staff	Additional number of temporary staff required to conduct refueling and major maintenance activities.	206 (413 total) [(b)]
Consumption of raw water	Average short-term consumptive use of water by the cooling-water systems (evaporation and drift losses) and service water systems, including potable and sanitary water use (if required).	5,850 gpm (water-cooled) 415 gpm (air-cooled)
Water discharge	Average characteristics of plant water discharges.	1,775 gpm (water-cooled) 415 gpm (air-cooled)
Water source	Source of any potable water or cooling water.	Groundwater [(c)]
Water discharge constituents	Chemical and radionuclide constituents of the plant discharges, and maximum and expected concentrations/activities in the discharge.	See Appendix C.
Waste streams	Volume of radioactive and nonradioactive wastes generated during routine plant operations.	See Appendix C. Appendix C.5 includes bounding values of radioactive solid waste generation, hazardous waste, nonhazardous waste, and gaseous waste.
Air emissions	Routine and periodic releases of criteria pollutants and greenhouse gases	See Appendix C.
Stack exit velocity	Exit velocity of the stack for dispersion calculations	58 ft/s
Auxiliary systems	Fuel source and size of auxiliary boilers, emergency power systems, and standby power systems	50 MWt oil fired; 15 MWe Sentry turbine
Operation noise	Maximum expected sound level produced by operation of cooling towers, measured at 1,000 ft from the noise source	65 dBA at site boundary
4.0 Fuel		
Fuel form	Form of fuel associated with the plant design	Molten salt, TRISO, uranium oxide, HALEU, U-Zr alloy. Emission release mechanisms from molten salt are different from LWRs; expect that molten salt will have upper bounding impacts compared to other fuel technologies.
Annual fuel requirements	Annual average fuel requirement (metric tons) per module	8 MT
Fuel source	Source location of the fuel or fuel feedstock	Offsite commercial source

Table 4.2. Small- to Medium-Sized Advanced Reactor PPE (continued)

Plant Design Parameter	Definition	PPE Value
Radionuclide inventory	Radionuclide inventory for irradiated fuel at time of shipment (curies/metric ton of uranium [Ci/MTU] by radionuclide)	See Appendix C
Refueling	Refueling frequency and MTU per refueling	Daily refueling of 10.6 kg enriched U and 18 kg Th; annual requirement 3.9 MT enriched U, 6.6 MT Th.
Fission product inventory	Annual activity, by radionuclide, contained in routine plant airborne effluent streams, excluding tritium	See Appendix C
5.0 Transportation		
Shipments of unirradiated fuel	Total number of shipments and MTU for unirradiated fuel shipped to reactor or site	432 shipments over the 80 yr life of the plant. 1,972 MTU total
Shipments of radioactive waste	Total number of shipments and volume of radioactive waste shipments from reactor/site	2,160 shipments over the 80 yr life of the plant. Volume of each shipment is 2.34 m^3.
Transport method	Method of transporting reactor, fresh fuel, and other large components to the site	Truck
Spent fuel disposition	Ultimate disposition of spent fuel.	3,944 irradiated fuel shipments over 80 yr life of the plant. Onsite storage, or offsite storage or disposal. Treatment, storage, and disposal in accordance with applicable legal requirements.
6.0 Decommissioning		
Prototype removal	Vendor plans to remove prototype from the INL site following demonstration.	Yes
Workforce	Estimated number of temporary staff required to conduct decommissioning activities	450 total
Duration	Duration of decommissioning activities onsite	10 yr
Waste generation	Amount of waste generated during decommissioning activities	Bounded by the waste streams evaluated in NUREG-0586

(a) Parameter value not included in vendor questionnaire. Construction noise value derived from Clinch River EIS (NRC 2019c).
(b) Note that this parameter assumes that periodic refueling would occur, in order to better bound potential workforce impacts; however, the refueling parameter included below assumes continuous refueling.
(c) Site parameter value not included in vendor questionnaire. Groundwater source is assumed to bound obtaining water from INL plant services. Also assumed a new surface water intake and connection would not be required.

5.0 Guidance for Use of the PPE

The use of the PPE to facilitate environmental impact assessments in future NEPA processes, and its limitations, are described below.

5.1 Use of the PPE in the NEPA Process

The intent of the PPE is to describe a surrogate plant that would bound the upper limits of most potential demonstration reactor designs, or that would provide a representative resource value. As such, this surrogate plant can serve as the input for a NEPA review, allowing for analysis of the largest range of impacts anticipated by any given reactor prototype. The PPE does not define whether these impacts are small, moderate, or large, nor does it define a threshold of NEPA significance. The PPE provides DOE with a way to assess and disclose the reasonably foreseeable impacts from the deployment of micro and small- to medium-sized advanced reactor prototypes at INL.

Typically, NEPA reviews consist of evaluating a proposed action's impacts on the environment at a specific location. A future NEPA review for NRIC could assume the surrogate plant was sited in an existing building or at one of the specified sites at INL. The parameters developed are those that have the potential to impact the usual resource areas evaluated in a NEPA review (land use, hydrology, ecology, etc.). For instance, the plant footprint would be used to identify the maximum land that would be committed to the surrogate plant deployment and to assess the impact on changes in land use and the potential to disturb native habitat. By assessing the impact of the surrogate plant parameters on the site, the NEPA review informs the decision-maker and the public of the impacts that would be reasonably foreseeable.

During the NEPA review for deployment of a specific reactor design conducted after the initial NEPA review, the impacts of the proposed reactor would be compared to the impacts of the surrogate reactor. Assuming that the impacts are bounded by the analyses of the surrogate reactor, no additional analysis would be needed. In this situation, a Supplement Analysis may be the appropriate NEPA compliance mechanism for the project. For specific parameters exceeding the values identified in the PPE, additional design-specific or site-specific NEPA analysis would be necessary. This would likely be completed through the development of a supplemental environmental assessment or EIS, depending on the significance of the values exceeding the PPE.

To better inform stakeholders and the public about the potential impacts of the range of reactor designs and sizes expected to be deployed, it is recommended that the surrogate plants defined by the two PPEs presented in this report be assessed as separate deployments in the proposed action in any future NEPA review.

5.2 Limitations

While the PPE contains useful and relevant information regarding advanced reactor technologies and deployment, the application of the PPE to a NEPA review has the following potential limitations:

1. Many parameters are not known by the vendors at their preliminary stages of design. As evidenced by the vendor responses to the PPE questionnaire, some vendors did not have information about the source of fuel, transportation of fresh fuel to and spent fuel from the site, decommissioning, and source term values. Because of the potential range of reactor sizes and

types and the early stage of plant designs for many vendors, it may not be possible to determine either a reasonable or bounding value for key plant parameters. For example, fission product inventories can be estimated, but routine releases and accident releases are design- and/or site-specific and must be considered as part of site-specific impact analysis. It was anticipated that many of these parameters would be unknown and therefore the information received from the vendors was supplemented with the other sources of information, as described in Section 3.0. This information was used to create estimates of PPE values based on best professional judgment.

2. Some parameters are dependent on the type of reactor design, and therefore developing a single parameter value for one type of design would not bound values for different designs. For purposes of developing a value for the PPE, the waste streams associated with MSRs differ from LWRs and could result in additional operational waste streams to be analyzed. Using the anticipated releases from a MSRs as the PPE value when evaluating another reactor design (i.e., high-temperature gas, liquid metal, nuclear battery, and heat pipe) would not provide useful information for a NEPA review. However, MSRs are included in the Summary PPE in Section 4.1 because of the unique nature of the fuel and potential releases. This provides a basis for development of some of the other PPE values, including refueling, waste, and average/annual fuel requirements.
3. Bounding of solid, liquid, and gaseous waste would be dependent on compliance with applicable Federal regulations (such as 40 CFR Part 61 Subpart H; DOE O 458.1 [DOE 2020d], including DOE-STD-1196 [DOE 2011] and DOE-STD-1153 [DOE 2019a]; 10 CFR Part 20 Appendix B; Resource Conservation and Recovery Act of 1976 (RCRA); Clean Air Act; Clean Water Act), State, local, and tribal regulations and permitting criteria. The best source of information for solid waste is developed from data about the transportation of shipments of radioactive waste as scaled from LWRs (Appendix C). However, the values for solid radioactive waste do not account for differences in the of advanced reactors and LWRs or any unique solid waste streams not found in a LWR. The best source of information for liquid radioactive effluent is developed from data about the constituents associated with the water discharges (Appendix C). Individual applications would have to provide additional information to address the generation and disposition of waste from the proposed action.
4. Some regulatory issues associated with advanced reactor deployment remain unresolved, including requirements associated with remote operation, emergency planning zones, and other issues differing from those associated with large LWRs. Resolution of these issues may affect the potential environmental impacts. For instance, the size of the emergency planning zone may affect the location of the nearest receptor.
5. The parameter values developed in the PPE are for deployment of a single reactor module. As part of vendor demonstrations, multiple modules may be deployed at the same time, or incrementally over time. The schedule for multiple module deployment will be needed to estimate the timing of impacts, and these schedules are not currently known. However, it is not necessarily the case that multiple deployments in a similar geographic area at the same or different times would result in additive impacts.
6. Any NEPA review based only on the PPE cannot fully cover deployment of advanced reactors at the INL site; supplemental NEPA analysis would be required to document that the impacts of any given proposed vendor demonstration would be within or similar to those discussed in the initial NEPA review. Some additional NEPA review could be necessary for parameters or actions that are not included in the PPEs or for deployment activities that are different than assumed. For example, if a demonstration project had design or operation features that did not fall within the

scope of the PPE considered in the initial NEPA review, then those features would be considered further and potentially evaluated as part of a supplemental NEPA analysis.

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Appendix A – Vendor Questionnaire

National Reactor Innovation Center Information Request - Environmental Impact Parameters

National Reactor Innovation Center | NRIC-20-GDE-0002 | June 15, 2020

NRIC needs your input to effectively enable your demonstration

In 2020, the National Reactor Innovation Center (NRIC) initiated projects to provide capable facilities to support reactor demonstrations. NRIC is seeking your input to gain a better understanding of the designs, features, and requirements of advanced reactor technologies in order to facilitate analysis of potential future deployment and implementation of these reactors at the INL site and elsewhere.

NRIC will utilize your input to develop generic, enveloped demonstration reactor parameters that will inform a potential National Environmental Policy Act (NEPA) review. To develop these parameters, NRIC will follow a process based on NEI 10-01 "Industry Guideline for Developing a Plant Parameter Envelope in Support of an Early Site Permit". This Plant Parameter Envelope will be used to identify environmental impacts of installation and operation of advanced reactor designs at the INL site and elsewhere, streamlining any future site-specific NEPA reviews and analysis for designs meeting these criteria. The bounding parameters developed in this process will serve to define a "surrogate" reactor that DOE would assess in a NEPA review.

Please provide your reactor design information in the attached form, using the entry boxes or drop-down choices as appropriate. If your organization is developing more than one design, please feel free to submit multiple forms.

NRIC recognizes that many of these questions may not be answerable at this stage, because:

- The design has not been finalized;
- A site location has not been determined; or
- The question is inapplicable to the design

If the information requested is not available or unknown at this time, please leave the response blank. However, any and all information (including estimates) that you can provide will be helpful.

Please note: **Your responses in this form should not contain any proprietary information.** If your company would like to submit information outside of this survey process that may be proprietary or business sensitive, please contact NRIC to discuss. Vendor responses and information provided in this survey will help NRIC to better understand advanced reactor technologies. The results of this survey will be anonymized and aggregated prior to any potential public dissemination or publication. NRIC may share this aggregated information with the Nuclear Regulatory Commission to inform their NEPA processes but individual questionnaire results will not be shared.

Deadline: Please submit your response(s) via email no later than July 10, 2020 to NRICppe@pnnl.gov.

NRIC

Pacific Northwest
NATIONAL LABORATORY

Plant Parameter Envelope Template – Vendor Questionnaire

Instructions: *Please enter the parameters for your reactor design in the designated fields. For your convenience, the following template has entry fields with drop-down choices for your design parameters where applicable. In some cases, text description or separate data tables are requested if available. If information is not known at this time, please leave the entry blank.*

Vendor Information

1. Company Name	Enter Company Name
2. Company Address	Enter Company Address
3. Point of Contact	Enter Name Enter Phone Number Enter Email
4. Does your company have interest in potentially siting a demonstration reactor at the INL site?	Choose an item.

Plant Design

5. What is your design type?	Choose an item or type your own.
6. How many units do you plan to install?	# units
7. What is the output of your design (per unit)?	# MWt # MWe
8. Is your reactor designed to be mobile?	Choose an item.
9. If the reactor is designed to be transportable, what are the total number of shipments and weight of reactor, fuel, and its packaging?	# Shipments # Combined Shipment Weight (MT) # Reactor Weight (MT) # Fuel Weight (MT) # Packaging Weight (MT)
10. Describe your power conversion system.	Click or tap here to enter text
11. Will offsite power sources be required to maintain functioning of structures, systems, and components important to safety following loss of onsite AC power? If so, what transmission voltage would be required from offsite power sources?	Choose an item. # kV
12. What support facilities (fuel storage and handling, waste treatment, etc.) are necessary for your plant design?	Click or tap here to enter text

Plant Structure and Footprint

13. What is the tallest structure and what is the maximum structure height (structure, ft)? What is the stack height?	Choose an item. # Structure (ft) # Stack (ft)
14. What is the maximum depth of excavation?	# ft # ft
15. What is the temporary disturbed acreage during construction, including parking and laydown?	# ac
16. What is the permanent disturbed acreage, including parking lots, ponds, substations and other plant support facilities?	# ac
17. Are there large quantities of any unique materials (perhaps items not normally utilized in general office or industrial buildings) that	Insert Material or choose not required # (material/MT)

will be utilized in plant construction (e.g., graphite)? If so, what are these anticipated volumes?	

Operational Parameters

18. What is the operational life for which the plant is designed? How long do you intend to operate the reactor prototype?	# years # years
19. Do you anticipate installing additional modules incrementally over time?	# (number of modules/periodicity)
20. What is the reactor heat transfer material (coolant)? How much is required initially/annually?	Choose an item. # Initially # Annually
21. What is the anticipated technology (or technologies) for the normal plant heat sink?	Choose an item.
22. What are the maximum and average daily water use requirements for plant cooling and service water systems, including potable and sanitary water use (if required)?	Choose an item. # Max water withdrawal (gpm) # Max consumptive use (gpm) # Average daily withdrawal (gpm) # Average daily use (gpm)
23. What are the expected characteristics of plant water discharges (if any)?	Choose an item. # Average flow rate (gpm) # Maximum flow rate (gpm) # Maximum seasonal temperatures (season/^{O}F)
24. What are the chemical and radionuclide constituents of the plant discharges, and maximum and expected concentrations/activities in the discharge (if available).	Choose an item.
25. What is the fuel source and size of auxiliary boilers, emergency power systems and standby power systems (if applicable) (fuel source, MW)?	Choose an item or type your own (Source/MW) Aux Boilers (fuel source, MW) (Source/MW) Emergency Power (fuel source, MW) (Source/MW) Standby Power (fuel source, MW)
26. How much hazardous, radioactive and mixed waste would be generated during operations, and where would it be dispositioned?	Choose an item. Hazardous MT/Location Choose an item. Radioactive MT/Location Choose an item. Mixed MT/Location ☐ Not available at this time
27. What is the stack exit velocity?	# Exit Velocity (ft/sec)
28. What amount of noise would be generated 50 ft from the source and at the site boundary?	# dB @ 50 ft # dB @ site boundary

Fuel

29. What is the form of the fuel associated with your design?	Click or tap here to enter text.
30. What is the annual average fuel requirement (metric tons) per module?	# MT
31. Where would fuel be obtained?	Click or tap here to enter text.
32. What is the total number of shipments and MTU for unirradiated fuel shipped to reactor or site?	# Total Shipments # MTU

33. Total number of shipments and volume of radioactive waste shipments from reactor/site?	# Total Shipments # M^3
34. What is the radionuclide inventory for irradiated fuel at time of shipment (Ci/MTU by radionuclide)?	Choose an item.
35. How will the reactor, fresh fuel and other large components be transported to the site?	Choose an item. Reactor Choose an item. Fresh Fuel Choose an item. Large Components
36. Is the reactor designed to be refueled? If so, at what frequency (yrs)? What MTU per refueling?	Choose an item or type your own # years # MTU/refueling
37. What are the source terms for routine releases (if any) per module and design-basis accidents?	Choose an item.
38. Are there any unique fuel storage or cooling requirements associated with the fuel?	Click or tap here to enter text.
39. How and where would spent fuel be dispositioned?	Click or tap here to enter text.

Workforce

40. How many workers will be onsite for construction?	# workers
41. What is the anticipated construction period?	# months
42. What is the number of total permanent staff to support operations?	# workers
43. What is the number of temporary staff during refueling (if planned)?	# workers
44. What is the number of temporary staff during additional module installation (if planned)?	# workers
45. What are the distances from radiation sources to the nearest involved worker?	# source/feet # source/feet

Decommissioning

46. Do you plan to decommission and remove the prototype from the INL site?	Choose an item.
47. What is the number of temporary staff during decommissioning (if planned)?	# workers
48. What is the number of months from start of decommissioning to completion (if planned)?	# months
49. How much waste would be generated during decommissioning (if planned)?	# radioactive (MT) # hazardous (MT) # mixed (MT)

Other

50. Do you have any other information about your design or comments that would be useful in developing a Plant Parameter Envelope?	Click or tap here to enter text.

Appendix B – INL Supporting Information

Idaho National Laboratory (INL) staff provided the following documents in support of the plant parameter envelope development. This information was used to inform PPE values and SPE characteristics at the INL site.

- INL (Idaho National Laboratory). 2011. Idaho National Laboratory Comprehensive Land Use and Environmental Stewardship Report. INL/EXT-05-00726, Revision 1.
- INL (Idaho National Laboratory). 2013. Site Suitability and Hazard Assessment Guide for Small Modular Reactors. INL/EXT-13-29749.
- INL (Idaho National Laboratory). 2017. Special Purpose Nuclear Reactor (5 MW) for Reliable Power at Remote Sites Assessment Report. INL/EXT-16-40741, Revision 1.
- INL (Idaho National Laboratory). 2019. Key Regulatory Issues in Nuclear Microreactor Transport and Siting. INL/EXT-19-55257.
- INL (Idaho National Laboratory). 2020a. Evaluation of Sites for Advanced Reactor Demonstrations at Idaho National Laboratory. INL/EXT-20-57821.

Appendix C – SME Reactor Plant Parameter Value Assessments

Appendix C describes the methodologies used by subject matter experts (SMEs) to estimate plant parameter envelope (PPE) values. These methodologies and assessments help define the PPE values associated with the surrogate plant.

Bounding plant parameters relevant to reactor design were determined via a review of publicly available documentation, including vendor websites and other literature, as well as vendor responses to the questionnaire presented in Appendix A. References reviewed beyond vendor websites include the following:

- Key Regulatory Issues in Nuclear Microreactor Transport and Siting (INL 2019)
- Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors: Westinghouse eVinci™ Micro-Reactor Licensing Modernization Project Demonstration (Maioli et al. 2019)
- Cost Competitiveness of Micro-Reactors for Remote Markets (NEI 2019a)
- Micro-Reactor Regulatory Issues (NEI 2019b)
- Roadmap for the Deployment of Micro-Reactors for U.S. Department of Defense Domestic Installations (NEI 2018)
- Oklo Power Combined Operating License Application for the Aurora at INL (Oklo 2020c)
- “DOE-NE Micro-Reactor RD&D Program Mission and Objectives” (Sowinski 2019)
- “Advanced Reactor Types Factsheet” (DOE 2020b)
- “Project Pele Overview: Mobile Nuclear Power for Future DoD Need” (Waksman 2020).
- “Status Report – Steam Cycle High-Temperature Gas-Cooled Reactor (SC-HTGR) (Framatome 2019)
- “Lead-Cooled Fast Reactor (LFR) Design: Safety, Neutronics, Thermal Hydraulics, Structural Mechanics, Fuel, Core, and Plant Design” (Cinotti et al. 2010)
- “Summary Description of the Fast Flux Test Facility” (Cabell 1980).

Because many of the proposed reactor designs are still early in the development phase, many design details have not yet been determined. Based on the compiled information and comparison to existing plant designs for larger reactors (such as small modular reactors [SMRs], experimental reactors [Fast Flux Test Facility, Molten Salt Reactor Experiment, etc.], and LWRs), professional judgment was employed to determine bounding values. Where necessary, conservative assumptions were made.

C.1 Land Use

Land use parameter values took into account input from the public scoping associated with the U.S. Nuclear Regulatory Commission (NRC) Advanced Nuclear Reactor Generic Environmental Impact Statement (ANR GEIS), internal research and analysis of advanced reactor plant designs, and input received from the advanced reactor vendors in response to the questionnaire.

For microreactors, among the vendor designs, the largest bounding value would be 8 ac of temporary disturbance and 7 ac of permanent disturbance. While the NRC's ANR GEIS is using a PPE value of 50 ac of temporary disturbance and 30 ac of permanent disturbance with a 50 ft maximum depth of excavation, the survey results and internal review found that most projects will use significantly less acreage with less excavation. Hence, the assumed land use requirements are slightly larger than the vendor information but smaller than the NRC's PPE values. Permanent disturbed acreage is assumed to be 8 ac or less, and temporary disturbed acreage is assumed to be 10 ac or less. Some reactors may require less acreage; for example, the total land use requirements associated with the Oklo Aurora reactor (Oklo 2020a) would be approximately 1 ac and would involve 0.3 ac of disturbed area.

For small- to medium-sized advanced reactors, the NRC ANR GEIS PPE values of 50 ac of temporary disturbance and 30 ac of permanent disturbance was instructive in developing the PPE values. However, the largest land use requirements provided in a vendor response were for 58 ac of temporary disturbance and 43 ac of permanent disturbance.

C.2 Water Demand

Water demand for a power plant includes cooling-water use, non-cooling process water use, and potable/sanitary water use. For a wet-cooling system (e.g., a wet mechanical draft cooling tower), total water demand will include the consumptive use for evaporative cooling and the nonconsumptive use for blowdown. For a dry-cooling system, water use for cooling will be minimal, but water will still be required for non-cooling process uses and for potable/sanitary uses.

For a power plant using a wet-cooling system, an estimate of the upper limit of consumptive water use for cooling can be made from the plant's thermal power output and electrical power generated in power conversion, and the assumption that all waste heat is dissipated to the environment through the vaporization of water. The amount of waste heat is the difference between the plant's thermal output and the work done (electricity generated). For example, a reactor that has an output of 60 MWt and a thermal efficiency of 37 percent would have an electrical power output of about 22 MWe. The cooling system in this case would need to dissipate 38 MW (60 MW – 22 MW).

The amount of heat required to evaporate a 1 kg mass of water is 2,256 kJ (i.e., the latent heat of vaporization of water is 2,256 kJ/kg). The mass rate of water evaporated by 1 MW (1,000 kJ/s) is therefore (1,000 kJ/s)/(2,256 kJ/kg) which equals 0.4433 kg/s. With continuous operation, this is about 38,300 L/d, which equates to 10,100 gal/d or 0.0157 ft^3/s of water evaporated. For a plant that has an output of 60 MWt and 22 MWe, dissipating 38 MW using evaporative cooling would require the consumption of (38 × 38,300 L/d), or about 1.5 ML/d, which is approximately 0.38 Mgd or 270 gpm.

Cooling requirements of a particular plant would depend on the efficiency of the power conversion system, but the above analysis would provide a maximum estimate of consumptive cooling-water use for any plant using a wet-cooling system. The actual amount of cooling water consumed would depend on the cooling technology used and whether the heat from the plant was used in other industrial processes instead of being dissipated to the environment.

Blowdown for wet-cooling was estimated to be one-third of the consumptive use (evaporation), which is consistent with the assumption that the cooling system is operated at four cycles of concentration. Non-cooling process water use was estimated as 7 percent of the total water demand, excluding the potable/sanitary water use. This was based on the use of non-circulating water system water for the Clinch River plant. Potable and sanitary water use was estimated to be 100 gpd for each member of the operations workforce.

Appendix E presents bounding water use values using this approach for microreactors that have an output of 60 MWt or less and small- to medium-sized advanced reactors that have outputs of 1,000 MWt or less.

C.3 Transportation

The number of shipments of unirradiated (fresh) fuel, irradiated (spent) fuel, and radioactive waste were estimated based on the shipment data presented in the Clinch River early site permit (ESP) (NUREG 2226, NRC 2019a) for a surrogate SMR (see Tables 6-4, 6-10, and 6-14 in the Clinch ESP). In the Clinch ESP, the surrogate SMR was developed by applying bounding parameters from four SMR designs (BWX Technologies mPower SMR, the Holtec SMR-160, the NuScale SMR, and the Westinghouse SMR). The bounding value for solid radioactive waste generation was 5,000 ft^3/yr and a bounding activity of 57,200 Ci/yr (NUREG 2226; NRC 2019a). Overall, the generating output of the surrogate SMR was 800 MWe and the capacity factor was 90 percent. The shipment data for the four SMRs was normalized to a generating output of 1,100 MWe and a capacity factor of 80 percent for analysis in the Clinch River ESP.

The PPE for transportation was developed by further normalizing to the thermal output of the reactor, thermal to electrical energy conversion efficiency, capacity factor, and duration of operations for one microreactor and one small- to medium-sized advanced reactor:

- 60 MWt microreactor, 0.33 thermal to electrical energy conversion efficiency, 0.95 capacity factor, 30-year duration of operations
- 1,000 MWt advanced non-LWR, 0.33 thermal to electrical energy conversion efficiency, 0.95 capacity factor, 80-year duration of operations.

Appendix E presents bounding values developed using this approach for the numbers of shipments associated with microreactors that have an output of 60 MWt or less and small- to medium-sized advanced reactors that have outputs of 1,000 MWt or less.

C.4 Workforce

As indicated in Section 3.3, there is no recent nuclear power industry experience upon which to base PPE estimates for workforce parameters, including construction, operations, outages for refueling or maintenance activities, or any additional module installation into array-type facilities. Thus, to provide a basis for initial estimates of these workforce parameters, SMEs used recent electric power generation industry experience and adapted information relevant to nuclear-fueled facilities that might alter that experience. For example, construction cost and workforce estimates are generally based on electric power ratings (MWe) rather than the thermal ratings (MWt) of power plants. Using values scaled from the power capacity of a plant requires additional assumptions about the thermal efficiency of the plants being used as data sources. The values developed in this section are assumed to apply to microreactors in the range of 60 MWt and for small- to medium-sized advanced reactors bounded by the 1,000 MWt PPE value. For the purposes of developing the PPE values and to provide a conservative bounding estimate, thermal efficiency is estimated to be 33 percent.

Natural gas combustion turbine (CT) plant construction was judged to be most analogous to the expected early plant construction effort to install microreactors in the 60 MWt range. Estimates from PJM Interconnection, LLC (PJM 2018) for the costs of new CT plants were used to derive proxy values for the expected workforce parameters associated with a 20 MWe SMR. Smaller CT plants (under 100 MWe units) most closely approximate the expected construction effort needed to install a 20 MWe SMR and would likely provide bounding values.

Construction staffing PPE values are based on the PJM (2018) report and are derived using the cost per kilowatt reported for CT construction, assuming a greenfield site. The bounding estimate for the PJM service area for CT construction labor cost is $104/kW (PJM 2018). Scaling this value for 20 MWe yields an aggregate construction labor cost of $2.08 million. Average staffing during the CT construction period is approximately 65 workers. However, the effort is weighted toward the final 11 months (PJM 2018). The modular nature of the SMR technology and the relatively small expected facility footprint would greatly reduce the construction period required, compared to typical CT construction. By assuming a construction period of a maximum of 6 months and scaling PJM costs by the 20 MWe SMR costs above, the average construction workforce would be approximately 12 workers, but the effort would be expected to be weighted to the final 3 months of construction, with general site preparation work occurring over the initial 3 months. Thus, a reasonable approximation of the peak construction workforce would be 20 workers.

Operations staffing would be greatly minimized through deployment of automation and sensor technologies. Based on gas plant recent experience compiled by Black and Veatch (Wagman 2017), a 565 MWe natural gas combined-cycle (NGCC) plant requires 27 full-time equivalent employees (FTEs) to operate on a typical 24-hour schedule. Increasing the plant size to 865 MWe requires only an additional six FTEs. Given the much larger capacity of the modern NGCC plant coupled with the use of nuclear-fueled technology in an SMR, it is assumed that 27 operations staff would be a bounding estimate for the 60 MWt first-of-a-kind implementation at Idaho National Laboratory (INL) of base-level operations. Presumably, the use of automation and sensor technologies would be maximized, but other safety- and security-related staffing would likely be required in addition to what might be expected at an NGCC plant.

To estimate the likely number of outage workers for the 60 MWt implementation at INL, SMEs reviewed recent experience at some plants in the existing fleet of large LWRs in geographically diverse areas of the United States. Specific cases include Byron 1 (Exelon 2020), Columbia Generating Station (Dobken and Markham 2017), Oconee 3 (Benson 2020), and Salem 1 (PSEG 2019). In these cases, the weighted average number of outage workers per megawatt-electrical unit of net capacity most recently equated to 1.04 workers/MWe. Scaling this experience to the 60 MWt size associated with the microreactor PPE would indicate that outage staffing would be approximately 21 workers. Given that the ANR technologies likely to be installed would have reduced maintenance requirements compared to large reactors, 21 workers could be considered a bounding estimate for outage staffing.

If an array-style design were implemented and capable of subsequently adding several 60 MWt modules, it is assumed that a workforce similar to the peak construction workforce would be needed for each module installation. Thus, 20 workers would be required to install a new module.

For the bounding case for a small- to medium-sized advanced reactor size of up to 1,000 MWt, workforce and related values are based on scaling down the values analyzed for the Clinch River ESP (NRC 2019a). Clinch River proposed a bounding value of 800 MWe (2,420 MWt) for a configuration of SMRs and represents the closest approximation available for NRIC consideration.

Assuming 33 percent thermal efficiency equates to approximately the 2,420 MWt used in the Clinch River case. Thus, workforce values reported for Clinch River were scaled using the factor: 1,000/2,420 = 0.413. This results in the maximum construction workforce onsite at one time of 909 workers at peak construction. Total construction staffing at this time would be 1,363 workers. The operations workforce would be 207 workers and the outage workforce would be 413 workers. It is assumed that the module replacement would occur concurrently with a typical refueling outage and use the same workforce of 413 workers.

Appendix E presents bounding workforce values for microreactors that have an output of 60 MWt or less and small- to medium-sized advanced reactors that have outputs of 1,000 MWt or less.

C.5 Waste

C.5.1 Nonhazardous, Hazardous, and Radiological Waste

Generation of nonhazardous, hazardous, and radiological waste materials would be expected at any nuclear reactor, regardless of the design. The Resource Conservation and Recovery Act (RCRA; 40 CFR Part 239–282) defines nonhazardous waste in Parts 239–259 and hazardous waste in Parts 260-273. DOE O 435.1 (DOE 2001) discusses radiological waste management.

Nonhazardous solid waste includes typical industrial waste such as metal, wood, and paper. A bounding estimate for nonhazardous waste could be 290 tons per month (NRC 2019a).

Examples of hazardous waste include lab packs, metals from shielding applications, and rags or wipes containing solvents. Universal waste is one class of hazardous waste that includes batteries, pesticides, equipment containing mercury, and lamps (bulbs) (40 CFR Part 273). RCRA also defines generator types for hazardous waste. This includes large quantity generators, small quantity generators, and very small quantity generators. Small quantity generators generate more than 100 kg, but less than 1,000 kg of hazardous waste per month. The ESP application for the Clinch River SMR expected the facility to qualify as a small quantity generator (TVA 2019). The ESP application for the Public Service Enterprise Group (PSEG) stated that PSEG maintains the program required of a small quantity generator (PSEG 2014). For nonradiological hazardous waste, it is assumed that hazardous waste amounts would remain below the criteria of a small quantity generator as defined by RCRA.

Examples of solid radioactive waste include low-level radioactive waste, such as radioactively contaminated protective clothing, tools, etc. The bounding value for solid radioactive waste generation was 5,000 ft^3/yr and a bounding activity of 57,200 Ci/yr for a 1,100 MWe generating output (NRC 2019a). This value does not reflect differences in solid radioactive waste streams between LWR and advanced reactors designs and does not account for any waste streams unique to advanced reactors. Scaling of waste bounding values is discussed in Section C.3 for transportation and scaled values are presented in Appendix E.

C.5.2 Gaseous Waste

Gaseous radioactive waste discharges will be controlled to the requirements of 10 CFR 20 and the ALARA principles of 10 CFR Part 50, Appendix I.

Typical gaseous radioactive wastes from large LWRs are released from vents on collection tanks and processing equipment and non-condensables in steam systems. The radioactive isotopes contained in these waste streams include fission products such as iodine, xenon and krypton, as well as activation products such as argon-41 and cobalt-60. The fission product inventories expected for advanced reactors are presented in Section C.7.3. Gaseous releases are typically collected and processed to decrease the radioactivity content to the point that they can be released to the environment through a controlled and monitored release point (plant vent or plant stack). The typical processing technique is one of holdup or delay to allow the short-lived activity to decay. Adsorption on activated charcoal or compression and storage are two methods used to create the necessary holdup time.

Specific systems for detecting minor leakage and ventilation systems for processing gases from plant systems will vary by prototype design. Microreactors would likely have no emissions of gaseous waste during routine operations. It is assumed that ventilation systems for small- to medium-sized advanced reactors will process gaseous releases by filtration, if needed, and direct the releases to a controlled and monitored release point. Part Gaseous radioactive waste discharges will be controlled to the requirements of Title 10 of the Code of Federal Regulations Part 20 (10 CFR Part 20) and the as low as reasonably achievable (ALARA) principles of 10 CFR Part 50, Appendix I. All emissions of gaseous radioactive effluents must comply with applicable regulatory limits established in 40 CFR Part 61 Subpart 61 Subpart H, and DOE O 458.1 (DOE 2020d), including DOE-STD-1196 (DOE 2011) and DOE-STD-1153 (DOE 2019a) (or 10 CFR Part 20 Appendix B depending on jurisdiction).

C.5.3 In-Vessel Solid Radioactive Waste

Molten salt reactors are assumed in this work to generate the greatest quantity of in-vessel solid radioactive waste, consisting of both fuel and primary loop coolant. In several molten salt reactor designs, the fuel is intimately mixed with the coolant. It is conservative to assume that no waste treatment will occur and therefore the full inventory of fuel and coolant must be treated as in-vessel solid radioactive waste. Estimations of the fuel and coolant waste masses were made by scaling the Thorcon and Molten Salt Reactor Experiment (MSRE) reactors, respectively, from their nominal power ratings to the maximum microreactor power rating of 60 MWt and small- to medium-sized advanced reactor power rating of 1,000 MWt.

Each Thorcon molten salt reactor power module is designed to produce 250 MWe (ThorCon 2020). According to data available on the company website, each module consumes 1,930 kg 19.7 percent enriched uranium and 3,290 kg thorium annually. This equates to 5.22 metric tons of combined uranium and thorium fuel waste per year; scaling this from 500 MWt to the maximum 60 MWt assumed for microreactors or 1,000 MWt for small- to medium-sized advanced reactors and assuming a 30- or 80-year demonstration period of operation, respectively, results in the conservative value of 19 and 836 MT fuel waste, respectively.

A similar scaling methodology was applied to calculate a bounding value for coolant waste mass. MSRE, an 8 MWt molten salt reactor, had an initial coolant loading of 15,300 lb. (7 MT) (ORNL 2015). Scaling this quantity to 60 MWt gives an initial coolant mass of 52.5 MT. Assuming the coolant would be replenished every 5 years and a 30-year demonstration period of operation, this results in 315 MT coolant waste. This waste could either take the form of high-level mixed waste or mixed transuranic waste depending upon whether there is any online fuel processing/waste management. A similar approach with an 80-year demonstration period was applied to obtain 13,920 MT of coolant waste for small- to medium-sized advanced reactors.

C.5.4 Liquid Waste

Radioactive waste system calculations are described in reactor final safety analysis reports for the design basis and realistic source terms for the primary and secondary coolants. The calculations are dependent on fuel design, steam generator design, reactor size, and other design factors. They are therefore specific to the reactor designs. All emissions of liquid radioactive effluents must comply with applicable regulatory limits established in DOE O 458.1 (DOE 2020d), including DOE-STD-1196 (DOE 2011) and DOE-STD-1153 (DOE 2019a) (or 10 CFR Part 20 Appendix B depending on the jurisdiction).

To develop reasonable estimates of waste volumes, SMEs reviewed the following documents:

- NuScale Design Control Document Final Safety Analysis Report (FSAR), Chapter 9, "Auxiliary Systems" (NuScale 2018a)
- NuScale Design Control Document Final Safety Analysis Report (FSAR), Chapter 11, "Radioactive Waste Management" (NuScale 2018b)
- Clinch River Nuclear Site Early Site Permit Application, Part 03—Environmental Report (TVA 2019)
- Final Safety Analysis Report (FSAR) – The Safety Case (Oklo 2020b) and Aurora Environmental Report—Combined License Stage (Oklo 2020a).

C.5.4.1 Liquid Waste Management System Description from NuScale (2018b)

The liquid radioactive waste management system (LRWS) is not safety related. NuScale inputs to LRWS include the following:

- primary coolant system letdown through chemical and volume control system and radwaste drain system (RWDS) equipment drains
- RWDS floor drains, solid radioactive waste system decant water, and the reactor component cooling-water system (RCCWS)
- detergent wastes from hand decontamination processes and personnel decontamination showers
- chemical wastes collected by the RWDS.

Liquid radioactive waste streams are treated and monitored before being discharged to the utility water system (UWS) discharge basin. Annual activity discharged through liquid effluents is calculated and provided in FSAR Table 11.2-5 (Ci/yr by isotope) (NuScale 2018b).

The UWS receives nonradioactive wastewater from other sources (see the UWS description below) that dilutes the LRWS discharge in the UWS basin (Section 9.2.9 [NuScale 2018a]). A dilution factor of 5.34 cfs of the LRWS discharge was assumed in the calculation of discharge concentrations to meet the 10 CFR Part 20 Appendix B, Table 2 limits at the point of discharge. (Sum of discharge concentration fraction of limits was 0.243, 96 percent due to tritium.) An additional dilution factor of 270 cfs (e.g., a river) was used in LADTAP II to calculate unrestricted area doses to meet ALARA design objective dose limits in 10 CFR Part 50 Appendix I. Site-specific dilution flows will be needed for a combined license application or an operating license application.

The UWS (Figure C.1) provides the distribution of clarified water to the fire protection water tank, demineralized water system, potable water system, reactor building, control building, annex building, radioactive waste building, turbine building, central utility building, and other plant users. The UWS also supplies raw water that has not been clarified to the two circulating water system cooling-tower basins and to the site cooling-water system cooling-tower basin for makeup water purposes. The water supplied by the UWS does not provide cooling functions. Raw water is the source of water for the UWS.

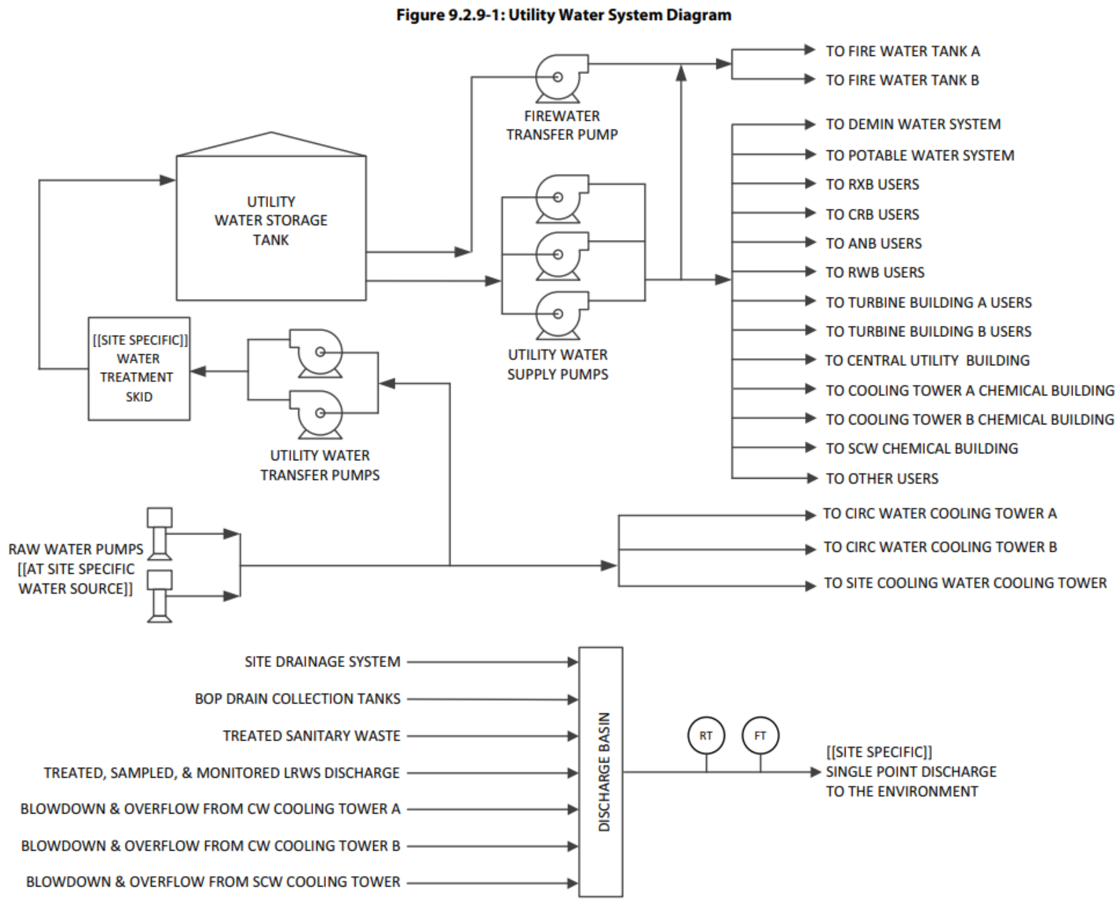

Figure C.1. Utility Water System Diagram

C.5.4.2 Liquid Waste Management from Clinch River (TVA 2019)

The Clinch River Environmental Report (ER) Section 3.5.1 describes liquid radioactive waste system releases and provides a table of annual total (Table 3.5-1) and single-reactor (Table 3.5-2) releases (Ci/yr by isotope) (TVA 2019).

ER Section 3.6 describes liquid nonradioactive waste system releases, which may include cooling water, wastewaters from operating systems (e.g., demineralized water system), floor and equipment drain waters, stormwater runoff, and sanitary sewer effluents (TVA 2019). Wastewater may contain residual chemicals and biocides used to avoid scaling or fouling of plant systems. The specific chemicals/biocides used and their concentrations in the discharge would be plant specific. The Clinch River ER provided anticipated constituents and their concentrations in Table C.1 (TVA 2019). For a larger plant with evaporative water cooling, the blowdown from the cooling system is the dominant portion of the liquid nonradioactive waste stream. The concentration of constituents in blowdown will depend on how the cooling system is operated (e.g., cycles of concentration) and the quality of the cooling-water source. Sanitary effluents would be similar to those from the existing workforce operations. For prototype reactors sited at INL, pretreatment of process and sanitary wastewaters would likely be required by existing discharge permits or would be similar to existing permit requirements if the discharges are authorized under a new permit.

Table C.1. Clinch River ESP Projected Blowdown Constituents and Concentrations

Constituent	Maximum Potential Concentration (ppm)[(a)]
Chlorine demand	1,000
Free available chlorine	0.5
Chromium	–
Copper	6
Iron	3.5
Zinc	0.6
Phosphate	7.2
Sulfate	3,500
Oil and grease	< 10
Total dissolved solids	17,000
Total suspended solids	150
Biological oxygen demand (BOD), 5-day	< 5
Calcium	260
Magnesium	85
Sodium	990
Manganese	0.1
Alkalinity as $CaCO_3$	150
Nitrate (NO_3)	52
Silicon dioxide (SiO_2)	150
pH Range	7.5–8.5

(a) Assumed four cycles of concentration.

C.5.4.3 Liquid Waste Management from Oklo (2020a, 2020b)

Oklo FSAR Section 1.2.2.4.2.3 states that “the Aurora does not use water for cooling, nor any other fluid that must be imported from offsite” (Oklo 2020b). ER Section 2.4.2 describes the heat rejection from the power conversion system to the atmosphere via radiators. The heating, ventilation, and air conditioning system uses standard commercial heat exchangers without using any water (Oklo 2020a). ER Section 2.5 states that “there are no permits required to ensure the Aurora INL site preparation and operation conforms with the Idaho Department of Environmental Quality regulations surrounding air and water quality.” ER Section 3.1.1.3 states, “the Aurora powerhouse does not use any water other than for incidental human use, which could be brought onsite and does not need be accessed from a local water line.” According to the ER, there are no process water uses and no process water discharges. Potable and sanitary water use would be the only considerations in this case, which the ER states could be satisfied using temporary facilities (bottled water and portable toilets).

Oklo FSAR Section 2.8.1 states that the power conversion system (PCS) is an off-the-shelf system, and that it can operate in turbine-bypass mode, in which heat is rejected to the environment (passively) (Oklo 2020b). Oklo FSAR Section 3.3.1.3 describes the PCS as using supercritical CO_2 as the working fluid (Rankine cycle). Heat from the reactor is transferred to the PCS using heat pipes fully contained within the reactor module. Potassium is the heat pipe working fluid (solid until operation melts the potassium and eventually vaporizes it). (According to the FSAR Section 5.6.2.5, the ultimate heat sink [the system of structures and components and associated assured water supply and atmospheric condition(s) credited for functioning as a heat sink to absorb reactor residual heat and essential station heat loads after a normal reactor shutdown or a shutdown following an accident or transient including a loss-of-coolant accident (NRC 2015)] involves passive heat removal to air via natural convection in the reactor cavity surrounding the reactor module [and subsequently conduction and radiation from the reactor module and the building]).

C.5.4.4 Recommendations for NRIC PPE

For a plant using evaporative cooling, liquid discharges will include cooling system blowdown. For most plants, discharge will include non-cooling process wastewater and potable/sanitary wastewater, with discharge rates assumed to be equivalent to the withdrawal rate (i.e., these water uses were assumed to be nonconsumptive).

Radionuclide releases in liquid wastes will be dependent on fuel design, steam generator design, reactor size, and other design factors. It is thus difficult to specify the constituents or inventory of radionuclides in the liquid radwaste system appropriate for use in the National Reactor Innovation Center (NRIC) PPE. However, it is safe to assume that any reactor that does have liquid radwaste will dilute that waste stream to meet the 10 CFR 20.1302 requirement to demonstrate that the annual average concentrations of radioactive material released in liquid effluents at the boundary of the unrestricted area (e.g., the point of discharge) do not exceed the values specified in 10 CFR Part 20 Appendix B, Table 2, Column 2. For a mixture of unidentified radionuclides, the lowest concentration value in Table 2 for any radionuclide not known to be absent from the effluent would be the applicable limit (Appendix B, Note 2). For a mixture of known radionuclides, the sum of the fractions (radionuclide effluent concentration divided by its limit) must be less than 1.0. For LWRs, the limiting radionuclide is typically tritium, but this may not be the case for non-LWR designs. For the NRIC PPE, an approach that would bound any liquid radwaste discharge would be to specify the PPE as the 10 CFR Part 20 Appendix B, Table 2, Column 2 effluent concentrations. The mixture limits would require that the actual discharge concentrations be less than these values.

Nonradioactive liquid waste would be dominated by the blowdown for a plant using evaporative cooling. For the NRIC PPE, it is assumed that nonradioactive liquid waste constituents and concentrations for a plant using an alternative cooling method (e.g., air-cooling) would be bounded by the blowdown from an evaporatively cooled plant. Blowdown concentrations will depend less on the size of the plant and more on the cycles of concentration at which the cooling system is operated; four cycles of concentration, which is not atypical for LWRs using surface water sources for cooling water, are assumed. Further, assuming that the additives to control scaling and biological growth used at Clinch River are typical, the NRIC PPE could use the Clinch River blowdown constituents and concentrations from the ER Table 3.6-1 (see above) (TVA 2019) for the nonradioactive liquid waste discharge. The specific mineralogical constituents of the blowdown at INL would vary from those listed in this table because of differences in the source waters. The Idaho Department of Environmental Quality (IDEQ 2018, p. 29) shows the presence of arsenic, barium, and chromium in INL groundwater; arsenic is present at a median concentration that would approach the drinking water standard at four cycles of concentration and thus arsenic should be added to the PPE (at a concentration of 0.01 mg/L, which is the drinking water standard). The INL groundwater is likely to be higher in conductivity and alkalinity than the Clinch River water, which could require fewer cycles of concentration to be used at INL or the use of different anti-scaling chemicals. This would reduce discharge concentrations and could introduce other discharge constituents.

C.5.5 Decommissioning Waste

At the end of the operating life of reactor prototypes, NRIC assumes the vendor and/or DOE would remove the facility from service, decommission the plant, and either remove or demolish plant structures and equipment. Decommissioning includes the reduction of residual radioactivity to a level that permits termination of an NRC license. It is assumed that a similar process would be required for termination of DOE's authorization of the prototype if an NRC license is not obtained. The regulations governing decommissioning of NRC-licensed power reactors are found in 10 CFR 50.75, 10 CFR 50.82, and 10 CFR 52.110. The radiological criteria for termination of the NRC license are in 10 CFR Part 20, Subpart E. Minimization of contamination and generation of radioactive waste requirements for facility design and procedures for operation are addressed in 10 CFR 20.1406. Additionally, 10 CFR 50.82 or 10 CFR 52.110, as applicable, provide that a NRC licensee shall not perform any decommissioning activities that result in significant environmental impacts not bounded by previously issued environmental review documents, such as NUREG-0586 "Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities" (NRC 2002).

Radioactive gaseous, liquid and solid wastes, nonradioactive waste and mixed waste could be generated during advanced reactor plant decommissioning, depending on the scope of the activities and the size and condition of the plant upon decommissioning. Decommissioning activities associated with small- to medium-sized advanced reactors would be similar in scope to activities required to decommission a LWR. Decommissioning activities for a modular reactor constructed in a factory and delivered to a site would be significantly less. Some smaller reactor prototypes may be removed in total and returned to the manufacturer.

The NRC evaluated the environmental impacts of decommissioning in NUREG-0586 "Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities" (NRC 2002). The anticipated volumes of wastes evaluated in NUREG-0586 were based on industry decommissioning experience as of 2002. Appendix G of NUREG-0586, "Radiation Protection Considerations for Nuclear Power Facility Decommissioning" summarizes effluent releases for operating facilities and decommissioning facilities. Appendix G includes occupational dose from reactors in the 60 MWe range for reference. Low-level waste volume estimates for decommissioning facilities are presented in Appendix K of NUREG-0586. The reactors considered in NUREG-0586 included non-LWR designs including fast breeder reactors such as FERMI Unit 1, and high-temperature gas-cooled reactors such as Peach Bottom Unit 1 and Fort St. Vrain.

The NRC's EIS for the Clinch River ESP (NRC 2019a) compared the impacts of decommissioning two or more SMRs with a combined capacity of 800 MWe to the impacts of decommissioning large LWRs described in NUREG-0586 (NRC 2002). The NRC found the following:

1. The quantities of Class C or greater than Class C wastes generated would be comparable to or less than the amounts of solid waste generated by reactors licensed before 2002.
2. The air-quality impacts of decommissioning are expected to be negligible at the end of the operating term.

3. Measures are readily available to avoid potential significant water-quality impacts from erosion or spills. The liquid radioactive waste system design includes features to limit release of radioactive material to the environment, such as pipe chases and tank collection basins. These features would minimize the amount of radioactive material in spills and leakage that would have to be addressed at decommissioning.

Microreactors will have significantly smaller reactor sizes, less infrastructure, and fewer balance-of-plant facilities than the LWRs described in NUREG-0586 (NRC 2002) and the SMRs described the Clinch River EIS (NRC 2019a). The plant facilities and infrastructure required for small- to medium-sized advanced reactors would be closer to, but smaller than those evaluated in the Clinch River EIS, based on the smaller bounding output of the proposed advanced reactors (1,000 MWt). Therefore, the amounts of solid waste and air emissions associated with decommissioning microreactors and small- to medium-sized advanced reactors would be bounded by the waste streams evaluated in NUREG-0586.

Demolition of plant structures and components that are not activated or otherwise contaminated would result in nonhazardous debris that could be disposed in commercial landfills.

C.6 Other Air Emissions

Criteria pollutants (nitrogen oxides [NO_x], carbon monoxide [CO], sulfur oxides [SO_x], hydrocarbons in the form of volatile organic compounds, and particulate matter) may be emitted from a prototype plant during construction activities, as part of operations including periodic or routine operation of balance-of-plant equipment, such as emergency power, evaporator heating, plant space heating, and/or feed water purification, and also from vehicles. Emissions may occur during operation of auxiliary boilers for heating and startup and emergency power supply system diesel generators and/or gas turbines. Because a specific ANR technology and supporting equipment have not been selected, detailed emission data are not available.

When a specific prototype is selected for deployment, modeling will be conducted, as required, to demonstrate that the project emissions will not result in exceedances of the National Ambient Air Quality Standards (NAAQSs). INL is not in a maintenance area, a nonattainment area, or a tribal nonattainment area administered by the U.S. Environmental Protection Agency (EPA 2020). Because INL is located in an attainment area for all NAAQS criteria pollutants, the proposed project is not subject to a Nonattainment New Source Review.

Regulatory limits were proposed by the NRC in the ANR GEIS public scoping effort as bounding parameters for air quality for advanced reactors, depending on the location. For reference, the NRC determined that emissions of criteria pollutants from one or more SMRs at the Clinch River site with a total installed capacity of up to 800 MWe would not have a noticeable impact on air quality (NRC 2019a). It is assumed that any new advanced reactor prototype constructed and operated at the INL site, up to and including a 1,000 MWt advanced reactor, would also comply with all regulatory requirements of the Clean Air Act, and therefore the de minimis levels of criteria pollutants would bound the potential emissions from any future reactor deployment. All emissions of gaseous radioactive effluents must comply with applicable regulatory limits established in 40 CFR Part 61 Subpart H and DOE O 458.1 (DOE 2020d), including DOE-STD-1196 (DOE 2011) and DOE-STD-1153 (DOE 2019a) (or 10 CFR Part 20 Appendix B, depending on the jurisdiction).

The NRC has evaluated greenhouse gas (GHG) emissions in previous National Environmental Policy Act (NEPA) reviews (for example, see NRC 2019a, Appendix K) for a 1,000 MWe reference plant. The emissions assumed for the reference plant included emissions during all phases of construction, operation, and decommissioning of the plant and included emissions from the uranium fuel cycle such as from mining and milling uranium ore. It is assumed that the GHG emissions from a microreactor or a small- to medium-sized advanced reactor, including any fuel cycle emissions, would be no greater than the emissions from the reference reactor when downscaled based on reactor size. In particular, GHG emissions from a 60 MWt reactor would be no greater than 2 percent of the emissions from the 1,000 MWe reference reactor and GHG emissions from a small- to medium-sized advanced reactor with output 1,000 MWt or less would be no greater than 33 percent of the emissions of the 1,000 MWe reference reactor.

Based on NRC's approach, Table C.2 presents estimated GHG emissions for a 60 MWt prototype and a 1,000 MWt prototype to be constructed at INL, operated and decommissioned, including emissions from the fuel cycle. Emissions for a 60 MWt plant assume 30 years of operation. Emissions for a 1,000 MWt plant assume 80 years of operation.

Table C.2. Nuclear Power Plant Lifetime Greenhouse Gas Footprints[a]

Source	NRC's Activity Duration (year)	Total Emissions 1,000 MWe Reference Plant[b]	Total Emissions Advanced Reactors	
			60 MWt Plant[c]	1,000 MWt Plant[d]
Preconstruction/Construction Equipment	7	39,000	780	12,870
Preconstruction/Construction Workforce	7	43,000	860	14,190
Plant Operations	40	181,000	2,715	119,460
Operations Workforce	40	136,000	2,040	89,760
Uranium Fuel Cycle	40	10,100,000	151,500	6,666,000
Decommissioning Equipment	10	19,000	380	6,270
Decommissioning Workforce	10	8,000	160	2,640
SAFSTOR Workforce	40	10,000	200	3,300
TOTAL		10,536,000	158,635	6,914,490

(a) Derived from NRC 2019a, Appendix K.
(b) Emissions are rounded to the nearest 1,000 MT CO_2e.
(c) Estimates for a 60 MWt plant are calculated as 2 percent of the reference plant estimates assuming 33 percent thermal efficiency and are based upon a 30-year plant operation cycle instead of NRC's assumed 40-year operation cycle.
(d) Emissions from a 1,000 MWt plant are calculated as 33 percent of the reference plant estimates assuming 33 percent thermal efficiency and are based upon a 80-year plant operation cycle instead of NRC's assumed 40-year operation cycle.

C.7 Fission Product Inventory

Each company that proposes a small- to medium-sized advanced reactor concept does so with a distinct and usually proprietary design. Fission product inventories were calculated based on generalized designs, described in further detail below. Once the generalized designs were selected, depletion models were run using TRITON (Transport Rigor Implemented with Time-dependent Operation for Neutronic depletion) in Standardized Computer Analyses for Licensing Evaluation (SCALE) to create library files for later use. The model for each category of reactor was the simplest unit-cell representation possible. For example, the LeadCold reactor only used one fuel assembly in two-dimensional geometry. Using a full three-dimensional design would provide the most precise results for the specific reactor, but that level of precision is not required for calculation of bounding core inventories.

TRITON runs a defined number of depletion cases based on the specified power level (MW/MTHM). Power levels and maximum burnup used for the depletion steps were representative of the reactors. For example, LeadCold has an average power of 3.3 MW/MTHM with a maximum center assembly burnup of 58 GWD/MTHM. The number of depletion steps for all models was balanced to achieve both a fast runtime and enough resolution to be confident in the results. Each model (except the molten salt reactor designs) was run at several different uranium enrichments to provide more flexibility in subsequent analyses. The molten salt reactor models were run with and without thorium fuel.

Radionuclide inventories were then calculated using ORIGEN (Oak Ridge Isotope Generation Code) from the TRITON libraries. All solid-fuel reactors were run at 10 percent, 15 percent, and 19.95 percent enrichment, 25 library positions, and depletion to maximum burnup from the TRITON calculations. The input material was only UO_2 with 1 MT of uranium. The input specific power was 20 MW/MTHM. The liquid-fuel reactors were run at a single library position that was closest to a k-effective of 1 from the TRITON runs. Input material was the respective fuel salt (1 MT heavy metal, split between uranium and thorium). If the run included thorium, a constant feed of ^{232}Th was added into the system at a rate of 10 kg/yr. The molten salt reactors were depleted for 20 years at 20 MW/MTHM.

During actual operation, a molten salt reactor is likely to have continuous feed of fuel and removal of fission products and neutron poisons. Therefore, the molten salt reactors were depleted for a much longer period than any of the other reactors. However, the ORIGEN model only had a feed of ^{232}Th in the thorium reactors, and no model incorporated the removal of fission products or neutron poisons; it is assumed that waste will remain onsite until decommissioning and is therefore available for release in an accident scenario.

After the depletion runs, ORIGEN performed decay-only calculations on the models out to 20 years, with small initial timesteps to account for short-lived isotopes. A decay time of 1 year was used for radionuclide inventory comparison.

C.7.1 Microreactors

Each company that proposes a microreactor concept does so with a distinct and usually proprietary design, and thus, establishing core models for each design is inefficient. Therefore, each reactor that had some public information available was categorized by general reactor type. The analysis then uses these general cases, instead of specific designs, to calculate the radionuclide inventory. Each design, detailed below, was based on either an existing design or a simplified representation.

- Helium-cooled prismatic reactor, with tri-structural isotropic (TRISO) fuel. This design was based on the High-Temperature Engineering Test Reactor (HTTR) (IAEA 2004; Bess et al. 2020).
- Helium-cooled pebble-bed reactor, with TRISO fuel. This design was based on the HTR-10 (IAEA 2004).
- Lead-cooled fast reactor, with UO_2 fuel. This design was based on the LeadCold reactor (Wallenius et al. 2018).
- Heat-pipe fast reactor, with UO_2 fuel. This design was based on the Special Purpose Reactor (INL 2017; Hernandez et al. 2018).
- Molten salt thermal reactor, with liquid fuel. This design was based on the ThorCon reactor (Fei et al. 2019; EPRI 2015).
- Molten salt fast reactor, with liquid fuel. This was a simplified design that separated out the fuel salt and the cooling salt (Betzler et al. 2016).

Some of these designs, while not specifically microreactors, were taken as a scaled representation to derive neutronic properties similar to those of an anticipated microreactor design of the same category. The LeadCold reactor is the only microreactor design that is available to the public.

C.7.2 Small- to Medium-Sized Advanced Reactors

Trying to establish models for each design can be difficult if not enough public information is available. However, reactors can be grouped based on input parameters so that the neutron spectra and self-shielding properties are similar. Therefore, one type of reactor, with the most publicly available information, can be used as an input library for ORIGEN. The different reactor types with public information used for this work are summarized below:

- Sodium fast reactor (SFR). This design is based on the General Electric (GE) Power Reactor Innovative Small Module (PRISM) (Triplett et al. 2010).
- Lead-cooled fast reactor (LFR). This design is based on the LeadCold reactor (Wallenius et al. 2018), but it was run at a higher specific power (MW/MTIHM) and out to a higher burnup.
- Boiling-water reactor (BWR). This design is based on the GE 10x10 BWR fuel distributed with SCALE.
- Pressurized water reactor (PWR). This design is based on the Westinghouse 17x17 PWR fuel distributed with SCALE (Wieselquist et al. 2020).
- Fluorine-lithium-beryllium (FLIBE) cooled pebble bed reactor (PBR). This design is based on the PB-FHR (Andreades et al. 2014).
- Helium-cooled PBR with TRISO fuel. This design is based on the HTR-10 (IAEA 2004).
- Helium-cooled prismatic reactor with TRISO fuel. This design is based on the HTTR (IAEA 2004).
- Molten salt thermal reactor with liquid fuel. This design is based on the Thorcon Reactor (Fei et al. 2019; EPRI 2015).
- Molten salt fast reactor with liquid fuel. This design is a simplified design that separated out the fuel salt and the cooling salt (Betzler et al. 2016).

With these groups, specific reactors can then be modeled with ORIGEN as long as they have enough publicly available information to provide accurate input parameters. However, some of the reactor designs above were also run as generic but possible configurations to account for possible small- to medium-sized advanced reactors using that specific technology.

C.7.3 Fission Product Inventory Comparison

Comparison of reactor core inventories based on reactor design was limited to comparison of activity for each radionuclide in the inventory. Without understanding the mechanisms or chemical form of release it must be assumed that a release of a specific radionuclide will be equivalent to the release of that same radionuclide from another reactor design (i.e., same mode of release, chemical composition, particle size, etc.). Assuming that individual radionuclides are released in comparable ways across reactor designs, radionuclide inventories can then be compared side-by-side based on activity.

This analysis can only be used to compare designs for individual radionuclide activities. This analysis does not address which combination of radionuclides would lead to a higher dose consequence to people. Dose conversion factors for radionuclides vary from radionuclide to radionuclide and trying to determine the combination of radionuclides that will lead to a higher dose is a very involved, complicated problem. This question should only be addressed when the source term release quantity, mechanism, and chemical form are known.

Fission product core inventories were compared across the different reactor types described above. The summary data set includes the reactor source term inventories organized alphabetically by radionuclide for all reactor designs included at Time 0 (end of operation) and Time 1 (1 year after end of operations). The molten salt reactors exhibited the highest activities for most radionuclides and had the highest total activities. From the summary data, a minimum, maximum, and average radionuclide activity can be deduced. However, not all radionuclides are mobile and available for release; Table C.3 and Table C.4 present a limited look at the radionuclides for microreactors that have potential mobility at Time 0 and Time 1. Calculations are based on observations of the Chernobyl and Fukushima accidents. The radionuclides compared here include noble gases as well as radionuclides known to be volatile or intermediately volatile as defined by Steinhauser et al. (2014). For reference, Table C.5 presents summary data for all radionuclides evaluated for microreactors. Similar information for small- to medium-sized advanced reactors is presented in Table C.6 (radionuclides with potential mobility at time 0) and Table C.7 (all radionuclides at time 0).

Table C.3. Minimum, Maximum, and Average Radionuclide Activity for Radionuclides with Potential Mobility at Time 0 (End of Operation) for Microreactors

	Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Noble Gases	Kr79	3.44E+03	7.77E+04	2.51E+04	183.1
	Kr81	2.40E+02	2.30E+06	3.28E+05	200.0
	Kr81m	3.13E+01	9.52E+02	3.24E+02	187.3
	Kr83m	9.55E+12	2.41E+13	1.45E+13	86.5
	Kr85	1.95E+14	2.36E+15	8.07E+14	169.5
	Kr85m	1.07E+14	2.93E+14	1.89E+14	93.3
	Kr87	1.59E+10	4.45E+10	3.06E+10	94.6
	Kr88	2.78E+13	7.85E+13	5.62E+13	95.3
	Rn217	1.39E+06	2.20E+06	1.79E+06	44.9
	Rn218	1.40E+03	1.27E+09	1.10E+08	200.0
	Rn219	9.62E+03	2.60E+09	1.97E+08	200.0
	Rn220	3.61E+06	8.73E+12	9.50E+11	200.0
	Rn222	1.24E+03	4.28E+06	5.19E+05	199.9
	Xe125	1.58E+03	6.63E+04	2.43E+04	190.7
	Xe127	3.05E+06	2.53E+10	3.20E+09	200.0
	Xe129m	5.86E+09	2.17E+13	3.13E+12	199.9
	Xe131m	2.12E+14	3.83E+14	2.52E+14	57.7
	Xe133	4.00E+16	5.72E+16	4.47E+16	35.4
	Xe133m	1.08E+15	1.80E+15	1.28E+15	49.6
	Xe135	1.20E+16	2.09E+16	1.47E+16	53.9
	Xe135m	5.59E+14	7.74E+14	6.08E+14	32.3
	Xe138	5.94E-15	7.90E-15	6.74E-15	28.4
Volatile	Cs131	7.54E+05	7.18E+07	1.39E+07	195.8
	Cs132	7.67E+09	8.57E+11	2.59E+11	196.5
	Cs134	1.53E+14	4.04E+16	9.92E+15	198.5

	Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
	Cs134m	1.85E+11	1.70E+13	4.58E+12	195.7
	Cs135	2.49E+10	3.11E+11	9.45E+10	170.4
	Cs135m	1.90E+04	8.77E+05	1.76E+05	191.5
	Cs136	2.90E+14	1.99E+16	3.37E+15	194.3
	Cs137	1.74E+15	2.21E+16	9.11E+15	170.9
	Cs138	7.07E+03	9.45E+03	7.98E+03	28.8
	H3	7.23E+12	3.89E+16	3.49E+15	199.9
	I123	1.01E+04	2.03E+05	6.42E+04	181.0
	I125	3.37E+02	1.15E+06	2.15E+05	199.9
	I126	2.26E+08	4.12E+10	1.20E+10	197.8
	I128	1.06E-04	1.26E-02	2.24E-03	196.7
	I129	3.62E+08	9.18E+09	2.57E+09	184.8
	I130	1.29E+13	1.56E+15	2.77E+14	196.7
	I131	1.82E+16	2.97E+16	2.12E+16	48.0
	I132	2.39E+16	3.55E+16	2.69E+16	39.3
	I132m	5.37E+08	5.13E+09	2.11E+09	162.1
	I133	1.98E+16	2.72E+16	2.18E+16	31.3
	I133m	3.18E+07	4.35E+07	3.65E+07	31.1
	I134	1.09E+09	1.39E+09	1.22E+09	23.9
	I135	3.25E+15	4.51E+15	3.54E+15	32.3
	Te121	3.02E+05	8.02E+08	9.18E+07	199.8
	Te121m	1.52E+05	3.96E+08	4.52E+07	199.8
	Te123m	3.05E+09	1.02E+13	1.36E+12	199.9
	Te125m	2.62E+13	2.12E+14	8.65E+13	156.0
	Te127	1.12E+15	3.67E+15	1.80E+15	106.2
	Te127m	1.43E+14	5.91E+14	2.46E+14	122.1
	Te129	5.25E+14	1.54E+15	8.00E+14	98.2
	Te129m	6.62E+14	1.95E+15	1.01E+15	98.5
	Te131	4.40E+14	1.37E+15	6.93E+14	102.8
	Te131m	1.68E+15	5.22E+15	2.64E+15	102.8
	Te132	2.32E+16	3.45E+16	2.61E+16	39.3
	Te133	6.77E+07	9.26E+07	7.77E+07	31.1
	Te133m	3.18E+08	4.35E+08	3.65E+08	31.1
	Te134	1.64E+06	2.04E+06	1.84E+06	21.8
Semi-Volatile	Ru103	2.20E+16	5.94E+16	3.11E+16	91.9
	Ru105	2.27E+14	1.19E+15	4.68E+14	136.0
	Ru106	3.38E+15	4.07E+16	1.25E+16	169.3
	Sr83	1.32E+02	1.16E+03	5.79E+02	159.1
	Sr85	6.47E+06	8.00E+08	1.52E+08	196.8
	Sr85m	1.40E+00	9.47E+01	1.89E+01	194.2
	Sr87m	2.68E+07	5.13E+10	7.76E+09	199.8

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Sr89	1.29E+16	3.42E+16	2.58E+16	90.4
Sr90	1.51E+15	1.72E+16	6.82E+15	167.8
Sr91	3.36E+15	6.92E+15	5.78E+15	69.3
Sr92	4.90E+13	8.75E+13	7.35E+13	56.4
Ba131	1.87E+03	4.81E+10	5.01E+09	200.0
Ba133	4.45E+06	1.78E+09	5.91E+08	199.0
Ba135m	1.67E+10	1.74E+14	2.72E+13	200.0
Ba136m	3.21E+13	2.20E+15	3.73E+14	194.3
Ba137m	1.65E+15	2.10E+16	8.62E+15	170.9
Ba139	2.60E+11	3.69E+11	2.95E+11	34.8
Ba140	3.48E+16	5.03E+16	3.96E+16	36.3
Ba141	6.67E-08	9.65E-08	7.53E-08	36.5
Total	9.05E+17	2.18E+18	1.30E+18	82.7

Table C.4. Minimum, Maximum, and Average Radionuclide Activity for Radionuclides with Potential Mobility at Time 1 (One Year After End of Operation) for Microreactors

	Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Noble Gases	Kr81	2.40E+02	2.30E+06	3.28E+05	200.0
	Kr83m	3.87E+05	1.08E+07	3.85E+06	186.1
	Kr85	1.83E+14	2.21E+15	7.57E+14	169.5
	Rn217	1.43E+05	4.26E+05	2.85E+05	99.3
	Rn218	4.02E-02	5.00E+03	6.84E+02	200.0
	Rn219	2.27E+04	3.34E+09	2.69E+08	200.0
	Rn220	1.25E+07	1.24E+13	1.25E+12	200.0
	Rn222	2.86E+03	4.63E+06	5.63E+05	199.8
	Xe127	2.98E+03	2.47E+07	3.12E+06	200.0
	Xe129m	2.68E-03	9.92E+00	1.43E+00	199.9
	Xe131m	3.53E+05	6.02E+05	4.16E+05	52.2
	Xe133	5.65E-05	8.06E-05	6.30E-05	35.1
Volatile	Cs131	7.16E-06	4.37E-04	9.14E-05	193.6
	Cs132	9.44E-08	1.06E-05	3.19E-06	196.5
	Cs134	1.10E+14	2.89E+16	7.10E+15	198.5
	Cs135	2.49E+10	3.11E+11	9.45E+10	170.4
	Cs136	1.37E+06	9.38E+07	1.59E+07	194.3
	Cs137	1.70E+15	2.16E+16	8.90E+15	170.9
	H3	6.84E+12	3.68E+16	3.29E+15	199.9
	I125	4.82E+00	1.64E+04	3.08E+03	199.9
	I126	7.58E-01	1.38E+02	4.04E+01	197.8

	Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
	I129	3.67E+08	9.19E+09	2.58E+09	184.7
	I131	4.12E+02	6.82E+02	4.83E+02	49.4
	Te121	3.29E+04	8.55E+07	9.76E+06	199.8
	Te121m	3.28E+04	8.52E+07	9.73E+06	199.8
	Te123m	3.68E+08	1.23E+12	1.63E+11	199.9
	Te125m	2.42E+13	1.76E+14	7.28E+13	151.6
	Te127	1.55E+13	5.91E+13	2.48E+13	116.8
	Te127m	1.59E+13	6.04E+13	2.53E+13	116.8
	Te129	2.29E+11	6.73E+11	3.48E+11	98.5
	Te129m	3.63E+11	1.07E+12	5.52E+11	98.5
Semi-Volatile	Ru103	3.55E+13	9.60E+13	5.02E+13	91.9
	Ru106	1.71E+15	2.06E+16	6.34E+15	169.3
	Sr85	1.32E+05	1.63E+07	3.11E+06	196.8
	Sr89	8.75E+13	2.32E+14	1.75E+14	90.3
	Sr90	1.47E+15	1.68E+16	6.65E+15	167.8
	Ba131	5.54E-07	1.81E-05	7.14E-06	188.1
	Ba133	4.17E+06	4.51E+10	4.90E+09	200.0
	Ba136m	1.51E+05	1.04E+07	1.76E+06	194.3
	Ba137m	1.61E+15	2.05E+16	8.43E+15	170.9
	Ba140	8.91E+07	1.29E+08	1.01E+08	36.3
	Total	4.21E+16	2.24E+17	1.01E+17	136.7

Table C.5. Minimum, Maximum, and Average Radionuclide Activity at Time 0 (End of Operation) for Microreactors

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Ac225	2.07E+03	3.13E+10	3.19E+09	200.0
Ac226	1.59E+02	2.26E+09	3.10E+08	200.0
Ac227	1.06E+04	2.61E+09	1.92E+08	200.0
Ac228	8.59E+02	2.26E+10	2.38E+09	200.0
Ag106	1.42E+03	4.97E+04	1.34E+04	188.9
Ag106m	2.59E+03	8.70E+04	2.25E+04	188.4
Ag107m	2.60E+05	6.90E+06	2.45E+06	185.5
Ag108	1.12E+08	7.42E+10	1.38E+10	199.4
Ag108m	2.46E+05	1.46E+08	2.98E+07	199.3
Ag109m	7.53E+14	2.55E+16	6.82E+15	188.5
Ag110	1.28E+13	2.17E+16	4.34E+15	199.8
Ag110m	3.39E+11	6.47E+14	1.30E+14	199.8
Ag111	2.81E+14	4.29E+15	1.18E+15	175.4
Ag111m	2.83E+14	4.10E+15	1.15E+15	174.2

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Ag112	1.62E+14	1.84E+15	5.58E+14	167.7
Ag113	9.09E+13	8.06E+14	2.62E+14	159.4
Ag113m	1.33E+14	1.18E+15	3.83E+14	159.4
Ag114	1.09E+14	7.26E+14	2.72E+14	147.8
Ag115	9.62E+13	4.40E+14	2.07E+14	128.3
Ag115m	5.81E+12	1.21E+13	7.25E+12	70.4
Ag116	1.04E+14	3.73E+14	1.94E+14	113.0
Ag116m	6.72E+12	2.54E+13	1.22E+13	116.3
Ag117	8.38E+13	2.70E+14	1.54E+14	105.1
Ag117m	1.47E+13	6.16E+13	2.95E+13	122.9
Ag118	5.89E+13	1.86E+14	1.17E+14	103.8
Ag118m	2.86E+13	8.65E+13	5.06E+13	100.5
Ag119	5.63E+13	1.50E+14	1.05E+14	90.8
Ag120	2.29E+13	7.77E+13	4.68E+13	108.8
Ag120m	4.73E+12	3.53E+13	2.07E+13	152.7
Ag121	1.79E+13	7.24E+13	3.82E+13	120.9
Ag122	2.34E+12	2.00E+13	8.49E+12	158.2
Ag122m	2.29E+12	1.77E+13	7.68E+12	154.3
Ag123	1.56E+12	1.84E+13	7.17E+12	168.8
Ag124	5.52E+11	8.71E+12	4.23E+12	176.2
Ag125	1.58E+10	1.83E+12	6.36E+11	196.6
Ag126	3.04E+09	3.59E+11	1.30E+11	196.6
Ag127	4.66E+08	5.50E+10	2.10E+10	196.6
Ag128	6.37E+07	7.52E+09	2.92E+09	196.6
Ag129	1.07E+07	2.90E+09	7.26E+08	198.5
Ag130	1.55E+08	3.95E+10	2.42E+10	198.4
Am239	3.61E+02	7.65E+07	5.29E+06	200.0
Am240	1.50E+06	5.63E+10	4.29E+09	200.0
Am241	1.08E+09	1.99E+14	1.88E+13	200.0
Am242	2.00E+10	2.61E+16	3.63E+15	200.0
Am242m	7.06E+06	3.39E+13	2.49E+12	200.0
Am243	3.12E+04	1.73E+13	3.23E+12	200.0
Am244	7.08E+05	4.31E+15	7.01E+14	200.0
Am244m	9.96E+06	6.45E+16	1.05E+16	200.0
Am245	5.87E+02	3.81E+12	5.64E+11	200.0
Am246	2.90E+04	3.83E+06	1.41E+06	197.0
Am246m	3.48E+05	3.54E+06	1.95E+06	164.3
As72	4.95E+02	2.77E+03	1.47E+03	139.3
As73	3.48E+04	1.40E+06	3.47E+05	190.3
As74	4.41E+06	2.08E+08	4.33E+07	191.7
As76	8.83E+10	1.77E+13	2.61E+12	198.0

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
As77	4.54E+13	1.35E+14	6.49E+13	99.1
As78	1.29E+14	2.98E+14	1.62E+14	79.5
As79	2.77E+14	7.44E+14	3.56E+14	91.4
As80	5.85E+14	1.29E+15	8.19E+14	75.0
As81	1.04E+15	2.01E+15	1.28E+15	63.3
As82	6.64E+14	1.60E+15	1.37E+15	82.6
As82m	1.97E+14	8.70E+14	3.32E+14	126.3
As83	1.01E+15	2.73E+15	1.90E+15	91.9
As84	5.96E+14	1.55E+15	1.22E+15	89.0
As85	2.33E+14	1.44E+15	1.04E+15	144.2
As86	9.41E+13	3.52E+15	2.27E+15	189.6
As87	1.91E+13	2.69E+14	1.90E+14	173.5
As88	3.34E+12	7.72E+14	4.77E+14	198.3
As89	3.63E+11	3.92E+12	1.75E+12	166.1
As90	1.99E+10	1.05E+11	4.99E+10	136.3
As91	1.01E+09	1.75E+10	6.70E+09	178.1
As92	1.67E+07	1.44E+09	5.01E+08	195.4
At217	2.07E+03	3.13E+10	3.19E+09	200.0
B12	1.25E+05	2.95E+09	5.16E+08	200.0
Ba131	1.99E+03	6.49E+04	2.56E+04	188.1
Ba133	4.46E+06	4.81E+10	5.24E+09	200.0
Ba135m	2.98E+10	3.11E+14	4.86E+13	200.0
Ba136m	3.39E+13	2.32E+15	3.94E+14	194.3
Ba137m	1.65E+15	2.10E+16	8.64E+15	170.8
Ba139	3.81E+16	5.45E+16	4.33E+16	35.4
Ba140	3.68E+16	5.31E+16	4.18E+16	36.3
Ba141	3.49E+16	5.05E+16	3.93E+16	36.7
Ba142	3.24E+16	4.79E+16	3.75E+16	38.6
Ba143	2.91E+16	3.63E+16	3.36E+16	22.3
Ba144	2.20E+16	2.76E+16	2.58E+16	22.6
Ba145	9.90E+15	1.35E+16	1.17E+16	31.1
Ba146	3.85E+15	6.77E+15	5.36E+15	55.1
Ba147	8.78E+14	1.94E+15	1.44E+15	75.2
Ba148	1.23E+14	3.41E+14	2.01E+14	93.9
Ba149	6.82E+12	4.80E+13	2.22E+13	150.2
Ba150	3.66E+11	5.13E+12	2.10E+12	173.4
Ba151	1.40E+10	8.48E+11	3.00E+11	193.5
Ba152	2.41E+08	1.35E+10	4.92E+09	193.0
Ba153	6.87E+06	6.48E+08	2.25E+08	195.8
Be10	1.09E+04	2.96E+11	4.30E+10	200.0

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Be11	1.13E+02	1.43E+12	1.62E+11	200.0
Be8	1.14E+04	9.78E+16	1.54E+16	200.0
Bi210	5.14E+02	6.53E+08	5.51E+07	200.0
Bi211	9.59E+03	2.60E+09	1.98E+08	200.0
Bi212	3.59E+06	8.74E+12	9.51E+11	200.0
Bi213	2.07E+03	3.12E+10	3.19E+09	200.0
Bi214	1.23E+03	4.28E+06	5.19E+05	199.9
Bk248	4.30E+05	3.47E+06	1.95E+06	155.9
Bk248m	2.60E+04	3.07E+07	8.15E+06	199.7
Bk249	8.25E+01	4.95E+12	6.55E+11	200.0
Bk250	5.88E+02	5.56E+13	7.39E+12	200.0
Bk251	5.45E+06	1.15E+10	3.10E+09	199.8
Br77	7.41E+03	1.85E+05	5.42E+04	184.6
Br77m	5.82E+03	1.43E+05	4.18E+04	184.3
Br78	9.00E+05	1.97E+07	7.24E+06	182.5
Br79m	4.88E+07	1.07E+09	4.07E+08	182.6
Br80	5.89E+09	3.55E+11	6.78E+10	193.5
Br80m	2.32E+09	1.11E+11	2.15E+10	191.8
Br82	1.34E+13	9.96E+14	1.75E+14	194.7
Br82m	1.18E+13	9.21E+14	1.61E+14	194.9
Br83	2.22E+15	5.57E+15	3.36E+15	86.1
Br84	3.33E+15	8.62E+15	5.82E+15	88.5
Br84m	1.11E+14	7.26E+14	2.30E+14	146.8
Br85	4.22E+15	1.14E+16	7.55E+15	91.9
Br86	5.30E+15	1.28E+16	1.00E+16	83.0
Br87	5.57E+15	1.28E+16	1.10E+16	78.5
Br88	4.40E+15	1.14E+16	9.22E+15	88.3
Br89	3.18E+15	7.79E+15	6.09E+15	84.1
Br90	1.97E+15	4.24E+15	3.28E+15	73.3
Br91	3.23E+14	1.74E+15	1.19E+15	137.5
Br92	6.44E+13	3.31E+14	1.89E+14	135.0
Br93	2.33E+13	1.06E+14	6.13E+13	127.9
Br94	2.66E+12	1.57E+13	7.85E+12	142.1
Br95	1.65E+10	1.30E+11	5.62E+10	154.9
Br96	5.96E+09	4.06E+10	1.93E+10	148.8
Br97	5.43E+07	7.78E+08	3.62E+08	173.9
C14	3.67E+06	5.53E+10	1.18E+10	200.0
C15	2.67E+07	2.50E+10	8.15E+09	199.6
Cd107	1.04E+03	1.15E+05	3.32E+04	196.4
Cd109	4.09E+06	3.54E+10	4.40E+09	200.0

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Cd111m	6.40E+08	2.16E+13	3.41E+12	200.0
Cd113m	1.25E+10	1.51E+12	3.23E+11	196.7
Cd115	9.70E+13	4.85E+14	2.20E+14	133.3
Cd115m	5.61E+12	3.61E+13	1.44E+13	146.2
Cd117	8.27E+13	2.78E+14	1.54E+14	108.2
Cd117m	2.09E+13	6.33E+13	3.63E+13	100.7
Cd118	8.83E+13	2.61E+14	1.66E+14	99.0
Cd119	5.91E+13	1.57E+14	1.06E+14	90.8
Cd119m	3.45E+13	9.03E+13	5.85E+13	89.4
Cd120	8.90E+13	2.28E+14	1.57E+14	87.6
Cd121	4.75E+13	1.26E+14	8.82E+13	90.5
Cd121m	2.78E+13	8.63E+13	5.18E+13	102.6
Cd122	8.73E+13	1.74E+14	1.33E+14	66.2
Cd123	5.13E+13	1.35E+14	8.45E+13	90.1
Cd124	1.52E+13	1.18E+14	7.19E+13	154.4
Cd125	4.05E+12	5.84E+13	3.18E+13	174.0
Cd126	1.60E+12	4.88E+13	3.46E+13	187.3
Cd127	6.27E+11	4.78E+13	3.12E+13	194.8
Cd128	1.97E+11	2.11E+13	1.35E+13	196.3
Cd129	3.71E+10	3.87E+12	6.40E+11	196.2
Cd130	1.30E+12	5.18E+14	3.16E+14	199.0
Cd131	2.66E+11	9.55E+13	5.83E+13	198.9
Cd132	2.04E+09	1.61E+11	5.78E+10	195.0
Ce137	2.03E+04	8.51E+05	2.16E+05	190.7
Ce139	7.63E+09	1.05E+12	2.99E+11	197.1
Ce139m	2.95E+09	4.41E+11	1.24E+11	197.3
Ce141	3.51E+16	5.21E+16	3.97E+16	39.0
Ce143	3.28E+16	4.76E+16	3.81E+16	36.7
Ce144	2.74E+16	3.91E+16	3.32E+16	35.4
Ce145	2.24E+16	2.95E+16	2.52E+16	27.2
Ce146	1.79E+16	2.32E+16	1.98E+16	25.6
Ce147	1.23E+16	1.70E+16	1.36E+16	32.2
Ce148	9.27E+15	1.14E+16	1.03E+16	20.4
Ce149	4.88E+15	6.88E+15	5.55E+15	34.0
Ce150	2.56E+15	3.68E+15	2.96E+15	36.1
Ce151	6.45E+14	1.19E+15	8.53E+14	59.7
Ce152	1.34E+14	3.27E+14	2.16E+14	83.6
Ce153	1.18E+13	6.04E+13	3.56E+13	134.6
Ce154	7.26E+11	7.75E+12	4.01E+12	165.7
Ce155	2.93E+10	7.76E+11	3.50E+11	185.4

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Ce156	1.14E+09	4.95E+10	2.11E+10	191.0
Ce157	3.19E+07	2.00E+09	8.21E+08	193.7
Cf248	3.75E+04	2.76E+07	7.87E+06	199.5
Cf249	7.56E+07	2.48E+09	1.07E+09	188.2
Cf250	3.60E+03	4.05E+11	8.86E+10	200.0
Cf251	2.41E+07	3.02E+09	8.25E+08	196.8
Cf252	1.19E+03	7.19E+12	1.50E+12	200.0
Cf253	2.89E+08	9.58E+11	2.51E+11	199.9
Cf254	1.20E+05	1.94E+09	5.11E+08	200.0
Cf255	2.40E+05	3.22E+06	1.73E+06	172.3
Cm241	6.04E+03	4.85E+09	4.68E+08	200.0
Cm242	6.50E+09	2.02E+16	2.83E+15	200.0
Cm243	4.88E+05	3.71E+13	3.30E+12	200.0
Cm244	1.31E+05	1.02E+16	1.58E+15	200.0
Cm245	1.48E+04	2.01E+12	3.58E+11	200.0
Cm246	2.21E+04	3.43E+12	5.52E+11	200.0
Cm247	2.58E+06	4.74E+07	1.66E+07	179.3
Cm248	1.66E+07	1.67E+09	4.87E+08	196.0
Cm249	6.70E+01	6.20E+13	6.69E+12	200.0
Cm251	2.51E+05	4.43E+08	1.14E+08	199.8
Co65	8.18E+05	3.49E+07	1.35E+07	190.8
Co66	5.67E+08	2.32E+10	5.23E+09	190.4
Co67	1.98E+09	2.73E+10	1.21E+10	173.0
Co68	2.19E+09	3.20E+10	1.39E+10	174.4
Co69	2.55E+09	2.97E+10	1.35E+10	168.3
Co70	1.67E+09	2.02E+10	9.41E+09	169.5
Co71	1.06E+09	1.31E+10	5.39E+09	170.2
Co72	3.91E+08	4.69E+09	1.95E+09	169.2
Co73	1.02E+08	1.72E+09	7.49E+08	177.6
Co74	1.33E+07	3.51E+08	1.19E+08	185.4
Co75	1.52E+06	5.37E+07	1.71E+07	189.0
Cr66	3.65E+04	2.34E+06	9.32E+05	193.8
Cr67	6.93E+03	7.03E+05	2.98E+05	196.1
Cs131	8.10E+05	7.71E+07	1.49E+07	195.8
Cs132	8.54E+09	9.54E+11	2.88E+11	196.5
Cs134	1.53E+14	4.04E+16	9.92E+15	198.5
Cs134m	5.59E+13	5.14E+15	1.39E+15	195.7
Cs135	2.48E+10	3.11E+11	9.45E+10	170.4
Cs135m	2.87E+12	1.32E+14	2.66E+13	191.5
Cs136	3.05E+14	2.10E+16	3.55E+15	194.3

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Cs136m	3.59E+13	7.30E+14	1.92E+14	181.2
Cs137	1.74E+15	2.21E+16	9.11E+15	170.9
Cs138	4.00E+16	5.37E+16	4.50E+16	29.3
Cs138m	1.34E+15	4.90E+15	2.41E+15	114.1
Cs139	3.73E+16	5.10E+16	4.22E+16	31.2
Cs140	3.19E+16	4.06E+16	3.62E+16	24.0
Cs141	2.52E+16	3.31E+16	2.79E+16	27.1
Cs142	1.52E+16	1.85E+16	1.69E+16	19.5
Cs143	7.16E+15	9.97E+15	8.81E+15	32.8
Cs144	2.16E+15	3.75E+15	2.88E+15	53.9
Cs145	4.47E+14	9.29E+14	6.24E+14	70.1
Cs146	4.62E+13	1.49E+14	8.52E+13	105.2
Cs147	7.73E+12	2.39E+13	1.54E+13	102.2
Cs148	1.05E+11	2.26E+12	8.80E+11	182.4
Cs149	2.66E+09	4.16E+10	1.70E+10	176.0
Cs150	5.76E+07	4.31E+09	1.53E+09	194.7
Cs151	7.73E+06	7.55E+08	2.61E+08	195.9
Cu66	6.49E+08	3.27E+10	6.35E+09	192.2
Cu67	2.71E+09	5.12E+10	1.65E+10	179.9
Cu68	5.68E+09	8.75E+10	2.94E+10	175.6
Cu68m	1.22E+08	4.76E+09	9.07E+08	190.0
Cu69	1.36E+10	1.56E+11	5.62E+10	168.1
Cu70	3.02E+10	2.99E+11	1.09E+11	163.3
Cu70m	6.87E+09	9.29E+10	3.04E+10	172.5
Cu71	6.75E+10	8.79E+11	2.58E+11	171.5
Cu72	1.60E+11	1.37E+12	4.53E+11	158.1
Cu73	3.74E+11	2.21E+12	7.93E+11	142.2
Cu74	4.49E+11	1.88E+12	7.99E+11	123.0
Cu75	3.28E+11	2.11E+12	9.18E+11	146.2
Cu76	1.46E+11	1.04E+12	6.33E+11	150.6
Cu77	5.47E+10	5.28E+11	3.12E+11	162.4
Cu78	1.66E+10	1.53E+11	8.57E+10	160.9
Cu79	4.10E+08	3.36E+10	1.27E+10	195.2
Cu80	1.87E+08	3.44E+09	1.38E+09	179.4
Dy159	1.29E+05	3.09E+08	5.08E+07	199.8
Dy165	1.77E+11	4.46E+14	4.84E+13	199.8
Dy165m	1.71E+09	2.84E+14	3.06E+13	200.0
Dy166	4.28E+10	3.21E+12	5.43E+11	194.7
Dy167	1.10E+10	6.15E+11	1.49E+11	193.0
Dy168	3.41E+09	3.40E+11	7.37E+10	196.0

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Dy169	1.18E+09	1.33E+11	3.02E+10	196.5
Dy170	3.03E+08	4.76E+10	1.14E+10	197.5
Dy171	9.70E+07	1.46E+10	3.98E+09	197.4
Dy172	7.67E+07	7.13E+09	2.81E+09	195.7
Er165	5.14E+04	2.56E+09	1.86E+08	200.0
Er167m	2.25E+09	6.90E+13	6.35E+12	200.0
Er169	1.42E+09	6.27E+12	5.19E+11	199.9
Er171	1.91E+08	6.20E+10	1.09E+10	198.8
Er172	1.44E+08	2.44E+10	6.80E+09	197.6
Es253	2.41E+08	5.16E+11	1.35E+11	199.8
Es254	3.50E+05	1.22E+09	3.13E+08	199.9
Es254m	1.88E+07	1.24E+11	3.21E+10	199.9
Es255	5.24E+04	1.79E+09	6.22E+08	200.0
Eu149	8.40E+01	1.34E+05	3.32E+04	199.7
Eu152	3.13E+10	9.23E+12	1.48E+12	198.6
Eu152m	7.71E+11	5.70E+13	1.17E+13	194.7
Eu154	6.53E+12	3.10E+15	4.39E+14	199.2
Eu154m	1.56E+12	5.93E+14	1.12E+14	198.9
Eu155	7.47E+13	1.27E+15	2.90E+14	177.8
Eu156	1.78E+14	3.98E+16	6.81E+15	198.2
Eu157	8.62E+13	1.58E+15	3.71E+14	179.4
Eu158	4.56E+13	5.23E+14	1.55E+14	168.0
Eu159	1.95E+13	2.75E+14	7.76E+13	173.4
Eu160	7.64E+12	1.31E+14	3.50E+13	177.9
Eu161	2.68E+12	5.75E+13	1.48E+13	182.2
Eu162	6.39E+11	1.32E+13	3.56E+12	181.5
Eu163	1.43E+11	4.34E+12	1.05E+12	187.2
Eu164	1.77E+10	7.39E+11	1.86E+11	190.6
Eu165	2.44E+09	9.77E+10	3.13E+10	190.3
Eu166	2.62E+08	1.13E+10	5.08E+09	191.0
Eu167	2.83E+07	2.35E+09	8.76E+08	195.2
F20	5.93E+04	2.09E+17	1.59E+16	200.0
Fe65	8.18E+05	3.49E+07	1.35E+07	190.8
Fe66	2.91E+08	7.49E+09	2.26E+09	185.0
Fe67	4.49E+08	9.04E+09	3.58E+09	181.1
Fe68	2.69E+08	5.48E+09	2.16E+09	181.3
Fe69	9.39E+07	1.96E+09	7.88E+08	181.7
Fe70	2.54E+07	5.41E+08	2.29E+08	182.1
Fe71	3.91E+06	1.45E+08	4.48E+07	189.5
Fe72	4.90E+05	2.55E+07	6.68E+06	192.5

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Fr221	2.07E+03	3.13E+10	3.19E+09	200.0
Fr223	2.81E+02	3.60E+07	3.98E+06	200.0
Ga68	1.70E+03	4.94E+05	1.08E+05	198.6
Ga70	4.67E+07	1.81E+10	2.73E+09	199.0
Ga72	2.37E+11	2.65E+12	7.57E+11	167.1
Ga72m	9.01E+09	9.42E+10	2.84E+10	165.1
Ga73	7.73E+11	5.96E+12	1.83E+12	154.1
Ga74	2.35E+12	1.34E+13	4.41E+12	140.5
Ga74m	8.07E+10	7.48E+11	2.28E+11	161.1
Ga75	7.08E+12	3.67E+13	1.17E+13	135.3
Ga76	1.63E+13	6.05E+13	2.73E+13	115.2
Ga77	2.98E+13	9.47E+13	5.39E+13	104.4
Ga78	4.86E+13	1.15E+14	8.70E+13	81.0
Ga79	4.29E+13	1.38E+14	1.07E+14	105.3
Ga80	2.84E+13	8.95E+13	6.70E+13	103.6
Ga81	1.32E+13	5.90E+13	4.32E+13	126.9
Ga82	3.18E+12	3.91E+13	2.83E+13	169.9
Ga83	5.85E+11	4.75E+12	2.24E+12	156.2
Ga84	1.33E+11	6.76E+13	4.15E+13	199.2
Ga85	3.22E+09	9.18E+10	3.40E+10	186.4
Ga86	5.87E+08	1.94E+11	1.20E+11	198.8
Gd151	6.50E+05	1.84E+08	3.68E+07	198.6
Gd153	2.18E+09	7.97E+12	9.19E+11	199.9
Gd159	2.00E+13	1.43E+15	2.54E+14	194.5
Gd161	3.20E+12	7.60E+13	1.94E+13	183.9
Gd162	1.32E+12	2.70E+13	7.33E+12	181.4
Gd163	5.23E+11	1.43E+13	3.50E+12	185.9
Gd164	1.60E+11	5.24E+12	1.26E+12	188.2
Gd165	4.24E+10	1.46E+12	3.61E+11	188.7
Gd166	1.13E+10	3.30E+11	9.92E+10	186.8
Gd167	1.17E+09	5.54E+10	1.98E+10	191.7
Gd168	2.06E+08	1.12E+10	5.45E+09	192.7
Gd169	3.16E+07	2.42E+09	9.81E+08	194.8
Ge71	1.75E+05	7.88E+08	7.13E+07	199.9
Ge71m	4.25E+04	6.33E+05	1.91E+05	174.9
Ge73m	7.61E+11	5.92E+12	1.81E+12	154.4
Ge75	7.20E+12	3.78E+13	1.20E+13	136.0
Ge75m	3.23E+11	2.18E+12	6.33E+11	148.2
Ge77	4.45E+13	1.29E+14	6.51E+13	97.7
Ge77m	7.43E+11	5.51E+12	1.70E+12	152.5

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Ge78	1.26E+14	2.90E+14	1.59E+14	78.7
Ge79	1.42E+14	4.04E+14	2.16E+14	96.3
Ge79m	8.31E+13	2.87E+14	1.14E+14	110.3
Ge80	3.87E+14	1.01E+15	6.82E+14	89.4
Ge81	3.93E+14	8.30E+14	7.26E+14	71.5
Ge81m	5.47E+12	2.45E+13	1.79E+13	127.0
Ge82	2.41E+14	8.63E+14	6.70E+14	112.8
Ge83	8.86E+13	2.95E+14	2.06E+14	107.5
Ge84	2.75E+13	2.17E+14	1.54E+14	154.9
Ge85	4.30E+12	3.77E+13	1.94E+13	159.1
Ge86	3.16E+12	3.58E+15	2.19E+15	199.6
Ge87	5.25E+10	1.30E+13	8.04E+12	198.4
Ge88	3.84E+09	3.08E+11	2.02E+11	195.1
Ge89	6.25E+07	3.87E+09	1.35E+09	193.6
H3	7.23E+12	3.89E+16	3.49E+15	199.9
He6	4.23E+03	3.73E+16	4.71E+15	200.0
Ho161	1.86E+03	1.57E+05	2.73E+04	195.3
Ho162	3.01E+04	2.45E+06	5.54E+05	195.1
Ho162m	3.00E+04	2.39E+06	5.36E+05	195.0
Ho163	3.26E+02	8.95E+02	5.66E+02	93.3
Ho163m	8.59E+04	5.41E+06	1.21E+06	193.7
Ho164	4.37E+06	2.71E+09	2.52E+08	199.4
Ho164m	3.00E+06	1.27E+09	1.27E+08	199.1
Ho166	5.63E+10	2.86E+14	2.85E+13	199.9
Ho166m	2.52E+05	7.42E+08	9.26E+07	199.9
Ho167	1.18E+10	1.65E+13	1.68E+12	199.7
Ho168	3.60E+09	3.68E+11	7.88E+10	196.1
Ho169	1.39E+09	1.69E+11	3.64E+10	196.7
Ho170	3.49E+08	6.36E+10	1.40E+10	197.8
Ho170m	4.60E+07	1.61E+10	2.62E+09	198.9
Ho171	1.72E+08	4.10E+10	8.30E+09	198.3
Ho172	1.29E+08	1.67E+10	5.61E+09	196.9
I123	3.56E+04	7.13E+05	2.26E+05	181.0
I125	3.41E+02	1.16E+06	2.17E+05	199.9
I126	2.38E+08	4.35E+10	1.27E+10	197.8
I128	2.34E+13	2.80E+15	4.96E+14	196.7
I129	3.62E+08	9.18E+09	2.57E+09	184.8
I130	4.94E+13	5.96E+15	1.06E+15	196.7
I130m	3.14E+13	3.85E+15	6.82E+14	196.8
I131	1.95E+16	3.16E+16	2.28E+16	47.3

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
I132	2.91E+16	4.43E+16	3.32E+16	41.5
I132m	8.69E+13	8.29E+14	3.42E+14	162.1
I133	4.28E+16	5.89E+16	4.71E+16	31.5
I133m	3.07E+15	6.03E+15	4.04E+15	65.0
I134	4.82E+16	6.51E+16	5.32E+16	29.9
I134m	2.50E+15	8.15E+15	4.37E+15	106.1
I135	4.09E+16	5.67E+16	4.45E+16	32.3
I136	1.60E+16	2.06E+16	1.80E+16	24.7
I136m	8.52E+15	1.44E+16	9.93E+15	51.6
I137	1.90E+16	2.51E+16	2.05E+16	27.8
I138	9.51E+15	1.33E+16	1.06E+16	33.3
I139	3.68E+15	5.65E+15	4.75E+15	42.1
I140	7.13E+14	1.54E+15	1.09E+15	73.3
I141	1.22E+14	4.09E+14	2.62E+14	108.1
I142	2.60E+13	6.41E+13	4.34E+13	84.5
I143	1.85E+11	6.63E+12	2.41E+12	189.1
I144	1.08E+10	1.02E+11	4.41E+10	161.7
In111	6.78E+01	7.11E+04	1.60E+04	199.6
In112	1.72E+03	1.08E+07	2.01E+06	199.9
In112m	1.39E+03	8.75E+06	1.62E+06	199.9
In113m	3.60E+01	8.36E+06	1.21E+06	200.0
In114	7.50E+08	2.95E+11	3.47E+10	199.0
In114m	4.37E+08	1.66E+11	1.99E+10	199.0
In115m	9.74E+13	4.85E+14	2.22E+14	133.1
In116	1.33E+12	1.80E+14	5.00E+13	197.1
In116m	2.18E+12	2.99E+14	8.27E+13	197.1
In117	6.34E+13	2.06E+14	1.16E+14	105.9
n117m	7.60E+13	2.55E+14	1.42E+14	108.2
In118	8.83E+13	2.62E+14	1.66E+14	99.1
In118m	1.74E+10	1.81E+11	6.05E+10	164.9
In119	4.57E+13	1.16E+14	7.64E+13	86.8
In119m	5.36E+13	1.43E+14	9.60E+13	91.0
In120	9.09E+13	2.34E+14	1.61E+14	88.2
In120m	3.43E+12	1.45E+13	6.98E+12	123.5
In121	6.38E+13	1.87E+14	1.15E+14	98.3
In121m	3.54E+13	9.47E+13	6.64E+13	91.2
In122	9.68E+13	2.20E+14	1.56E+14	77.6
In122m	1.81E+13	1.01E+14	4.56E+13	139.1
In123	5.57E+13	2.37E+14	1.12E+14	123.9
In123m	4.97E+13	1.04E+14	7.53E+13	70.7

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
In124	9.71E+13	1.74E+14	1.36E+14	56.9
In124m	2.10E+13	1.40E+14	6.39E+13	148.1
In125	6.04E+13	1.70E+14	1.01E+14	95.3
In125m	4.77E+13	1.68E+14	9.37E+13	111.6
In126	6.76E+13	1.65E+14	1.04E+14	83.5
In126m	1.95E+13	1.55E+14	6.97E+13	155.2
In127	2.02E+14	2.99E+14	2.69E+14	38.9
In127m	5.31E+13	2.32E+14	1.19E+14	125.6
In128	1.21E+14	2.36E+14	1.59E+14	64.6
In128m	9.96E+13	2.36E+14	1.45E+14	81.2
In129	6.52E+13	2.34E+14	1.67E+14	112.7
In129m	6.51E+13	2.28E+14	1.55E+14	111.2
In130	7.90E+13	5.91E+14	4.00E+14	152.8
In130m	5.15E+13	1.51E+14	1.07E+14	98.2
In131	5.62E+13	9.14E+13	7.57E+13	47.7
In131m	3.58E+13	6.25E+13	4.64E+13	54.3
In132	1.99E+13	8.40E+13	4.75E+13	123.3
In133	1.05E+12	7.40E+12	3.20E+12	150.2
In134	2.58E+10	4.51E+11	1.70E+11	178.3
In135	2.10E+08	1.59E+10	5.58E+09	194.8
Kr100	3.37E+07	6.81E+09	4.27E+09	198.0
Kr79	5.53E+03	1.25E+05	4.04E+04	183.1
Kr79m	2.78E+03	6.25E+04	2.03E+04	183.0
Kr81	2.40E+02	2.30E+06	3.28E+05	200.0
Kr81m	1.72E+07	4.78E+10	6.62E+09	199.9
Kr83m	2.30E+15	5.83E+15	3.40E+15	86.9
Kr85	1.95E+14	2.36E+15	8.07E+14	169.5
Kr85m	4.32E+15	1.19E+16	7.66E+15	93.3
Kr87	7.57E+15	2.12E+16	1.45E+16	94.7
Kr88	9.72E+15	2.74E+16	1.96E+16	95.3
Kr89	1.13E+16	2.93E+16	2.40E+16	88.8
Kr90	1.04E+16	2.87E+16	2.40E+16	93.8
Kr91	6.75E+15	2.09E+16	1.67E+16	102.5
Kr92	3.57E+15	1.10E+16	8.56E+15	101.8
Kr93	1.05E+15	3.60E+15	2.64E+15	109.6
Kr94	3.35E+14	1.08E+15	6.48E+14	105.1
Kr95	3.22E+13	1.60E+14	7.95E+13	132.8
Kr96	5.00E+13	2.34E+14	1.65E+14	129.6
Kr97	1.08E+12	9.03E+12	3.23E+12	157.4
Kr98	3.25E+10	9.96E+12	6.16E+12	198.7

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Kr99	2.06E+08	1.95E+10	6.76E+09	195.8
La135	9.69E+04	2.52E+06	8.95E+05	185.2
La137	8.12E+03	1.90E+06	4.90E+05	198.3
La140	3.71E+16	6.40E+16	4.43E+16	53.1
La141	3.51E+16	5.14E+16	3.96E+16	37.7
La142	3.34E+16	5.07E+16	3.87E+16	41.1
La143	3.25E+16	4.62E+16	3.77E+16	34.9
La144	2.90E+16	3.80E+16	3.36E+16	26.9
La145	2.09E+16	2.61E+16	2.38E+16	21.8
La146	9.18E+15	1.15E+16	1.02E+16	22.1
La146m	4.19E+15	5.64E+15	4.80E+15	29.6
La147	5.59E+15	8.00E+15	6.60E+15	35.6
La148	2.12E+15	3.13E+15	2.58E+15	38.2
La149	5.06E+14	1.09E+15	7.63E+14	72.9
La150	6.92E+13	2.56E+14	1.52E+14	114.9
La151	7.05E+12	4.62E+13	2.33E+13	147.0
La152	3.62E+11	5.50E+12	2.46E+12	175.3
La153	1.63E+10	5.87E+11	2.31E+11	189.2
La154	4.02E+08	2.54E+10	9.61E+09	193.8
La155	1.02E+07	7.18E+08	2.89E+08	194.4
Li8	1.26E+15	4.58E+17	1.36E+17	198.9
Mn66	7.52E+06	3.21E+08	1.24E+08	190.8
Mn67	6.04E+06	2.87E+08	1.04E+08	191.7
Mn68	9.87E+05	6.03E+07	2.32E+07	193.6
Mn69	1.26E+05	1.03E+07	3.44E+06	195.2
Mo101	3.45E+16	5.15E+16	3.83E+16	39.5
Mo102	2.93E+16	5.15E+16	3.44E+16	55.1
Mo103	2.18E+16	5.55E+16	3.00E+16	87.3
Mo104	1.42E+16	4.93E+16	2.28E+16	110.8
Mo105	8.02E+15	3.67E+16	1.52E+16	128.2
Mo106	3.68E+15	2.35E+16	8.56E+15	145.8
Mo107	1.17E+15	8.91E+15	3.17E+15	153.6
Mo108	3.35E+14	2.99E+15	1.06E+15	159.7
Mo109	1.04E+14	3.57E+14	2.00E+14	109.7
Mo110	7.24E+12	8.29E+13	3.89E+13	167.9
Mo111	6.99E+11	1.82E+13	6.88E+12	185.2
Mo112	6.28E+10	2.41E+12	8.61E+11	189.8
Mo113	3.27E+09	2.79E+11	9.74E+10	195.4
Mo114	4.57E+08	1.21E+10	5.01E+09	185.5
Mo115	1.18E+07	9.32E+08	3.76E+08	195.0

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Mo93	3.65E+03	3.65E+03	3.65E+03	0.0
Mo93m	1.30E+03	4.62E+04	1.63E+04	189.0
Mo99	3.94E+16	5.35E+16	4.30E+16	30.3
N16	1.13E+11	4.83E+16	4.50E+15	200.0
Na24	2.73E+06	2.73E+06	2.73E+06	0.0
Na24m	2.10E+06	2.10E+06	2.10E+06	0.0
Nb100	3.79E+16	5.03E+16	4.05E+16	28.1
Nb100m	1.97E+15	5.32E+15	3.05E+15	92.0
Nb101	3.33E+16	4.97E+16	3.66E+16	39.5
Nb102	1.86E+16	2.73E+16	2.07E+16	37.9
Nb102m	5.48E+15	1.32E+16	7.34E+15	83.0
Nb103	1.33E+16	2.97E+16	1.77E+16	76.1
Nb104	2.67E+15	8.37E+15	4.38E+15	103.3
Nb104m	2.11E+15	7.26E+15	3.47E+15	109.9
Nb105	1.86E+15	6.33E+15	3.23E+15	109.1
Nb106	1.34E+14	1.36E+15	5.52E+14	164.0
Nb107	2.35E+13	2.70E+14	1.17E+14	168.0
Nb108	1.53E+12	2.75E+13	1.26E+13	179.0
Nb109	1.26E+12	5.08E+12	3.09E+12	120.7
Nb110	1.72E+10	2.88E+11	1.39E+11	177.5
Nb111	3.14E+09	2.64E+11	9.19E+10	195.3
Nb112	4.04E+07	2.71E+09	9.43E+08	194.1
Nb113	4.41E+06	4.94E+08	1.70E+08	196.5
Nb91	1.73E+02	3.61E+02	2.75E+02	70.5
Nb92m	7.58E+03	3.00E+06	5.10E+05	199.0
Nb93m	1.33E+09	1.48E+11	4.45E+10	196.4
Nb94	1.46E+06	3.99E+07	1.28E+07	185.9
Nb94m	4.89E+09	5.85E+10	2.65E+10	169.2
Nb95	3.60E+16	4.87E+16	4.08E+16	30.0
Nb95m	3.89E+14	5.31E+14	4.42E+14	30.8
Nb96	1.44E+13	1.66E+14	6.42E+13	168.0
Nb97	3.63E+16	5.13E+16	4.04E+16	34.1
Nb97m	3.44E+16	4.83E+16	3.82E+16	33.6
Nb98	3.67E+16	4.80E+16	3.98E+16	26.6
Nb98m	1.94E+14	3.65E+14	2.50E+14	61.1
Nb99	2.32E+16	3.06E+16	2.51E+16	27.7
Nb99m	1.55E+16	2.21E+16	1.73E+16	35.4
Nd140	1.56E+02	1.53E+05	3.84E+04	199.6
Nd141	1.99E+07	7.99E+10	8.98E+09	199.9
Nd141m	4.04E+06	1.72E+10	1.91E+09	199.9

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Nd144	6.56E+01	6.76E+01	6.66E+01	3.1
Nd147	1.41E+16	1.93E+16	1.56E+16	31.2
Nd149	7.24E+15	1.28E+16	8.59E+15	55.5
Nd151	2.96E+15	7.19E+15	4.06E+15	83.3
Nd152	1.92E+15	5.03E+15	2.68E+15	89.7
Nd153	1.08E+15	3.15E+15	1.59E+15	98.0
Nd154	4.82E+14	1.85E+15	8.05E+14	117.1
Nd155	1.46E+14	7.66E+14	3.03E+14	136.1
Nd156	4.06E+13	2.91E+14	1.05E+14	151.1
Nd157	5.14E+12	7.99E+13	2.47E+13	175.8
Nd158	6.28E+11	1.45E+13	4.29E+12	183.4
Nd159	3.77E+10	1.41E+12	4.30E+11	189.6
Nd160	1.86E+09	7.95E+10	2.82E+10	190.9
Nd161	6.05E+07	2.91E+09	1.16E+09	191.8
Ne23	4.34E+05	7.57E+08	2.60E+08	199.8
Ni65	8.19E+05	3.50E+07	1.36E+07	190.8
Ni66	6.48E+08	3.25E+10	6.33E+09	192.2
Ni67	2.70E+09	4.98E+10	1.63E+10	179.4
Ni68	5.53E+09	8.15E+10	2.83E+10	174.6
Ni69	1.18E+10	1.23E+11	4.77E+10	164.9
Ni70	2.46E+10	2.24E+11	8.50E+10	160.4
Ni71	3.26E+10	3.48E+11	1.10E+11	165.7
Ni72	5.11E+10	3.48E+11	1.24E+11	148.8
Ni73	3.04E+10	2.12E+11	9.39E+10	149.7
Ni74	1.20E+10	9.40E+10	4.72E+10	154.6
Ni75	2.97E+09	3.11E+10	1.58E+10	165.1
Ni76	5.91E+08	8.70E+09	4.71E+09	174.5
Ni77	9.76E+07	1.32E+09	6.56E+08	172.4
Ni78	1.21E+07	2.05E+08	8.80E+07	177.6
Np235	1.21E+05	1.41E+08	2.75E+07	199.7
Np236	1.14E+03	1.12E+06	2.15E+05	199.6
Np236m	9.89E+08	3.73E+11	8.94E+10	198.9
Np237	1.97E+09	9.70E+10	2.08E+10	192.1
Np238	1.81E+14	4.44E+16	9.42E+15	198.4
Np239	1.37E+17	8.51E+17	4.23E+17	144.7
Np240	7.62E+12	5.91E+14	1.30E+14	194.9
Np240m	1.31E+13	1.03E+15	2.21E+14	195.0
Np241	5.21E+04	1.96E+08	2.12E+07	199.9
O19	7.39E+09	3.52E+15	3.27E+14	200.0
Pa229	3.11E+02	6.14E+06	3.54E+06	200.0

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Pa230	1.61E+04	2.13E+09	2.94E+08	200.0
Pa231	5.54E+04	2.58E+10	2.47E+09	200.0
Pa232	8.71E+08	7.67E+14	6.18E+13	200.0
Pa233	2.33E+09	4.48E+17	5.43E+16	200.0
Pa234	2.60E+07	6.58E+15	6.17E+14	200.0
Pa234m	9.39E+09	6.92E+15	6.48E+14	200.0
Pa235	7.79E+04	3.61E+10	9.70E+09	200.0
Pb207m	3.15E+07	1.37E+08	8.44E+07	125.3
Pb209	2.14E+03	3.16E+10	3.22E+09	200.0
Pb210	5.25E+02	6.22E+08	5.16E+07	200.0
Pb211	9.59E+03	2.60E+09	1.98E+08	200.0
Pb212	3.58E+06	8.74E+12	9.51E+11	200.0
Pb214	1.23E+03	4.28E+06	5.19E+05	199.9
Pd101	2.54E+03	1.82E+05	6.57E+04	194.5
Pd103	3.34E+08	5.33E+11	5.98E+10	199.7
Pd107	4.70E+08	4.09E+10	8.93E+09	195.5
Pd107m	1.61E+11	6.10E+13	1.44E+13	198.9
Pd109	7.53E+14	2.55E+16	6.82E+15	188.5
Pd109m	1.75E+11	2.49E+14	5.54E+13	199.7
Pd111	2.85E+14	3.93E+15	1.12E+15	173.0
Pd111m	7.32E+10	5.77E+13	9.44E+12	199.5
Pd112	1.62E+14	1.82E+15	5.53E+14	167.4
Pd113	1.39E+14	1.21E+15	3.97E+14	159.0
Pd114	1.04E+14	7.22E+14	2.68E+14	149.9
Pd115	9.30E+13	4.31E+14	2.02E+14	129.1
Pd116	7.28E+13	3.46E+14	1.67E+14	130.4
Pd117	6.85E+13	2.04E+14	1.23E+14	99.5
Pd118	2.48E+13	1.04E+14	6.42E+13	122.7
Pd119	4.14E+12	4.63E+13	2.31E+13	167.2
Pd120	6.52E+12	3.29E+13	1.85E+13	133.8
Pd121	3.69E+11	7.97E+12	2.99E+12	182.3
Pd122	4.94E+10	2.28E+12	8.08E+11	191.5
Pd123	6.62E+09	4.90E+11	1.73E+11	194.7
Pd124	1.77E+09	8.08E+10	2.95E+10	191.4
Pm144	2.87E+02	2.87E+04	5.64E+03	196.0
Pm145	3.55E+04	3.32E+07	6.40E+06	199.6
Pm146	1.54E+08	2.33E+11	4.47E+10	199.7
Pm147	5.56E+15	1.22E+16	9.04E+15	74.4
Pm148	2.11E+14	6.37E+15	2.29E+15	187.2
Pm148m	1.65E+14	3.77E+15	8.99E+14	183.2

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Pm149	7.35E+15	1.85E+16	9.67E+15	86.3
Pm150	1.58E+12	4.50E+14	5.71E+13	198.6
Pm151	2.97E+15	7.24E+15	4.08E+15	83.7
Pm152	1.93E+15	5.11E+15	2.71E+15	90.3
Pm152m	1.36E+13	1.42E+14	5.25E+13	165.0
Pm153	1.17E+15	3.60E+15	1.77E+15	101.8
Pm154	5.42E+14	2.18E+15	9.28E+14	120.3
Pm154m	4.31E+13	3.30E+14	1.23E+14	153.8
Pm155	2.77E+14	1.52E+15	5.82E+14	138.3
Pm156	1.22E+14	8.96E+14	3.09E+14	152.0
Pm157	4.15E+13	4.69E+14	1.42E+14	167.5
Pm158	1.01E+13	1.73E+14	4.67E+13	177.9
Pm159	1.83E+12	4.35E+13	1.15E+13	183.8
Pm160	1.98E+11	6.80E+12	1.77E+12	188.7
Pm161	1.78E+10	7.49E+11	2.02E+11	190.7
Pm162	4.12E+08	1.32E+10	6.18E+09	187.9
Pm163	2.15E+07	1.39E+09	5.46E+08	193.9
Po210	3.65E+02	6.19E+08	5.62E+07	200.0
Po211	3.39E+02	7.17E+06	1.17E+06	200.0
Po212	2.30E+06	5.60E+12	6.09E+11	200.0
Po213	2.03E+03	3.06E+10	3.12E+09	200.0
Po214	1.44E+04	2.07E+09	1.72E+08	200.0
Po215	9.59E+03	2.60E+09	1.97E+08	200.0
Po216	3.60E+06	8.74E+12	9.51E+11	200.0
Po218	1.23E+03	4.28E+06	5.19E+05	199.9
Pr139	8.42E+02	1.06E+07	1.32E+06	200.0
Pr140	4.86E+09	1.00E+12	2.18E+11	198.1
Pr142	1.26E+14	1.83E+16	3.19E+15	197.3
Pr142m	4.40E+13	6.37E+15	1.11E+15	197.3
Pr143	3.28E+16	4.73E+16	3.81E+16	36.0
Pr144	2.74E+16	3.93E+16	3.33E+16	35.7
Pr144m	2.67E+14	5.70E+14	3.65E+14	72.4
Pr145	2.24E+16	2.95E+16	2.52E+16	27.3
Pr146	1.80E+16	2.34E+16	1.99E+16	25.9
Pr147	1.39E+16	1.82E+16	1.52E+16	26.7
Pr148	1.01E+16	1.31E+16	1.10E+16	25.9
Pr148m	2.81E+14	1.82E+15	7.60E+14	146.6
Pr149	7.13E+15	1.06E+16	7.96E+15	39.3
Pr150	4.20E+15	7.73E+15	5.11E+15	59.1
Pr151	2.31E+15	4.68E+15	2.90E+15	67.9

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Pr152	8.61E+14	2.37E+15	1.26E+15	93.4
Pr153	2.69E+14	9.49E+14	4.75E+14	111.6
Pr154	3.94E+13	2.63E+14	1.09E+14	147.8
Pr155	5.65E+12	5.32E+13	2.35E+13	161.6
Pr156	4.53E+11	8.10E+12	3.46E+12	178.8
Pr157	3.04E+10	8.87E+11	4.01E+11	186.8
Pr158	1.41E+09	5.57E+10	2.54E+10	190.2
Pr159	4.62E+07	2.08E+09	1.04E+09	191.3
Pu236	1.74E+08	7.91E+10	1.85E+10	199.1
Pu237	2.57E+07	7.17E+10	9.10E+09	199.9
Pu237m	1.04E+07	2.33E+10	3.66E+09	199.8
Pu238	1.20E+12	3.31E+15	4.61E+14	199.9
Pu239	4.35E+12	1.12E+14	3.45E+13	185.0
Pu240	5.36E+11	4.39E+13	1.53E+13	195.2
Pu241	1.39E+12	3.54E+16	5.10E+15	200.0
Pu242	2.05E+05	1.03E+12	1.71E+11	200.0
Pu243	8.60E+08	7.87E+16	1.36E+16	200.0
Pu244	8.65E+02	1.12E+06	6.09E+05	199.7
Pu245	1.87E+04	3.81E+12	6.08E+11	200.0
Ra222	1.52E+03	2.06E+09	1.72E+08	200.0
Ra223	9.59E+03	2.60E+09	1.97E+08	200.0
Ra224	3.60E+06	8.74E+12	9.51E+11	200.0
Ra225	2.18E+03	3.35E+10	3.35E+09	200.0
Ra226	1.25E+03	4.28E+06	5.19E+05	199.9
Ra227	3.97E+04	1.20E+09	9.68E+07	200.0
Ra228	1.01E+09	1.35E+09	1.18E+09	28.1
Rb100	3.14E+11	2.46E+14	1.50E+14	199.5
Rb101	6.00E+09	4.87E+10	2.11E+10	156.2
Rb102	5.88E+07	5.00E+09	1.73E+09	195.3
Rb81	1.25E+03	3.78E+04	1.08E+04	187.3
Rb83	9.80E+06	2.73E+08	9.75E+07	186.1
Rb84	2.42E+08	3.68E+10	9.24E+09	197.4
Rb86	8.59E+12	8.83E+14	1.84E+14	196.1
Rb86m	1.11E+12	1.08E+14	2.27E+13	195.9
Rb87	4.18E+05	7.12E+06	2.34E+06	177.8
Rb88	1.02E+16	2.94E+16	2.01E+16	96.8
Rb89	1.29E+16	3.42E+16	2.59E+16	90.3
Rb90	1.14E+16	2.96E+16	2.52E+16	89.0
Rb90m	4.09E+15	9.94E+15	5.60E+15	83.3
Rb91	1.65E+16	3.44E+16	3.01E+16	70.1

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Rb92	1.60E+16	2.98E+16	2.61E+16	60.1
Rb93	1.27E+16	2.32E+16	1.96E+16	58.9
Rb94	7.33E+15	1.19E+16	9.76E+15	47.2
Rb95	3.59E+15	5.69E+15	4.75E+15	45.3
Rb96	7.46E+14	2.12E+15	1.39E+15	96.0
Rb97	1.35E+14	2.94E+14	2.18E+14	74.1
Rb98	1.95E+13	1.07E+14	5.47E+13	138.5
Rb99	4.65E+11	1.60E+13	5.78E+12	188.7
Rh101	6.10E+04	1.44E+06	4.03E+05	183.8
Rh101m	2.30E+04	6.15E+06	1.09E+06	198.5
Rh102	1.82E+09	3.09E+11	6.78E+10	197.7
Rh102m	4.86E+08	8.43E+10	1.82E+10	197.7
Rh103m	2.22E+16	5.98E+16	3.13E+16	91.8
Rh104	4.40E+14	6.19E+16	1.51E+16	197.2
Rh104m	3.48E+13	4.70E+15	1.15E+15	197.1
Rh105	8.97E+15	4.24E+16	1.82E+16	130.2
Rh105m	2.64E+15	1.39E+16	5.46E+15	136.1
Rh106	3.38E+15	4.74E+16	1.32E+16	173.3
Rh106m	1.65E+11	7.05E+14	7.66E+13	199.9
Rh107	2.69E+15	3.17E+16	9.80E+15	168.7
Rh108	1.29E+15	2.23E+16	6.41E+15	178.2
Rh108m	5.20E+12	1.38E+14	4.26E+13	185.5
Rh109	7.47E+14	1.54E+16	4.32E+15	181.5
Rh110	4.58E+12	2.29E+14	5.63E+13	192.2
Rh110m	4.61E+14	7.53E+15	2.13E+15	176.9
Rh111	2.83E+14	3.82E+15	1.09E+15	172.3
Rh112	1.61E+14	1.64E+15	5.18E+14	164.2
Rh113	1.32E+14	1.02E+15	3.55E+14	154.3
Rh114	7.16E+13	4.87E+14	1.91E+14	148.7
Rh115	3.66E+13	2.02E+14	9.82E+13	138.8
Rh116	1.03E+13	7.50E+13	3.95E+13	151.9
Rh117	3.95E+12	2.67E+13	1.45E+13	148.5
Rh118	4.19E+11	1.21E+13	4.80E+12	186.5
Rh119	4.59E+10	3.09E+12	1.14E+12	194.1
Rh120	2.05E+10	6.92E+11	2.48E+11	188.5
Rh121	1.06E+09	1.13E+11	3.96E+10	196.3
Rh122	1.22E+08	1.42E+10	5.04E+09	196.6
Rn217	1.39E+06	2.19E+06	1.79E+06	44.6
Rn218	1.52E+03	2.06E+09	1.72E+08	200.0
Rn219	9.59E+03	2.60E+09	1.97E+08	200.0

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Rn220	3.60E+06	8.74E+12	9.51E+11	200.0
Rn222	1.23E+03	4.28E+06	5.19E+05	199.9
Ru103	2.24E+16	6.05E+16	3.16E+16	91.9
Ru105	9.31E+15	4.90E+16	1.93E+16	136.1
Ru106	3.38E+15	4.07E+16	1.25E+16	169.3
Ru107	2.67E+15	3.13E+16	9.66E+15	168.5
Ru108	1.28E+15	2.22E+16	6.36E+15	178.1
Ru109	7.33E+14	1.46E+16	4.13E+15	180.9
Ru110	4.56E+14	7.30E+15	2.07E+15	176.5
Ru111	2.62E+14	3.10E+15	9.14E+14	168.8
Ru112	1.25E+14	1.05E+15	3.62E+14	157.5
Ru113	6.29E+13	4.68E+14	1.72E+14	152.6
Ru114	2.10E+13	1.58E+14	6.36E+13	153.2
Ru115	3.05E+12	3.40E+13	1.51E+13	167.1
Ru116	3.47E+11	5.77E+12	2.90E+12	177.3
Ru117	3.15E+10	1.03E+12	4.16E+11	188.1
Ru118	2.64E+09	2.22E+11	7.91E+10	195.3
Ru119	2.02E+08	2.21E+10	7.85E+09	196.4
Ru120	2.02E+07	2.14E+09	7.56E+08	196.3
Sb118	7.83E+02	3.94E+04	8.89E+03	192.2
Sb118m	1.24E+03	6.42E+04	1.42E+04	192.5
Sb119	2.93E+05	1.71E+07	3.62E+06	193.3
Sb120	2.38E+07	4.04E+09	1.02E+09	197.7
Sb120m	2.14E+07	1.89E+09	5.20E+08	195.5
Sb122	2.39E+12	1.73E+14	3.74E+13	194.5
Sb122m	1.62E+11	1.11E+13	2.42E+12	194.2
Sb124	1.69E+12	2.15E+14	4.09E+13	196.9
Sb124m	7.95E+10	2.97E+12	7.44E+11	189.6
Sb125	1.26E+14	9.19E+14	3.80E+14	151.7
Sb126	5.78E+12	1.99E+13	9.54E+12	110.0
Sb126m	9.10E+12	1.55E+13	1.18E+13	51.8
Sb127	1.23E+15	4.11E+15	2.03E+15	107.7
Sb128	1.62E+14	7.04E+14	3.13E+14	125.1
Sb128m	2.42E+15	5.70E+15	3.29E+15	80.7
Sb129	4.02E+15	1.16E+16	6.16E+15	97.1
Sb129m	7.52E+13	1.39E+14	9.15E+13	59.3
Sb130	4.95E+15	7.78E+15	5.74E+15	44.5
Sb130m	6.75E+15	9.67E+15	7.47E+15	35.5
Sb131	1.65E+16	2.24E+16	1.81E+16	30.5
Sb132	1.20E+16	1.82E+16	1.33E+16	41.4

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Sb132m	5.84E+15	7.61E+15	6.57E+15	26.2
Sb133	1.19E+16	1.45E+16	1.31E+16	19.6
Sb134	2.06E+15	3.09E+15	2.52E+15	40.2
Sb134m	2.02E+15	3.00E+15	2.50E+15	38.9
Sb135	8.03E+14	1.71E+15	1.16E+15	72.4
Sb136	6.91E+13	2.71E+14	1.41E+14	118.8
Sb137	1.15E+13	4.31E+14	2.76E+14	189.6
Sb138	2.80E+11	4.01E+12	1.58E+12	173.9
Sb139	1.15E+10	2.87E+11	1.08E+11	184.5
Se75	4.31E+04	5.57E+07	7.49E+06	199.7
Se77m	1.55E+11	3.30E+12	4.70E+11	182.0
Se79	1.43E+09	3.42E+10	8.83E+09	183.9
Se79m	2.71E+14	7.41E+14	3.51E+14	92.8
Se81	1.15E+15	2.22E+15	1.38E+15	63.5
Se81m	8.60E+13	2.28E+14	1.23E+14	90.5
Se83	1.85E+15	4.82E+15	2.90E+15	89.1
Se83m	2.18E+14	5.28E+14	2.87E+14	83.1
Se84	3.15E+15	8.21E+15	5.65E+15	88.9
Se85	3.08E+15	7.19E+15	5.87E+15	80.1
Se86	2.53E+15	8.44E+15	6.76E+15	107.7
Se87	1.32E+15	4.92E+15	3.81E+15	115.2
Se88	4.64E+14	2.44E+15	1.74E+15	136.1
Se89	1.05E+14	5.84E+14	3.44E+14	138.9
Se90	3.90E+13	1.62E+14	9.43E+13	122.2
Se91	1.83E+12	2.03E+13	8.72E+12	166.9
Se92	1.44E+11	1.99E+12	8.04E+11	173.1
Se93	1.46E+10	1.02E+11	5.15E+10	149.7
Se94	2.96E+08	6.11E+09	2.49E+09	181.5
Sm145	8.34E+03	1.76E+07	3.66E+06	199.8
Sm146	1.25E+02	1.93E+04	4.92E+03	197.4
Sm147	3.70E+04	7.31E+05	2.75E+05	180.7
Sm151	1.17E+13	1.76E+14	6.30E+13	174.9
Sm153	1.28E+15	3.68E+16	8.89E+15	186.5
Sm155	3.03E+14	2.40E+15	7.60E+14	155.1
Sm156	1.67E+14	1.19E+15	4.17E+14	150.9
Sm157	8.27E+13	8.10E+14	2.54E+14	163.0
Sm158	4.30E+13	4.78E+14	1.42E+14	167.0
Sm159	1.44E+13	2.13E+14	5.91E+13	174.6
Sm160	4.28E+12	7.67E+13	2.03E+13	178.9
Sm161	7.92E+11	1.98E+13	4.93E+12	184.6

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Sm162	8.10E+10	1.51E+12	4.72E+11	179.7
Sm163	6.23E+09	1.95E+11	6.14E+10	187.6
Sm164	4.23E+08	1.61E+10	7.28E+09	189.7
Sm165	2.74E+07	1.83E+09	7.12E+08	194.1
Sn113	1.07E+03	3.70E+05	1.24E+05	198.8
Sn113m	1.45E+03	2.90E+05	8.97E+04	198.0
Sn117m	2.22E+11	1.58E+12	5.64E+11	150.5
Sn119m	3.33E+12	8.72E+12	5.42E+12	89.6
Sn121	9.43E+13	2.66E+14	1.71E+14	95.2
Sn121m	3.32E+11	4.89E+12	1.75E+12	174.6
Sn123	1.24E+13	8.82E+13	3.38E+13	150.6
Sn123m	1.08E+14	3.62E+14	1.97E+14	108.0
Sn125	1.14E+14	4.44E+14	2.14E+14	118.1
Sn125m	1.56E+14	4.90E+14	2.66E+14	103.6
Sn126	2.83E+09	9.57E+10	2.63E+10	188.5
Sn127	7.54E+14	2.51E+15	1.23E+15	107.6
Sn127m	4.26E+14	1.44E+15	7.23E+14	108.8
Sn128	2.35E+15	5.24E+15	3.12E+15	76.2
Sn128m	1.12E+15	2.58E+15	1.51E+15	78.6
Sn129	2.12E+15	6.24E+15	3.38E+15	98.6
Sn129m	1.33E+15	2.44E+15	1.61E+15	59.3
Sn130	3.75E+15	4.59E+15	4.06E+15	20.2
Sn130m	3.71E+15	4.71E+15	4.08E+15	23.8
Sn131	2.81E+15	3.70E+15	3.24E+15	27.3
Sn131m	2.69E+15	3.54E+15	3.11E+15	27.2
Sn132	3.05E+15	4.49E+15	3.93E+15	38.4
Sn133	5.36E+14	1.13E+15	7.77E+14	71.5
Sn134	7.05E+13	2.45E+14	1.41E+14	110.8
Sn135	3.73E+12	2.04E+13	9.32E+12	138.1
Sn136	1.04E+11	1.33E+12	5.22E+11	171.1
Sn137	7.14E+09	1.32E+11	8.71E+10	179.4
Sr100	1.14E+14	4.54E+14	2.66E+14	119.8
Sr101	1.15E+13	6.33E+13	3.37E+13	138.7
Sr102	7.87E+11	6.44E+12	2.72E+12	156.4
Sr103	1.48E+10	1.56E+11	6.66E+10	165.3
Sr104	9.30E+08	1.91E+10	6.98E+09	181.4
Sr105	3.68E+07	3.61E+09	1.25E+09	196.0
Sr83	2.21E+02	1.94E+03	9.67E+02	159.1
Sr85	6.54E+06	8.09E+08	1.54E+08	196.8
Sr85m	3.60E+06	2.43E+08	4.84E+07	194.2

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Sr87m	9.32E+09	1.89E+13	2.86E+12	199.8
Sr89	1.31E+16	3.46E+16	2.61E+16	90.4
Sr90	1.51E+15	1.72E+16	6.82E+15	167.8
Sr91	1.89E+16	3.89E+16	3.24E+16	69.3
Sr92	2.27E+16	4.06E+16	3.40E+16	56.4
Sr93	2.81E+16	4.40E+16	3.69E+16	44.2
Sr94	2.85E+16	4.02E+16	3.56E+16	33.9
Sr95	2.54E+16	3.37E+16	3.08E+16	28.0
Sr96	1.90E+16	2.59E+16	2.24E+16	31.0
Sr97	8.37E+15	1.26E+16	1.04E+16	40.1
Sr98	3.41E+15	6.35E+15	4.94E+15	60.2
Sr99	5.83E+14	1.63E+15	1.03E+15	94.7
Tb155	1.85E+03	4.01E+05	5.23E+04	198.2
Tb156	9.95E+04	5.92E+06	1.23E+06	193.4
Tb156m	9.53E+03	5.46E+05	1.11E+05	193.1
Tb157	2.66E+04	1.86E+06	4.01E+05	194.4
Tb158	7.57E+05	3.49E+07	8.60E+06	191.5
Tb158m	9.43E+06	3.80E+09	5.13E+08	199.0
Tb160	7.01E+11	5.53E+14	8.38E+13	199.5
Tb161	3.19E+12	2.52E+14	4.07E+13	195.0
Tb162	1.35E+12	2.89E+13	7.83E+12	182.2
Tb163	5.85E+11	1.61E+13	3.94E+12	186.0
Tb164	2.07E+11	7.05E+12	1.67E+12	188.6
Tb165	8.21E+10	2.94E+12	7.02E+11	189.1
Tb166	3.37E+10	1.07E+12	2.89E+11	187.8
Tb167	6.69E+09	3.74E+11	9.30E+10	193.0
Tb168	1.35E+09	1.28E+11	3.28E+10	195.8
Tb169	3.42E+08	2.88E+10	9.73E+09	195.3
Tb170	6.94E+07	5.10E+09	2.39E+09	194.6
Tb171	1.48E+07	1.37E+09	5.37E+08	195.7
Tc100	1.05E+15	4.20E+16	1.18E+16	190.2
Tc101	3.45E+16	5.16E+16	3.83E+16	39.6
Tc102	2.93E+16	5.17E+16	3.45E+16	55.2
Tc102m	5.26E+13	1.91E+14	9.71E+13	113.6
Tc103	2.24E+16	5.70E+16	3.08E+16	87.2
Tc104	1.50E+16	5.29E+16	2.43E+16	111.5
Tc105	9.26E+15	4.74E+16	1.87E+16	134.6
Tc106	4.92E+15	3.83E+16	1.30E+16	154.5
Tc107	2.46E+15	2.56E+16	7.93E+15	165.0
Tc108	9.17E+14	1.25E+16	3.54E+15	172.7

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Tc109	4.61E+14	5.70E+15	1.63E+15	170.1
Tc110	1.26E+14	9.80E+14	3.47E+14	154.5
Tc111	3.41E+13	1.62E+14	8.39E+13	130.5
Tc112	4.97E+12	4.07E+13	1.91E+13	156.4
Tc113	1.00E+12	1.18E+13	4.87E+12	168.8
Tc114	2.20E+11	2.70E+12	1.27E+12	169.9
Tc115	9.35E+09	4.35E+11	2.13E+11	191.6
Tc116	6.36E+08	4.47E+10	1.76E+10	194.4
Tc117	2.34E+07	1.88E+09	6.82E+08	195.1
Tc118	3.97E+06	4.55E+08	1.57E+08	196.5
Tc97m	1.03E+06	5.09E+07	1.35E+07	192.1
Tc98	1.04E+04	9.50E+05	2.12E+05	195.7
Tc99	2.51E+11	2.80E+12	1.11E+12	167.1
Tc99m	3.50E+16	4.70E+16	3.83E+16	29.2
Te121	3.08E+05	8.19E+08	9.37E+07	199.8
Te121m	1.53E+05	3.98E+08	4.54E+07	199.8
Te123m	3.07E+09	1.03E+13	1.37E+12	199.9
Te125m	2.62E+13	2.12E+14	8.64E+13	156.0
Te127	1.22E+15	3.95E+15	1.95E+15	105.3
Te127m	1.40E+14	5.90E+14	2.45E+14	123.2
Te129	3.84E+15	1.12E+16	5.90E+15	98.2
Te129m	6.72E+14	1.98E+15	1.02E+15	98.5
Te131	1.72E+16	2.53E+16	1.95E+16	37.8
Te131m	2.75E+15	8.60E+15	4.34E+15	103.0
Te132	2.87E+16	4.28E+16	3.24E+16	39.3
Te133	2.16E+16	2.93E+16	2.37E+16	30.3
Te133m	2.11E+16	2.89E+16	2.42E+16	31.2
Te134	3.84E+16	4.79E+16	4.31E+16	21.8
Te135	1.99E+16	2.44E+16	2.14E+16	20.1
Te136	7.42E+15	9.94E+15	8.51E+15	29.0
Te137	1.79E+15	3.30E+15	2.56E+15	59.6
Te138	3.35E+14	8.69E+14	5.36E+14	88.7
Te139	3.90E+13	1.41E+14	7.53E+13	113.5
Te140	4.91E+12	1.01E+14	6.82E+13	181.4
Te141	2.46E+11	4.98E+12	1.84E+12	181.1
Te142	1.29E+10	9.18E+10	3.94E+10	150.7
Th226	1.52E+03	2.06E+09	1.72E+08	200.0
Th227	9.94E+03	2.59E+09	1.96E+08	200.0
Th228	3.69E+06	8.71E+12	9.47E+11	200.0
Th229	1.60E+03	5.04E+09	4.25E+08	200.0

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Th230	2.47E+05	8.07E+08	8.79E+07	199.9
Th231	3.72E+09	7.88E+14	6.47E+13	200.0
Th232	4.23E+01	1.64E+09	1.09E+09	200.0
Th233	5.50E+08	4.62E+17	5.56E+16	200.0
Th234	5.50E+09	1.10E+10	9.58E+09	66.7
Tl207	9.56E+03	2.59E+09	1.97E+08	200.0
Tl208	1.29E+06	3.14E+12	3.42E+11	200.0
Tl209	6.65E+02	6.87E+08	1.25E+08	200.0
Tm167	5.64E+02	3.53E+04	9.59E+03	193.7
Tm168	1.40E+03	1.07E+07	1.20E+06	199.9
Tm170	1.95E+08	3.43E+12	2.66E+11	200.0
Tm171	1.36E+08	2.14E+11	1.92E+10	199.7
Tm172	1.53E+08	2.30E+11	2.30E+10	199.7
U230	1.22E+03	1.86E+08	2.46E+07	200.0
U231	4.62E+03	1.22E+10	9.13E+08	200.0
U232	2.19E+07	2.08E+13	1.95E+12	200.0
U233	1.45E+05	1.09E+13	8.55E+11	200.0
U234	6.10E+09	1.94E+12	3.59E+11	198.7
U235	3.92E+05	1.46E+10	6.26E+09	200.0
U235m	2.42E+14	3.17E+16	4.44E+15	197.0
U236	8.20E+09	4.84E+10	2.32E+10	142.0
U237	4.50E+15	5.28E+16	1.65E+16	168.6
U238	5.51E+09	1.10E+10	9.58E+09	66.6
U239	1.37E+17	8.52E+17	4.24E+17	144.6
U240	8.64E+02	1.12E+06	6.08E+05	199.7
Xe125	4.23E+03	1.77E+05	6.50E+04	190.7
Xe125m	1.12E+03	4.69E+04	1.72E+04	190.7
Xe127	3.11E+06	2.58E+10	3.26E+09	200.0
Xe127m	5.38E+05	4.34E+09	5.50E+08	200.0
Xe129m	6.34E+09	2.35E+13	3.39E+12	199.9
Xe131m	2.12E+14	3.86E+14	2.53E+14	58.2
Xe133	4.13E+16	5.94E+16	4.63E+16	36.0
Xe133m	1.16E+15	2.05E+15	1.41E+15	55.4
Xe134m	2.29E+14	1.22E+15	5.82E+14	136.9
Xe135	7.08E+15	5.95E+16	3.28E+16	157.5
Xe135m	7.92E+15	1.43E+16	9.78E+15	57.6
Xe137	3.83E+16	5.47E+16	4.30E+16	35.5
Xe138	3.62E+16	4.82E+16	4.11E+16	28.4
Xe139	2.69E+16	3.31E+16	3.07E+16	20.7
Xe140	1.77E+16	2.32E+16	2.08E+16	26.6

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Xe141	5.85E+15	8.91E+15	7.57E+15	41.5
Xe142	1.92E+15	3.63E+15	2.80E+15	61.6
Xe143	2.12E+14	5.16E+14	3.33E+14	83.6
Xe144	3.51E+13	1.35E+14	6.87E+13	117.1
Xe145	6.08E+11	1.25E+13	5.10E+12	181.4
Xe146	7.26E+10	9.01E+11	3.64E+11	170.2
Xe147	6.97E+09	2.49E+10	1.45E+10	112.4
Y100	3.69E+15	6.46E+15	4.90E+15	54.6
Y101	1.21E+15	2.77E+15	1.93E+15	78.6
Y102	2.81E+14	1.72E+15	1.27E+15	143.9
Y103	1.75E+13	1.35E+14	5.95E+13	154.1
Y104	2.70E+12	1.73E+13	7.76E+12	145.9
Y105	3.66E+10	3.18E+12	1.12E+12	195.5
Y106	1.25E+08	8.28E+09	3.27E+09	194.0
Y107	1.05E+07	1.19E+09	4.05E+08	196.5
Y108	5.56E+04	1.58E+07	2.83E+06	198.6
Y87	6.44E+05	1.60E+07	6.03E+06	184.6
Y87m	3.35E+03	3.61E+04	1.62E+04	166.0
Y88	5.94E+08	8.51E+10	2.20E+10	197.2
Y89m	1.30E+12	3.39E+12	2.54E+12	89.0
Y90	1.54E+15	1.77E+16	7.07E+15	168.0
Y90m	1.20E+11	2.94E+12	9.91E+11	184.3
Y91	1.90E+16	3.88E+16	3.26E+16	68.5
Y91m	1.11E+16	2.29E+16	1.91E+16	69.3
Y92	2.30E+16	4.13E+16	3.45E+16	57.1
Y93	2.91E+16	4.64E+16	3.79E+16	45.9
Y93m	1.01E+16	1.62E+16	1.31E+16	46.6
Y94	3.28E+16	4.76E+16	3.92E+16	36.9
Y95	3.50E+16	4.68E+16	3.98E+16	28.9
Y96	2.09E+16	2.74E+16	2.42E+16	26.8
Y96m	1.13E+16	1.88E+16	1.35E+16	50.2
Y97	1.80E+16	2.08E+16	1.94E+16	14.4
Y97m	1.09E+16	1.39E+16	1.20E+16	24.4
Y98	1.20E+16	1.46E+16	1.32E+16	20.0
Y98m	7.53E+15	1.06E+16	8.19E+15	33.9
Y99	1.23E+16	1.56E+16	1.40E+16	23.5
Yb169	3.54E+02	1.16E+07	1.12E+06	200.0
Yb169m	7.10E+04	7.10E+04	7.10E+04	0.0
Zn69	1.36E+10	1.61E+11	5.69E+10	168.9
Zn69m	4.97E+07	1.85E+09	3.23E+08	189.5

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Zn71	6.89E+10	9.09E+11	2.65E+11	171.8
Zn71m	5.92E+09	1.26E+11	3.04E+10	182.1
Zn72	2.32E+11	2.34E+12	7.07E+11	163.9
Zn73	7.52E+11	5.69E+12	1.75E+12	153.3
Zn74	2.20E+12	1.21E+13	4.01E+12	138.5
Zn75	4.73E+12	2.56E+13	8.58E+12	137.7
Zn76	6.15E+12	3.26E+13	1.61E+13	136.6
Zn77	4.97E+12	2.80E+13	1.93E+13	139.6
Zn78	4.01E+12	2.75E+13	1.89E+13	149.1
Zn79	1.31E+12	1.25E+13	8.49E+12	162.1
Zn80	3.46E+11	2.54E+12	1.59E+12	152.1
Zn81	6.13E+09	3.81E+11	1.54E+11	193.7
Zn82	9.21E+09	1.34E+11	7.25E+10	174.3
Zn83	3.25E+08	9.08E+09	3.37E+09	186.2
Zr100	3.50E+16	4.52E+16	3.75E+16	25.3
Zr101	1.91E+16	2.53E+16	2.10E+16	28.0
Zr102	1.12E+16	1.50E+16	1.33E+16	28.8
Zr103	3.07E+15	5.24E+15	3.95E+15	52.4
Zr104	5.49E+14	1.39E+15	9.11E+14	87.1
Zr105	1.49E+14	6.89E+14	5.14E+14	128.8
Zr106	2.68E+11	1.15E+13	3.27E+12	190.9
Zr107	1.54E+10	4.63E+11	2.20E+11	187.1
Zr108	6.82E+08	2.90E+10	1.32E+10	190.8
Zr109	1.12E+09	4.45E+09	2.77E+09	119.4
Zr110	9.76E+06	1.80E+08	8.90E+07	179.4
Zr88	8.06E+02	2.73E+04	7.94E+03	188.5
Zr89	1.62E+07	9.93E+09	1.47E+09	199.3
Zr89m	5.82E+05	9.38E+08	1.12E+08	199.8
Zr90m	5.84E+09	1.06E+12	2.33E+11	197.8
Zr93	3.36E+10	4.42E+11	1.78E+11	171.8
Zr95	3.60E+16	4.88E+16	4.08E+16	30.2
Zr97	3.61E+16	5.07E+16	4.02E+16	33.5
Zr98	3.61E+16	4.69E+16	3.91E+16	26.2
Zr99	3.57E+16	4.65E+16	3.94E+16	26.3
Total	4.13E+18	7.91E+18	5.25E+18	62.8

Table C.6. Minimum, Maximum and Average Radionuclide Activity for Radionuclides with Potential Mobility at Time 0 (End of Operation) for Small- to Medium-Sized Advanced Reactors

	Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Noble Gases	Kr100	6.71E+07	3.75E+10	8.22E+09	199.3
	Kr79	1.87E+04	4.64E+06	9.02E+05	198.4
	Kr79m	9.34E+03	2.35E+06	5.63E+05	198.4
	Kr81	6.65E+03	3.57E+05	1.17E+05	192.7
	Kr81m	1.54E+08	4.65E+10	1.22E+10	198.7
	Kr83m	1.73E+15	3.58E+16	1.08E+16	181.6
	Kr85	4.08E+14	2.18E+15	1.20E+15	137.0
	Kr85m	3.87E+15	7.81E+16	2.29E+16	181.1
	Kr87	7.48E+15	1.45E+17	4.22E+16	180.4
	Kr88	1.01E+16	1.95E+17	5.59E+16	180.4
	Kr89	1.26E+16	2.39E+17	6.76E+16	180.0
	Kr90	1.24E+16	2.37E+17	6.64E+16	180.1
	Kr91	8.06E+15	1.63E+17	4.64E+16	181.2
	Kr92	4.23E+15	8.31E+16	2.46E+16	180.6
	Kr93	1.22E+15	2.41E+16	7.88E+15	180.7
	Kr94	3.42E+14	6.31E+15	2.33E+15	179.4
	Kr95	2.92E+13	9.50E+14	3.07E+14	188.1
	Kr96	6.12E+13	1.67E+15	4.35E+14	185.8
	Kr97	1.10E+12	1.01E+14	2.59E+13	195.7
	Kr98	7.81E+10	5.46E+13	1.16E+13	199.4
	Kr99	1.86E+08	1.19E+11	3.02E+10	199.4
	Rn217	1.10E+06	1.10E+06	1.10E+06	0.0
	Rn218	1.13E+04	3.83E+08	3.50E+07	200.0
	Rn219	1.73E+05	9.97E+09	1.66E+09	200.0
	Rn220	2.47E+07	2.05E+13	1.71E+12	200.0
	Rn222	2.80E+05	6.75E+05	4.78E+05	82.8
	Xe125	7.80E+04	2.17E+07	5.57E+06	198.6
	Xe125m	2.06E+04	5.75E+06	2.88E+06	198.6
	Xe127	2.63E+07	3.80E+11	4.83E+10	200.0
	Xe127m	4.41E+06	6.92E+10	9.03E+09	200.0
	Xe129m	5.74E+10	4.49E+13	9.09E+12	199.5
	Xe131m	1.36E+14	3.50E+15	1.13E+15	185.0
	Xe133	2.50E+16	6.77E+17	2.07E+17	185.8
	Xe133m	7.65E+14	2.14E+16	6.72E+15	186.2
	Xe134m	2.95E+14	1.39E+16	3.74E+15	191.7
	Xe135	7.01E+15	3.44E+17	8.74E+16	192.0
	Xe135m	5.25E+15	1.58E+17	4.70E+16	187.1
	Xe137	2.30E+16	6.09E+17	1.85E+17	185.4

	Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
	Xe138	2.22E+16	5.68E+17	1.70E+17	185.0
	Xe139	1.69E+16	4.05E+17	1.19E+17	183.9
	Xe140	1.20E+16	2.59E+17	7.57E+16	182.2
	Xe141	4.90E+15	9.33E+16	2.87E+16	180.0
	Xe142	1.79E+15	3.23E+16	1.04E+16	179.0
	Xe143	1.70E+14	3.56E+15	1.36E+15	181.7
	Xe144	2.56E+13	7.85E+14	2.87E+14	187.4
	Xe145	5.86E+11	7.53E+13	2.38E+13	196.9
	Xe146	5.43E+10	5.36E+12	1.59E+12	196.0
	Xe147	7.07E+09	1.40E+11	5.56E+10	180.7
	Cs131	3.62E+06	1.27E+09	3.31E+08	198.9
	Cs132	1.93E+10	2.46E+13	4.72E+12	199.7
	Cs134	1.14E+15	8.37E+16	2.56E+16	194.6
	Cs134m	1.61E+14	4.31E+16	9.74E+15	198.5
	Cs135	3.27E+10	3.33E+11	1.13E+11	164.2
	Cs135m	7.79E+12	9.82E+14	2.49E+14	196.8
	Cs136	8.85E+14	5.47E+16	1.33E+16	193.6
	Cs136m	7.44E+13	3.61E+15	1.09E+15	191.9
	Cs137	4.48E+15	2.29E+16	1.38E+16	134.5
	Cs138	2.43E+16	6.26E+17	1.89E+17	185.1
	Cs138m	1.15E+15	4.47E+16	1.28E+16	190.0
	Cs139	2.28E+16	5.80E+17	1.74E+17	184.9
	Cs140	1.99E+16	4.77E+17	1.43E+17	184.0
	Cs141	1.56E+16	3.76E+17	1.14E+17	184.1
	Cs142	9.72E+15	2.24E+17	6.65E+16	183.4
	Cs143	5.41E+15	1.13E+17	3.41E+16	181.8
	Cs144	1.78E+15	3.46E+16	1.14E+16	180.4
	Cs145	3.24E+14	6.80E+15	2.53E+15	181.8
	Cs146	3.44E+13	8.80E+14	3.70E+14	185.0
	Cs147	8.74E+12	1.55E+14	5.62E+13	178.6
	Cs148	9.05E+10	1.35E+13	4.04E+12	197.3
	Cs149	2.29E+09	2.48E+11	7.95E+10	196.3
	Cs150	5.94E+07	2.62E+10	6.89E+09	199.1
	Cs151	6.85E+06	4.59E+09	1.15E+09	199.4
	H3	1.80E+13	1.85E+16	2.92E+15	199.6
	I123	6.01E+04	7.56E+06	2.04E+06	196.8
	I125	1.45E+04	4.81E+07	7.28E+06	199.9
	I126	6.47E+08	1.39E+12	2.13E+11	199.8
	I128	7.90E+13	1.15E+16	2.84E+15	197.3
	I129	1.18E+09	6.47E+09	3.49E+09	138.3
Volatile	I130	1.65E+14	1.73E+16	5.11E+15	196.2

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
I130m	1.05E+14	1.11E+16	3.29E+15	196.3
I131	1.24E+16	3.45E+17	1.05E+17	186.1
I132	1.83E+16	5.00E+17	1.52E+17	185.9
I132m	1.63E+14	9.18E+15	2.37E+15	193.0
I133	2.59E+16	6.70E+17	2.05E+17	185.1
I133m	2.20E+15	6.72E+16	2.01E+16	187.3
I134	2.91E+16	7.55E+17	2.28E+17	185.1
I134m	2.27E+15	9.16E+16	2.53E+16	190.3
I135	2.46E+16	6.47E+17	1.97E+17	185.4
I136	1.07E+16	2.42E+17	7.46E+16	183.0
I136m	5.38E+15	1.65E+17	4.82E+16	187.4
I137	1.19E+16	2.92E+17	8.83E+16	184.3
I138	6.62E+15	1.52E+17	4.76E+16	183.3
I139	3.12E+15	5.86E+16	1.81E+16	179.8
I140	6.43E+14	1.23E+16	4.30E+15	180.1
I141	1.58E+14	2.44E+15	9.10E+14	175.7
I142	2.49E+13	4.92E+14	1.77E+14	180.7
I143	1.47E+11	4.03E+13	1.06E+13	198.5
I144	8.53E+09	6.16E+11	2.06E+11	194.5
Te121	1.35E+06	1.26E+10	1.67E+09	200.0
Te121m	6.86E+05	5.48E+09	7.02E+08	199.9
Te123m	3.75E+10	1.97E+13	3.24E+12	199.2
Te125m	5.13E+13	5.41E+14	2.26E+14	165.3
Te127	1.04E+15	3.29E+16	1.00E+16	187.8
Te127m	1.47E+14	2.85E+15	8.10E+14	180.4
Te129	3.17E+15	1.06E+17	3.09E+16	188.4
Te129m	5.53E+14	1.79E+16	5.23E+15	188.0
Te131	1.09E+16	2.88E+17	8.86E+16	185.4
Te131m	2.15E+15	7.63E+16	2.20E+16	189.0
Te132	1.80E+16	4.74E+17	1.46E+17	185.4
Te133	1.34E+16	3.37E+17	1.03E+17	184.7
Te133m	1.28E+16	3.35E+17	9.97E+16	185.3
Te134	2.40E+16	5.80E+17	1.72E+17	184.1
Te135	1.25E+16	2.91E+17	8.80E+16	183.5
Te136	5.51E+15	1.03E+17	3.20E+16	179.6
Te137	1.68E+15	3.00E+16	9.70E+15	178.9
Te138	2.70E+14	5.09E+15	2.04E+15	179.8
Te139	2.89E+13	8.28E+14	3.21E+14	186.5
Te140	6.18E+12	5.88E+14	1.47E+14	195.8
Te141	1.72E+11	2.98E+13	7.90E+12	197.7
Te142	9.34E+09	5.43E+11	1.69E+11	193.2

	Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
	Ba131	3.02E+04	6.14E+06	1.79E+06	198.0
	Ba133	9.39E+07	4.06E+10	6.45E+09	199.1
	Ba135m	1.15E+12	1.83E+14	5.75E+13	197.5
	Ba136m	9.83E+13	6.06E+15	1.48E+15	193.6
	Ba137m	4.24E+15	2.19E+16	1.32E+16	135.0
	Ba139	2.32E+16	5.94E+17	1.80E+17	185.0
	Ba140	2.24E+16	5.67E+17	1.71E+17	184.8
	Ba141	2.12E+16	5.29E+17	1.60E+17	184.6
	Ba142	1.99E+16	4.96E+17	1.49E+17	184.6
	Ba143	1.82E+16	4.34E+17	1.28E+17	183.9
	Ba144	1.40E+16	3.21E+17	9.36E+16	183.3
	Ba145	6.97E+15	1.41E+17	4.22E+16	181.1
	Ba146	3.49E+15	5.98E+16	1.82E+16	177.9
	Ba147	9.56E+14	1.55E+16	4.92E+15	176.7
	Ba148	9.25E+13	1.97E+15	8.07E+14	182.0
	Ba149	5.27E+12	2.86E+14	1.02E+14	192.8
	Ba150	3.05E+11	3.06E+13	9.77E+12	196.1
	Ba151	1.26E+10	5.15E+12	1.34E+12	199.0
	Ba152	2.71E+08	8.18E+10	2.28E+10	198.7
	Ba153	6.32E+06	3.94E+09	9.94E+08	199.4
	Ru103	1.84E+16	6.33E+17	1.81E+17	188.7
	Ru105	1.00E+16	5.33E+17	1.39E+17	192.6
	Ru106	5.85E+15	2.18E+17	6.48E+16	189.5
	Ru107	3.27E+15	3.37E+17	8.10E+16	196.2
	Ru108	1.91E+15	2.35E+17	5.46E+16	196.8
	Ru109	1.19E+15	1.55E+17	3.56E+16	197.0
	Ru110	6.99E+14	7.57E+16	1.73E+16	196.3
	Ru111	3.42E+14	3.19E+16	7.36E+15	195.8
	Ru112	1.63E+14	1.10E+16	2.74E+15	194.2
	Ru113	7.76E+13	5.01E+15	1.27E+15	193.9
	Ru114	2.71E+13	1.73E+15	4.70E+14	193.9
	Ru115	4.63E+12	3.64E+14	1.05E+14	195.0
	Ru116	5.37E+11	4.43E+13	1.72E+13	195.2
	Ru117	3.90E+10	6.52E+12	2.09E+12	197.6
	Ru118	2.84E+09	1.33E+12	3.66E+11	199.1
	Ru119	1.95E+08	1.31E+11	3.56E+10	199.4
	Ru120	1.78E+07	1.26E+10	3.41E+09	199.4
	Ru97	3.36E+04	3.36E+04	3.36E+04	0.0
Semi-Volatile	Sr100	1.45E+14	2.59E+15	9.24E+14	178.8
	Sr101	1.35E+13	3.57E+14	1.15E+14	185.5
	Sr102	6.99E+11	3.77E+13	1.09E+13	192.7

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Sr103	1.18E+10	9.50E+11	3.08E+11	195.1
Sr104	6.62E+08	1.14E+11	3.01E+10	197.7
Sr105	3.29E+07	2.19E+10	5.50E+09	199.4
Sr83	1.21E+03	4.43E+04	1.78E+04	189.4
Sr85	1.07E+07	3.07E+10	3.90E+09	199.9
Sr85m	5.66E+06	8.87E+09	1.20E+09	199.7
Sr87m	4.74E+10	1.63E+14	2.01E+13	199.9
Sr89	1.35E+16	2.84E+17	7.87E+16	181.8
Sr90	3.55E+15	1.64E+16	9.58E+15	128.6
Sr91	1.74E+16	3.45E+17	1.00E+17	180.7
Sr92	1.84E+16	3.79E+17	1.11E+17	181.5
Sr93	2.01E+16	4.32E+17	1.28E+17	182.3
Sr94	1.94E+16	4.30E+17	1.26E+17	182.8
Sr95	1.74E+16	3.77E+17	1.12E+17	182.3
Sr96	1.35E+16	2.74E+17	8.23E+16	181.2
Sr97	6.55E+15	1.24E+17	3.79E+16	180.0
Sr98	3.28E+15	5.40E+16	1.74E+16	177.1
Sr99	5.64E+14	9.89E+15	3.84E+15	178.4

Table C.7. Minimum, Maximum, and Average Radionuclide Activity at Time 0 (End of Operation) for Small- to Medium-Sized Advanced Reactors

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Ac225	2.58E+04	1.58E+10	1.31E+09	200.0
Ac226	5.09E+03	1.57E+08	2.24E+07	200.0
Ac227	2.21E+04	1.04E+10	2.09E+09	200.0
Ac228	6.88E+03	8.78E+10	9.76E+09	200.0
Ag106	4.54E+03	6.30E+05	2.10E+05	197.1
Ag106m	9.58E+03	8.79E+05	2.76E+05	195.7
Ag107m	4.24E+05	1.79E+08	3.52E+07	199.1
Ag108	7.57E+08	1.93E+12	2.78E+11	199.8
Ag108m	2.57E+06	1.95E+09	2.55E+08	199.5
Ag109m	1.32E+15	2.67E+17	5.44E+16	198.0
Ag110	1.18E+14	1.99E+17	3.40E+16	199.8
Ag110m	4.98E+12	1.51E+15	3.96E+14	198.7
Ag111	3.90E+14	4.41E+16	9.58E+15	196.5
Ag111m	3.89E+14	4.16E+16	9.24E+15	196.3
Ag112	2.15E+14	1.79E+16	4.02E+15	195.2
Ag113	1.05E+14	7.48E+15	1.74E+15	194.5
Ag113m	1.53E+14	1.09E+16	2.54E+15	194.5
Ag114	1.14E+14	6.92E+15	1.73E+15	193.5
Ag115	8.91E+13	4.36E+15	1.23E+15	192.0
Ag115m	3.51E+12	7.92E+13	2.48E+13	183.0
Ag116	9.42E+13	4.00E+15	1.16E+15	190.8
Ag116m	6.37E+12	2.72E+14	7.59E+13	190.8
Ag117	7.35E+13	2.92E+15	8.92E+14	190.2
Ag117m	1.46E+13	6.75E+14	1.92E+14	191.5
Ag118	5.13E+13	2.03E+15	6.62E+14	190.1
Ag118m	2.49E+13	9.54E+14	2.94E+14	189.8
Ag119	4.63E+13	1.65E+15	5.63E+14	189.1
Ag120	1.79E+13	5.86E+14	2.34E+14	188.2
Ag120m	5.66E+12	3.64E+14	1.25E+14	193.9
Ag121	1.32E+13	4.57E+14	1.80E+14	188.8
Ag122	1.72E+12	1.21E+14	3.91E+13	194.4
Ag122m	1.68E+12	1.08E+14	3.54E+13	193.9
Ag123	1.20E+12	1.09E+14	3.11E+13	195.6
Ag124	6.64E+11	4.81E+13	1.40E+13	194.6
Ag125	1.41E+10	1.08E+13	2.89E+12	199.5
Ag126	2.73E+09	2.11E+12	5.78E+11	199.5
Ag127	4.19E+08	3.23E+11	9.06E+10	199.5
Ag128	5.72E+07	4.42E+10	1.25E+10	199.5
Ag129	1.08E+07	7.82E+09	2.30E+09	199.4

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Ag130	3.23E+08	2.17E+11	4.52E+10	199.4
Am239	2.61E+05	1.12E+08	2.33E+07	199.1
Am240	2.80E+08	1.51E+11	2.27E+10	199.3
Am241	4.78E+11	4.70E+13	8.48E+12	196.0
Am242	9.36E+12	2.76E+16	6.51E+15	199.9
Am242m	1.46E+10	1.66E+12	4.67E+11	196.5
Am243	8.04E+07	1.44E+13	3.78E+12	200.0
Am244	1.91E+09	2.98E+16	4.31E+15	200.0
Am244m	2.68E+10	4.48E+17	6.47E+16	200.0
Am245	5.16E+04	1.38E+13	1.75E+12	200.0
Am246	2.16E+04	3.49E+06	1.02E+06	197.5
Am246m	2.00E+04	3.22E+06	9.39E+05	197.5
Am247	1.69E+05	1.69E+05	1.69E+05	0.0
As72	1.89E+03	1.03E+05	3.98E+04	192.8
As73	1.34E+05	7.88E+06	2.10E+06	193.3
As74	7.34E+06	1.89E+09	4.35E+08	198.5
As76	2.25E+11	2.25E+13	7.33E+12	196.0
As77	3.52E+13	6.57E+14	2.21E+14	179.7
As78	8.72E+13	1.81E+15	5.82E+14	181.6
As79	1.87E+14	3.84E+15	1.26E+15	181.4
As80	4.20E+14	9.05E+15	2.74E+15	182.3
As81	6.54E+14	1.51E+16	4.56E+15	183.3
As82	7.43E+14	1.39E+16	3.96E+15	179.7
As82m	1.57E+14	4.83E+15	1.46E+15	187.4
As83	1.06E+15	1.86E+16	5.66E+15	178.4
As84	7.04E+14	1.18E+16	3.71E+15	177.4
As85	2.73E+14	8.47E+15	2.64E+15	187.5
As86	1.23E+14	1.99E+16	4.53E+15	197.6
As87	2.29E+13	1.61E+15	4.23E+14	194.4
As88	6.95E+12	4.24E+15	9.01E+14	199.3
As89	4.22E+11	2.27E+13	6.24E+12	192.7
As90	1.43E+10	6.29E+11	2.05E+11	191.1
As91	9.93E+08	1.04E+11	2.77E+10	196.2
As92	1.43E+07	8.75E+09	2.20E+09	199.3
At217	2.58E+04	1.58E+10	1.31E+09	200.0
Au200	1.78E+05	1.78E+05	1.78E+05	0.0
B12	4.21E+05	1.39E+09	2.78E+08	199.9
Ba131	3.02E+04	6.14E+06	1.79E+06	198.0
Ba133	9.39E+07	4.06E+10	6.45E+09	199.1
Ba135m	1.15E+12	1.83E+14	5.75E+13	197.5
Ba136m	9.83E+13	6.06E+15	1.48E+15	193.6

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Ba137m	4.24E+15	2.19E+16	1.32E+16	135.0
Ba139	2.32E+16	5.94E+17	1.80E+17	185.0
Ba140	2.24E+16	5.67E+17	1.71E+17	184.8
Ba141	2.12E+16	5.29E+17	1.60E+17	184.6
Ba142	1.99E+16	4.96E+17	1.49E+17	184.6
Ba143	1.82E+16	4.34E+17	1.28E+17	183.9
Ba144	1.40E+16	3.21E+17	9.36E+16	183.3
Ba145	6.97E+15	1.41E+17	4.22E+16	181.1
Ba146	3.49E+15	5.98E+16	1.82E+16	177.9
Ba147	9.56E+14	1.55E+16	4.92E+15	176.7
Ba148	9.25E+13	1.97E+15	8.07E+14	182.0
Ba149	5.27E+12	2.86E+14	1.02E+14	192.8
Ba150	3.05E+11	3.06E+13	9.77E+12	196.1
Ba151	1.26E+10	5.15E+12	1.34E+12	199.0
Ba152	2.71E+08	8.18E+10	2.28E+10	198.7
Ba153	6.32E+06	3.94E+09	9.94E+08	199.4
Be10	2.29E+05	1.69E+11	4.51E+10	200.0
Be11	6.65E+05	1.41E+12	3.18E+11	200.0
Be8	4.57E+05	1.47E+17	3.60E+16	200.0
Bi206	9.33E+06	9.33E+06	9.33E+06	0.0
Bi207	6.24E+09	6.24E+09	6.24E+09	0.0
Bi208	5.71E+09	5.71E+09	5.71E+09	0.0
Bi210	6.46E+03	4.44E+15	8.88E+14	200.0
Bi210m	2.58E+09	2.58E+09	2.58E+09	0.0
Bi211	1.73E+05	9.56E+11	1.61E+11	200.0
Bi212	2.47E+07	2.05E+13	1.71E+12	200.0
Bi213	2.58E+04	1.58E+10	1.31E+09	200.0
Bi214	2.80E+05	6.75E+05	4.78E+05	82.8
Bk248	1.24E+05	1.90E+05	1.57E+05	42.3
Bk248m	6.86E+05	9.77E+06	3.42E+06	173.8
Bk249	6.86E+05	4.16E+11	6.57E+10	200.0
Bk250	2.84E+05	6.39E+12	9.06E+11	200.0
Bk251	9.86E+06	3.78E+09	6.70E+08	199.0
Br77	3.11E+04	2.05E+06	5.23E+05	194.0
Br77m	2.40E+04	1.59E+06	4.63E+05	194.0
Br78	3.28E+06	2.20E+08	5.80E+07	194.1
Br79m	1.85E+08	1.20E+10	3.05E+09	193.9
Br80	1.30E+10	9.44E+11	3.22E+11	194.6
Br80m	4.11E+09	3.26E+11	1.13E+11	195.0
Br82	3.21E+13	3.00E+15	8.91E+14	195.8
Br82m	2.89E+13	2.75E+15	8.16E+14	195.8

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Br83	1.73E+15	3.55E+16	1.07E+16	181.5
Br84	3.08E+15	5.93E+16	1.76E+16	180.2
Br84m	8.96E+13	3.55E+15	1.03E+15	190.1
Br85	3.84E+15	7.69E+16	2.24E+16	181.0
Br86	5.32E+15	1.02E+17	2.96E+16	180.3
Br87	6.04E+15	1.12E+17	3.23E+16	179.5
Br88	5.25E+15	9.28E+16	2.66E+16	178.6
Br89	3.78E+15	6.11E+16	1.88E+16	176.7
Br90	2.20E+15	3.39E+16	1.09E+16	175.7
Br91	3.79E+14	9.81E+15	3.36E+15	185.1
Br92	7.43E+13	1.88E+15	6.20E+14	184.8
Br93	1.86E+13	6.57E+14	2.49E+14	189.0
Br94	2.95E+12	1.76E+14	5.33E+13	193.4
Br95	1.21E+10	7.95E+11	2.46E+11	194.0
Br96	7.55E+09	2.37E+11	7.22E+10	187.6
Br97	7.98E+07	5.88E+09	2.16E+09	194.6
C14	2.55E+06	2.00E+11	2.78E+10	200.0
C15	5.75E+07	2.16E+11	3.95E+10	199.9
Cd107	4.15E+04	2.49E+07	4.73E+06	199.3
Cd109	4.04E+07	2.88E+11	3.87E+10	199.9
Cd111m	1.45E+10	7.09E+13	1.18E+13	199.9
Cd113m	7.09E+09	1.53E+13	1.88E+12	199.8
Cd115	8.94E+13	4.68E+15	1.30E+15	192.5
Cd115m	5.34E+12	2.76E+14	8.00E+13	192.4
Cd117	7.39E+13	3.02E+15	9.20E+14	190.4
Cd117m	1.82E+13	6.96E+14	2.15E+14	189.8
Cd118	7.58E+13	2.88E+15	9.33E+14	189.7
Cd119	4.93E+13	1.75E+15	5.75E+14	189.1
Cd119m	2.89E+13	1.01E+15	3.22E+14	188.9
Cd120	7.35E+13	2.55E+15	8.42E+14	188.8
Cd121	3.89E+13	1.34E+15	4.66E+14	188.7
Cd121m	2.47E+13	9.68E+14	2.98E+14	190.1
Cd122	6.53E+13	1.74E+15	6.29E+14	185.5
Cd123	4.37E+13	8.40E+14	3.41E+14	180.2
Cd124	1.82E+13	6.38E+14	2.09E+14	188.9
Cd125	4.89E+12	3.16E+14	9.56E+13	193.9
Cd126	2.08E+12	2.74E+14	7.91E+13	197.0
Cd127	9.33E+11	2.66E+14	6.35E+13	198.6
Cd128	3.19E+11	1.17E+14	2.70E+13	198.9
Cd129	2.64E+10	4.00E+12	1.25E+12	197.4
Cd130	3.59E+12	2.84E+15	5.89E+14	199.5

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Cd131	6.98E+11	5.23E+14	1.09E+14	199.5
Cd132	2.40E+09	9.94E+11	2.64E+11	199.0
Ce137	9.36E+04	7.66E+06	1.89E+06	195.2
Ce139	2.07E+10	2.56E+13	4.47E+12	199.7
Ce139m	7.65E+09	1.05E+13	2.08E+12	199.7
Ce141	2.13E+16	5.22E+17	1.60E+17	184.3
Ce143	2.03E+16	4.92E+17	1.48E+17	184.2
Ce144	1.86E+16	3.61E+17	1.18E+17	180.4
Ce145	1.38E+16	3.34E+17	1.00E+17	184.1
Ce146	1.10E+16	2.66E+17	8.08E+16	184.1
Ce147	7.68E+15	1.93E+17	5.96E+16	184.7
Ce148	5.82E+15	1.36E+17	4.10E+16	183.6
Ce149	3.29E+15	7.84E+16	2.43E+16	183.9
Ce150	1.83E+15	4.18E+16	1.30E+16	183.2
Ce151	4.81E+14	1.33E+16	4.03E+15	186.0
Ce152	1.04E+14	3.55E+15	1.08E+15	188.6
Ce153	1.10E+13	5.86E+14	1.94E+14	192.6
Ce154	8.67E+11	6.37E+13	2.26E+13	194.6
Ce155	4.81E+10	4.89E+12	1.92E+12	196.1
Ce156	2.31E+09	3.10E+11	1.14E+11	197.0
Ce157	7.42E+07	1.25E+10	4.32E+09	197.6
Cf248	5.71E+04	8.61E+05	4.95E+05	175.1
Cf249	1.60E+03	5.40E+07	1.27E+07	200.0
Cf250	1.69E+04	1.30E+10	2.43E+09	200.0
Cf251	9.10E+05	9.91E+07	2.13E+07	196.4
Cf252	8.75E+03	4.45E+10	8.49E+09	200.0
Cf253	1.63E+08	1.26E+10	2.48E+09	194.9
Cf254	3.26E+05	2.77E+07	5.43E+06	195.3
Cf255	0.00E+00	0.00E+00	0.00E+00	0
Cm240	9.93E+03	7.36E+05	3.81E+05	194.7
Cm241	6.81E+06	1.33E+10	4.72E+09	199.8
Cm242	5.95E+12	1.16E+16	3.54E+15	199.8
Cm243	2.22E+09	9.45E+12	2.51E+12	199.9
Cm244	1.61E+09	6.04E+15	1.25E+15	200.0
Cm245	4.99E+04	1.57E+12	2.41E+11	200.0
Cm246	6.44E+02	7.49E+11	1.24E+11	200.0
Cm247	2.74E+05	6.41E+06	1.62E+06	183.6
Cm248	2.39E+06	4.32E+07	1.09E+07	179.1
Cm249	2.26E+04	9.71E+12	1.24E+12	200.0
Cm251	3.29E+04	2.03E+08	3.58E+07	199.9
Co65	6.28E+05	2.13E+08	5.66E+07	198.8

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Co66	4.98E+08	5.87E+10	1.79E+10	196.6
Co67	1.44E+09	1.69E+11	4.74E+10	196.6
Co68	1.67E+09	2.04E+11	5.86E+10	196.8
Co69	1.96E+09	2.00E+11	5.84E+10	196.1
Co70	1.33E+09	1.40E+11	4.17E+10	196.2
Co71	7.77E+08	7.01E+10	2.17E+10	195.6
Co72	2.76E+08	2.59E+10	7.51E+09	195.8
Co73	9.68E+07	1.00E+10	2.79E+09	196.2
Co74	1.32E+07	1.55E+09	4.44E+08	196.6
Co75	1.67E+06	2.17E+08	6.25E+07	197.0
Cr66	2.83E+04	1.42E+07	3.83E+06	199.2
Cr67	6.51E+04	4.26E+06	1.29E+06	194.0
Cs131	3.62E+06	1.27E+09	3.31E+08	198.9
Cs132	1.93E+10	2.46E+13	4.72E+12	199.7
Cs134	1.14E+15	8.37E+16	2.56E+16	194.6
Cs134m	1.61E+14	4.31E+16	9.74E+15	198.5
Cs135	3.27E+10	3.33E+11	1.13E+11	164.2
Cs135m	7.79E+12	9.82E+14	2.49E+14	196.8
Cs136	8.85E+14	5.47E+16	1.33E+16	193.6
Cs136m	7.44E+13	3.61E+15	1.09E+15	191.9
Cs137	4.48E+15	2.29E+16	1.38E+16	134.5
Cs138	2.43E+16	6.26E+17	1.89E+17	185.1
Cs138m	1.15E+15	4.47E+16	1.28E+16	190.0
Cs139	2.28E+16	5.80E+17	1.74E+17	184.9
Cs140	1.99E+16	4.77E+17	1.43E+17	184.0
Cs141	1.56E+16	3.76E+17	1.14E+17	184.1
Cs142	9.72E+15	2.24E+17	6.65E+16	183.4
Cs143	5.41E+15	1.13E+17	3.41E+16	181.8
Cs144	1.78E+15	3.46E+16	1.14E+16	180.4
Cs145	3.24E+14	6.80E+15	2.53E+15	181.8
Cs146	3.44E+13	8.80E+14	3.70E+14	185.0
Cs147	8.74E+12	1.55E+14	5.62E+13	178.6
Cs148	9.05E+10	1.35E+13	4.04E+12	197.3
Cs149	2.29E+09	2.48E+11	7.95E+10	196.3
Cs150	5.94E+07	2.62E+10	6.89E+09	199.1
Cs151	6.85E+06	4.59E+09	1.15E+09	199.4
Cu66	6.02E+08	6.52E+10	2.05E+10	196.3
Cu67	2.10E+09	2.06E+11	6.14E+10	196.0
Cu68	4.78E+09	3.77E+11	1.20E+11	195.0
Cu68m	1.63E+08	1.03E+10	3.89E+09	193.7
Cu69	1.25E+10	7.67E+11	2.62E+11	193.6

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Cu70	2.92E+10	1.45E+12	5.46E+11	192.1
Cu70m	8.96E+09	5.55E+11	1.82E+11	193.6
Cu71	6.62E+10	3.21E+12	1.20E+12	191.9
Cu72	1.35E+11	5.21E+12	1.98E+12	189.9
Cu73	2.70E+11	6.86E+12	2.85E+12	184.8
Cu74	3.23E+11	7.10E+12	2.77E+12	182.6
Cu75	3.80E+11	7.18E+12	2.60E+12	179.9
Cu76	1.66E+11	5.54E+12	1.65E+12	188.3
Cu77	6.17E+10	2.85E+12	8.16E+11	191.5
Cu78	1.91E+10	8.89E+11	2.56E+11	191.6
Cu79	4.16E+08	2.06E+11	5.31E+10	199.2
Cu80	2.19E+08	2.06E+10	5.46E+09	195.8
Dy157	7.06E+03	1.24E+05	7.25E+04	178.5
Dy159	7.08E+05	2.73E+10	2.72E+09	200.0
Dy165	4.45E+11	8.18E+14	1.21E+14	199.8
Dy165m	1.42E+10	5.17E+14	7.51E+13	200.0
Dy166	4.22E+10	2.05E+13	3.69E+12	199.2
Dy167	1.28E+10	3.48E+12	8.94E+11	198.5
Dy168	5.13E+09	1.89E+12	4.04E+11	198.9
Dy169	2.23E+09	7.11E+11	1.61E+11	198.8
Dy170	6.87E+08	2.42E+11	5.55E+10	198.9
Dy171	1.53E+08	6.64E+10	1.80E+10	199.1
Dy172	8.00E+07	4.48E+10	1.25E+10	199.3
Er163	2.01E+04	2.01E+04	2.01E+04	0.0
Er165	3.14E+05	1.88E+09	4.39E+08	199.9
Er167m	5.59E+09	2.95E+13	4.60E+12	199.9
Er169	2.84E+09	1.87E+12	3.72E+11	199.4
Er171	3.71E+08	2.32E+11	4.64E+10	199.4
Er172	2.03E+08	1.01E+11	2.94E+10	199.2
Es253	7.05E+07	3.40E+09	7.36E+08	191.9
Es254	9.19E+04	6.81E+06	1.41E+06	194.7
Es254m	1.42E+07	2.18E+09	4.08E+08	197.4
Es255	3.31E+05	3.58E+07	6.77E+06	196.3
Eu149	1.48E+04	1.88E+06	4.83E+05	196.9
Eu152	1.71E+10	6.12E+12	1.42E+12	198.9
Eu152m	3.33E+12	1.03E+14	2.32E+13	187.5
Eu154	7.65E+13	2.41E+15	7.78E+14	187.7
Eu154m	5.16E+12	4.16E+15	8.29E+14	199.5
Eu155	1.36E+14	3.57E+15	7.97E+14	185.3
Eu156	2.53E+14	2.31E+17	4.55E+16	199.6
Eu157	1.13E+14	4.08E+16	6.45E+15	198.9

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Eu158	5.43E+13	5.59E+15	1.27E+15	196.1
Eu159	2.26E+13	2.75E+15	6.31E+14	196.7
Eu160	8.59E+12	1.25E+15	2.77E+14	197.3
Eu161	3.06E+12	5.34E+14	1.15E+14	197.7
Eu162	5.81E+11	1.06E+14	2.40E+13	197.8
Eu163	1.38E+11	3.02E+13	6.39E+12	198.2
Eu164	2.08E+10	4.58E+12	1.05E+12	198.2
Eu165	3.50E+09	5.79E+11	1.67E+11	197.6
Eu166	3.42E+08	8.27E+10	2.52E+10	198.4
Eu167	2.83E+07	1.49E+10	4.00E+09	199.2
F20	2.38E+05	3.66E+17	6.24E+16	200.0
Fe65	6.28E+05	2.13E+08	5.66E+07	198.8
Fe66	2.17E+08	2.87E+10	8.10E+09	197.0
Fe67	3.43E+08	5.51E+10	1.45E+10	197.5
Fe68	1.99E+08	3.35E+10	8.92E+09	197.6
Fe69	6.57E+07	1.21E+10	3.28E+09	197.8
Fe70	1.78E+07	3.37E+09	9.35E+08	197.9
Fe71	2.74E+06	5.95E+08	1.72E+08	198.2
Fe72	3.79E+05	8.33E+07	2.46E+07	198.2
Fr221	2.58E+04	1.58E+10	1.31E+09	200.0
Fr222	9.39E+03	9.39E+03	9.39E+03	0.0
Fr223	2.14E+03	1.44E+08	4.81E+07	200.0
Ga68	3.35E+03	1.59E+07	2.54E+06	199.9
Ga70	1.38E+08	8.34E+10	1.55E+10	199.3
Ga72	2.18E+11	9.15E+12	3.43E+12	190.7
Ga72m	8.78E+09	3.83E+11	1.38E+11	191.0
Ga73	6.25E+11	1.93E+13	7.29E+12	187.5
Ga74	1.77E+12	4.47E+13	1.69E+13	184.8
Ga74m	8.62E+10	4.15E+12	1.24E+12	191.9
Ga75	5.04E+12	1.05E+14	3.86E+13	181.7
Ga76	1.29E+13	2.36E+14	8.51E+13	179.2
Ga77	3.00E+13	4.90E+14	1.66E+14	176.9
Ga78	5.09E+13	8.53E+14	2.66E+14	177.5
Ga79	5.08E+13	9.67E+14	3.06E+14	180.1
Ga80	3.36E+13	6.25E+14	1.97E+14	179.6
Ga81	1.56E+13	3.75E+14	1.19E+14	184.0
Ga82	3.82E+12	2.29E+14	6.63E+13	193.4
Ga83	6.83E+11	2.79E+13	7.98E+12	190.4
Ga84	4.30E+11	3.70E+14	7.77E+13	199.5
Ga85	2.91E+09	5.47E+11	1.42E+11	197.9
Ga86	1.47E+09	1.07E+12	2.28E+11	199.4

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Gd151	4.12E+05	3.04E+09	3.83E+08	199.9
Gd153	4.01E+10	9.86E+12	1.41E+12	198.4
Gd155m	3.67E+10	7.10E+10	5.38E+10	63.7
Gd159	2.55E+13	8.43E+15	1.44E+15	198.8
Gd161	3.93E+12	6.77E+14	1.51E+14	197.7
Gd162	1.24E+12	2.22E+14	5.11E+13	197.8
Gd163	4.76E+11	1.05E+14	2.23E+13	198.2
Gd164	1.50E+11	3.59E+13	7.89E+12	198.3
Gd165	4.24E+10	9.65E+12	2.21E+12	198.3
Gd166	1.27E+10	2.27E+12	6.21E+11	197.8
Gd167	1.50E+09	3.24E+11	1.03E+11	198.2
Gd168	4.83E+08	7.99E+10	2.90E+10	197.6
Gd169	6.09E+07	1.55E+10	4.88E+09	198.4
Ge69	6.53E+02	2.17E+05	9.17E+04	198.8
Ge71	7.89E+05	1.84E+09	2.44E+08	199.8
Ge71m	9.99E+04	6.93E+06	1.87E+06	194.3
Ge73m	6.16E+11	1.91E+13	7.21E+12	187.5
Ge75	5.12E+12	1.08E+14	3.98E+13	181.9
Ge75m	2.54E+11	7.17E+12	2.44E+12	186.3
Ge77	3.45E+13	6.34E+14	2.17E+14	179.3
Ge77m	6.64E+11	2.50E+13	7.72E+12	189.6
Ge78	8.58E+13	1.75E+15	5.63E+14	181.3
Ge79	1.19E+14	2.16E+15	7.05E+14	179.2
Ge79m	5.53E+13	1.23E+15	4.11E+14	182.8
Ge80	3.54E+14	6.96E+15	2.07E+15	180.6
Ge81	3.93E+14	7.56E+15	2.19E+15	180.2
Ge81m	6.47E+12	1.56E+14	4.92E+13	184.0
Ge82	2.87E+14	6.25E+15	1.81E+15	182.5
Ge83	1.04E+14	1.73E+15	6.35E+14	177.3
Ge84	3.25E+13	1.16E+15	3.88E+14	189.1
Ge85	4.94E+12	2.17E+14	6.28E+13	191.1
Ge86	1.67E+13	1.96E+16	4.06E+15	199.7
Ge87	1.12E+11	7.12E+13	1.53E+13	199.4
Ge88	5.63E+09	1.71E+12	4.09E+11	198.7
Ge89	4.85E+07	2.33E+10	5.93E+09	199.2
H3	1.80E+13	1.85E+16	2.92E+15	199.6
He6	8.63E+05	5.58E+16	1.59E+16	200.0
Hg203	1.37E+10	1.37E+10	1.37E+10	0.0
Hg205	5.43E+10	5.43E+10	5.43E+10	0.0
Hg206	3.07E+08	3.07E+08	3.07E+08	0.0
Ho161	8.27E+03	1.85E+06	4.60E+05	198.2

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Ho161m	0.00E+00	0.00E+00	#DIV/0!	#DIV/0!
Ho162	7.90E+04	1.67E+07	3.38E+06	198.1
Ho162m	7.64E+04	1.62E+07	3.30E+06	198.1
Ho163	1.71E+03	4.66E+03	2.83E+03	92.6
Ho163m	1.66E+05	3.55E+07	7.08E+06	198.1
Ho164	8.83E+06	4.10E+09	1.07E+09	199.1
Ho164m	5.81E+06	2.24E+09	5.68E+08	199.0
Ho166	1.10E+11	3.24E+14	4.74E+13	199.9
Ho166m	1.15E+06	2.74E+08	8.04E+07	198.3
Ho167	1.62E+10	2.13E+13	3.35E+12	199.7
Ho168	5.44E+09	2.03E+12	4.31E+11	198.9
Ho169	2.59E+09	8.86E+11	1.94E+11	198.8
Ho170	8.06E+08	3.15E+11	6.74E+10	199.0
Ho170m	1.11E+08	7.36E+10	1.19E+10	199.4
Ho171	3.18E+08	1.85E+11	3.69E+10	199.3
Ho172	1.66E+08	7.15E+10	2.45E+10	199.1
I123	6.01E+04	7.56E+06	2.04E+06	196.8
I125	1.45E+04	4.81E+07	7.28E+06	199.9
I126	6.47E+08	1.39E+12	2.13E+11	199.8
I128	7.90E+13	1.15E+16	2.84E+15	197.3
I129	1.18E+09	6.47E+09	3.49E+09	138.3
I130	1.65E+14	1.73E+16	5.11E+15	196.2
I130m	1.05E+14	1.11E+16	3.29E+15	196.3
I131	1.24E+16	3.45E+17	1.05E+17	186.1
I132	1.83E+16	5.00E+17	1.52E+17	185.9
I132m	1.63E+14	9.18E+15	2.37E+15	193.0
I133	2.59E+16	6.70E+17	2.05E+17	185.1
I133m	2.20E+15	6.72E+16	2.01E+16	187.3
I134	2.91E+16	7.55E+17	2.28E+17	185.1
I134m	2.27E+15	9.16E+16	2.53E+16	190.3
I135	2.46E+16	6.47E+17	1.97E+17	185.4
I136	1.07E+16	2.42E+17	7.46E+16	183.0
I136m	5.38E+15	1.65E+17	4.82E+16	187.4
I137	1.19E+16	2.92E+17	8.83E+16	184.3
I138	6.62E+15	1.52E+17	4.76E+16	183.3
I139	3.12E+15	5.86E+16	1.81E+16	179.8
I140	6.43E+14	1.23E+16	4.30E+15	180.1
I141	1.58E+14	2.44E+15	9.10E+14	175.7
I142	2.49E+13	4.92E+14	1.77E+14	180.7
I143	1.47E+11	4.03E+13	1.06E+13	198.5
I144	8.53E+09	6.16E+11	2.06E+11	194.5

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
In111	2.03E+03	6.84E+05	1.90E+05	198.8
In112	8.42E+03	3.31E+08	4.19E+07	200.0
In112m	6.80E+03	2.67E+08	3.38E+07	200.0
In113m	2.14E+03	3.65E+07	7.68E+06	200.0
In114	1.14E+09	2.41E+12	2.64E+11	199.8
In114m	7.12E+08	1.31E+12	1.46E+11	199.8
In115m	8.94E+13	4.89E+15	1.33E+15	192.8
In116	4.83E+12	1.72E+15	3.72E+14	198.9
In116m	7.94E+12	2.86E+15	6.16E+14	198.9
In117	5.62E+13	2.25E+15	6.88E+14	190.2
In117m	6.79E+13	2.77E+15	8.45E+14	190.4
In118	7.59E+13	2.88E+15	9.34E+14	189.7
In118m	2.46E+10	1.30E+12	3.95E+11	192.6
In119	3.79E+13	1.30E+15	4.17E+14	188.7
In119m	4.48E+13	1.59E+15	5.23E+14	189.1
In120	7.53E+13	2.63E+15	8.63E+14	188.9
In120m	3.58E+12	1.49E+14	4.35E+13	190.6
In121	5.57E+13	2.11E+15	6.56E+14	189.7
In121m	2.97E+13	1.06E+15	3.59E+14	189.1
In122	7.57E+13	2.28E+15	7.80E+14	187.2
In122m	2.03E+13	1.07E+15	3.00E+14	192.6
In123	4.82E+13	1.82E+15	5.81E+14	189.7
In123m	3.60E+13	8.67E+14	3.38E+14	184.1
In124	7.02E+13	1.62E+15	6.10E+14	183.4
In124m	2.44E+13	1.37E+15	4.01E+14	193.0
In125	4.78E+13	1.46E+15	5.00E+14	187.3
In125m	4.00E+13	1.39E+15	4.81E+14	188.8
In126	5.18E+13	1.48E+15	5.02E+14	186.5
In126m	2.35E+13	1.41E+15	4.23E+14	193.4
In127	1.58E+14	3.15E+15	9.43E+14	181.0
In127m	5.01E+13	2.22E+15	6.59E+14	191.2
In128	9.45E+13	2.71E+15	8.24E+14	186.5
In128m	8.20E+13	2.68E+15	7.95E+14	188.1
In129	7.56E+13	1.44E+15	4.66E+14	180.0
In129m	7.54E+13	1.33E+15	4.44E+14	178.5
In130	9.39E+13	3.83E+15	9.35E+14	190.4
In130m	4.49E+13	1.68E+15	5.64E+14	189.6
In131	4.47E+13	9.70E+14	2.82E+14	182.4
In131m	2.66E+13	7.06E+14	2.14E+14	185.5
In132	2.45E+13	4.86E+14	1.74E+14	180.8
In133	7.60E+11	4.42E+13	1.36E+13	193.2

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
In134	1.87E+10	2.70E+12	7.46E+11	197.2
In135	1.80E+08	9.58E+10	2.43E+10	199.3
Kr100	6.71E+07	3.75E+10	8.22E+09	199.3
Kr79	1.87E+04	4.64E+06	9.02E+05	198.4
Kr79m	9.34E+03	2.35E+06	5.63E+05	198.4
Kr81	6.65E+03	3.57E+05	1.17E+05	192.7
Kr81m	1.54E+08	4.65E+10	1.22E+10	198.7
Kr83m	1.73E+15	3.58E+16	1.08E+16	181.6
Kr85	4.08E+14	2.18E+15	1.20E+15	137.0
Kr85m	3.87E+15	7.81E+16	2.29E+16	181.1
Kr87	7.48E+15	1.45E+17	4.22E+16	180.4
Kr88	1.01E+16	1.95E+17	5.59E+16	180.4
Kr89	1.26E+16	2.39E+17	6.76E+16	180.0
Kr90	1.24E+16	2.37E+17	6.64E+16	180.1
Kr91	8.06E+15	1.63E+17	4.64E+16	181.2
Kr92	4.23E+15	8.31E+16	2.46E+16	180.6
Kr93	1.22E+15	2.41E+16	7.88E+15	180.7
Kr94	3.42E+14	6.31E+15	2.33E+15	179.4
Kr95	2.92E+13	9.50E+14	3.07E+14	188.1
Kr96	6.12E+13	1.67E+15	4.35E+14	185.8
Kr97	1.10E+12	1.01E+14	2.59E+13	195.7
Kr98	7.81E+10	5.46E+13	1.16E+13	199.4
Kr99	1.86E+08	1.19E+11	3.02E+10	199.4
La135	3.83E+05	2.84E+07	7.18E+06	194.7
La137	1.67E+05	2.17E+06	8.73E+05	171.5
La140	2.37E+16	5.95E+17	1.82E+17	184.7
La141	2.13E+16	5.33E+17	1.62E+17	184.6
La142	2.04E+16	5.14E+17	1.54E+17	184.7
La143	2.01E+16	4.89E+17	1.46E+17	184.2
La144	1.81E+16	4.30E+17	1.27E+17	183.8
La145	1.31E+16	3.06E+17	9.11E+16	183.6
La146	6.06E+15	1.26E+17	3.80E+16	181.7
La146m	2.57E+15	6.65E+16	1.98E+16	185.1
La147	4.07E+15	9.12E+16	2.86E+16	182.9
La148	1.58E+15	3.31E+16	1.05E+16	181.8
La149	3.77E+14	1.01E+16	3.45E+15	185.5
La150	5.51E+13	2.10E+15	7.51E+14	189.8
La151	6.06E+12	2.85E+14	1.17E+14	191.7
La152	3.95E+11	3.37E+13	1.27E+13	195.4
La153	2.06E+10	3.60E+12	1.13E+12	197.7
La154	6.22E+08	1.56E+11	4.66E+10	198.4

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
La155	2.43E+07	4.44E+09	1.51E+09	197.8
Li8	1.38E+09	8.08E+17	4.50E+17	200.0
Mg27	1.10E+05	1.10E+05	1.10E+05	0.0
Mn66	5.77E+06	1.96E+09	5.20E+08	198.8
Mn67	4.48E+06	1.73E+09	4.48E+08	199.0
Mn68	7.70E+05	3.66E+08	9.76E+07	199.2
Mn69	9.92E+04	4.88E+07	1.37E+07	199.2
Mo101	2.18E+16	5.90E+17	1.83E+17	185.8
Mo102	1.98E+16	5.89E+17	1.73E+17	187.0
Mo103	1.75E+16	6.23E+17	1.73E+17	189.1
Mo104	1.29E+16	5.45E+17	1.45E+17	190.7
Mo105	8.13E+15	4.02E+17	1.04E+17	192.1
Mo106	4.29E+15	2.54E+17	6.31E+16	193.4
Mo107	1.45E+15	9.52E+16	2.32E+16	194.0
Mo108	4.55E+14	3.20E+16	7.77E+15	194.4
Mo109	8.79E+13	3.81E+15	1.07E+15	191.0
Mo110	6.86E+12	4.82E+14	1.29E+14	194.4
Mo111	9.49E+11	1.08E+14	2.79E+13	196.5
Mo112	5.45E+10	1.44E+13	3.72E+12	198.5
Mo113	2.72E+09	1.66E+12	4.36E+11	199.3
Mo114	6.24E+08	9.10E+10	2.70E+10	197.3
Mo115	3.03E+07	5.83E+09	2.02E+09	197.9
Mo91	4.83E+11	4.83E+11	4.83E+11	0.0
Mo93	1.92E+03	1.32E+13	4.40E+12	200.0
Mo93m	6.80E+03	7.72E+13	6.43E+12	200.0
Mo99	2.38E+16	6.17E+17	1.97E+17	185.2
N16	1.12E+11	7.26E+16	1.40E+16	200.0
Na22	2.03E+10	2.03E+10	2.03E+10	0.0
Na24	1.70E+05	4.04E+17	1.35E+17	200.0
Na24m	1.31E+05	3.10E+17	1.03E+17	200.0
Na25	7.70E+09	7.70E+09	7.70E+09	0.0
Nb100	2.29E+16	5.81E+17	1.77E+17	184.9
Nb100m	1.56E+15	6.04E+16	1.69E+16	189.9
Nb101	2.08E+16	5.68E+17	1.72E+17	185.8
Nb102	1.25E+16	3.12E+17	9.56E+16	184.6
Nb102m	4.00E+15	1.49E+17	4.11E+16	189.5
Nb103	1.05E+16	3.33E+17	9.53E+16	187.7
Nb104	2.29E+15	9.18E+16	2.56E+16	190.2
Nb104m	1.88E+15	7.95E+16	2.12E+16	190.8
Nb105	1.65E+15	6.93E+16	1.92E+16	190.7
Nb106	1.84E+14	1.43E+16	3.76E+15	194.9

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Nb107	3.52E+13	2.84E+15	7.85E+14	195.1
Nb108	3.05E+12	2.88E+14	8.34E+13	195.8
Nb109	1.48E+12	2.63E+13	9.17E+12	178.7
Nb110	1.87E+10	1.70E+12	4.49E+11	195.6
Nb111	2.63E+09	1.60E+12	4.02E+11	199.3
Nb112	3.16E+07	1.62E+10	4.20E+09	199.2
Nb113	3.94E+06	2.91E+09	7.72E+08	199.5
Nb90	7.28E+07	7.28E+07	7.28E+07	0.0
Nb90m	5.03E+07	5.03E+07	5.03E+07	0.0
Nb91	7.07E+10	7.07E+10	7.07E+10	0.0
Nb91m	5.43E+12	5.43E+12	5.43E+12	0.0
Nb92	5.82E+06	5.82E+06	5.82E+06	0.0
Nb92m	6.02E+04	7.18E+14	5.99E+13	200.0
Nb93m	1.17E+10	9.17E+11	1.04E+11	195.0
Nb94	5.97E+06	1.37E+09	1.48E+08	198.3
Nb94m	7.68E+09	1.17E+13	1.25E+12	199.7
Nb95	2.42E+16	5.39E+17	1.59E+17	182.8
Nb95m	2.61E+14	5.78E+15	1.72E+15	182.7
Nb96	1.46E+13	7.20E+15	1.67E+15	199.2
Nb97	2.22E+16	5.52E+17	1.67E+17	184.5
Nb97m	2.10E+16	5.21E+17	1.58E+17	184.5
Nb98	2.21E+16	5.56E+17	1.70E+17	184.7
Nb98m	1.18E+14	3.50E+15	1.04E+15	187.0
Nb99	1.41E+16	3.56E+17	1.08E+17	184.8
Nb99m	9.40E+15	2.52E+17	7.71E+16	185.6
Nd140	1.96E+03	6.13E+06	1.26E+06	199.9
Nd141	1.84E+08	8.51E+11	1.20E+11	199.9
Nd141m	3.79E+07	1.81E+11	2.55E+10	199.9
Nd147	8.64E+15	2.05E+17	6.44E+16	183.8
Nd149	4.74E+15	1.36E+17	4.09E+16	186.5
Nd151	2.26E+15	7.84E+16	2.23E+16	188.8
Nd152	1.50E+15	5.56E+16	1.52E+16	189.5
Nd153	9.01E+14	3.41E+16	9.16E+15	189.7
Nd154	4.38E+14	1.98E+16	5.07E+15	191.3
Nd155	1.49E+14	8.10E+15	2.02E+15	192.8
Nd156	4.77E+13	3.08E+15	7.46E+14	193.9
Nd157	7.09E+12	8.28E+14	1.89E+14	196.6
Nd158	9.12E+11	1.50E+14	3.36E+13	197.6
Nd159	6.80E+10	1.45E+13	3.33E+12	198.1
Nd160	4.04E+09	8.17E+11	2.04E+11	198.0
Nd161	1.57E+08	2.98E+10	8.01E+09	197.9

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Ne23	9.83E+07	5.13E+14	1.71E+14	200.0
Ni65	6.28E+05	2.16E+08	5.72E+07	198.8
Ni66	6.00E+08	6.51E+10	2.05E+10	196.3
Ni67	2.08E+09	2.05E+11	6.08E+10	196.0
Ni68	4.57E+09	3.65E+11	1.15E+11	195.1
Ni69	1.03E+10	6.53E+11	2.17E+11	193.8
Ni70	2.20E+10	1.12E+12	4.03E+11	192.3
Ni71	2.67E+10	1.19E+12	4.52E+11	191.2
Ni72	3.62E+10	1.22E+12	4.38E+11	188.5
Ni73	2.94E+10	9.41E+11	2.96E+11	187.9
Ni74	1.36E+10	4.64E+11	1.41E+11	188.6
Ni75	3.35E+09	1.55E+11	4.49E+10	191.5
Ni76	6.60E+08	4.85E+10	1.25E+10	194.6
Ni77	1.10E+08	7.88E+09	2.06E+09	194.5
Ni78	1.83E+07	1.21E+09	3.20E+08	194.0
Np234	6.45E+02	2.60E+05	1.33E+05	199.0
Np235	1.62E+06	3.27E+09	5.00E+08	199.8
Np236	8.82E+03	1.10E+07	2.29E+06	199.7
Np236m	5.26E+09	1.10E+13	2.16E+12	199.8
Np237	8.56E+09	1.25E+11	3.99E+10	174.4
Np238	1.20E+15	4.70E+17	9.53E+16	199.0
Np239	8.00E+16	9.23E+18	2.45E+18	196.6
Np240	8.34E+12	3.25E+16	5.63E+15	199.9
Np240m	1.44E+13	5.48E+16	9.48E+15	199.9
Np241	2.19E+06	3.27E+09	7.36E+08	199.7
O19	9.27E+09	5.32E+15	1.03E+15	200.0
Pa229	6.70E+02	7.52E+06	3.76E+06	200.0
Pa230	1.33E+05	3.21E+09	3.22E+08	200.0
Pa231	3.75E+05	1.28E+11	1.28E+10	200.0
Pa232	2.45E+08	9.77E+14	8.15E+13	200.0
Pa233	1.38E+10	8.78E+17	7.32E+16	200.0
Pa234	4.52E+07	1.67E+15	1.39E+14	200.0
Pa234m	9.60E+09	1.75E+15	1.46E+14	200.0
Pa235	4.14E+05	1.49E+11	1.93E+10	200.0
Pb203	2.16E+12	2.16E+12	2.16E+12	0.0
Pb205	2.38E+08	2.38E+08	2.38E+08	0.0
Pb207m	4.09E+03	1.75E+14	5.83E+13	200.0
Pb209	4.07E+04	4.78E+14	3.98E+13	200.0
Pb210	6.47E+03	4.24E+07	1.03E+07	199.9
Pb211	1.73E+05	9.97E+09	1.66E+09	200.0
Pb212	2.47E+07	2.05E+13	1.71E+12	200.0

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Pb214	2.80E+05	6.75E+05	4.78E+05	82.8
Pd101	8.54E+04	1.52E+07	3.87E+06	197.8
Pd103	2.70E+09	4.91E+12	9.27E+11	199.8
Pd107	2.29E+09	3.39E+10	1.44E+10	174.7
Pd107m	6.01E+11	9.66E+14	1.47E+14	199.8
Pd109	1.33E+15	2.67E+17	5.45E+16	198.0
Pd109m	1.65E+12	2.33E+15	3.88E+14	199.7
Pd111	3.92E+14	3.84E+16	8.79E+15	196.0
Pd111m	2.32E+11	2.71E+14	5.98E+13	199.7
Pd112	2.15E+14	1.70E+16	3.89E+15	195.0
Pd113	1.60E+14	1.13E+16	2.64E+15	194.4
Pd114	1.10E+14	6.88E+15	1.72E+15	193.7
Pd115	8.67E+13	4.28E+15	1.20E+15	192.1
Pd116	7.36E+13	3.68E+15	1.06E+15	192.2
Pd117	5.82E+13	2.21E+15	6.89E+14	189.7
Pd118	2.23E+13	9.88E+14	3.53E+14	191.2
Pd119	4.49E+12	3.26E+14	1.25E+14	194.6
Pd120	7.40E+12	1.84E+14	6.25E+13	184.5
Pd121	2.84E+11	4.74E+13	1.37E+13	197.6
Pd122	3.85E+10	1.35E+13	3.67E+12	198.9
Pd123	5.33E+09	2.88E+12	7.80E+11	199.3
Pd124	1.56E+09	4.73E+11	1.29E+11	198.7
Pm144	2.23E+03	9.62E+05	2.07E+05	199.1
Pm145	4.79E+05	7.65E+08	1.20E+08	199.7
Pm146	1.43E+08	3.01E+12	5.13E+11	200.0
Pm147	7.17E+15	4.54E+16	1.70E+16	145.4
Pm148	3.03E+14	9.72E+16	2.21E+16	198.8
Pm148m	2.52E+14	1.77E+16	4.30E+15	194.4
Pm149	4.72E+15	2.23E+17	5.84E+16	191.7
Pm150	2.34E+12	1.26E+16	1.89E+15	199.9
Pm151	2.27E+15	7.82E+16	2.23E+16	188.7
Pm152	1.51E+15	5.67E+16	1.55E+16	189.6
Pm152m	2.67E+13	1.82E+15	4.43E+14	194.2
Pm153	1.00E+15	3.91E+16	1.05E+16	190.0
Pm154	5.04E+14	2.34E+16	5.97E+15	191.6
Pm154m	6.54E+13	3.55E+15	9.08E+14	192.8
Pm155	2.95E+14	1.61E+16	4.06E+15	192.8
Pm156	1.49E+14	9.48E+15	2.31E+15	193.8
Pm157	5.25E+13	4.85E+15	1.11E+15	195.7
Pm158	1.23E+13	1.77E+15	3.83E+14	197.2
Pm159	2.42E+12	4.43E+14	9.49E+13	197.8

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Pm160	2.92E+11	6.84E+13	1.46E+13	198.3
Pm161	2.94E+10	7.51E+12	1.62E+12	198.4
Pm162	5.54E+08	9.74E+10	3.17E+10	197.7
Pm163	2.49E+07	8.89E+09	2.56E+09	198.9
Po208	5.06E+06	5.06E+06	5.06E+06	0.0
Po209	1.78E+09	1.78E+09	1.78E+09	0.0
Po210	5.78E+03	4.36E+15	1.09E+15	200.0
Po211	4.79E+02	5.83E+09	1.46E+09	200.0
Po211m	4.76E+07	4.76E+07	4.76E+07	0.0
Po212	1.58E+07	1.31E+13	1.09E+12	200.0
Po213	2.53E+04	1.54E+10	1.29E+09	200.0
Po214	1.13E+04	3.84E+08	3.51E+07	200.0
Po215	1.73E+05	9.97E+09	1.66E+09	200.0
Po216	2.47E+07	2.05E+13	1.71E+12	200.0
Po218	2.80E+05	6.75E+05	4.78E+05	82.8
Pr139	1.39E+05	1.32E+08	3.34E+07	199.6
Pr140	1.23E+10	1.52E+13	3.00E+12	199.7
Pr142	3.76E+14	4.86E+16	1.39E+16	196.9
Pr142m	1.31E+14	1.70E+16	4.86E+15	196.9
Pr143	1.95E+16	4.89E+17	1.46E+17	184.6
Pr144	1.86E+16	3.69E+17	1.20E+17	180.9
Pr144m	1.79E+14	1.12E+16	2.45E+15	193.7
Pr145	1.38E+16	3.34E+17	1.00E+17	184.1
Pr146	1.11E+16	2.67E+17	8.12E+16	184.1
Pr147	8.46E+15	2.12E+17	6.44E+16	184.6
Pr148	6.21E+15	1.52E+17	4.64E+16	184.3
Pr148m	3.91E+14	2.04E+16	5.33E+15	192.5
Pr149	4.53E+15	1.21E+17	3.67E+16	185.6
Pr150	2.97E+15	8.69E+16	2.53E+16	186.8
Pr151	1.68E+15	5.23E+16	1.49E+16	187.6
Pr152	7.06E+14	2.58E+16	6.97E+15	189.3
Pr153	2.34E+14	1.01E+16	2.73E+15	190.9
Pr154	4.35E+13	2.75E+15	7.06E+14	193.8
Pr155	7.33E+12	5.53E+14	1.52E+14	194.8
Pr156	7.57E+11	8.43E+13	2.32E+13	196.4
Pr157	6.56E+10	9.18E+12	2.64E+12	197.2
Pr158	3.64E+09	5.77E+11	1.67E+11	197.5
Pr159	1.44E+08	1.88E+10	6.34E+09	197.0
Pt200	7.33E+04	7.33E+04	7.33E+04	0.0
Pu236	1.58E+09	1.17E+12	1.95E+11	199.5
Pu237	4.38E+08	1.21E+12	1.75E+11	199.9

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Pu237m	1.51E+08	4.50E+11	9.07E+10	199.9
Pu238	3.30E+13	1.77E+15	5.23E+14	192.7
Pu239	4.42E+12	1.85E+14	5.26E+13	190.7
Pu240	1.16E+13	1.65E+14	3.93E+13	173.8
Pu241	1.33E+14	3.48E+16	8.41E+15	198.5
Pu242	1.05E+08	8.06E+11	2.76E+11	199.9
Pu243	4.57E+11	6.40E+17	1.02E+17	200.0
Pu244	2.45E+03	5.90E+05	2.25E+05	198.3
Pu245	5.16E+04	1.37E+13	1.74E+12	200.0
Ra222	1.13E+04	3.83E+08	3.50E+07	200.0
Ra223	1.73E+05	9.97E+09	1.66E+09	200.0
Ra224	2.47E+07	2.05E+13	1.71E+12	200.0
Ra225	2.63E+04	1.60E+10	1.34E+09	200.0
Ra226	2.80E+05	6.75E+05	4.78E+05	82.7
Ra227	3.59E+03	3.84E+07	4.68E+06	200.0
Ra228	1.40E+10	1.40E+10	1.40E+10	0.0
Rb100	1.31E+12	1.35E+15	2.80E+14	199.6
Rb101	6.18E+09	2.83E+11	8.25E+10	191.5
Rb102	5.03E+07	3.02E+10	7.64E+09	199.3
Rb81	5.93E+03	4.25E+05	1.42E+05	194.5
Rb83	4.57E+07	2.34E+09	6.43E+08	192.3
Rb84	6.10E+08	5.60E+11	1.16E+11	199.6
Rb86	2.31E+13	5.45E+15	1.15E+15	198.3
Rb86m	2.88E+12	6.79E+14	1.46E+14	198.3
Rb87	1.08E+06	4.48E+06	2.84E+06	122.2
Rb88	1.02E+16	2.00E+17	5.76E+16	180.6
Rb89	1.34E+16	2.60E+17	7.42E+16	180.4
Rb90	1.36E+16	2.50E+17	7.04E+16	179.4
Rb90m	2.59E+15	6.27E+16	1.84E+16	184.1
Rb91	1.63E+16	3.17E+17	9.10E+16	180.4
Rb92	1.46E+16	2.87E+17	8.30E+16	180.7
Rb93	1.16E+16	2.18E+17	6.43E+16	179.8
Rb94	6.31E+15	1.12E+17	3.47E+16	178.7
Rb95	3.15E+15	5.69E+16	1.78E+16	179.0
Rb96	8.10E+14	1.32E+16	4.91E+15	176.8
Rb97	1.47E+14	2.29E+15	7.20E+14	175.8
Rb98	1.60E+13	5.67E+14	1.85E+14	189.0
Rb99	3.66E+11	9.83E+13	2.59E+13	198.5
Rh101	1.52E+05	6.27E+07	9.29E+06	199.0
Rh101m	6.97E+04	7.79E+08	1.12E+08	200.0
Rh102	4.47E+09	7.01E+12	1.10E+12	199.7

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Rh102m	1.88E+09	1.19E+12	1.85E+11	199.4
Rh103m	1.82E+16	6.27E+17	1.79E+17	188.7
Rh104	1.57E+15	5.48E+17	1.21E+17	198.9
Rh104m	1.24E+14	4.17E+16	9.20E+15	198.8
Rh105	9.90E+15	3.63E+17	1.13E+17	189.4
Rh105m	2.84E+15	1.51E+17	3.95E+16	192.6
Rh106	5.98E+15	3.45E+17	8.38E+16	193.2
Rh106m	5.43E+11	1.39E+16	2.06E+15	200.0
Rh107	3.30E+15	3.41E+17	8.22E+16	196.2
Rh108	1.92E+15	2.37E+17	5.49E+16	196.8
Rh108m	9.86E+12	1.46E+15	3.67E+14	197.3
Rh109	1.23E+15	1.62E+17	3.71E+16	197.0
Rh110	1.18E+13	1.89E+15	3.80E+14	197.5
Rh110m	7.11E+14	7.76E+16	1.77E+16	196.4
Rh111	3.90E+14	3.77E+16	8.54E+15	195.9
Rh112	2.14E+14	1.59E+16	3.76E+15	194.7
Rh113	1.55E+14	1.02E+16	2.49E+15	194.0
Rh114	8.42E+13	5.14E+15	1.35E+15	193.5
Rh115	3.94E+13	2.18E+15	6.43E+14	192.9
Rh116	1.27E+13	8.14E+14	2.57E+14	193.8
Rh117	3.91E+12	2.05E+14	8.11E+13	192.5
Rh118	4.81E+11	7.63E+13	2.41E+13	197.5
Rh119	5.45E+10	1.87E+13	5.41E+12	198.8
Rh120	1.49E+10	4.08E+12	1.12E+12	198.5
Rh121	9.59E+08	6.63E+11	1.79E+11	199.4
Rh122	1.10E+08	8.31E+10	2.26E+10	199.5
Rn217	1.10E+06	1.10E+06	1.10E+06	0.0
Rn218	1.13E+04	3.83E+08	3.50E+07	200.0
Rn219	1.73E+05	9.97E+09	1.66E+09	200.0
Rn220	2.47E+07	2.05E+13	1.71E+12	200.0
Rn222	2.80E+05	6.75E+05	4.78E+05	82.8
Ru103	1.84E+16	6.33E+17	1.81E+17	188.7
Ru105	1.00E+16	5.33E+17	1.39E+17	192.6
Ru106	5.85E+15	2.18E+17	6.48E+16	189.5
Ru107	3.27E+15	3.37E+17	8.10E+16	196.2
Ru108	1.91E+15	2.35E+17	5.46E+16	196.8
Ru109	1.19E+15	1.55E+17	3.56E+16	197.0
Ru110	6.99E+14	7.57E+16	1.73E+16	196.3
Ru111	3.42E+14	3.19E+16	7.36E+15	195.8
Ru112	1.63E+14	1.10E+16	2.74E+15	194.2
Ru113	7.76E+13	5.01E+15	1.27E+15	193.9

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Ru114	2.71E+13	1.73E+15	4.70E+14	193.9
Ru115	4.63E+12	3.64E+14	1.05E+14	195.0
Ru116	5.37E+11	4.43E+13	1.72E+13	195.2
Ru117	3.90E+10	6.52E+12	2.09E+12	197.6
Ru118	2.84E+09	1.33E+12	3.66E+11	199.1
Ru119	1.95E+08	1.31E+11	3.56E+10	199.4
Ru120	1.78E+07	1.26E+10	3.41E+09	199.4
Ru97	3.36E+04	3.36E+04	3.36E+04	0.0
Sb118	3.32E+03	6.30E+05	1.91E+05	197.9
Sb118m	5.13E+03	1.06E+06	3.14E+05	198.1
Sb119	9.96E+05	9.72E+07	2.89E+07	195.9
Sb120	5.34E+07	1.24E+11	2.15E+10	199.8
Sb120m	4.22E+07	5.69E+10	1.00E+10	199.7
Sb122	7.11E+12	1.06E+15	2.64E+14	197.3
Sb122m	4.67E+11	6.96E+13	1.74E+13	197.3
Sb124	5.66E+12	6.78E+14	1.73E+14	196.7
Sb124m	1.64E+11	1.79E+13	4.55E+12	196.4
Sb125	2.23E+14	2.53E+15	1.08E+15	167.6
Sb126	4.88E+12	2.84E+14	8.02E+13	193.2
Sb126m	6.10E+12	1.87E+14	6.08E+13	187.4
Sb127	1.07E+15	3.87E+16	1.10E+16	189.2
Sb128	1.54E+14	6.15E+15	1.73E+15	190.2
Sb128m	1.80E+15	5.47E+16	1.64E+16	187.3
Sb129	3.33E+15	1.11E+17	3.23E+16	188.4
Sb129m	4.94E+13	1.42E+15	4.28E+14	186.5
Sb130	3.20E+15	8.94E+16	2.70E+16	186.2
Sb130m	4.12E+15	1.07E+17	3.32E+16	185.2
Sb131	1.04E+16	2.57E+17	7.95E+16	184.5
Sb132	7.77E+15	2.09E+17	6.30E+16	185.7
Sb132m	4.13E+15	8.34E+16	2.59E+16	181.1
Sb133	7.87E+15	1.75E+17	5.29E+16	182.8
Sb134	1.59E+15	3.24E+16	1.05E+16	181.3
Sb134m	1.62E+15	3.23E+16	1.03E+16	180.8
Sb135	6.24E+14	1.26E+16	4.68E+15	181.1
Sb136	5.08E+13	1.62E+15	6.08E+14	187.9
Sb137	1.52E+13	2.44E+15	5.57E+14	197.5
Sb138	2.11E+11	2.40E+13	7.04E+12	196.5
Sb139	9.43E+09	1.72E+12	4.84E+11	197.8
Se75	5.30E+05	1.41E+08	3.09E+07	198.5
Se77m	1.24E+11	3.09E+12	1.01E+12	184.6
Se79	3.99E+09	1.73E+10	1.04E+10	125.0

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Se79m	1.83E+14	3.77E+15	1.24E+15	181.5
Se81	6.97E+14	1.66E+16	5.06E+15	183.8
Se81m	5.88E+13	1.90E+15	5.70E+14	188.0
Se83	1.52E+15	3.00E+16	9.15E+15	180.8
Se83m	1.30E+14	3.49E+15	1.04E+15	185.6
Se84	3.02E+15	5.72E+16	1.70E+16	180.0
Se85	3.15E+15	5.99E+16	1.73E+16	180.0
Se86	3.03E+15	6.51E+16	1.82E+16	182.2
Se87	1.58E+15	3.61E+16	1.02E+16	183.3
Se88	5.50E+14	1.52E+16	4.51E+15	186.0
Se89	1.22E+14	3.34E+15	1.06E+15	185.9
Se90	4.57E+13	9.19E+14	3.12E+14	181.0
Se91	2.39E+12	1.18E+14	3.25E+13	192.1
Se92	1.67E+11	1.18E+13	3.18E+12	194.4
Se93	1.06E+10	5.86E+11	1.61E+11	192.9
Se94	3.28E+08	4.13E+10	1.25E+10	196.9
Sm145	4.16E+05	4.70E+08	6.73E+07	199.6
Sm146	1.54E+03	1.00E+05	4.40E+04	193.9
Sm147	9.67E+04	9.93E+05	3.59E+05	164.5
Sm151	1.10E+13	4.10E+14	1.12E+14	189.5
Sm153	1.12E+15	2.95E+17	6.17E+16	198.5
Sm155	3.39E+14	2.16E+16	5.30E+15	193.8
Sm156	2.05E+14	1.27E+16	3.14E+15	193.6
Sm157	1.06E+14	8.40E+15	2.00E+15	195.0
Sm158	5.00E+13	4.91E+15	1.13E+15	196.0
Sm159	1.71E+13	2.15E+15	4.82E+14	196.8
Sm160	5.03E+12	7.52E+14	1.64E+14	197.3
Sm161	9.58E+11	1.91E+14	4.01E+13	198.0
Sm162	8.32E+10	1.24E+13	3.06E+12	197.3
Sm163	8.51E+09	1.27E+12	3.40E+11	197.3
Sm164	5.59E+08	1.11E+11	3.64E+10	198.0
Sm165	3.07E+07	1.19E+10	3.33E+09	199.0
Sn113	1.30E+05	2.15E+07	6.44E+06	197.6
Sn113m	1.11E+05	2.50E+07	7.57E+06	198.2
Sn117m	2.05E+11	1.19E+13	3.23E+12	193.2
Sn119m	2.79E+12	6.38E+13	2.43E+13	183.2
Sn121	8.16E+13	2.95E+15	9.55E+14	189.2
Sn121m	1.15E+12	7.32E+12	3.04E+12	145.8
Sn123	1.43E+13	6.51E+14	1.94E+14	191.4
Sn123m	8.79E+13	2.93E+15	9.89E+14	188.3
Sn125	1.07E+14	4.42E+15	1.26E+15	190.6

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Sn125m	1.28E+14	4.25E+15	1.37E+15	188.3
Sn126	1.03E+10	7.14E+10	3.38E+10	149.7
Sn127	6.37E+14	2.36E+16	6.69E+15	189.5
Sn127m	3.70E+14	1.37E+16	3.92E+15	189.5
Sn128	1.72E+15	5.07E+16	1.53E+16	186.9
Sn128m	8.24E+14	2.47E+16	7.44E+15	187.1
Sn129	1.76E+15	6.01E+16	1.77E+16	188.6
Sn129m	8.71E+14	2.50E+16	7.54E+15	186.5
Sn130	2.38E+15	5.47E+16	1.67E+16	183.3
Sn130m	2.41E+15	5.55E+16	1.72E+16	183.3
Sn131	2.05E+15	4.15E+16	1.32E+16	181.1
Sn131m	1.97E+15	3.99E+16	1.27E+16	181.2
Sn132	2.62E+15	5.26E+16	1.67E+16	181.0
Sn133	4.34E+14	8.40E+15	3.05E+15	180.4
Sn134	7.01E+13	1.43E+15	5.26E+14	181.3
Sn135	2.67E+12	1.21E+14	3.79E+13	191.4
Sn136	7.50E+10	7.98E+12	2.24E+12	196.3
Sn137	8.81E+09	6.70E+11	2.15E+11	194.8
Sr100	1.45E+14	2.59E+15	9.24E+14	178.8
Sr101	1.35E+13	3.57E+14	1.15E+14	185.5
Sr102	6.99E+11	3.77E+13	1.09E+13	192.7
Sr103	1.18E+10	9.50E+11	3.08E+11	195.1
Sr104	6.62E+08	1.14E+11	3.01E+10	197.7
Sr105	3.29E+07	2.19E+10	5.50E+09	199.4
Sr83	1.21E+03	4.43E+04	1.78E+04	189.4
Sr85	1.07E+07	3.07E+10	3.90E+09	199.9
Sr85m	5.66E+06	8.87E+09	1.20E+09	199.7
Sr87m	4.74E+10	1.63E+14	2.01E+13	199.9
Sr89	1.35E+16	2.84E+17	7.87E+16	181.8
Sr90	3.55E+15	1.64E+16	9.58E+15	128.6
Sr91	1.74E+16	3.45E+17	1.00E+17	180.7
Sr92	1.84E+16	3.79E+17	1.11E+17	181.5
Sr93	2.01E+16	4.32E+17	1.28E+17	182.3
Sr94	1.94E+16	4.30E+17	1.26E+17	182.8
Sr95	1.74E+16	3.77E+17	1.12E+17	182.3
Sr96	1.35E+16	2.74E+17	8.23E+16	181.2
Sr97	6.55E+15	1.24E+17	3.79E+16	180.0
Sr98	3.28E+15	5.40E+16	1.74E+16	177.1
Sr99	5.64E+14	9.89E+15	3.84E+15	178.4
Tb155	8.67E+03	1.79E+06	4.58E+05	198.1
Tb156	2.77E+05	3.09E+07	8.47E+06	196.4

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Tb156m	2.45E+04	3.96E+06	9.56E+05	197.5
Tb157	1.07E+05	5.14E+06	1.03E+06	191.9
Tb158	8.11E+05	4.70E+08	6.71E+07	199.3
Tb158m	2.29E+07	4.47E+10	7.90E+09	199.8
Tb160	2.63E+12	1.08E+15	2.42E+14	199.0
Tb161	4.07E+12	1.83E+15	3.06E+14	199.1
Tb162	1.28E+12	3.28E+14	6.46E+13	198.5
Tb163	5.30E+11	1.18E+14	2.52E+13	198.2
Tb164	1.92E+11	4.79E+13	1.05E+13	198.4
Tb165	7.78E+10	1.95E+13	4.37E+12	198.4
Tb166	3.33E+10	7.37E+12	1.88E+12	198.2
Tb167	8.38E+09	2.08E+12	5.35E+11	198.4
Tb168	2.73E+09	7.54E+11	1.77E+11	198.6
Tb169	7.65E+08	1.70E+11	5.09E+10	198.2
Tb170	1.42E+08	3.34E+10	1.17E+10	198.3
Tb171	1.76E+07	8.51E+09	2.43E+09	199.2
Tc100	2.82E+15	3.57E+17	9.14E+16	196.9
Tc101	2.18E+16	5.91E+17	1.83E+17	185.8
Tc102	1.98E+16	5.91E+17	1.73E+17	187.0
Tc102m	5.01E+13	2.19E+15	5.96E+14	191.0
Tc103	1.79E+16	6.41E+17	1.78E+17	189.1
Tc104	1.38E+16	5.85E+17	1.56E+17	190.8
Tc105	9.93E+15	5.19E+17	1.33E+17	192.5
Tc106	5.81E+15	4.15E+17	1.02E+17	194.5
Tc107	2.85E+15	2.74E+17	6.52E+16	195.9
Tc108	1.25E+15	1.32E+17	2.97E+16	196.2
Tc109	5.87E+14	6.03E+16	1.35E+16	196.1
Tc110	1.48E+14	1.01E+16	2.39E+15	194.2
Tc111	3.17E+13	1.67E+15	4.83E+14	192.6
Tc112	4.17E+12	2.52E+14	9.44E+13	193.5
Tc113	7.68E+11	7.15E+13	2.23E+13	195.7
Tc114	2.99E+11	2.29E+13	7.95E+12	194.8
Tc115	3.07E+10	3.62E+12	1.31E+12	196.6
Tc116	1.20E+09	2.77E+11	8.84E+10	198.3
Tc117	2.68E+07	1.13E+10	3.20E+09	199.1
Tc118	3.57E+06	2.67E+09	7.15E+08	199.5
Tc96	8.68E+04	8.68E+04	8.68E+04	0.0
Tc97m	4.30E+06	1.06E+09	2.43E+08	198.4
Tc98	4.46E+04	5.75E+06	1.14E+06	196.9
Tc99	6.66E+11	2.79E+12	1.72E+12	122.9
Tc99m	2.09E+16	5.42E+17	1.75E+17	185.2

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Te121	1.35E+06	1.26E+10	1.67E+09	200.0
Te121m	6.86E+05	5.48E+09	7.02E+08	199.9
Te123m	3.75E+10	1.97E+13	3.24E+12	199.2
Te125m	5.13E+13	5.41E+14	2.26E+14	165.3
Te127	1.04E+15	3.29E+16	1.00E+16	187.8
Te127m	1.47E+14	2.85E+15	8.10E+14	180.4
Te129	3.17E+15	1.06E+17	3.09E+16	188.4
Te129m	5.53E+14	1.79E+16	5.23E+15	188.0
Te131	1.09E+16	2.88E+17	8.86E+16	185.4
Te131m	2.15E+15	7.63E+16	2.20E+16	189.0
Te132	1.80E+16	4.74E+17	1.46E+17	185.4
Te133	1.34E+16	3.37E+17	1.03E+17	184.7
Te133m	1.28E+16	3.35E+17	9.97E+16	185.3
Te134	2.40E+16	5.80E+17	1.72E+17	184.1
Te135	1.25E+16	2.91E+17	8.80E+16	183.5
Te136	5.51E+15	1.03E+17	3.20E+16	179.6
Te137	1.68E+15	3.00E+16	9.70E+15	178.9
Te138	2.70E+14	5.09E+15	2.04E+15	179.8
Te139	2.89E+13	8.28E+14	3.21E+14	186.5
Te140	6.18E+12	5.88E+14	1.47E+14	195.8
Te141	1.72E+11	2.98E+13	7.90E+12	197.7
Te142	9.34E+09	5.43E+11	1.69E+11	193.2
Th226	1.45E+04	3.83E+08	3.85E+07	200.0
Th227	1.73E+05	9.92E+09	1.65E+09	200.0
Th228	2.45E+07	2.04E+13	1.70E+12	200.0
Th229	2.10E+03	7.25E+09	1.45E+09	200.0
Th230	1.90E+04	4.39E+08	4.27E+07	200.0
Th231	3.09E+08	1.38E+15	1.15E+14	200.0
Th232	2.31E+10	2.31E+10	2.31E+10	0.0
Th233	1.63E+09	8.82E+17	7.35E+16	200.0
Th234	8.32E+09	1.12E+10	9.82E+09	29.6
Tl202	1.40E+08	1.40E+08	1.40E+08	0.0
Tl204	3.55E+10	3.55E+10	3.55E+10	0.0
Tl206	1.75E+12	1.75E+12	1.75E+12	0.0
Tl207	1.73E+05	9.75E+11	1.64E+11	200.0
Tl208	8.86E+06	7.36E+12	6.14E+11	200.0
Tl209	5.68E+02	3.47E+08	1.16E+08	200.0
Tm167	2.50E+03	2.27E+05	6.81E+04	195.6
Tm168	5.93E+03	9.37E+07	1.23E+07	200.0
Tm170	9.29E+08	3.13E+11	8.17E+10	198.8
Tm171	2.99E+08	5.86E+10	1.59E+10	198.0

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Tm172	2.29E+08	2.17E+11	4.93E+10	199.6
U230	1.03E+04	2.53E+08	3.18E+07	200.0
U231	6.51E+03	2.88E+09	2.62E+08	200.0
U232	5.37E+07	3.86E+13	3.22E+12	200.0
U233	8.52E+04	2.04E+13	1.70E+12	200.0
U234	1.21E+09	6.24E+11	1.19E+11	199.2
U235	1.41E+06	8.01E+09	2.95E+09	199.9
U235m	7.79E+12	1.81E+16	2.95E+15	199.8
U236	1.34E+10	6.77E+10	3.40E+10	133.9
U237	7.09E+15	7.55E+17	1.46E+17	196.3
U238	8.32E+09	1.12E+10	9.82E+09	29.9
U239	8.00E+16	9.34E+18	2.47E+18	196.6
U240	2.45E+03	5.86E+05	2.24E+05	198.3
Xe125	7.80E+04	2.17E+07	5.57E+06	198.6
Xe125m	2.06E+04	5.75E+06	2.88E+06	198.6
Xe127	2.63E+07	3.80E+11	4.83E+10	200.0
Xe127m	4.41E+06	6.92E+10	9.03E+09	200.0
Xe129m	5.74E+10	4.49E+13	9.09E+12	199.5
Xe131m	1.36E+14	3.50E+15	1.13E+15	185.0
Xe133	2.50E+16	6.77E+17	2.07E+17	185.8
Xe133m	7.65E+14	2.14E+16	6.72E+15	186.2
Xe134m	2.95E+14	1.39E+16	3.74E+15	191.7
Xe135	7.01E+15	3.44E+17	8.74E+16	192.0
Xe135m	5.25E+15	1.58E+17	4.70E+16	187.1
Xe137	2.30E+16	6.09E+17	1.85E+17	185.4
Xe138	2.22E+16	5.68E+17	1.70E+17	185.0
Xe139	1.69E+16	4.05E+17	1.19E+17	183.9
Xe140	1.20E+16	2.59E+17	7.57E+16	182.2
Xe141	4.90E+15	9.33E+16	2.87E+16	180.0
Xe142	1.79E+15	3.23E+16	1.04E+16	179.0
Xe143	1.70E+14	3.56E+15	1.36E+15	181.7
Xe144	2.56E+13	7.85E+14	2.87E+14	187.4
Xe145	5.86E+11	7.53E+13	2.38E+13	196.9
Xe146	5.43E+10	5.36E+12	1.59E+12	196.0
Xe147	7.07E+09	1.40E+11	5.56E+10	180.7
Y100	2.88E+15	6.06E+16	2.07E+16	181.9
Y101	1.15E+15	2.05E+16	7.28E+15	178.8
Y102	3.34E+14	1.14E+16	3.19E+15	188.6
Y103	1.32E+13	8.11E+14	2.72E+14	193.6
Y104	2.35E+12	1.01E+14	3.18E+13	190.9
Y105	3.91E+10	1.95E+13	5.04E+12	199.2

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Y106	5.53E+08	6.81E+10	2.09E+10	196.8
Y107	1.00E+07	7.02E+09	1.89E+09	199.4
Y108	8.51E+04	2.85E+07	1.05E+07	198.8
Y87	2.76E+06	1.79E+08	4.67E+07	193.9
Y87m	1.45E+04	6.73E+06	2.32E+06	199.1
Y88	1.50E+09	1.38E+12	2.85E+11	199.6
Y89m	1.57E+12	2.77E+13	8.76E+12	178.5
Y90	3.60E+15	1.78E+16	1.06E+16	132.6
Y90m	2.19E+11	4.01E+13	9.10E+12	197.8
Y91	1.75E+16	3.79E+17	1.06E+17	182.4
Y91m	1.03E+16	2.03E+17	5.89E+16	180.8
Y92	1.85E+16	3.84E+17	1.12E+17	181.6
Y93	2.05E+16	4.43E+17	1.32E+17	182.3
Y93m	7.05E+15	1.53E+17	4.55E+16	182.4
Y94	2.11E+16	4.78E+17	1.42E+17	183.1
Y95	2.17E+16	5.04E+17	1.51E+17	183.5
Y96	1.43E+16	2.98E+17	8.94E+16	181.7
Y96m	6.82E+15	1.87E+17	5.65E+16	185.9
Y97	1.13E+16	2.53E+17	7.65E+16	182.9
Y97m	6.65E+15	1.65E+17	4.98E+16	184.5
Y98	8.22E+15	1.75E+17	5.44E+16	182.0
Y98m	4.71E+15	1.21E+17	3.70E+16	185.0
Y99	8.93E+15	1.86E+17	5.80E+16	181.7
Yb169	1.70E+04	1.55E+07	2.20E+06	199.6
Yb169m	6.70E+03	1.71E+06	4.98E+05	198.4
Zn69	1.26E+10	7.78E+11	2.65E+11	193.6
Zn69m	7.73E+07	5.23E+09	1.63E+09	194.2
Zn71	6.82E+10	3.34E+12	1.24E+12	192.0
Zn71m	8.46E+09	5.69E+11	1.68E+11	194.1
Zn72	2.14E+11	8.63E+12	3.27E+12	190.3
Zn73	5.99E+11	1.79E+13	6.86E+12	187.0
Zn74	1.62E+12	3.74E+13	1.48E+13	183.4
Zn75	3.78E+12	6.87E+13	2.55E+13	179.1
Zn76	7.20E+12	1.20E+14	4.30E+13	177.3
Zn77	5.80E+12	1.48E+14	4.78E+13	184.9
Zn78	4.75E+12	1.50E+14	4.54E+13	187.8
Zn79	1.55E+12	6.44E+13	2.10E+13	190.6
Zn80	4.03E+11	1.45E+13	4.47E+12	189.2
Zn81	7.45E+09	2.39E+12	6.51E+11	198.8
Zn82	1.09E+10	7.41E+11	2.15E+11	194.2
Zn83	3.06E+08	5.41E+10	1.41E+10	197.8

Radionuclide	Minimum Activity (Bq/MTHM)	Maximum Activity (Bq/MTHM)	Average Activity (Bq/MTHM)	Percent Difference (Min/Max)
Zr100	2.14E+16	5.28E+17	1.60E+17	184.5
Zr101	1.27E+16	2.94E+17	9.07E+16	183.4
Zr102	8.49E+15	1.76E+17	5.45E+16	181.6
Zr103	2.29E+15	4.89E+16	1.67E+16	182.1
Zr104	4.13E+14	1.24E+16	4.40E+15	187.1
Zr105	1.79E+14	4.88E+15	1.36E+15	185.9
Zr106	1.04E+12	1.19E+14	2.71E+13	196.6
Zr107	3.25E+10	3.23E+12	1.31E+12	196.0
Zr108	1.97E+09	2.19E+11	7.89E+10	196.4
Zr109	1.34E+09	2.37E+10	8.43E+09	178.6
Zr110	1.11E+07	1.10E+09	2.92E+08	196.0
Zr88	4.55E+03	1.37E+09	1.52E+08	200.0
Zr89	1.44E+08	1.12E+13	1.05E+12	200.0
Zr89m	8.43E+06	3.09E+12	2.63E+11	200.0
Zr90m	1.42E+10	1.41E+13	3.22E+12	199.6
Zr93	1.01E+11	3.75E+11	2.40E+11	115.0
Zr95	2.42E+16	5.34E+17	1.58E+17	182.7
Zr97	2.21E+16	5.48E+17	1.66E+17	184.5
Zr98	2.18E+16	5.46E+17	1.67E+17	184.7
Zr99	2.15E+16	5.41E+17	1.64E+17	184.7

Appendix D – NRC Advanced Reactor GEIS Values

The following draft plant parameter envelope (PPE) and site parameter envelope (SPE) tables were presented to the public in May 2020 during the U.S. Nuclear Regulatory Commission's (NRC's) scoping process associated with the Advanced Nuclear Reactor (ANR) Generic Environmental Impact Statement (GEIS). These tables are subject to change based upon additional NRC analysis.

Table D.1. NRC Advanced Reactor GEIS Draft Plant Parameter Envelope

Parameter	Value/Description	Assumptions
Site size	100 ac	1. Meets NRC Siting Regulations 2. Stand-alone site or designated portion of larger site (e.g., Government reservation, military base, or existing power plant site) 3. Complies with applicable zoning 4. Not inconsistent with any comprehensive plans or other land use plans
Permanent footprint of disturbance	30 ac	1. No prime farmland, or not adjacent to actively used farmland 2. No wetlands, floodplains, surface water features, riparian habitat, climax or old-growth vegetation, or dedicated conservation land
Temporary footprint of disturbance	Additional 20 ac	1. Restored to original grade and seeded or planted with indigenous vegetation once construction is complete. 2. Meets assumptions for permanent footprint
Offsite right-of-way	1,000 ft × 100 ft (new right-of-way) or unlimited length (within or adjacent to existing right-of-way)	1. Meets assumptions for site size 2. Does not cross or pass adjacent to parks, wildlife refuges, or conservation lands 3. Does not cross Wild and Scenic River or National Heritage River, or river of similar state designation
Cooling and service water intake	1000 Gallons per minute (gpm)	1. If water-cooled, maximum amount of water removed from surface water bodies for cooling-water makeup
Consumptive water use	400 gpm	1. Consumption through evaporative loss during the cooling process

Table D.1. NRC Advanced Reactor GEIS Draft Plant Parameter Envelope (continued)

Parameter	Value/Description	Assumptions
Plant water discharge	600 gpm	1. Amount of water discharged to waterbody after use for plant purposes including cooling (i.e., blowdown), and service water system. 2. Also includes other discharges from potable and sanitary systems (if applicable).
Blowdown temperature and constituent concentrations	Within applicable Clean Water Act limits	1. Discharge results mainly from plant blowdown. 2. Discharges are regulated under a clean water act permit and meets established discharge limits for temperature and for quantity of waste and concentration of each constituent.
Potable and sanitary water use and discharge	5 gpm	1. If groundwater is used, pumping rates fall within permittable limits. 2. If municipal water and sewage is used, usage amount is available and within capacity of the system.
Emissions from construction equipment and standby power equipment during operations	Criteria pollutants are less than Clean Air Act de minimis levels.	1. Clean Air Act requires a conformity determination for maintenance or nonattainment areas that exceed de minimis values. Not applicable to attainment areas.
Megawatts thermal (MWt)	60 MWt	1. Total thermal power generated by all units on site; can be more than one unit, however total thermal power is 60 MWt.
The operational life for which the plant is designed	80 yr	1. Bounding value. Assumes 40-year license with two 20-year license renewals for operational life
Building height	50 ft	1. Tallest structure, other than meteorology tower
Foundation embedment	50 ft	1. The depth from finished grade to the bottom of the basemat for the most deeply embedded power-block structure
Maximum number of construction workers	150 people	1. Maximum number of construction workforce, half of whom in-migrate to the host county is less than 5% of total host county populations
The number of total permanent staff to support operations	50 people	1. Maximum operations workforce all of whom in- migrate to the host county.
The additional number of temporary staff for refueling outage	100 people	1. No refueling workers in-migrate to the host county.
Noise generation	65 dBA	1. At site boundary
Station capacity factor	95% or greater	1. The percentage of time that a plant is capable of providing power to the grid

Table D.1. NRC Advanced Reactor GEIS Draft Plant Parameter Envelope (continued)

Parameter	Value/Description	Assumptions
The normal plant operating cycle length	2 to 20 yr	1. Different designs have different operating cycle lengths
Electrical output in megawatts-electric (MWe)	20 MWe	1. Most nuclear steam supply system designs are approximately 33 to 37 percent efficient applying a Rankine cycle without superheated steam. 2. It is acceptable that if the efficiency is higher than the bounding value can be slightly higher than 20 MWe.

Table D.2 NRC Advanced Reactor GEIS Draft Site Parameter Envelope

Parameter	Value/Description	Assumptions
Water for sanitary and potable water uses	Up to 5 gpm supply provided by municipal systems or groundwater resources.	1. If groundwater is used, pumping rates fall within permittable limits and the aquifer supplying water must support the required amount at a rate that is sustainable and does not impact offsite uses or users. 2. If municipal water supply is used, usage amount is available and within capacity of the system. 3. Sanitary discharge to sewage treatment plant is within available capacity and permittable. The plant is allowed to hook up to municipal water and sewage system with sufficient capacity.
Surface water availability	If plant uses surface water, monthly minimum flow is 75 cubic feet per second (cfs).	1. Not applicable if plant is air-cooled. 2. Maximum average plant water withdrawals are less than 3% percent of minimum monthly flow of water body. 3. Water availability is demonstrated by state-issued withdrawal permit. 4. Withdrawals do not prevent the maintenance of applicable instream flow requirements. 5. Water rights are obtainable, if needed, and amount is available without impact to other uses and users. 6. Large water bodies such as the oceans, Great Lakes are presumed to have sufficient water availability. 7. Coastal Zone Management Act consistency determination obtained.
Surface water discharge	If plant is discharging to surface water, monthly minimum flow is 75 cfs	1. Not applicable if plant is air-cooled. 2. Maximum average plant discharge is small in comparison to monthly minimum flow of water body (<3 percent) and thermal and chemical components within the discharge would be diluted quickly. 3. Discharge is in accordance with state/local permits. 4. Altered current patterns and salinity gradients would be localized. 5. Large water bodies such as the oceans, Great Lakes are presumed to have sufficient water capacity for dilution as long as restrictions on localized impacts are met.

Table D.2 NRC Advanced Reactor GEIS Draft Site Parameter Envelope (continued)

Parameter	Value/Description	Assumptions
Groundwater- availability and quality	Pumping rate of < 100 gpm (regardless of proposed purpose)	1. Pumping rate is sustainable, is a small percent of flow within the aquifer and does not impact availability to offsite uses and users 2. Withdrawal rates are within limits which are permittable by applicable state or local agencies 3. Groundwater usage does not impact quality within the aquifer
Air quality	Attainment, maintenance area, or nonattainment	1. Emission of criteria pollutants are less than de minimis levels
Economics	Annual property tax for the proposed project is less than ten percent of the total property tax revenue of the host county	1. Overnight construction cost of the proposed project is no more than $500 million USD

Appendix E – PPE Data Sources and Methodology

These tables describe the sources of information that were used to develop the bounding value, as well as the source or rationale for the identified value. The plant parameter envelope (PPE) table references certain values and tables that are included in Appendix C; these locations are noted in the table as applicable. The bounding value in both Table E.1 and Table E.2 are denoted using **bold** font.

Table E.1 Microreactor PPE Data Sources and Methodology[(a)]

PPE Section	Parameter	Information Sources: NRC ANR GEIS Value (from Appendix D)	Information Sources: Internal Research Value (from Appendices B and C)	Information Sources: Vendor Bounding Value (from Appendix A)	Regulatory Limit Value	Microreactor Bounding Value	Source/Rationale
Plant Design	What is your design type?	None provided	HTGR, MSR, LMR, heat pipe, and nuclear battery	HTGR, MSR, LMR, and nuclear battery	Not applicable	**Not Applicable for bounding parameters**	Plant type itself is not relevant to the environmental analysis; parameters therein that have an environmental interface are considered.
Plant Design	How many units do you plan to install?	None provided	Not evaluated	1	Not applicable	**1**	While the PPE considers installation of one unit, multiple units may be proposed, for instance to demonstrate the capability of following increases in electricity demand over time. This would have potential impacts on the extent and timing of resource analyses, including cumulative impacts.
Plant Design	What is the output of your design (per unit)?	60 MWt/20MWe	13 MWe	50 MWt 17 MWe	Not applicable	**60 MWt**	Value is bounded by a larger microreactor that maximizes the difference between thermal and electrical output and will generally lead to greater resource needs, such as cooling water. Therefore, NRC's proposed microreactor limit was selected as the bounding value.
Plant Design	Is your reactor designed to be mobile?	None provided	Not evaluated	No	Not applicable	**No**	While the PPE representative value indicates that the reactor is not mobile, some designs may include mobile reactors. If so, there would be additional transportation and workforce related issues that would have to be considered.
Plant Design	If the reactor is designed to be transportable, what are the total number of shipments and weight of reactor, fuel, and its packaging?	None provided	Not evaluated	15 shipments, including the reactor, fuel, and core assembly	Not applicable	**30 shipments, including the reactor, fuel, and core assembly**	The largest value among the microreactor vendor responses, scaled to a 60 MWt reactor.

Table E.1. Microreactor PPE Data Sources and Methodology[(a)] (continued)

PPE Section	Parameter	Information Sources: NRC ANR GEIS Value (from Appendix D)	Information Sources: Internal Research Value (from Appendices B and C)	Information Sources: Vendor Bounding Value (from Appendix A)	Regulatory Limit Value	Microreactor Bounding Value	Source/Rationale
Plant Design (continued)	Describe your power conversion system.	None provided	Not evaluated	Varied; including Rankine Cycle, Brayton Cycle, and Air cooled DOWTHERM ™ heat transfer fluid		**N/A**	Power conversion does not itself have an environmental nexus.
	Will offsite power sources be required to maintain functioning of structures, systems, and components important to safety following loss of onsite AC power? If so, what transmission voltage would be required from offsite power sources?	Required. Offsite ROW 1,000 ft x 100 ft (new) or within or adjacent to existing ROW	Required, assuming compliance with General Design Criteria 17	None provided	General Design Criteria 17	**Required. Offsite ROW 1,000 ft x 100 ft (new) or within or adjacent to existing ROW**	The requirement for access to the existing onsite INL transmission system would bound all designs. Both substation and transmission interconnections are assumed to be required. Length and breadth of transmission line right-of-way and size of the switchyard will depend on final site location.
	What support facilities (fuel storage and handling, waste treatment, etc.) are necessary for your plant design?	None provided	Fuel storage and handling; waste treatment; reactor pre-heating and metal melting; control building; power conversion	Varied	Not applicable	**Fuel storage and handling; waste treatment; reactor pre-heating and metal melting; control building; power conversion**	Representative support facilities as informed by SME analysis and review of publicly available information on microreactors, small- to medium-sized advanced reactors, and vendor responses. The existence of these structures themselves does not have an environmental nexus; however, the land use requirements and resource needs associated with these facilities will have an environmental nexus and should be considered.
Plant Structure and Footprint	What is the tallest structure and what is the maximum structure height (structure, ft)? What is the stack height?	50 ft (structure) None provided (stack)	28 ft structure 45 ft stack height	28 ft structure 45 ft stack	Not applicable	**28 ft structure** **50 ft stack height**	Selected largest values from vendor responses and NRC ANR GEIS PPE to better bound potential visual, scenic, and land use impacts

Table E.1. Microreactor PPE Data Sources and Methodology[(a)] (continued)

PPE Section	Parameter	Information Sources: NRC ANR GEIS Value (from Appendix D)	Information Sources: Internal Research Value (from Appendices B and C)	Information Sources: Vendor Bounding Value (from Appendix A)	Regulatory Limit Value	Microreactor Bounding Value	Source/Rationale
Plant Structure and Footprint (continued)	What is the maximum depth of excavation?	50 ft	Not Evaluated	20 ft	Not applicable	**20 ft**	Selected largest value consistent with vendor responses. The NRC ANR GEIS value appears larger than necessary for the planned microreactor deployments.
	What is the temporary disturbed acreage during construction, including parking and laydown?	50 ac	10 ac	8 ac	Not applicable	**18 ac**	Selected values that bound vendor responses, with slight rounding up to account for potential larger projects. The NRC ANR GEIS value appears larger than necessary for the planned microreactor deployments.
	What is the permanent disturbed acreage, including parking lots, ponds, substations, and other plant support facilities?	30 ac	8 ac	7 ac	Not applicable	**8 ac**	Selected values that bound vendor responses. The NRC ANR GEIS value appears larger than necessary for the planned microreactor deployments.
	What is the maximum expected sound level due to construction activities, measured at 50 ft from the noise source?	None provided	101 dB at 50 ft	Question not asked	Not applicable	**101 dB at 50 ft**	Questionnaire did not include this parameter. SME estimate is from the Clinch River EIS PPE (NRC 2019c).
	Are there large quantities of any unique materials (perhaps items not normally used in general office or industrial buildings) that will be used in plant construction (e.g., graphite)? If so, what are these anticipated volumes?	None provided	Not evaluated	160 tons/15 m^3 lead; borated poly; graphite; sodium	Not applicable	**160 tons/15 m^3 lead; borated poly; graphite; sodium; 52.5 MT molten salt**	Responses did not pose any particular environmental challenges. Any particular unique materials are necessarily specific to a given design proposed for deployment. For purposes of impact analysis, the SME estimate of unique materials should bound applicable resource impacts.
Operational Parameters	What is the operational life for which the plant is designed? How long do you intend to operate the reactor prototype?	80 yr 2 to 20 yr operating cycle length	Not evaluated	30 yr 10 yr	Not applicable	**30 yr** **10 yr**	Selected longest vendor response value. Prototype deployment at INL would likely be shorter than the 80 yr operational period chosen by NRC for a commercial reactor.

Table E.1. Microreactor PPE Data Sources and Methodology[(a)] (continued)

PPE Section	Parameter	Information Sources: NRC ANR GEIS Value (from Appendix D)	Information Sources: Internal Research Value (from Appendices B and C)	Information Sources: Vendor Bounding Value (from Appendix A)	Regulatory Limit Value	Microreactor Bounding Value	Source/Rationale
Operational Parameters (continued)	Do you anticipate installing additional modules incrementally over time?	None provided	Not evaluated	No	Not applicable	**No**	While the PPE assumes that additional modules would not be added over time, particularly for demonstration projects, it is possible that multiple modules could be proposed for certain microreactor applications. This would have additive implications as well as potential cumulative impacts.
	What is the reactor heat transfer material (coolant)? How much is required initially/annually?	Water	52.5 MT molten salt initial loading 150 MT lead initial loading	Liquid lead 73 tons Initially 0 tons Annually	Not applicable	**52.5 MT molten salt initial loading** **150 MT lead initial loading**	Molten salt value obtained by scaling Molten Salt Reactor Experiment (MSRE) (8 MWt) coolant quantity to 60 MWt (ORNL 2015) Lead value obtained by scaling and rounding vendor response (30 MWt) to 60 MWt
	What is the anticipated technology (or technologies) for the normal plant heat sink?	None provided	Mechanical draft cooling tower	Varied	Not applicable	**Mechanical draft cooling tower**	It is anticipated that mechanical draft cooling towers, in general, will have the most resource-intensive type of plant heat sink.
	What are the maximum and average daily water use requirements for plant cooling and service water systems, including potable and sanitary water use (if required)?	1,000 gpm	335 gpm (average) For air-cooled reactors, 25 gpm	450 gpm (average)	Not applicable	**450 gpm (average)** For air-cooled reactors, 25 gpm	The bounding vendor value was chosen as the PPE value because it exceeds the SME calculated value. For air-cooled reactors, the PPE water use includes non-cooling uses, which were based upon scaling non-cooling-water use from the Clinch River EIS, and potable/sanitary use assumed to be 100 gpd per member of the vendor-provided or estimated operations work force.
	What are the expected characteristics of plant water discharges (if any)?	600 gpm	102 gpm For air-cooled reactors, 25 gpm	400 gpm	Not applicable	**400 gpm** For air-cooled reactors, 25 gpm	The vendor value was chosen as the PPE value for consistency with the water demand estimate. For air-cooled reactors, the PPE discharge includes non-cooling system wastewater and potable/sanitary wastewater, assumed to be equivalent to the water use rate for these purposes.

Table E.1. Microreactor PPE Data Sources and Methodology[(a)] (continued)

PPE Section	Parameter	Information Sources: NRC ANR GEIS Value (from Appendix D)	Information Sources: Internal Research Value (from Appendices B and C)	Information Sources: Vendor Bounding Value (from Appendix A)	Regulatory Limit Value	Microreactor Bounding Value	Source/Rationale
Operational Parameters (continued)	Blowdown temperature and constituent concentrations	Within applicable Clean Water Act limits	Not evaluated	Not Evaluated	Within applicable Clean Water Act limits	**Within applicable Clean Water Act limits**	Questionnaire did not include this parameter. Discharges mainly from plant blowdown are regulated under a Clean Water Act permit.
	What are the chemical and radionuclide constituents of the plant discharges, and maximum and expected concentrations/activities in the discharge (if available)?	Within applicable Clean Water Act limits	See Clinch River Table - Projected Blowdown Constituents and Concentrations (Table C.2).	None Provided	DOE O 458.1 (DOE-STD-1196 and DOE-STD-1153) (or 10 CFR Part 20 Appendix B) for both liquid and gaseous effluents and 40 CFR Part 61 Subpart H for gaseous effluents	**See Clinch River Table - Projected Blowdown Constituents and Concentrations, Table C.2.**	Clinch River ER provided anticipated constituents and concentrations associated with blowdown, which would be assumed to be the dominant portion of liquid nonradioactive waste. Not all of these constituents would be relevant to each microreactor design, but these values represent a reasonable estimate for values that could be included in a surrogate plant.
	What is the fuel source and size of auxiliary boilers, emergency power systems and standby power systems (if applicable) (fuel source, MW)?	None provided	Not evaluated	Diesel 50-150 kW Standby Power	Not applicable	**Two diesel 50–150 kW standby power generators**	The largest value from vendor responses was selected. Two generators are assumed for redundancy to power plant safety systems in the event of loss of offsite power.
	Emissions from construction equipment and standby power equipment during operations	Criteria pollutants are less than Clean Air Act de minimus levels	Not evaluated	Not evaluated	Criteria pollutants are less than Clean Air Act de minimus levels	**Criteria pollutants are less than Clean Air Act de minimus levels**	Questionnaire did not include this parameter. Clean Air Act requires a conformity determination for maintenance or nonattainment areas that exceed de minimus values. Not applicable to attainment areas, so this would be bounding for INL.

Table E.1. Microreactor PPE Data Sources and Methodology[(a)] (continued)

PPE Section	Parameter	Information Sources: NRC ANR GEIS Value (from Appendix D)	Information Sources: Internal Research Value (from Appendices B and C)	Information Sources: Vendor Bounding Value (from Appendix A)	Regulatory Limit Value	Microreactor Bounding Value	Source/Rationale
Operational Parameters (continued)	How much hazardous, radioactive, and mixed waste would be generated during operations, and where would it be dispositioned?	None provided	19 MT radioactive waste (fuel) 315 MT molten salt (mixed) Hazardous waste generation amount would be within the criteria of a small quantity generator	None provided	Small quantity generators produce more than 100 kilograms, but less than 1,000 kilograms of hazardous waste a month.	**19 MT radioactive waste (fuel)** **315 MT molten salt (mixed)**	Reference molten salt reactor consumes 1,930 kg of 19.7 percent enriched U and 3,290 kg of Th annually; scaled from reference reactor's power (500 MWt) to microreactor power (60 MWt). Assumes 30 yr demonstration. Initial loading of molten salt value taken by scaling MSRE (8 MWt) coolant quantity to 60 MWt (ORNL 2015). Assuming MSRE initial loading of 52.5 MT of molten salt would be replenished every 5 yr, two loadings would be needed for the assumed 80 yr demonstration. The molten salt spent fuel and coolant would be classified as either high-level mixed waste or mixed transuranic waste (depending on the spent fuel processing). These waste volumes reflect waste that would be generated from within the reactor vessel. For estimates of total radioactive waste generation (excluding spent fuel) see estimates of the total number of shipments and volume of radioactive waste. RCRA requires waste management for hazardous waste and sets a volume amount of generations of no more than 1,000 kg a month. This volume could be used as a bounding value.
	What is the stack exit velocity?	None Provided	Not evaluated	10 ft/s	Not applicable	**10 ft/s**	The largest value from vendor responses was selected.
	What amount of noise would be generated 50 ft from the source and at the site boundary?	65 dBA at site boundary	Not evaluated	None provided	Not applicable	**65 dBA at site boundary**	The value from the NRC estimate in ANR GEIS was selected and it is consistent with NRC Environmental Standard Review Plans (NRC 2013).

Table E.1. Microreactor PPE Data Sources and Methodology[(a)] (continued)

PPE Section	Parameter	Information Sources: NRC ANR GEIS Value (from Appendix D)	Information Sources: Internal Research Value (from Appendices B and C)	Information Sources: Vendor Bounding Value (from Appendix A)	Regulatory Limit Value	Microreactor Bounding Value	Source/Rationale
Fuel	What is the form of the fuel associated with your design?	None provided	Molten salt	Molten salt	Not applicable	**Molten salt**	Fuel types could include UO2, MOX, Metal (U, U alloys, Pu-containing alloys), TRISO, molten salt, uranium nitride, uranium carbide, QUADRISO, cermet, accident-tolerant fuel. Emission release mechanisms from molten salt are different from LWRs; expect that molten salt will have upper bounding impacts compared to other fuel technologies.
	What is the annual average fuel requirement (metric tons) per module?	None provided	0.5 MT (5 MT initial fuel loading)	0.5 MT	Not applicable	**0.5 MT (5 MT initial fuel loading)**	The largest value from vendor responses was selected.
	Where would fuel be obtained?	None provided	Not evaluated	Existing DOE supply	Not applicable	**Offsite commercial source**	Multiple vendors assumed that the fuel would come from an existing DOE supply at INL, while other vendors would source the fuel from offsite commercial sources. For purposes of developing a surrogate reactor, the PPE assumes that fuel would be obtained from offsite sources.
	What is the total number of shipments and MTU for unirradiated fuel shipped to reactor or site?	None provided	10 shipments over the 30 yr life of the plant. 45 MTU total	1 shipment, 3 MTU	Not applicable	**10 shipments over the 30 yr life of the plant.** **45 MTU total**	Unirradiated fuel shipments scaled to 60 MWt from surrogate SMR from Clinch River ESP (NUREG-2226, NRC 2019a, Table 6-4). MTU scaled to 1,000 MWt from Clinch River ESP (NUREG-2226, NRC 2019a, Table 6-10).
	Total number of shipments and volume of radioactive waste shipments from reactor/site?	None provided	49 shipments over the 30 yr life of the plant. Volume of each shipment is 2.34 m^3.	1 shipment, 14 m^3	Not applicable	**49 shipments over the 30 yr life of the plant. Volume of each shipment is 2.34 m^3. Total volume = 113 m^3**	Radioactive waste shipments and volume scaled to 60 MWt from surrogate SMR from Clinch River ESP (NUREG-2226, NRC 2019a, Table 6-14). These values are used as a bounding measure of radioactive waste generation (excluding spent fuel) but do not account for differences in design or unique waste streams from advanced reactors.

Table E.1. Microreactor PPE Data Sources and Methodology[(a)] (continued)

PPE Section	Parameter	Information Sources: NRC ANR GEIS Value (from Appendix D	Information Sources: Internal Research Value (from Appendices B and C)	Information Sources: Vendor Bounding Value (from Appendix A)	Regulatory Limit Value	Microreactor Bounding Value	Source/Rationale
Fuel (continued)	What is the radionuclide inventory for irradiated fuel at time of shipment (Ci/MTU by radionuclide)?	None provided	See Fission Product Inventory (Appendix C.7)	None provided	Not applicable	**Fission Product Inventory (Appendix C.7)**	See Appendix C.7
	How will the reactor, fresh fuel and other large components be transported to the site?	None provided	Truck	Truck or rail	Not applicable	**Truck**	Truck transportation is assumed based upon internal research value.
	Is the reactor designed to be refueled? If so, at what frequency (year)? What MTU per refueling?	None provided	5 MTU [full core refueling]	Yes, online and continuous refueling	Not applicable	**Yes, 5 MTU (full core refueling), online and continuous refueling**	Assumed that the reactor would be refueled in order to develop a more robust bounding impact. Online and continuous refueling was assumed, which may increase impacts associated with radioactive and nonradioactive emissions.
	What are the source terms for routine releases (if any) per module and design-basis accidents?	None provided	See Fission Product Inventory (Appendix C.7)	None provided	Not applicable	**Fission Product Inventory (Appendix C.7)**	See Appendix C.7. The analysis uses general cases, instead of specific designs, to calculate the radionuclide inventory.
	Are there any unique fuel storage or cooling requirements associated with the fuel?	None provided	Not Evaluated	None	Not applicable	**None**	No unique fuel storage or cooling requirements identified by microreactor vendors.
	How and where would spent fuel be dispositioned?	None provided	89 shipments of irradiated fuel over the 30 yr life of the plant. Onsite storage, or offsite storage or disposal	Onsite storage	Not applicable	**89 irradiated fuel shipments over 30 yr life of the plant.** **Offsite storage or disposal. Treatment, storage, and disposal in accordance with applicable legal requirements.**	HALEU and all spent fuel used for the Oklo application would stay at the INL site post-demonstration. (Oklo 2020c) Irradiated fuel shipments scaled to 60 MWt from surrogate SMR from Clinch River ESP (NUREG-2226, NRC 2019a, Table 6-10). The Clinch River ESP assumed that fuel would be dispositioned to Yucca Mountain. This assumption is not carried forward into this PPE, but the number of shipments is scaled as a bounding value.

Table E.1. Microreactor PPE Data Sources and Methodology[(a)] (continued)

PPE Section	Parameter	Information Sources: NRC ANR GEIS Value (from Appendix D	Information Sources: Internal Research Value (from Appendices B and C)	Information Sources: Vendor Bounding Value (from Appendix A)	Regulatory Limit Value	Microreactor Bounding Value	Source/Rationale
Workforce	How many workers will be onsite for construction?	150	150	None provided	Not applicable	**150**	The largest number of workers from NRC ANR GEIS was selected to bound impacts.
	What is the anticipated construction period?	None provided	6 months	24 months	Not applicable	**24 months**	The largest value from the vendor responses was selected and is consistent with the SME estimate.
	What is the number of total permanent staff to support operations?	50	27	None provided	Not applicable	**50**	The largest value from NRC ANR GEIS was selected to bound impacts.
	What is the number of temporary staff during refueling (if planned)?	100	21	None provided	Not applicable	**100**	The largest value from the NRC ANR GEIS was selected to bound impacts.
	What is the number of temporary staff during additional module installation (if planned)?	None provided	20	None provided	Not applicable	**N/A**	Assumed a single module for purposes of this PPE, thus no temporary staff are needed.
	What are the distances from radiation sources to the nearest involved worker?	None provided	500 ft	500 ft	Not applicable	**500 ft**	Internal research estimate is consistent with vendor response.

Table E.1. Microreactor PPE Data Sources and Methodology[(a)] (continued)

PPE Section	Parameter	Information Sources: NRC ANR GEIS Value (from Appendix D)	Information Sources: Internal Research Value (from Appendices B and C)	Information Sources: Vendor Bounding Value (from Appendix A)	Regulatory Limit Value	Microreactor Bounding Value	Source/Rationale
Decommissioning	Do you plan to decommission and remove the prototype from the INL site?	None provided	Not evaluated	Yes	Not applicable	**Yes**	It is assumed that the prototype would be decommissioned to bound impacts associated with land use, fuel, transportation, and workforce.
	What is the number of temporary staff during decommissioning (if planned)?	None provided	150	None provided	Not applicable	**150**	It is assumed that the number of staff needed during decommissioning would be similar to those needed during construction.
	What is the number of months from start of decommissioning to completion (if planned)?	None provided	Not evaluated	18 months	Not applicable	**18 months**	Selected the largest value from vendor responses.
	How much waste would be generated during decommissioning (if planned)?	None provided	Bounded by the waste streams evaluated in NUREG-0586	None provided	Not applicable	**Bounded by the waste streams evaluated in NUREG-0586**	The anticipated volumes of wastes evaluated in NUREG-0586 were based on industry decommissioning experience as of 2002. Appendix G of NUREG-0586, “Radiation Protection Considerations for Nuclear Power Facility Decommissioning” summarizes effluent releases for operating facilities and decommissioning facilities. Low-level waste volume estimates for decommissioning facilities are presented in Appendix K of NUREG-0586.

AC = alternating current; ANR = advanced nuclear reactor; CFR = *Code of Federal Regulations*; DOE = U.S. Department of Energy; ; EIS = environmental impact statement; ER = Environmental Report; ESP = early site permit; GDC = General Design Criteria; GEIS = generic environmental impact statement; HALEU = high-assay low-enriched uranium; HTGR = high-temperature gas-cooled reactor; INL = Idaho National Laboratory; LMR = liquid metal reactor; LWR = light-water reactor; MSR = molten salt reactor; MSRE = Molten Salt Reactor Experiment; NRC = U.S. Nuclear Regulatory Commission; PPE = plant parameter envelope; RCRA = Resource Conservation and Recovery Act; ROW = right-of-way; SME = subject matter expert; SMR = small modular reactor.

Table E.2 Small- to Medium-Sized Advanced Reactor PPE Data Sources and Methodology

PPE Section	Parameter	Information Sources: NRC ANR GEIS Value[a] (From Appendix D)	Internal Research Value (From Appendices B and C)	Vendor Bounding Value (From Appendix A)	Versatile Test Reactor Draft EIS Value	Regulatory Limit Value	Small- to Medium-Sized Advanced Reactor Bounding Value	Source/Rationale
Plant Design	What is your design type?	None provided		HTGR, BWR, LMR	GE Hitachi PRISM design	Not applicable	**Not Applicable for bounding parameters**	Plant type itself is not relevant to the environmental analysis; parameters therein that have an environmental interface are considered.
	How many units do you plan to install?	None provided	Not evaluated	1-4 units	1	Not applicable	**1**	While the PPE considers installation of one unit, multiple units may be proposed, for instance to demonstrate the capability of following increases in electricity demand over time. This would have potential impacts on the extent and timing of resource analyses, including cumulative impacts.
	What is the output of your design (per unit)?	60 MWt/20MWe	Not evaluated	950 MWt	300 MWt	Not applicable	**1,000 MWt**	Based upon the largest vendor response, 1,000 MWt was selected as a bounding value to account for potentially larger plants.
	Is your reactor designed to be mobile?	None provided	Not evaluated	No	No	Not applicable	**No**	Reactor is not assumed to be mobile.
	If the reactor is designed to be transportable, what are the total number of shipments and weight of reactor, fuel, and its packaging?	None provided	Not evaluated	N/A	N/A	Not applicable	**N/A**	Reactor is not assumed to be transportable.
	Describe your power conversion system.	None provided	Not evaluated	Rankine Cycle	N/A		**N/A**	Power conversion does not itself have an environmental nexus.
	Will offsite power sources be required to maintain functioning of structures, systems, and components important to safety following loss of onsite AC power? If so, what transmission voltage would be required from offsite power sources?	Required. Offsite ROW 1,000 ft x 100 ft (new) or within or adjacent to existing ROW	Required, assuming compliance with General Design Criteria 17	Not required	Yes, two 230 kV transmission lines available	General Design Criteria 17	**Two 230 kV transmission lines required. Offsite ROW 1,000 ft x 100 ft (new) or within or adjacent to existing ROW**	GDC 17 requires two offsite sources of power. The requirement for access to the existing onsite INL transmission system would bound all designs. Both substation and transmission interconnections are assumed to be required. Length and breadth of transmission line right-of-way and size of the switchyard will depend on final site location.

Table E.2. Small- to Medium-Sized Advanced Reactor PPE Data Sources and Methodology (continued)

PPE Section	Parameter	Information Sources: NRC ANR GEIS Value[(a)] (From Appendix D)	Information Sources: Internal Research Value (From Appendices B and C)	Information Sources: Vendor Bounding Value (From Appendix A)	Information Sources: Versatile Test Reactor Draft EIS Value	Information Sources: Regulatory Limit Value	Small- to Medium-Sized Advanced Reactor Bounding Value	Source/Rationale
Plant Design (continued)	What support facilities (fuel storage and handling, waste treatment, etc.) are necessary for your plant design?	None provided	Not evaluated	Cooling-water system; switchyard/ transformers; chemical/gas/ fuel storage, potable water supply; wastewater system, including retention basins and associated discharge equipment; liquid radwaste system; fire protection and emergency response buildings; Administration/ Maintenance Building(s); Security Facility; Chemistry and Meteorology Facility; Radioactive Waste Storage Facility (Region/ Country Dependent); various offsite facilities	Feedstock preparation facility, fuel fabrication facility, experiment support areas, post-irradiation examination facility, spent fuel treatment facility, onsite spent fuel pad	Not applicable	**Multiple support facilities**	Vendor responses included representative facilities that may be required associated with any given design.

Table E.2. Small- to Medium-Sized Advanced Reactor PPE Data Sources and Methodology (continued)

PPE Section	Parameter	Information Sources: NRC ANR GEIS Value[(a)] (From Appendix D)	Internal Research Value (From Appendices B and C)	Vendor Bounding Value (From Appendix A)	Versatile Test Reactor Draft EIS Value	Regulatory Limit Value	Small- to Medium-Sized Advanced Reactor Bounding Value	Source/Rationale
Plant Structure and Footprint	What is the tallest structure and what is the maximum structure height (structure, ft)? What is the stack height?	50 ft (structure) None Provided (stack)	Not evaluated	75 ft structure 87 ft stack height	90 ft (experiment support area) 190 ft (cooling chimneys)	Not applicable	**75 ft structure** **87 ft stack height**	Selected largest values from vendor responses to better bound potential visual, scenic, and land use impacts. Although the VTR is larger, aboveground PPE estimates are generally consistent; the height of the aboveground portion of the VTR stack is generally consistent with the vendor-provided estimates.
	What is the maximum depth of excavation?	50 ft	Not evaluated	155 ft	93 ft	Not applicable	**155 ft**	Selected largest value consistent with vendor responses.
	What is the temporary disturbed acreage during construction, including parking and laydown?	50 ac	60 ac	58 ac	100 ac	Not applicable	**100 ac**	Selected largest value consistent with vendor responses and the VTR estimated acreage.
	What is the permanent disturbed acreage, including parking lots, ponds, substations, and other plant support facilities?	30 ac	50 ac	43 ac	25 ac	Not applicable	**50 ac**	Selected largest value consistent with vendor responses, rounded up to consider potential additional acreage needed for air-cooling.
	What is the maximum expected sound level due to construction activities, measured at 50 ft from the noise source?	None provided	101 dB at 50 ft	Question not asked	Imperceptible at the INL site boundary and the closest receptor	Not applicable	**101 dB at 50 ft**	Questionnaire did not include this parameter. SME estimate is from the Clinch River EIS PPE (NRC 2019c).

Table E.2. Small- to Medium-Sized Advanced Reactor PPE Data Sources and Methodology (continued)

PPE Section	Parameter	Information Sources: NRC ANR GEIS Value(a) (From Appendix D)	Information Sources: Internal Research Value (From Appendices B and C)	Information Sources: Vendor Bounding Value (From Appendix A)	Information Sources: Versatile Test Reactor Draft EIS Value	Information Sources: Regulatory Limit Value	Small- to Medium-Sized Advanced Reactor Bounding Value	Source/Rationale
Plant Structure and Footprint (continued)	Are there large quantities of any unique materials (perhaps items not normally utilized in general office or industrial buildings) that will be utilized in plant construction (e.g., graphite)? If so, what are these anticipated volumes?	None provided	230 MT Graphite, 65 m^3 lead, 2,020 m^3 sodium	Graphite, 280 m^3 lead	No information provided	Not applicable	**280 m^3 lead**	Responses did not pose any particular environmental challenges. Any particular unique materials are necessarily specific to a given design proposed for deployment. For purposes of impact analysis, the SME estimate of unique materials should bound applicable resource impacts.
Operational Parameters	What is the operational life for which the plant is designed? How long do you intend to operate the reactor prototype?	80 yr 2 to 20 yr operating cycle length	Not evaluated	80 yr 80 yr	60 yr	Not applicable	**80 yr** **80 yr**	Selected longest vendor response value. Prototype deployment at INL would likely be shorter than the 80 yr operational period chosen by NRC for a commercial reactor.
	Do you anticipate installing additional modules incrementally over time?	None provided	Not evaluated	Yes	No	Not applicable	**Yes**	Vendor responses stated that multiple modules may be installed. This would have additive implications as well as potential cumulative impacts.
	What is the reactor heat transfer material (coolant)? How much is required initially/annually?	Water	870 MT molten salt, 65 m^3 lead, 2,020 m^3 sodium	Liquid metal (e.g., sodium, lead-bismuth); gas (e.g., helium); water	Sodium	Not applicable	**Various**	Molten salt value taken by scaling reference MSR (8 MWt) coolant quantity to 1,000 MWt (ORNL 2015) Lead value taken by scaling reference LFR (280 MWt) coolant quantity to 1,000 MWt (Cinotti et al. 2010) Sodium value taken by scaling reference SFR (400 MWt) coolant quantity to 1,000 MWt (Cabell 1980)

Table E.2. Small- to Medium-Sized Advanced Reactor PPE Data Sources and Methodology (continued)

PPE Section	Parameter	Information Sources: NRC ANR GEIS Value[a] (From Appendix D)	Information Sources: Internal Research Value (From Appendices B and C)	Information Sources: Vendor Bounding Value (From Appendix A)	Information Sources: Versatile Test Reactor Draft EIS Value	Information Sources: Regulatory Limit Value	Small- to Medium-Sized Advanced Reactor Bounding Value	Source/Rationale
Operational Parameters (continued)	What is the anticipated technology (or technologies) for the normal plant heat sink?	None provided	Mechanical draft cooling tower	Mechanical draft cooling tower or air-cooled condenser	Air-cooled heat exchangers	Not applicable	**Mechanical draft cooling tower**	It is anticipated that mechanical draft cooling towers, in general, will have the most resource-intensive type of plant heat sink.
	What are the maximum and average daily water use requirements for plant cooling and service water systems, including potable and sanitary water use (if required)?	1,000 gpm	5,850 gpm For air-cooled reactors, 415 gpm	7,500 gpm maximum; 4,200 gpm average	6.8 million gallons/yr/365 = about 18,000 gallons per day	Not applicable	**For water-cooled reactors, 5,850 gpm (maximum)** **5,850 gpm (average)** **For air-cooled reactors, 415 gpm**	Cooling-water use was estimated assuming the bounding reactor operates at 33 percent thermal efficiency, which is considered a lower bound on the efficiency. The 7,500 gpm vendor value seems excessively high given the power output and efficiency of the reactor. This water use estimate would also bound those reactor designs that are only using process heat rather than electricity. For air-cooled reactors, the PPE water use includes non-cooling uses, which were based upon scaling non-cooling-water use from Clinch River EIS, and potable/sanitary use assumed to be 100 gpd per member of the vendor-provided or estimated operations work force.

Table E.2. Small- to Medium-Sized Advanced Reactor PPE Data Sources and Methodology (continued)

PPE Section	Parameter	Information Sources: NRC ANR GEIS Value[(a)] (From Appendix D)	Information Sources: Internal Research Value (From Appendices B and C)	Information Sources: Vendor Bounding Value (From Appendix A)	Information Sources: Versatile Test Reactor Draft EIS Value	Information Sources: Regulatory Limit Value	Small- to Medium-Sized Advanced Reactor Bounding Value	Source/Rationale
Operational Parameters (continued)	What are the expected characteristics of plant water discharges (if any)?	600 gpm	1,775 gpm For air-cooled reactors, 415 gpm	Various, including no discharges anticipated	4.4 million gallons annually (about 8.4 gpm if continuous), including the volume required for personnel use and sanitation, fire protection water, and demineralized water. No water required for reactor operation	Not applicable	**1,775 gpm** **For air-cooled reactors, 415 gpm**	Discharge includes blowdown from the cooling towers with contributions from non-cooling systems and potable/sanity uses. The blowdown rate depends on the cycles of concentration during tower operation, which was assumed to be four. Two cycles of concentration would result in a larger discharge rate, but four cycles of concentration was selected for the PPE because this maximizes nonradioactive concentrations in the discharge. Minimizing the liquid discharge rate is likely to be desirable at INL. For air-cooled reactors, the PPE discharge includes non-cooling system wastewater and potable/sanitary wastewater, assumed to be equivalent to the water use rate for these purposes.
	Blowdown Temperature and Constituent Concentrations	Within applicable Clean Water Act limits	Not evaluated	Question not asked	No information provided	Within applicable Clean Water Act limits	**Within applicable Clean Water Act limits**	Questionnaire did not include this parameter. Discharges mainly from plant blowdown are regulated under a Clean Water Act permit.
	What are the chemical and radionuclide constituents of the plant discharges, and maximum and expected concentrations/activities in the discharge (if available)?	Within applicable Clean Water Act limits	See Clinch River Table - Projected Blowdown Constituents and Concentrations (Table C.2)	TBD	No information provided	DOE O 458.1 (DOE-STD-1196 and DOE-STD-1153) (or 10 CFR Part 20 Appendix B) for both liquid and gaseous effluents and 40 CFR Part 61 Subpart H for gaseous effluents	**See Clinch River Table - Projected Blowdown Constituents and Concentrations, Table C.2.**	Radionuclides in liquid discharge will be dependent on the specific reactor. Discharge can be assumed to be diluted to meet the 10 CFR Part 20 Appendix B, Table 2 limits at the point of discharge. Nonradioactive constituents of the liquid discharge will be determined by the source water used for cooling, by the cycles of concentration used in cooling-tower operation, and by additives used in plant processes. The Clinch River discharge was assumed to be representative with some consideration of the typical source water at INL. The microreactor PPE assumed four cycles of concentration, which would also be a reasonable bounding

assumption for the small- to medium-sized advanced reactors.

Table E.2. Small- to Medium-Sized Advanced Reactor PPE Data Sources and Methodology (continued)

PPE Section	Parameter	Information Sources: NRC ANR GEIS Value[(a)] (From Appendix D)	Information Sources: Internal Research Value (From Appendices B and C)	Information Sources: Vendor Bounding Value (From Appendix A)	Information Sources: Versatile Test Reactor Draft EIS Value	Information Sources: Regulatory Limit Value	Small- to Medium-Sized Advanced Reactor Bounding Value	Source/Rationale
Operational Parameters (continued)	What is the fuel source and size of auxiliary boilers, emergency power systems and standby power systems (if applicable) (fuel source, MW)?	None provided	50 MWt oil fired; 15 MWe Sentry turbine	4 MWe natural gas or diesel auxiliary boiler; 1 MWe standby power (gas, diesel, battery)	4.7 million ft^3 of propane per year	Not applicable	**50 MWt oil fired; 15 MWe Sentry turbine**	Selected based upon a review of publicly available documentation, including vendor websites and other literature.
	Emissions from construction equipment and standby power equipment during operations	Criteria pollutants are less than Clean Air Act de minimus levels	Not Evaluated	Not asked	Well below EPA PSD permitting threshold of 250 tons per year for a criteria pollutant	Criteria pollutants are less than Clean Air Act de minimus levels	**Criteria pollutants are less than Clean Air Act de minimus levels**	Questionnaire did not include this parameter. Clean Air Act requires a conformity determination for maintenance or nonattainment areas that exceed de minimus values. Not applicable to attainment areas, so this would be bounding for INL.

Table E.2. Small- to Medium-Sized Advanced Reactor PPE Data Sources and Methodology (continued)

PPE Section	Parameter	NRC ANR GEIS Value[a] (From Appendix D)	Internal Research Value (From Appendices B and C)	Vendor Bounding Value (From Appendix A)	Versatile Test Reactor Draft EIS Value	Regulatory Limit Value	Small- to Medium-Sized Advanced Reactor Bounding Value	Source/Rationale
		Information Sources						
Operational Parameters (continued)	How much hazardous, radioactive, and mixed waste would be generated during operations, and where would it be dispositioned?	None Provided	836 MT radioactive waste (fuel) 13,920 MT molten salt (mixed)	Various	45 spent driver fuel assemblies per year, initially stored onsite 18,460 total shipments	Not applicable	**836 MT radioactive waste (fuel)** **13,920 MT molten salt (mixed)**	Reference molten salt reactor consumes 1,930 kg of 19.7 percent enriched U and 3,290 kg of Th annually; scaled from reference reactor's nominal power (500 MWt) to 1,000 MWt. Assumes 30 yr demonstration. Initial loading of molten salt value taken by scaling reference MSR (8 MWt) coolant quantity to 1,000 MWt (ORNL 2015). Assuming initial loading of 870 MT of molten salt would be replenished every 5 yr, 16 loadings would be needed for assumed 80 yr demonstration. The molten salt spent fuel and coolant would be classified as either high-level mixed waste or mixed transuranic waste (depending on the spent fuel processing) Vendor responses indicate that fuel would come from existing INL feedstock. As a result, the PPE assumes that disposition would remain at INL. These waste volumes reflect waste that would be generated from within the reactor vessel. For estimates of total radioactive waste generation (excluding spent fuel), see estimates of the total number of shipments and volume of radioactive waste.
	What is the stack exit velocity?	None provided	Not evaluated	58 ft/s	No information provided	Not applicable	**58 ft/s**	The largest value from vendor responses was selected.
	What amount of noise would be generated 50 ft from the source and at the site boundary?	65 dBA at site boundary	Not evaluated	70 dBA at 50 ft from cooling tower; <55 dBA at site boundary	Imperceptible at the INL site boundary and the closest receptor	Not applicable	**65 dBA at site boundary**	The value from the NRC estimate in ANR GEIS was selected and it is consistent with NRC Environmental Standard Review Plans (NRC 2013).

Table E.2. Small- to Medium-Sized Advanced Reactor PPE Data Sources and Methodology (continued)

PPE Section	Parameter	Information Sources: NRC ANR GEIS Value[a] (From Appendix D)	Information Sources: Internal Research Value (From Appendices B and C)	Information Sources: Vendor Bounding Value (From Appendix A)	Information Sources: Versatile Test Reactor Draft EIS Value	Regulatory Limit Value	Small- to Medium-Sized Advanced Reactor Bounding Value	Source/Rationale
Fuel	What is the form of the fuel associated with your design?	None provided	Molten salt	Various, including metal fuel, TRISO in graphite blocks or pebbles, uranium oxide.	Uranium-plutonium-zirconium alloy fuel (U-20Pu-10Zr)	Not applicable	**Molten Salt**	Molten salt, TRISO, Uranium Oxide, HALEU, U-Zr alloy. Emission release mechanisms from molten salt are different from LWRs; expect that molten salt will have upper bounding impacts compared to other fuel technologies.
Fuel	What is the annual average fuel requirement (metric tons) per module?	None provided	8 MT	4.9 MT	1.8 MT annually (~1.3-1.4 MT uranium, ~0.4-0.54 MT plutonium, plus 10% Zr)		**8 MT**	The largest value from vendor responses was selected, scaled to 1000 MWt.
Fuel	Where would fuel be obtained?	None provided	Not evaluated	Various	U from within DOE complex and from commercial vendors, Pu from within the DOE complex. Fuel manufactured at either INL or SRS	Not applicable	**Offsite commercial source**	Multiple vendors assumed that the fuel would come from an existing DOE supply at INL, while other vendors would source the fuel from offsite commercial sources. For purposes of developing a surrogate reactor, the PPE assumes that fuel would be obtained from offsite sources.
Fuel	What is the total number of shipments and MTU for unirradiated fuel shipped to reactor or site?	None provided	432 shipments over the 80 yr life of the plant. 1,972 MTU total.	255 fuel blocks per module/year; 10 shipments per module/year	460-550 kg Pu and 1.61-1.92 MTU per year needed for feedstock. 3 assemblies per shipment. 22 shipments for initial loading, then 15 shipments annually = 922 total unirradiated fuel shipments for 60 yr operational life	Not applicable	**432 shipments over the 80 yr life of the plant.** **1,972 MTU total.**	Unirradiated fuel shipments scaled to 1,000 MWt from surrogate SMR from Clinch River ESP (NUREG-2226, NRC 2019a, Table 6-4). MTU scaled to 1,000 MWt from Clinch River ESP (NUREG-2226, NRC 2019a, Table 6-10). VTR refueling will occur more frequently than for demonstration reactors considered in this PPE to support its research mission. While the VTR value would bound impacts, this value is not representative of the anticipated fueling cycle for anticipated advanced reactor designs.

Table E.2. Small- to Medium-Sized Advanced Reactor PPE Data Sources and Methodology (continued)

PPE Section	Parameter	Information Sources: NRC ANR GEIS Value[a] (From Appendix D)	Information Sources: Internal Research Value (From Appendices B and C)	Information Sources: Vendor Bounding Value (From Appendix A)	Information Sources: Versatile Test Reactor Draft EIS Value	Information Sources: Regulatory Limit Value	Small- to Medium-Sized Advanced Reactor Bounding Value	Source/Rationale
Fuel (continued)	Total number of shipments and volume of radioactive waste shipments from reactor/site?	None provided	2,160 shipments over the 80 yr life of the plant. Volume of each shipment is 2.34 m^3. Total volume = 4,981 m^3.	~4 Total Shipments, ~24 M3	Up to 423 shipments annually = 25,380 shipments for 60 yr operational life	Not applicable	**2,160 shipments over the 80 yr life of the plant. Volume of each shipment is 2.34 m^3.** **Total volume= 4,981 m^3.**	Radioactive waste shipments and volume scaled to 1,000 MWt from surrogate SMR from Clinch River ESP (NUREG-2226, NRC 2019a, Table 6-14). These values are used a bounding measure of radioactive waste generation (excluding spent fuel) but do not account for differences in design or unique waste streams from advanced reactors Similar to the shipments of unirradiated fuel, the VTR parameter for radioactive waste shipments exceeds the PPE bounding parameter related to NRIC prototypes, based upon a different research and mission goal. While the VTR value would bound impacts, this value is not representative of the anticipated radioactive waste shipments for anticipated advanced reactor designs.
	What is the radionuclide inventory for irradiated fuel at time of shipment (Ci/MTU by radionuclide)?	None provided	See Fission Product Inventory (Appendix C.7)	No information provided	No information provided	Not applicable	**Fission Product Inventory (Appendix C.7)**	See Appendix C.7

Table E.2. Small- to Medium-Sized Advanced Reactor PPE Data Sources and Methodology (continued)

PPE Section	Parameter	Information Sources: NRC ANR GEIS Value[a] (From Appendix D)	Internal Research Value (From Appendices B and C)	Vendor Bounding Value (From Appendix A)	Versatile Test Reactor Draft EIS Value	Regulatory Limit Value	Small- to Medium-Sized Advanced Reactor Bounding Value	Source/Rationale
Fuel (continued)	How will the reactor, fresh fuel and other large components be transported to the site?	None provided		Truck or rail	Unirradiated fuel to be shipped via Secure Transportation Asset (STA; truck transport only)	Not applicable	**Truck**	Truck transportation is assumed for both microreactors and small- to medium-sized advanced reactors
	Is the reactor designed to be refueled? If so, at what frequency (year)? What MTU per refueling?	None Provided	Daily refueling of 10.6 kg enriched U and 18 kg Th; annual requirement 3.9 MT enriched U, 6.6 MT Th.	Various, depending on whether counting total amount of fuel or annual fuel consumption	Typically <17 driver fuel assemblies (~1/4 of core) replaced per refueling/ three ~100-day operating cycles per year/~0.6 MT Th per refueling Up to 45 driver fuel assemblies per year/ ~1.8 MT Th	Not applicable	**Daily refueling of 10.6 kg enriched U and 18 kg Th; annual requirement 3.9 MT enriched U, 6.6 MT Th.**	Assumed that the reactor would be refueled in order to develop a more robust bounding impact. Online and continuous refueling assumed, which may increase impacts associated with radioactive and nonradioactive emissions. Continuous refueling quantity scaled from reference MSR (500 MWt) to 1,000 MWt.
	What are the source terms for routine releases (if any) per module and design-basis accidents?	None provided	See Fission Product Inventory (Appendix C.7)	TBD	No information provided	Not applicable	**Fission Product Inventory (Appendix C.7)**	See Appendix C.7. The analysis uses general cases, instead of specific designs, to calculate the radionuclide inventory.

Table E.2. Small- to Medium-Sized Advanced Reactor PPE Data Sources and Methodology (continued)

PPE Section	Parameter	Information Sources: NRC ANR GEIS Value[a] (From Appendix D)	Information Sources: Internal Research Value (From Appendices B and C)	Information Sources: Vendor Bounding Value (From Appendix A)	Information Sources: Versatile Test Reactor Draft EIS Value	Regulatory Limit Value	Small- to Medium-Sized Advanced Reactor Bounding Value	Source/Rationale
Fuel (continued)	Are there any unique fuel storage or cooling requirements associated with the fuel?	None provided	Sodium or lead pool depending on coolant type; separate storage area required for liquid metal reactors	TBD	Cool in-vessel for ~1 year, then wash off external sodium and transfer to cask on spent fuel pad to cool further. Fuel then chopped, consolidated, sodium removed, and diluted (likely with scrap metal from driver fuel assembly). Mixture packaged in containers, placed in storage casks, and stored on spent fuel pad until shipped offsite	Not applicable	**Sodium or lead pool depending on coolant type; separate storage area required with liquid metal reactors**	No unique fuel storage or cooling requirements identified by small- to medium-sized advanced reactor vendors. Fuel from liquid metal reactors will require compatible pool/storage vessel.
	How and where would spent fuel be dispositioned?	None provided	3,944 shipments of irradiated fuel over the 80 yr life of the plant. Onsite storage, or offsite storage or disposal	Various; cask stored for future disposition, either onsite, intermediate offsite storage, bore-hole, or permanent repository. Recycling is possible.	Ultimately an offsite storage or disposal facility	Not applicable	**3,944 shipments of irradiated fuel over the 80 yr life of the plant. Onsite storage, or offsite storage or disposal**	HALEU and all spent fuel used for the Oklo application would stay at the INL site post-demonstration. (Oklo 2020c) Irradiated fuel shipments scaled to 1,000 MWt from surrogate SMR from Clinch River ESP (NUREG-2226, NRC 2019a, Table 6-10).

Table E.2. Small- to Medium-Sized Advanced Reactor PPE Data Sources and Methodology (continued)

PPE Section	Parameter	Information Sources: NRC ANR GEIS Value[a] (From Appendix D)	Information Sources: Internal Research Value (From Appendices B and C)	Information Sources: Vendor Bounding Value (From Appendix A)	Information Sources: Versatile Test Reactor Draft EIS Value	Information Sources: Regulatory Limit Value	Small- to Medium-Sized Advanced Reactor Bounding Value	Source/Rationale
Workforce	How many workers will be onsite for construction?	150	909 maximum onsite at one time construction workforce; maximum total construction workforce of 1,363 at peak	900	1,400	Not applicable	**909 maximum onsite at one time construction workforce; maximum total construction workforce of 1,400 at peak**	Scaled down the values analyzed for the Clinch River ESP to a 1,000 MWt reactor.
	What is the anticipated construction period?	None provided	45 months	54 months	51 months	Not applicable	**54 months**	Consistent with vendor response
	What is the number of total permanent staff to support operations?	50	207	420	600	Not applicable	**207**	Scaled down the values analyzed for the Clinch River ESP to a 1,000 MWt reactor. The VTR is a research reactor that will involve a larger staff to support research operations; this is a higher value than would be expected for demonstration prototypes. The construction workforce estimate is consistent with the VTR data (DOE 2020c). and bounds the VTR value peak. The VTR value of 600 also includes the workforce associated with activities outside of reactor operations (e.g., fuel fabrication, post-irradiation examination of experiments, experiment design and support staff) that are not relevant to the demonstration reactors considered in this PPE.

Table E.2. Small- to Medium-Sized Advanced Reactor PPE Data Sources and Methodology (continued)

PPE Section	Parameter	Information Sources: NRC ANR GEIS Value[(a)] (From Appendix D)	Information Sources: Internal Research Value (From Appendices B and C)	Information Sources: Vendor Bounding Value (From Appendix A)	Information Sources: Versatile Test Reactor Draft EIS Value	Regulatory Limit Value	Small- to Medium-Sized Advanced Reactor Bounding Value	Source/Rationale
Workforce (continued)	What is the number of temporary staff during refueling (if planned)?	100	413	<125	No information provided	Not applicable	**413**	Scaled down the values analyzed for the Clinch River ESP to a 1,000 MWt reactor.
	What is the number of temporary staff during additional module installation (if planned)?	None provided	413	900	N/A	Not applicable	**413**	Scaled down the values analyzed for the Clinch River ESP to a 1,000 MWt reactor.
	What are the distances from radiation sources to the nearest involved worker?	None provided	500 ft	~20 m	Nearest uninvolved worker: 330 ft	Not applicable	**~20 m**	Consistent with vendor response
Decommissioning	Do you plan to decommission and remove the prototype from the INL site?	None provided	Not evaluated	TBD	No information provided	Not applicable	**Yes**	It is assumed that the prototype would be decommissioned to bound impacts associated with land use, fuel, transportation, and workforce.
	What is the number of temporary staff during decommissioning (if planned)?	None provided	Not evaluated	450	No information provided	Not applicable	**450**	Consistent with vendor response
	What is the number of months from start of decommissioning to completion (if planned)?	None provided	Not evaluated	10 yr	No information provided	Not applicable	**10 yr**	Selected the largest value from vendor responses.
	How much waste would be generated during decommissioning (if planned)?	None provided	Bounded by the waste streams evaluated in NUREG-0586		No information provided	Not applicable	Bounded by the waste streams evaluated in NUREG-0586	The anticipated volumes of wastes evaluated in NUREG-0586 were based on industry decommissioning experience as of 2002. Appendix G of NUREG-0586, “Radiation Protection Considerations for Nuclear Power Facility Decommissioning” summarizes effluent releases for operating facilities and decommissioning facilities. Low-level waste volume estimates for decommissioning facilities are presented in Appendix K of NUREG-0586.

Table E.2. Small- to Medium-Sized Advanced Reactor PPE Data Sources and Methodology (continued)

		Information Sources						
PPE Section	Parameter	NRC ANR GEIS Value[(a)] (From Appendix D)	Internal Research Value (From Appendices B and C)	Vendor Bounding Value (From Appendix A)	Versatile Test Reactor Draft EIS Value	Regulatory Limit Value	Small- to Medium-Sized Advanced Reactor Bounding Value	Source/Rationale

(a) Note that because NRC's ANR GEIS PPE/SPE generally focuses on microreactors, many of the parameters are not applicable to these small- to medium-sized advanced reactors. However, some of the parameters would provide appropriate bounding values regardless of the size of the reactor and are therefore included in this table.

AC = alternating current; ANR = advanced nuclear reactor; BWR = boiling-water reactor; CFR = Code of Federal Regulations; DOE = U.S. Department of Energy; EIS = environmental impact statement; ER = Environmental Report; ESP = early site permit; GDC = General Design Criteria; GEIS = generic environmental impact statement; HALEU = high-assay low-enriched uranium; HTGR = High-Temperature Gas-Cooled Reactor; INL = Idaho National Laboratory; LFR = Lead-Cooled Fast Reactor; LMR = Liquid Metal Reactor; LWR = light-water reactor; MSR = Molten Salt Reactor; MSRE = Molten Salt Reactor Experiment; NRC = U.S. Nuclear Regulatory Commission; PPE = plant parameter envelope; RCRA = Resource Conservation and Recovery Act; ROW = right-of-way; SME = subject matter expert; SMR = small modular reactor; TBD = to be developed; Th = Thorium; TRISO = tri-structural isotropic.