SI Appendix to Krall *et al.*, "Nuclear Waste from Small Modular Reactors

Section 1: Relations between neutron leakage and reactor power, criticality, and size

In assessing the relation between reactor size and nuclear waste generation, it is useful to understand the parameters driving the nuclear reactions within the core. The equations presented here are summarized from Ch. 3 of Glasstone & Sesonske (1994). The thermal power output is driven by the rate at which fission reactions are occurring, which, in turn, depends upon the nuclear properties, the distribution, and the total mass of isotopes in the core (e.g. the cross-sections of the fuel, coolant, and moderator isotopes), as well as the flux of the neutrons being released through fission reactions. Essentially,

Equation 1:
$$Power_{thermal} = \frac{V \cdot \Sigma_f \cdot \boldsymbol{\Phi}}{3.1 \cdot 10^{10}}\ watts$$

where $\sum_f$ is the macroscopic fission cross section of the fissile nuclides (i.e. the microscopic fission cross section multiplied by density of fissile material in the core -$N \cdot \sigma_f$), which, when multiplied by the volume of the fuel (V), reflects the probability that an incoming neutron triggers a fission event in the material. The Φ is the neutron flux in $s^{-1} \cdot cm^{-2}$.

A reactor is "critical" when its fuel can sustain a fission chain reaction without undergoing significant power—or reactivity—excursions. That is, the neutron flux is temporally constant.

Equation 2:
$$\boldsymbol{\Phi(t)} = \Phi_i e^{\frac{(k-1)t}{l}}$$

Equation 3:
$$\frac{d\Phi}{dt} = \frac{(\boldsymbol{k}-1)\Phi(t)}{l}$$

Eqn 2 & 3 shows that conditions for criticality are met when the effective multiplication factor, *k*, is equal to one. That is, the neutron flux does not increase or decrease over time with successive generations of neutrons and so is in steady state, i.e. $\frac{d\emptyset}{dt} = 0$. The effective multiplication factor, *k*, is a function of the rate of neutron supply through fission reactions, as balanced by neutron consumption by fuel elements or loss by leakage from the core periphery:

Equation 4:
$$\boldsymbol{k} = \eta f p \epsilon \boldsymbol{P_{NL}}$$

The unspecified variables in Eqn 2-4 are defined in Table S1. The following discussion will focus on the probability of non-leakage (P_{NL}), or the probability that a neutron will not leak from the periphery of the core. The other variables in Eqn 4 are defined in Table S1. The probability of non-leakage P_{NL}, which reflects the efficiency of neutron utilization by fuel elements, is an important consideration for SMR design because it is decreased in smaller reactor systems. Glasstone & Sesonske (1994) express the relation between P_{NL} and reactor size for a simple, homogeneous reactor as:

Equation 5:
$$\boldsymbol{P_{NL}} = \frac{1}{1 + L^2 B^2}$$

where L^2 is the diffusion length of neutrons in a particular medium (a function of the total, macroscopic cross-sections for fuel, coolant, and moderator materials), and B^2 is the spatial distribution of the neutron flux, which depends on the properties of the reactor materials and the reactor geometry (i.e. dimensions and and surface area-to-volume ratio). For a cylinder, which is the common shape for power reactor cores, with a cross-sectional radius *R* and height *H:*

Equation 6:
$$B^2 = \left(\frac{2.405}{R}\right)^2 + \left(\frac{\pi}{H}\right)^2$$

Typically for reactors, $H = 2.4 \cdot R$, so that Equation 6 simplifies to:

Equation 7:

$$B^2 = \frac{7.5}{R^2}$$

Ultimately, Equations 5 – 7 show that the probability of leakage is proportional to the square of the neutron diffusion length (L^2) and inversely proportional to the reactor radius (R^2).

Equation 8:

$$P_L = 1 - P_{NL} = \frac{1}{\frac{R^2}{7.5L^2} + 1}$$

The P_L for **two reactors of distinct size but similar fuel–moderator–coolant compositions** (i.e. $L_{SMR} = L_{GW}$, where subscripts "SMR" and "GW" denote small- and gigawatt-scale reactors, respectively), can be compared by dividing $P_{L,SMR}$ by $P_{L,GW}$, which will be simplified in Equation 10:

Equation 9:

$$\frac{P_{L,SMR}}{P_{L,GW}} = \frac{R_{GW}^2 + 7.5L^2}{R_{SMR}^2 + 7.5L^2}$$

Since the neutron diffusion length in normal water (L = 0.0275 m) is much smaller than the radius of a LWR core (0.86 and 1.3 m for iPWR and standard PWR, respectively). In other words, $L_{water} << R$ so that Eqn 9 can be further simplified to:

Equation 10:

$$\frac{P_{L,SMR}}{P_{L,GW}} \sim \left(\frac{R_{GW}}{R_{SMR}}\right)^2$$

Although L, ideally, should ideally reflect the diffusion length in a system that includes the fuel and structural material, in addition to the moderating water, neglecting L in Eqn. 9 leads to an adequate approximation for comparison between a gigawatt-scale PWR and an iPWR: ($\frac{P_{L,SMR}}{P_{L,GW}}$ = 0.07/0.03 = 2.33) ~ ($\left(\frac{R_{GW}}{R_{SMR}}\right)^2$ = $(1.3/0.86)^2$ = 2.29). Ultimately, this shows that, compared to the gigawatt-scale reactor, SMR leakage grows quadratically with decreasing core radius. This is significant because P_L factors into the effective multiplication factor for these reactors (*k*) and because *k*, in turn, has an exponential impact on the neutron flux and, therefore, the ability to sustain corecriticality.

The neutron diffusion length (L) and, consequently, the P_L is less certain for those non-water SMR designs that are not yet finalized. In general, L is larger for graphite- than for water-moderated reactors; nevertheless, the negative correlation between P_L and reactor size holds for molten salt reactor designs (Robertson et al., 1971). On the other hand, P_L for liquid metal-cooled fast reactors, will be primarily driven by the large L under unmoderated conditions, such that P_L for fast reactors is large, at least 4% but up to 25% (Qvist, 2013) regardless of the reactor size. Hence, neutron leakage will be significantly enhanced in both water and non-water SMRs. Although SMR designs invoke neutron reflectors and elevated fuel enrichments to counteract the adverse effects that the small reactor size has on core reactivity and fuel burnup, these design changes have important consequences for SNF and LILW management systems.

Table S1: Meaning of variables

Variable	Definition	Variable	Definition
V	Volume of fuel in the core	$N{\cdot}\sigma_f$	Product of atomic density of material and probability that a nuclide will absorb a neutron and subsequently fission
Σ_f	Macroscopic fission cross section ($N{\cdot}\ \sigma_f$)	η	Number of neutrons released by fission per neutron absorbed by the fuel
Φ	Neutron flux	l	Average time between the emission of a fission neutron and its absorption

k	Coefficient of reactivity (ratio of neutrons absorbed in one generation to neutron absorbed in preceding generation)	p	Resonance escape probability (probability that collisions with moderator will slow neutrons down to the thermal energy spectrum)
P_L, P_{NL}	Leakage, non-leakage probability	ϵ	Fast fission factor (correction for neutrons made available through fission induced by fast neutrons in a thermal reactor)
B	Buckling (a measure of the curvature of the spatial distribution of the neutron flux)		
L	Neutron diffusion length	f	Thermal utilization factor (ratio of thermal neutron absorbed in fuel to total thermal neutrons absorbed, incl. parasitic captures)
H, R	Height, Radius of core		

Reference:

Glasstone, S. & Sesonske, A. (1994). *Nuclear Reactor Engineering* (4th ed., Vol. 1). Chapman & Hall.

Robertson, R.C. et al., 1971. Conceptual Design Study of a Single-Fluid Molten Salt Breeder Reactor ORNL-4541. Union Carbide Nuclear Division, Oak Ridge National Laboratory. http://moltensalt.org/references/static/downloads/pdf/ORNL-4541.pdf

Qvist, S. A. (2013). *Safety and core design of large liquid-metal cooled fast breeder reactors* (Doctoral dissertation, UC Berkeley). https://escholarship.org/content/qt64g748hn/qt64g748hn.pdf

Section 2

Waste volume calculations to support Section 4

The calculation of SNF, long- and short-lived LILW volumes for each of the three SMRs, as well as a gigawatt-scale PWR for comparison, is described here. In addition, SNF burnup and mass are calculated for the NuScale iPWR and the AP1000 gigawatt-scale PWR. In general, energy-equivalent SNF volumes, in m^3/GW_{th}-yr, are calculated by dividing the volume of the active core by its thermal power multiplied by the residence time of fuel in the core. Long- and short-lived LILW volumes for the three SMR designs are calculated from the design specifications of their reactor vessels and other near-core components or coolants, typically in the shape of a (hollow) cylinder. Justification for the classification these materials as SNF, long- or short-lived LILW is provided in the main text.

Pressurized Water Reactors (PWRs)

To determine the energy-equivalent SNF volume for a Westinghouse **AP1000**, a core volume of 31 m^3 was calculated from the height of an active core (*i.e.* the length of a fuel rod; 4.27 m; Table S2) and equivalent core diameter (3.04 m; Westinghouse, 2011). The residence time of fuel in the 3400 MW_{th} reactor is 4.5 years. Hence:

$$\frac{31\ m^3}{3.4\ GW_{th} \times 4.5\ years} = 2\ m^3/GW_{th} \cdot year$$

Fuel burnup calculations are shown in Table S2 and long- and short-lived LILW volume calculations in Table S3.

Near-core reactor components exposed to a neutron fluence $>10^{21}$ neutrons/cm^2 will be activated to levels corresponding to long-lived LILW or Greater-than-Class-C, as defined by the NRC (Main text Section 4.3.1; Mancini *et al.*, 1994). For the current PWR fleet, these include the core shroud, barrel, and upper and lower grid plates; whereas components further than 25 cm from the edge of the active core, including the thermal shield and the pressure vessel, may not reach long-lived LILW activation levels (Figure S1; Figure 1; Sandia National Laboratories, 2008; Mancini *et al.*, 1994; Love *et al.*, 1995). Nevertheless, some countries—Sweden, for instance—plan to dispose of thermal shields and pressure vessels as long-lived LILW (*e.g.*, Herschend, 2013). Component volumes for the calculation of long-lived LILW from a 3400 MW_{th} PWR calculation, as shown in Table S3, were obtained from (Mancini *et al.*, 1994).

Short-lived LILW volumes for a 3400 MW_{th} PWR were calculated by converting mass data provided by Hansson (2013) to volumes, applying densities of 2.5 and 8 g/cm^3 to concrete and steel, respectively (Table S3).

NuScale iPWR

Although NuScale has recently increased the power capacity of their iPWR module to 250 MW_{th} (77 MW_{elec}; NuScale Power LLC, 2020b), the dimensions used for this analysis are those for the 160 MW_{th} iPWR specified in the NRC license application (50 MW_{elec}; NuScale Power LLC, 2020c). Each reactor core would be loaded with 37 reduced-length fuel assemblies (~2 m-long; Table S2).

Since fuel burnup details were redacted from the publicly available license application of the NuScale reactor, a burnup of ~34 MWd/kg is here calculated using the fuel rod dimensions, linear power density, and reactor operating parameters provided for this iPWR (Table S2; NuScale Power LLC, 2020c). For the SNF volume calculation, a core radius and height of 0.88 m and 2 m, respectively, were used to determine

the NuScale iPWR core volume (4.87 m^3). Provided a reactor power of 160 MW_{th} and a six year-long fuel residence time, an energy-equivalent SNF volume of 5.1 m^3/GW_{th}·year is derived (Tables A2.1-.2).

Although pure uranium will occupy a smaller volume of the core than as calculated above, our approach to calculating the fuel volume accounts for the entire contents of a SNF storage or disposal canisters, including:

- the fuel matrix (*e.g.* UO_2 is less dense than is pure uranium),
- the materials that mechanically stabilize the fuel matrix (zirconium-cladding and stainless-steel assemblies), and
- the void space between fuel rods stabilized in a single assembly

This approach is valid, because SNF canister designs typically call for packaging of in-tact assemblies.

The volume of long-lived LILW that might arise from decommissioning a 160 MW_{th} iPWR was estimated from the component dimensions listed in Table S3. The iPWR core barrel and pressure vessel were treated as hollow cylinders with a height equivalent to that of the fuel assembly (2.4 m; Table S3, Figure 1). The volume of the iPWR neutron reflector was derived by subtracting the area of the active core (1.7 m^2) from the area of a solid circle (radius 0.88 m) and multiplying the result by the reflector height (2.3 m; Table S3, Figure 1). The thickness of the iPWR core support plates were assumed to be similar to those of the 3400 MW_{th} PWR (Mancini *et al.*, 1994), and this was multiplied by the radial area of the reflector to obtain a volume. In total, one iPWR will generate 2.7 m^3 or 5.1 m^3 of long-lived LILW, depending on whether the pressure vessel will be activated to long-lived LILW levels (Section 4.3.1). This is equivalent to 0.29 m^3/GW_{th}-yr and 0.53 m^3/GW_{th}-yr, respectively (Table S3).

Short-lived LILW from a NuScale iPWR will be dominated by steel from the pressure and containment vessels (Figure S2), in particular the >20 m-tall regions above the reactor core that may not be neutron-activated but will become contaminated by radionuclides in the primary coolant (Section 4.4.1). The iPWR containment vessel, though not in direct contact with the primary coolant during normal operation, will be submerged in a reactor pool filled with water that, due to discharge of primary coolant during refuelling outages, will contain more radioactivity than the SNF cooling pond of PWR (NuScale, 2020a). In addition to becoming contaminated by radionuclides in the reactor pool, the iPWR containment vessel will become neutron-activated, although neutron flux data is unavailable for this component. Ultimately, the iPWR pressure and containment vessels will generate 17 and 43 m^3 of short-lived decommissioning LILW, respectively, equivalent to 6.9 m^3/GW_{th}-yr (Table S3).

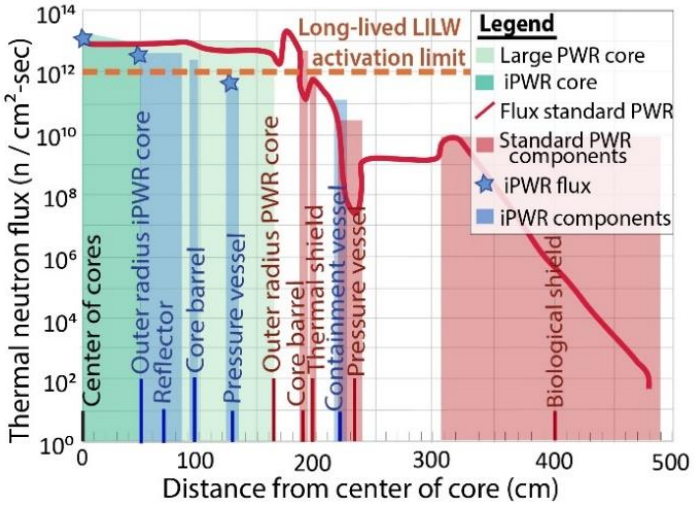

Figure S1: Radial neutron flux modelled for a standard PWR (Mancini et al., 1994), alongside fluxes (where available) for various points in an iPWR (NuScale, 2020b), indicating that near-core iPWR components may be activated to levels similar to those of a standard PWR.

Figure S2: To-scale vertical cross-section of one of the twelve, 160 MW_{th} iPWR modules that will sit in a joint reactor pool at a notional NuScale power station.

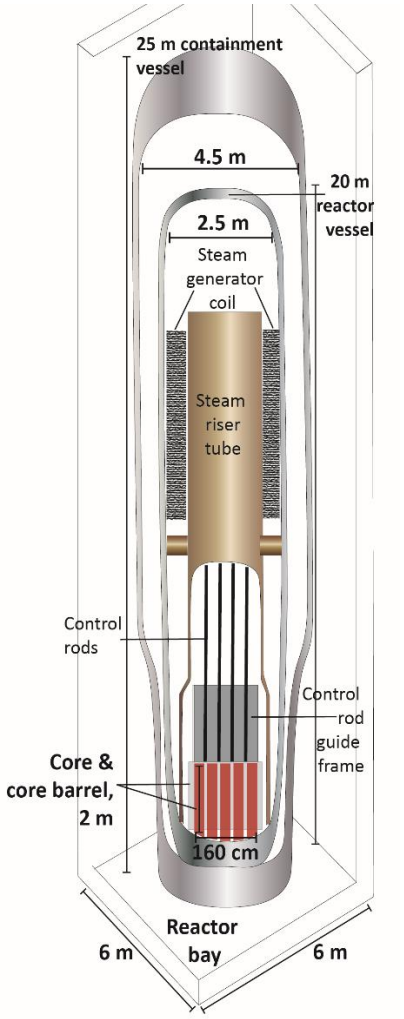

Table S2: Parameters and equations used to calculate fuel burnup and annual SNF discharge from the NuScale and AP1000 reactors using design parameters listed in their respective license applications, available through the NRC.

Specification	**NuScale[1]**	**AP1000 PWR[2]**
Avg. Linear Power Density (LPD, kW/m)	8.2	19
Length Fuel Rod (L_{rod} , m)	2	4.3
Rods per assembly (R)	264	264
Assemblies per core (A)	37	157
Thermal output core (MW_{th}) = LPD x L_{rod} x R x A	**160**	**3325**
Total number of irradiation cycles (I, cycles / assembly)	3	3
Cycle length (C_D, days / cycle)	730	548
Mass of assembly (M_{asmbl}, kg / assembly)	283	611
Burnup (MWd / kg) = (MWth x I x C_D) / (A x M_{asmbl})	**34**	**57**
Energy-equivalent SNF discharge (MT / GW_{th}-year) = 365 / Burnup	**11**	**6.4**
Mass of fuel (UO_2) in core (M_{core}, ton) = M_{asmbl} x A	10.5	96
Number reactor modules per station (RM)	12	1
Cycle length (C_Y, years / cycle)	2	1.5
Annual SNF discharge (tonnes / (station-year)) = (M_{core} x RM) / (C_Y x I)	**21**	**21**

[1]Most figures found in Table 4.4-2 of NuScale Power, LLC (2020c)

[2]Most figures found in Table 4.1-1 of https://www.nrc.gov/docs/ML0715/ML071580895.pdf

Table S3: Design parameters for NuScale iPWR and gigawatt-scale PWR (3400 MWth), used to calculate volumes of short- and long-lived LILW.

Design parameters for LILW estimation	**NuScale[1,2]**	**Standard PWR[3]**
Height of active core, h_{core} (fuel assembly length, cm)	244	420
Radial area of active core, A_{core} (m^2)	1.7	8.9
Reflector / Shroud height, h_{refl} (cm)	233	N/A
Reflector / Shroud radius, r_{refl} (cm)	88.2	171
Core support plate thickness, h_{SP} (cm)	5.1	5.1
Core barrel inner / outer radius, r_{barrel} (cm)	91.5/99	182/188
Pressure vessel inner / outer radius, r_{PV} (cm)[2]	122/134	220/241
Upper pressure vessel height, $h_{PV,upper}$ (m)	20	
Containment vessel radius, $r_{con,\ inner}$ / $r_{con,\ outer}$ (m)	2.16/2.29	n/a
Containment vessel height, h_{con} (m)	25	n/a
Long-lived LILW volumes (m^3)	**NuScale (calculated)**	**Standard PWR[4]**
Upper + lower core support plate: $2 \cdot h_{SP} \cdot \pi \cdot r^2_{refl}$	0.294	0.855
Reflector / Shroud: $h_{refl} \cdot (\pi \cdot r^2_{refl} - A_{core})$	1.71	1.99
Core barrel: $h_{core} \cdot \pi \cdot (r^2_{barrel,\ outer} - r^2_{barrel,\ inner})$	0.745	3.49
Pressure vessel, lower: $h_{core} \cdot \pi \cdot (r^2_{PV,outer} - r^2_{PV,\ inner})$	2.3	see below
Total volume (min/max, m^3)	2.75 / 5.05[5]	6.3
Electric-equivalent (m^3/ MWe)	0.055 / 0.10	5.8E-03
Energy-equivalent (m^3/GW_{th}-year)	0.29 / 0.53	3.1E-02

Short-lived LILW volumes	Volume	Mass[7]	Volume[7]	Mass[8]
Pressure vessel[6]: $h_{PV,upper} \cdot \pi \cdot (r^2_{PV,inner} - r^2_{PV,outer})$	17	138	63	500
Containment vessel[6]: $h_{con} \cdot \pi \cdot (r^2_{con,\ inner} - r^2_{con,\ outer})$	43	343	n/a	n/a
Activated concrete, primarily biological shield	n/a	n/a	304	760
Primary systems (pipes, valves, pumps, electrical, etc.)	n/a	n/a	236	1885
Total (volume or mass, m^3, tonnes)	60	480	600	3600
Electric-equivalent (m^3/MWe or tonnes/MWe)	1.2	10	0.55	3.4
Energy-equivalent (m^3/GW_{th}-year or tonnes/GW_{th}-year)	6.9	56	3.3	19

[1]Table 4.1-2 in NuScale (2020c)

[2]Table 5.3-1 in NuScale (2020a)

[3] Table 15 in Mancini *et al.* (1994)

[4]Volumes taken from Table 13 of Mancini *et al.* (1994), rather than calculated from above dimensions

[5]Pressure vessel included and excluded in minimum and maximum estimates, respectively

[6]For short-lived LILW calculation, iPWR pressure vessel height taken as the 20 m above the active core

[7]Volume and mass conversions assume density of 8 and 2.5 g/cm^3 for steel and concrete, respectively

[8]Table L-3 in Hansson (2013)

Sodium-cooled fast reactor

A methodology similar to that described above was employed to calculate the waste volumes, by type and reactor (Figure 5), for the molten salt and sodium reactors described in Section 4.

The **Toshiba 4S** (30 MWth) was selected for the fast-spectrum SMR analysis because an adequate level of design detail was made available in the pre-license application submitted by Toshiba to the U.S. NRC (Toshiba Corporation, 2008; Figure S3.b). An approach similar to that described above was taken to calculate the volume of fuel discharged from this reactor in the quoted fuel cycle, with a fuel residence time of *30 years*. The diameter and height of the active core are 0.95 m and 2.5 m, respectively, thus the core volume is 1.77 m^3.

Hence, the energy-equivalent volume of fuel discharged by a 4S reactor will be:

$$\frac{1.77\ m^3}{0.030\ GW_{th} \times 30\ years} = 2.0\ m^3/(GW_{th} \cdot year)$$

As is the case for the (i)PWR, this calculation accounts for the fuel cladding and assemblies that will contain the irradiated fuel.

The reflector and shielding assemblies are assumed to become long-lived LILW through neutron activation. Their volume was calculated by subtracting the volume of the active core (1.77 m^3) from the volume calculated from the diameter of the reactor vessel (3.5 m) and the height of the active core (24 m^3; Figures A2.3.b-c). The reflector and shielding assemblies will have a lifetime similar to that of the fuel residence time (30 years):

$$\frac{(24.0 - 1.77)\ m^3}{0.030\ GW_{th} \times 30\ years} = 24.7\ m^3/(GW_{th} \cdot year)$$

The volume of the sodium primary coolant was not provided in the Toshiba 4S application. It was here calculated by assuming that the sodium occupies the entire, above-core volume of the reactor vessel (height 24 m, diameter 3.5 m). We neglect the fraction that fills the void space between the fuel and the shielding assemblies because the core height (2.5 m) is much shorter than the primary coolant loop (24 m). We further assume that the lifetime of the coolant will approach that stated for a 4S power station (*i.e.* 60 years). That is, when the reactor is refueled after 30 years, we assume that the sodium coolant can be used for the second core-loading.

$$\frac{\left[24 \cdot \pi \cdot \left(\frac{3.5}{2}\right)^2 - 24.1\right] m^3}{0.030\ GW_{th} \times 60\ years} = 115\ m^3/(GW_{th} \cdot year)$$

Finally, the volume of LILW from the 4S reactor vessel was estimated by using the same dimensions listed above, except that 3.5 m was used as the outer diameter of a cylinder that is 2.5 cm-thick. The volume of 6.5 m^3 was converted to an energy equivalent of 7.3 m^3 / GW_{th}-year, given the stated reactor vessel lifetime of 30 years (Toshiba Corporation and Central Research Institute of Electric Power Industry, 2013). The 4S reactor vessel is classified as short-lived LILW in Figures 2 & 4 because it is likely to be contaminated by radionuclides carried in the sodium primary coolant, whereas neutron activation may be limited by flux attenuation of the reflector and shielding assemblies immediately surrounding the core.

Molten Salt Reactors

Unlike the water- and sodium-cooled reactors, molten salt reactors have a liquid fuel that flows throughout the entire vessel, which also contains a graphite moderator/reflector. Hence, the volume of the graphite moderator for the 400 MWth **Terrestrial IMSR** was calculated by assuming that it occupies the entire volume of the *active* core (3.4 m diameter, 4 m height → 37 m^3; Terrestrial Energy, 2016; Figure S3.a). To estimate the volume of the liquid fuel-coolant, 37 m^3 was subtracted from that of the entire reactor vessel (3.5 m diameter, 7 m height → 67 m^3) to arrive at 31 m^3. Since the lifetime of the fuel and graphite in the IMSR are 7 years, the energy equivalent volumes of graphite and fuel reduce to 13 and 11 m^3/GW_{th}-year, respectively.

The lifetime of the IMSR reactor vessel is, likewise, 7 years. This vessel has an outer diameter of 3.55 m, an inner diameter or 3.5 m, and a height of 7 m. Hence, the energy-equivalent volume of long-lived LILW generated by the reactor vessel will amount to 1.0 m^3/GW_{th}-year.

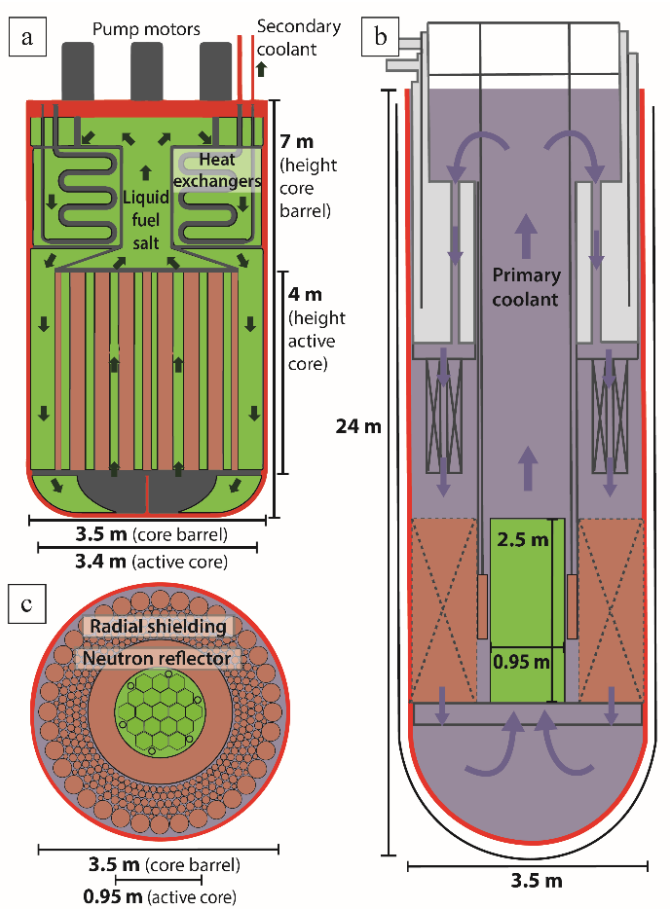

Figure S3: To-scale illustrations of two non-water-cooled, or "advanced", SMR designs, including (a) the 400 MW_{th} Integral Molten Salt Reactor design by Terrestrial Energy, (b) the 30 MW_{th} 4S design by Toshiba, a sodium-cooled fast reactor, and (c) a cross-section showing the geometry of radial shielding and neutron reflector assemblies around the 4S core. The components are color-coded according to the waste categorizations shown in Figure 5.

Uncertainties

These waste estimates are an adequate means to compare cross-compare fuel cycles but are not nearly accurate enough to plan for reactor decommissioning, waste packaging, and the long-term safety assessment of a notional repository. In particular, the mined repository volume will far exceed those stated above due to the addition of packaging and barrier materials.

We have also neglected the waste that will arise from the reactor internals. For instance, SMRs will have heat exchangers located within the reactor vessel, typically above the active core and in contact with the primary coolant.

In addition, the graphite moderator and reflector-shielding subassemblies will not be solid cylinders, because they must be designed to accommodate the through-flow of reactor coolant. In particular, the graphite moderator in a molten salt reactor will have holes drilled through it, so our approach over-estimates the graphite waste volumes.

Finally, our analysis neglects the waste streams that will arise from chemically treating the reactive sodium and fluoride-salt SMR coolants and cleansing the reactor vessels of these substances. Ultimately, our analysis might serve as a lower-limit waste estimation for the analyzed non-LWR SMRs.

Calculation of SNF composition, decay heat, radiotoxicity, fissile isotope concentration

To cross-compare SMRs and standard LWR in terms of metrics relevant to SNF disposal, the isotopic composition of representative SNF—i.e. that which would result from the initial enrichment and burnup combinations quoted by SMR designers—was needed. The appropriate data drawn from the open literature and from calculation tools and data libraries available with the Oak Ridge National Laboratory Scale-ORIGEN software (Skutnik et al., 2015). The Origami module distributed with this software package, given simple input parameters like initial fuel enrichment and final burnup, can generate SNF isotopic data for water-moderated reactors (i.e. PWRs and iPWRs). SNF isotopic composition were projected out to important future time intervals by coupling the Origami results to the Origen module.

The Origami module, however, lacked SNF isotopic data for molten salt reactors and for liquid metal-cooled fast reactors. Literature data for molten salt reactor SNF compositions were limited to designs that call for higher initial fuel enrichments (e.g. ≥20% ^{235}U and/or ^{239}Pu, Ahmad et al., 2015) than the Terrestrial Energy IMSR-400 design assessed in this study. Choe et al. (2018) presented model results for plutonium isotopes in IMSR-400 SNF but, lacking data for fission products and other transuranic isotopes, these results could not be used to calculate the decay power and radiotoxicity of the SNF. To navigate the incomplete data for molten salt reactors, Origami was used to generate SNF isotopics for PWR fuel enriched to 3 wt% ^{235}U irradiated to a burnup of 14 MWd/kg[1]. This is similar to the fuel cycle specifications provided by Choe et al. (2018), except that an IMSR-400 would be moderated by graphite rather than light water. Nevertheless, the fissile isotope composition of the SNF depleted in the Origami module (^{235}U + ^{239}Pu = 2.1 wt%) was similar to that calculated from the results provided by Choe et al. (2018; 1.6 wt%). Therefore, it is sufficient to extrapolate results from LWR depletion calculations to a graphite-moderated, thermal-spectrum molten salt reactor in order to perform a high-level assessment of the metrics relevant to SNF disposal.

However, an LWR-based depletion model is less appropriate to apply to a fast-spectrum reactor, such as the Toshiba 4S. To illustrate the SNF disposal consequences for this SMR, the relevant data was compiled from IAEA (2007) and from Kuwagaki et al. (2020).

[1] Choe *et al*. (2018) analyze two types of IMSR fuel cycles: a "once-through" and a "feed-seed-breed" cycle. Our analysis focuses on the former, because it is most similar to the fuel cycle described in pre-application materials submitted to the NRC (*e.g.* Terrestrial Energy USA, 2020). The two references describe a fuel cycle using 2wt% ^{235}U-enriched start-up fuel, to which 5wt% ^{235}U-enriched make-up fuel is added until the fuel salt volume increases by 50% relative to the initial loading. Hence, the total fuel volume contains two-parts 2 wt% ^{235}U and one-part 5 wt% ^{235}U fuel. A 3 wt%-enriched fuel is used for the Origami fuel enrichment parameter because it is the solution (p) to the mixing equation 0.02*(2/3) + 0.05*(1/3) = p*(1); p = 0.03.

The results, obtained from the afore-discussed sources and used to create Figure 4, are shown in Table S4. The SNF fissile concentration and decay power were extracted directly from the Origami output files for the relevant fuel compositions and burnups, whereas the radiotoxicity was calculated by converting the activity of the waste (in Bq units) to radiotoxicity (in Sv units) using the ICRP dose conversion factors (Eckerman et al., 2012). Additional details on this conversion are provided in Table S5. Decay heat and radiotoxicity are shown at 100 and 10,000 years, respectively, similar to the timing of peak buffer temperatures and of corrosion-accelerated canister failure in the safety assessments for common geologic repository designs (Sections 5.1.2 and 5.1.3).

Table S4: Fuel cycle parameters (enrichment and burnup) used as inputs for the depletion calculations and resulting SNF re-criticality, heat, and radiotoxicity for input into Figure 7 of main text. SNF isotopics for the PWR, iPWR, and IMSR-400 fuel cycles were obtained from the Origami-ORIGEN simulations, whereas results for a 4S-like, fast neutron SMR were found in the literature.

	Enrichment - Burnup (wt % - MWd/kg)	SNF fissile concentration ($^{235}U+^{239}Pu$, wt%)	Decay power at 100 yr post-discharge (watts/GW-yr)	Radiotoxicity at 10ky post-discharge (Sv/GW-yr)
PWR	4.5 - 50	1.3	3850	$3.9 \cdot 10^7$
iPWR	5 - 33	2.3	3000	$4.6 \cdot 10^7$
IMSR-400	3 - 14	2.1[1]	2740	$6.7 \cdot 10^7$
4S	18 - 34	17[2]	5830[3]	$(1.8 \cdot 10^8)$[4]

[1]Reflects wt% of fissile isotopes in the total mass of heavy metal fuel isotopes (primarily ^{238}U), as processed into a stable UO_{2+X} waste form (Section 6.1.4) and neglecting mass contribution from non-fuel isotopes in the Li-F salt.
[2]Table XIV-4 in IAEA (2007)
[3] Figure 6 in Kuwagaki et al. (2020)*
[4] Figure 11 in Kuwagaki et al. (2020)*
* Note that Kuwagaki et al. (2020) model a Pu-fueled fast reactor with a burnup of 63 MWd/kg, whereas the 4S fuel quoted by Toshiba cycle calls for ^{235}U-fuel and a lower burnup, which may entail lower decay power and long-term radiotoxicity.

Table S5: Calculation of total radiotoxicity at 10,000 years post-discharge for SNF of burnup 50, 33, and 14 MWd/kg, selected to represent that associated with standard PWRs, iPWRs, and small molten salt reactors. The most active isotopes present in the various fuels at 10,000 years were extracted from the Origami results and multiplied by an isotope-specific dose conversion factor obtained from Eckerman et al. (2012). The results were then summed and converted to energy-equivalent, total radiotoxicity (in Sv/GW-yr) by normalizing the total radiotoxicity against the SNF burnup.

Nuclide	**Dose Conversion Factor (DCF, Sv/Bq)**	**Activity (Bq/tonne SNF)**			**Sv/tonne SNF (DCF x Activity, Sv/tonne SNF)**		
		50 BU	**33 BU**	**14 BU**	**50 BU**	**33 BU**	**14 BU**
U-234	4.9E-08	1.5E+11	8.8E+10	5.3E+10	7.3E+03	4.3E+03	2.6E+03
Np-237	1.1E-07	7.2E+10	4.4E+10		7.9E+03	4.8E+03	
Np-239	8.0E-10	7.7E+11	1.5E+11		6.2E+02	1.2E+02	
Pu-239	2.5E-07	1.1E+13	1.1E+13	7.5E+12	2.8E+06	2.6E+06	1.9E+06
Pu-240	2.5E-07	9.4E+12	5.8E+12	2.8E+12	2.4E+06	1.4E+06	7.0E+05
Pu-241	4.7E-09	3.3E+10	2.0E+09		1.5E+02	9.2E+00	
Pu-242	2.4E-07	1.5E+11	4.6E+10		3.7E+04	1.1E+04	
Am-241	2.0E-07	3.3E+10	2.0E+09	4.4E+07	6.5E+03	4.0E+02	8.8E+00
Am-243	2.0E-07	7.7E+11	1.5E+11		1.5E+05	2.9E+04	
Tc-99	7.8E-10	8.1E+11	5.1E+11	2.3E+11	6.3E+02	4.0E+02	1.8E+02

Zr-93	2.8E-10	1.1E+11	6.9E+10	3.0E+10	3.1E+01	1.9E+01	8.5E+00
Nb-93	1.2E-10	1.1E+11	6.7E+10	3.0E+10	1.3E+01	8.1E+00	3.6E+00
Pa-233	8.7E-10	7.2E+10	4.4E+10		6.2E+01	3.8E+01	
		Total Radiotoxicity (Sv/tonne SNF)			5.4E+06	4.1E+06	2.6E+06
		Energy-equivalent radiotoxicity (Sv/GW-yr)			3.9E+07	5.8E+07	6.7E+07

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Section 3

DOE/OR/01-2793&D1

Feasibility of In Situ Decommissioning of Fuel and Flush Salt at the Molten Salt Reactor Experiment, Oak Ridge National Laboratory, Oak Ridge, Tennessee

This document is approved for public release per review by:

Teresa D. Fancher	9/11/18
UCOR Classification & Information Control Office	Date

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Feasibility of In Situ Decommissioning of Fuel and Flush Salt at the Molten Salt Reactor Experiment, Oak Ridge National Laboratory, Oak Ridge, Tennessee

Date Issued—October 2018

Prepared for the
U.S. Department of Energy
Office of Environmental Management

URS | CH2M Oak Ridge LLC
under contract DE-SC-0004645

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CONTENTS

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FIGURES

TABLES

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ACRONYMS

CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CLSM	controlled low-strength material
DOE	U.S. Department of Energy
EM	Office of Environmental Management
EPA	U.S. Environmental Protection Agency
ESD	Explanation of Significant Differences
FDT	fuel drain tank
FFT	fuel flush tank
FS	Feasibility Study
ISD	in situ decommissioning
LUC	land use control
MSRE	Molten Salt Reactor Experiment
ORNL	Oak Ridge National Laboratory
ROD	Record of Decision
SRS	Savannah River Site
TRU	transuranic
WIPP	Waste Isolation Pilot Plant

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1. INTRODUCTION

The purpose of this report is to evaluate the feasibility of in situ decommissioning (ISD) of the fuel and flush salts at the Molten Salt Reactor Experiment (MSRE) located at the Oak Ridge National Laboratory (ORNL) in Oak Ridge, Tennessee. ISD is the entombment of a radiologically or chemically contaminated facility that remains under institutional control of the U.S. Government. The ISD concept has been a recognized decommissioning option by the U.S. Nuclear Regulatory Commission since the 1970s. The U.S. Department of Energy (DOE) and Office of Environmental Management (EM) have begun evaluating more sites for an ISD option since the successful implementation at Idaho, South Carolina, and Washington State. Currently, the *Record of Decision for Interim Action to Remove Fuel and Flush Salts from the Molten Salt Reactor Experiment Facility at the Oak Ridge National Laboratory, Oak Ridge, Tennessee* (DOE/OR/02-1671&D2) (Fuel and Flush Salts ROD) requires the fuel and flush salts to be removed and stored at ORNL. This evaluation is needed to gain a better understanding of the risks and benefits of ISD versus removal of the fuel and flush salts.

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2. BACKGROUND

MSRE is located at ORNL in Melton Valley about seven miles south of Oak Ridge, Tennessee (Fig. 1). MSRE was a graphite-moderated, liquid-fueled reactor that operated from June 1965 to December 1969 to investigate the practicality of a large-scale molten salt breeder reactor concept. The fuels tested by MSRE when operational were trace amounts of plutonium (< 1 Ci) and fluoride salts of uranium (U)-235 and U-233 (Fuel and Flush Salts ROD).

Fig. 1. Molten Salt Reactor Experiment aerial view.

The MSRE reactor loop consisted of a reactor vessel, primary heat exchanger, pump, and associated piping located in the heavily shielded reactor cell (Fig. 2). When the reactor was shut down for extended periods, the molten fuel salt was drained to two fuel drain tanks (FDTs), FDT-1 and FDT-2, located in the heavily shielded drain tank cell adjacent to the reactor cell. The reactor system was then flushed with a flush salt to remove residual pockets of fuel salt. The flush salt, which became cross contaminated with a small amount of residual fuel salt after each flush, was drained to the fuel flush tank (FFT), also located in the drain tank cell.

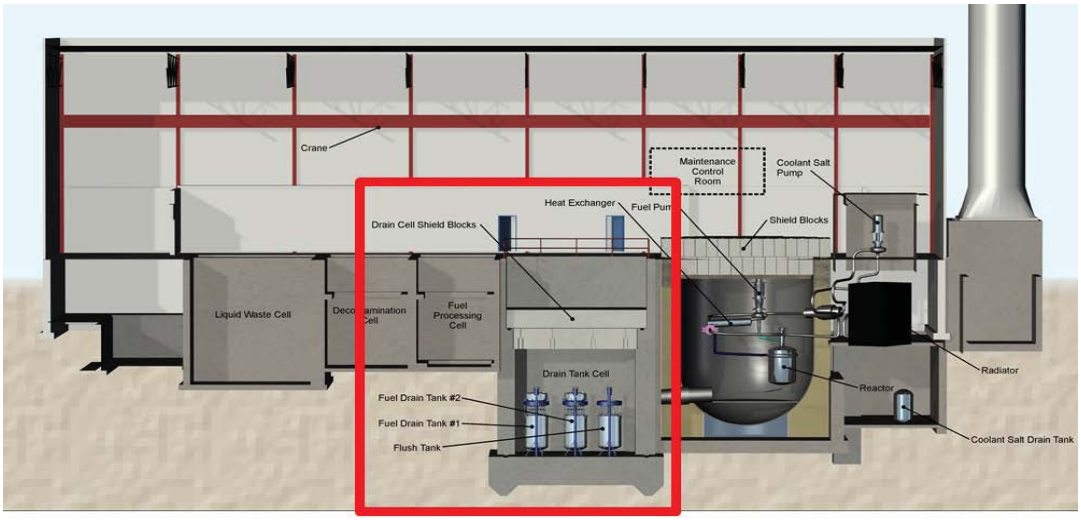

Fig. 2. Molten Salt Reactor Experiment cross section.

When reactor operations ceased in December 1969, the fuel salt was drained in approximately equal volumes into FDT-1 and FDT-2. The reactor loop was flushed, and the flush salt, containing less than 2% of the uranium, was drained to the FFT. The three tanks are shown in Fig. 3. The salts were then allowed to cool and solidify such that the reactor could no longer be operated. Density of the solidified blocks of rock salt in the tanks is very close to ordinary concrete (DOE-OR-01-2496, *Engineering Evaluation of Options for Molten Salt Reactor Experiment Defueled Coolant Salts*). The most recent forecast of the solidified salts indicate that over 98% of the emissions, including secondary buildup, arise from the 13,000 Ci of cesium (Cs)-137 + barium (Ba)-137m + strontium (Sr)-90 + yttrium (Y)-90 that remain in the salt today and result in a 1000 R/hr exposure rate near the salt tanks. The 8-ft-tall FDTs have a solid block of salt at a depth of about 30 in. from the bottom of the tanks. Fluorine (F_2) gas that is radiolytically generated by decay constantly builds in the large headspaces of the salt tanks and is removed periodically onto chemical traps.

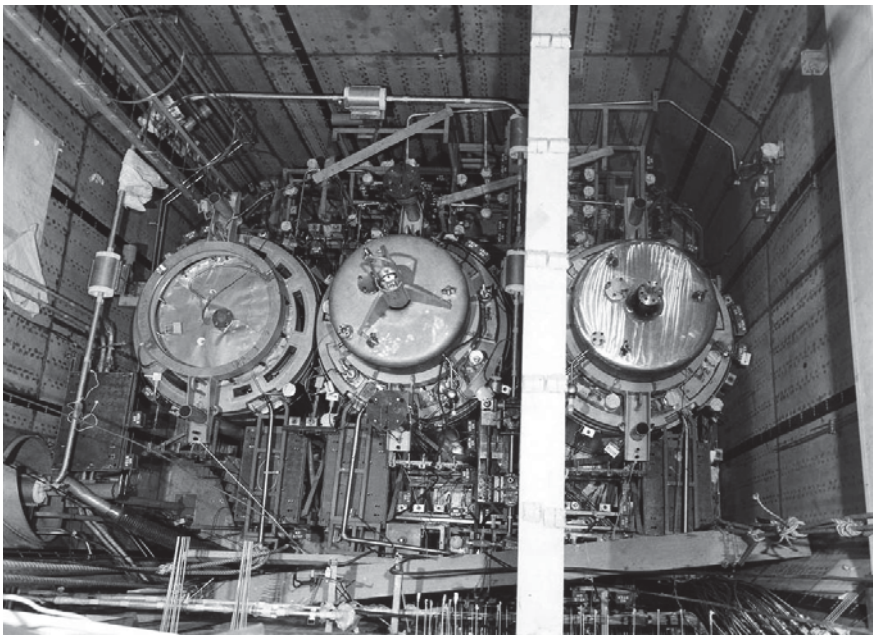

Fig. 3. Three salt tanks in the drain tank cell.

In 1994, a significant migration of uranium from the stored fuel and flush salts was discovered. Gas samples were obtained from the off-gas system piping in the vent house in March 1994. The gas samples indicated high concentrations of F_2 and uranium hexafluoride (UF_6) gases in the off-gas system piping. Gamma scanning in the charcoal bed cell in May 1994 indicated a significant deposit of uranium in the auxiliary charcoal bed. The migration of the uranium posed significant safety concerns. Plans were made for returning MSRE to a safe shutdown condition, including removing the F_2 and UF_6 gases and subliming (transformation of solid to gas form) any solid UF_6 deposits.

As a result of these problems, the following three Comprehensive Environmental Response, Compensation, and Liability Act of 1980 (CERCLA) response actions have been implemented:

- Reactive gases were removed from the drain tanks and connected piping systems and equipment using vacuum under a time-critical removal action. This removal action was completed in February 1999, other than the continued removal of newly generated gases, and documented in the *Removal Action Report on the Molten Salt Reactor Experiment Time-Critical Removal Action at Oak Ridge National Laboratory, Oak Ridge, Tennessee* (DOE/OR/01-1623&D2).
- The uranium fluoride deposit and carbon fluoride compounds were removed from the auxiliary charcoal bed under the non-time-critical *Action Memorandum for Uranium Deposit Removal at the Molten Salt*

Reactor Experiment Facility at the Oak Ridge National Laboratory, Oak Ridge, Tennessee (DOE/OR/02-1488&D2) and the *Removal Action Work Plan for Uranium Deposit Removal at the Molten Salt Reactor Experiment Facility at Oak Ridge National Laboratory, Oak Ridge, Tennessee* (DOE/OR/01-1735&D2). This removal action was completed in January 2002, and documented in the *Removal Action Report for Uranium Deposit Removal at the Molten Salt Reactor Experiment at Oak Ridge National Laboratory, Oak Ridge, Tennessee,* (DOE/OR/01-1918&D2).

- Under the Fuel and Flush Salts ROD, the following four tasks were completed:
  - — The salts were melted and chemically treated
  - — The molten salts were fluorinated to remove uranium
  - — The uranium was condensed into cold traps and transferred to sodium fluoride (NaF) traps
  - — NaF traps loaded with the uranium were placed into storage containers and transferred to ORNL Building 3019A for interim storage

The *Explanation of Significant Differences for the Record of Decision for Interim Action to Remove Fuel and Flush Salts from the Molten Salt Reactor Experiment Facility at the Oak Ridge National Laboratory, Oak Ridge, Tennessee* (DOE/OR/01-2088&D2) (ESD) deleted the requirement to convert the separated U-233 to an oxide form, substituting storage of the removed uranium fuel in Building 3019A at ORNL as the remedial action. Thus, requirements relative to uranium defueling were completed when the uranium was delivered to Building 3019A in NaF traps, as described in the *Phased Construction Completion Report for the Removal and Transfer of Uranium from the Molten Salt Reactor Experiment at the Oak Ridge National Laboratory, Oak Ridge, Tennessee* (DOE/OR/01-2256&D1). Removal of the defueled salts is part of the scope of this remedial action, but was deferred to a later date due to the inability to transfer the salt using the designed method.

After uranium was removed from the salt by fluorination, final defueling was declared to be sufficient, as of May, 2008. Once the salt was defueled, actions to transfer the salts to shielded canisters for storage at the ORNL Solid Waste Storage Area 5 were attempted. However, due to an inability to transfer the molten salt using the original transfer process design and equipment, this activity never occurred. The defueled salt resides today in three tanks fabricated from Hastelloy N alloy, located in the drain tank cell. Based on the following reasons, the salts will be left in place until an alternate path can be developed:

- The salt has been sufficiently defueled and will no longer support significant UF_6 generation.
- The salt is a dry solid, stored in a safe and easily monitored configuration.
- The salt tanks are stored in an extremely robust underground structure.
- The substantial cost of removal requires deferral of other critical remediation work that poses a greater and more immediate risk to workers and to the environment.

Since disposition for disposal of the salts has not yet been formally approved, this evaluation of ISD is being performed.

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3. STATE OF IN SITU DECOMMISSIONING AT DOE SITES

A summary of recent DOE ISD efforts are documented in *DOE EM Strategy and Experience for In Situ Decommissioning* (September 2009). The objective of this document is to provide a common understanding of ISD as practiced by DOE, covering regulations relevant to ISD across the complex. In addition to the successful ISD case studies presented in the DOE EM Strategy document, it also evaluated an additional 125 legacy facilities within the DOE complex that could be future candidates for ISD. The evaluation determined that an ISD end-state option for these 125 legacy facilities could result in a cumulative cost savings to DOE, ranging from $1.5 billion to $3 billion.

Summarized in the table below are the most recent ISD activities performed at DOE sites across the complex. In all cases presented below regarding legacy reactors, the fuel was removed prior to ISD. Land use for all the facilities presented below were considered "Site retained by U.S. DOE in perpetuity." The ISD end-state option was selected over other methods due to cost and risks associated with the complete removal and disposal of the facilities.

Table 1. State of ISD at other DOE sites

Year	Facilities/DOE site	Note	Decommission option
2009–2011	105-C Disassembly Basin/SRS	Water in basin was evaporated. Rod hangers were disposed in basin, which was filled with grout. The above-grade structure remains in place.	ISD Land Use Control-Monitored Natural Attenuation
	105-P Reactor/SRS	Early Action Record of Decision approving ISD using the CERCLA remedial process. Construction debris, heat exchangers, and some equipment were disposed of as LLW in SRS SWDF. Below-grade portion of facility was physically stabilized with concrete. Reactor vessels were located below grade, but above water table and were also filled with concrete or grout. Above-grade structure was left in place, openings sealed, and new roofs installed.	
	105-R Reactor/SRS		
2011	Heavy Water Components Test Reactor/SRS	Reactor vessel removed and disposed at the SRS LLW facility. Below-grade structure, including piping and equipment filled with grout/concrete, and a concrete cap installed over Containment Building footprint.	ISD
2007	Super Kukla Facility/NNSS	Reactor core and components removed. Lead shielding wall left in place and grouted to reduce worker risk from removal.	ISD
2012–2015	Experimental Breeder Reactor II/INL	Reactor and reactor vessel grouted in place instead of removing to protect workers from industrial hazards and radiological risks. Above-grade structure removed and cap placed over building footprint.	ISD

Table 1. State of ISD at other DOE sites (cont.)

Year	Facilities/DOE site	Note	Decommission option
2011	U Canyon/Hanford	CERCLA remedial action. To avoid complication of obtaining regulatory certification, decision to not use the canyon for disposal of anything other than what was in the CERCLA Area of Concern. Tank D-10 contained TRU (>100 nCi/ gram) and was removed from the AOC and packaged for WIPP.	ISD
2010	CPP-601/627/640 Fuel Reprocessing Complex/INL	CERCLA non-time-critical removal action WASTE: (a) the presence of chemically contaminated, abandoned isolated piping runs that could not be flushed, (b) waste remaining in the tank system at closure, and (c) lead that was impractical to remove, all resulting in hazardous constituents left in place. Radionuclide contamination and lead that was not removed during decommissioning.	CPP-601 closed as a land fill.
1992	BORAX No. 1/INL	Deliberate destruction test of BORAX Reactor created high radiation field. Reactor debris, uranium fuel residue, irradiated metal scrap, and contaminated soil/debris buried at reactor site.	ISD

AOC = area of contamination
BORAX = Boiling Water Reactor Experiment
CPP = Chemical Processing Plan (facility prefix used at INL)
INL = Idaho National Laboratory
LLW = low-level waste
NNSS = Nevada National Security Site
SRS = Savannah River Site
SWDF = Solid Waste Disposal Facility
TRU = transuranic
WIPP = Waste Isolation Pilot Plant

The selection of an ISD option at ORNL is not a new decision and has occurred at least three times in the past to significantly reduce worker exposure and environmental impacts during decommissioning of a facility. Examples of ORNL projects that were successfully decommissioned using ISD are summarized below:

- Building 3001 Canal—Water was used for shielding from residual radioactive material in the Building 3001 Canal. The water was removed from the canal and replaced with a controlled, low-strength material (CLSM) to provide shielding as required by the *Action Memorandum for Building 3001 Canal, Oak Ridge National Laboratory, Oak Ridge, Tennessee* (DOE/OR/02-1533&D2) and documented in the *Removal Action Report on the Building 3001 Canal at Oak Ridge National Laboratory, Oak Ridge, Tennessee* (DOE/OR/01-1599&D2). The CLSM provides stable shielding for residual contamination, replaces the water shielding, seals potential leaks, and is removable with traditional soil-removing techniques.
- Old Hydrofracture Facility—CLSM was used to fill the tanks, well risers, and stabilize the sediment in the impoundment to reduce worker exposure to contaminants as required by the *Action Memorandum for the Old Hydrofracture Facility Tanks and Impoundment at the Oak Ridge National Laboratory, Oak Ridge, Tennessee* (DOE/OR/01-1751&D3) and documented in the *Removal Action Report for the*

Old Hydrofracture Facility Tanks and Impoundment at the Oak Ridge National Laboratory, Oak Ridge, Tennessee (DOE/OR/01-1098&D2).

- Metal Recovery Facility—The below-slab drains were plugged with grout, and the below-slab portions of the canal and dissolver pit were filled with a mixture of low-strength cement and gravel to demobilize remaining contaminants in place as required by the *Action Memorandum for the Demolition of the Metal Recovery Facility, Building 3505, at the Oak Ridge National Laboratory, Oak Ridge, Tennessee* (DOE/OR/01-1843&D2) and documented in the *Removal Action Report for the Metal Recovery Facility, Building 3505, at the Oak Ridge National Laboratory, Oak Ridge, Tennessee* (DOE/OR/01-2000&D2/R1).
- Gunite and Associated Tanks— An interim remedial action was performed from 1997 to 2000 for removal of the tank liquids and residual solids to the extent practical, resulting in the removal of 423,000 gal of supernate/sludge containing over 88,000 Ci or radioactivity. A residual volume of material, estimated at 7581 gal with 3920 Ci or radioactivity, was left in the tanks to be stabilized. The Gunite and Associated Tank shells and risers were stabilized with grout/flowable fill as required by the *Action Memorandum to Stabilize the Gunite and Associated Tanks at the Oak Ridge National Laboratory, Oak Ridge, Tennessee* (DOE/OR-01-1957&D2) and documented in the *Removal Action Report for Gunite and Associated Tanks Stabilization Project, Oak Ridge National Laboratory, Oak Ridge, Tennessee* (DOE/OR/01-2010&D1).

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4. EVALUATION OF IN SITU DECOMMISSIONING AT MSRE

4.1. IN SITU DECOMMISSIONING APPROACH

ISD of the salt tanks within the drain tank cell will be performed by pouring engineered grout, or other containment media into the drain tank cell to entomb the tanks, piping, and associated equipment. Prior to entombment of the salt tanks, a getter material would need to be installed within the salt tanks to eliminate or reduce the generation of fluorine gas. If the fluorine gas cannot be eliminated by use of a getter material, a direct vent and purge system will need to be designed and installed. A direct vent and purge system would not prevent entombment of the salt tanks within the drain tank cell. Entombment of the reactor cell and all other below-grade cells would also be proposed as part of the ISD strategy for MSRE.

4.2. RATIONALE FOR CONSIDERING IN SITU DECOMMISSIOING

ISD with land use controls (LUCs) was the CERCLA remedial decision for the radiologically contaminated reactor complexes at the Savannah River Site (SRS). ISD technology stabilizes the contamination within the reactor structures to prevent direct human exposure, limits contaminant migration to groundwater, and prevents animal intrusion exposure to radiological and hazardous components. The application of ISD as a remedial action is typically viewed as a final remedy, but can be used as an interim remedy. All previous decisions to use ISD with LUCs at DOE sites involved public participation through the CERCLA process. Various assessments were used to determine the overall contaminant inventory within the reactors and demonstrate CERCLA acceptance. At SRS, these assessments were captured and evaluated within the *RCRA Facility Investigation/Remedial Investigation (RFI/RI) Report with Baseline Risk Assessment and Corrective Measures Study/Feasibility Study (CMS/FS) for the P-Area Operable Unit (U)* (WSRC-RP-2007-4032). Contaminant migration groundwater models were developed and used to support the base case (No Action) and the various ISD alternatives.

The ESD requires the fuel and flush salts to be removed and stored at an approved ORNL facility. However, neither the *Feasibility Study for Fuel and Flush Salt Removal from the Molten Salt Reactor Experiment at the Oak Ridge National Laboratory, Oak Ridge, Tennessee* (DOE/OR/02-1559&D2) nor the *Interim Action Proposed Plan for Fuel and Flush Salt Disposition from the Molten Salt Reactor Experiment, Oak Ridge National Laboratory, Oak Ridge, Tennessee* (DOE/OR/02-1601&D3) (Proposed Plan) included ISD as an alternative for consideration. Because ISD was not included as an alternative for the Fuel and Flush Salts ROD or ESD, an updated feasibility study (FS), including ISD, is appropriate for the following reasons:

- ISD has been successfully implemented at other DOE sites with acceptance by the public and the regulators.
- The *Record of Decision for Interim Actions for the Melton Valley Watershed at the Oak Ridge National Laboratory, Oak Ridge, Tennessee* (DOE/OR/01-1826&D3) (Melton Valley ROD) identifies the end use for the portion of Melton Valley containing MSRE as Industrial, with the associated LUCs. Since this area is in the middle of ORNL, contiguous to several capped burial grounds, it will remain under control of the Federal Government. Therefore, ISD is appropriate for the end use in the Melton Valley area.
- The public can participate in the evaluation of ISD for fuel and flush salts through the CERCLA decision process.
- If ISD can be demonstrated to be protective of human health and the environment, the remedy is less costly and eliminates the worker health and safety concerns with removal.

4.3. WASTE CLASSIFICATION

The remaining fuel and flush salts within the tanks at MSRE will not be characterized until a remedial action is selected. Based on historical records and process knowledge, it is assumed that the salts would be classified as transuranic (TRU) waste if removed for disposal. However, sampling of the salts will need to occur to accurately analyze the material and determine the correct waste classification. Under the assumption that the salts would be classified as TRU waste, the primary disposal facility selected would be the Waste Isolation Pilot Plant (WIPP) in Carlsbad, NM. Disposal of the MSRE salts at WIPP will require a determination that the salts are defense-related TRU waste to meet the licensing requirements for WIPP. Currently, there is no disposal container licensed for use at WIPP that matches the geometry or payload weight for intact removal and disposal of the entire FFT's and FDT.

If an ISD option is selected to leave the salts in place, then a performance assessment per 40 *CFR* Part 191 (*Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes*) may be required. The performance assessment would be used to demonstrate CERCLA acceptance and to assess the impacts on the public and the environment. It is likely that some of the requirements defined in DOE Order 435.1 (*Radioactive Waste Management*) could be applicable as well.

4.4. EVALUATION CRITERIA

Below is an evaluation of the feasibility of ISD for the fuel and flush salts at MSRE in terms of implementability, protection of human health and the environment, cost, and LUCs:

- **Implementability—**Pouring engineered grout, or other containment media into the cell to entomb the fuel and flush tanks can be implemented with conventional processes and equipment that will ensure worker safety. Challenges regarding ISD as a disposal option for the MSRE fuel and flush salts are based on the hazards associated with long-term entombment and the regulatory pathway.
  - — Hazards associated with long-term entombment: Prior to fuel removal, the long lifetime of the U-233 was the primary concern. While removal of the uranium fuel has reduced the anticipated radioactive longevity of total fission products and daughters from tens of thousands of years to approximately 1000+ years of activities above 10 Ci, a number of technical and regulatory hurdles remain, such as hydrogeological or seismic considerations, potential chemical interactions that may occur over time between the groundwater and soil at the site and the grout, and the effects of corrosive gas(es) produced by continued radiolysis of the salts. The grouting media must be resistant to the effects of radiation from known waste and fission products in the salts. The technical feasibility of removing obstacles from the drain tank cell and sealing openings through the cell walls, if it were to be used for secondary containment, will need to be considered during the design phase, but is considered feasible based on historical knowledge. A getter can also be added to the tanks to absorb any free F_2 gas generated or a direct vent off-gas treatment system. This will place the highly radioactive salt in a safe storage configuration, and the MSRE Facility can be demolished without removal of the salts.
  - — Regulatory pathway: From a CERCLA perspective, the Flush and Fuel Salts ROD (DOE/OR/02-1671&D2) will have to be revised with approval from the U.S. Environmental Protection Agency (EPA), the Tennessee Department of Environment and Conservation, and input from the public. In order to revise the Flush and Fuel Salts ROD, the FS will have to be revised to include ISD as an alternative. If ISD is found to be the preferred alternative, the revised ROD will rescind the requirement to remove the salt from MSRE. This CERCLA process for revising a ROD is well understood and has been performed at the Oak Ridge Reservation many times.

- **Protection of human health and the environment—**Protection of human health and the environment has the following two concerns:

  — Contaminant releases to groundwater and surface water: ORNL is in the White Oak Creek watershed, which drains approximately 16.8 km^2 (6.5 $mile^2$) of land. Surface water near MSRE is primarily drained by two unnamed drainages, one originating to the west and the other to the northeast of the facility. These drainages divert water into the westerly flowing Melton Branch south of MSRE. Melton Branch is one of the largest tributaries of White Oak Creek. Bedrock beneath MSRE is a dark calcareous shale, which is part of the Conasauga Group that underlies the Melton Valley. The Conasauga shale formation is heterogeneous and relatively low in permeability, although flow velocities of a few feet per week are possible in the soil/weathered overburden. The overburden averages 20 ft thick and consists of a blanket of topsoil generally less than 1 ft thick on top of weathered shale. Groundwater levels in the vicinity of MSRE are currently suppressed by pumping a sump located to the southeast of the reactor cell. The interaction with groundwater would be resolved by entombing the salt tanks and equipment in an engineered grout under an ISD option. Additionally, groundwater and surface water monitoring will be performed to ensure there have been no releases.

  — Airborne releases: The FS presents the exposure estimates for radiological and chemical contaminants at MSRE. The ecological risk assessment evaluated the potential for adverse effects on the environment from exposure to contaminants from the drain tank cell. Adverse effects to terrestrial plants and wildlife would be expected in the event of the failure of the off-gas system equipment. However, an ISD approach would require a direct vent system for the gasses or installation of a chemical getter that eliminates generation of the gas.

  — The human health risk assessment in the FS focused on a near-term release scenario and a long-term release scenario. The near-term release scenario assumed that MSRE was abandoned and that surveillance and maintenance activities had been discontinued. The potential release was assumed to involve a failure of the off-gas system, which resulted in a release of F_2 and UF_6 gases that had migrated and accumulated within the system. The exposure and risk were estimated for a receptor at an onsite location and an offsite location who was in the line of travel of the plume as it passed. For both locations, the estimated risk exceeded the EPA target risk range of 10^{-4} to 10^{-6}. The long-term release scenario involves water infiltration into the drain tank cell, followed by a criticality accident. Gaseous fission products produced during the criticality accident were assumed to escape the drain tank cell to the outside atmosphere where atmospheric dispersion would transport them to the receptor. For the onsite receptor, the risk exceeded the EPA target risk range, while the estimated risk for an offsite receptor was within the target range. The off-gas collection and treatment system or installation of a chemical getter would prevent these releases.

Although ISD does not reduce the remaining radiological inventory for MSRE, it may allow for the re-classification of the facility with regards to DOE orders that guide nuclear safety requirements. A nuclear safety analysis would be performed to determine if the facility can be re-classified as radiological and a Hazard Assessment Document prepared for approval. Once ISD is complete, implementation of the new safety basis documents can be performed. This re-classification would document the reduced safety hazard to workers and the environment.

- **Estimated Cost—**After defueling occurred in 2008, several alternatives were considered for disposition of the defueled coolant salts within the drain and flush tanks. Maintaining the facility as is for 50 years, entombment of the salt tanks, and in-tack removal and disposal of the salt tanks were options evaluated in the *Engineering Evaluation of Options for Molten Salt Reactor Experiment Defueled Coolant Salts* (DOE/OR/01-2496&D1). A comparison of alternatives that were presented in the engineering evaluation are provided in Table 2.

Table 2. Alternative cost comparison

Option	Alternative	Performance merits	Issues	Project duration/project cost in millions	End state after entire project duration
Keep salts at MSRE	Maintain as-is for 50 years	Technically feasible and easy to implement	Change in facility maintenance strategy and increase in long-term costs. MSRE remains a nuclear facility classification. Corrosion of equipment and piping need to be addressed. Evaluation to show salt tank integrity is adequate would need to be performed.	Not a project; maintenance for 50-yr duration $497 Cost	100-year-old facility has been maintained safe, operable, and habitable.
	ISD – Entombment of Fuel and Flush Salt Tanks	Technically feasible	Performance assessment and feasibility study will need to be performed. Approval through CERCLA process needs to be obtained. Installation of a getter material or off-gas system required to address fluorine gas generation.	36–63 months estimated for design and implementation of ISD Estimated cost $50 to $77	Salt and tanks are entombed in drain tank cell. Above-grade structure removed. Land use controls Implemented.
Remove salts and tanks from MSRE	Intact-Tank Removal of FDTs and FFTs, disposal at WIPP	No salt transfer required Reduces tank corrosion concerns	Disconnection and removal of tanks would require remote-handled technics and significant shielding for personal protection. No approved Type B disposal container for WIPP currently exist that accommodates geometry, payload, or shipping dose rate for FDTs and FFTs.	62–80 months estimated for development of new Type B container, removal, packaging, and disposal at WIPP. Estimated cost $110 to $180	Majority (not all) of radiological inventory removed from facility. Facility will require decommissioning and demolition.

- **Land Use Controls—**as stated previously, MSRE is located in Melton Valley. The Melton Valley ROD identifies the end use for the portion of Melton Valley containing MSRE as Industrial, with the associated LUCs. Since this area is in the middle of ORNL, contiguous to capped burial grounds, it will be under the control of the Federal Government. Therefore, ISD is appropriate for the end use in this area.

4.5. SUMMARY OF EVALUATION

A summary of advantages and disadvantages regarding the option of ISD for the MSRE salts is included in Table 3.

Table 3. ISD evaluation summary for MSRE

Criteria	Advantages	Disadvantages
Implementability	• ISD can be implemented technically and administratively. • CERCLA requirements to revise the ROD can be performed. • A performance assessment will evaluate long-term performance of ISD. • Public participation is required.	Land use controls will be required to maintain protectiveness.
Protection of human health and the environment	• The drain and flush tanks will be entombed in an engineered grout and protective of the environment. • The off-gas will be collected and treated or use of a chemical getter will be implemented. • The risk to the workers is less than removal. • Environmental monitoring will be performed to identify any release to groundwater.	The long-term performance of ISD is difficult to evaluate. The estimated risk from airborne releases may be outside the acceptable risk range if the off-gas system fails.
Cost and schedule	• Cost and schedule for ISD is less than previous alternatives.	None
Land use controls	• Land use controls are already defined in the Melton Valley ROD	Land use controls will be required to maintain protectiveness.

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5. CONCLUSIONS

The ISD approach has been successfully implemented at various DOE sites across the complex and has significantly reduced the risk to human health and the environment by avoiding removal of waste. It is recommended that complex-wide lessons learned and experience be utilized in development of an FS to evaluate ISD as an alternative for the MSRE fuel and flush salts. The ISD approach for MSRE involving entombment of the fuel and flush salts is deemed feasible, but additional engineering evaluations will need to be conducted to address the F_2 gas generation and potential removal of drain tank components. The evaluation team recommends initiating an FS through the CERCLA process to amend the Fuel and Flush Salts ROD.

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